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Developments in the transport of radioactive waste

Proceedings of a Seminar held in Vienna, 21–25 February 1994

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FOREWORD

The International Atomic Energy Agency's Safety Series No. 6, Regulations for the Safe Transport of Radioactive Material, is being revised with completion planned for 1996. One of the areas under review in Safety Series No. 6 is that of low specific activity materials and surface contaminated objects (LSA/SCO). Material classified as LSA/SCO for transport purposes is primarily radioactive waste, but can also include material at the front end of the nuclear fuel cycle and sealed sources.

Considering the status of the revision process, a Seminar on Developments in Radioactive Waste Transport was held. The objective of this Seminar was to gather the information needed to assist in the revision of Safety Series No. 6, and to promote a dialogue between operators and regulators of both the transport and waste management fields.

Amendments to Safety Series No. 6 under consideration aim at establishing regulatory provisions that maintain the requisite level of safety specified by the current version in a manner that avoids unwarranted restrictions produced for disposal or long term interim storage. These amendments may have a wide impact on a wide range of aspects related to waste management. Prior to implementing any changes to Safety Series No. 6, it is necessary to determine how the amendments can best fit with waste management plans and projections.

CONTENTS

Summary of the Seminar 9

IAEA ACTIVITY IN RADIOACTIVE WASTE TRANSPORT (Session 1)

The transport safety programme 13
H.A. Selling
Progress towards the 1996 Edition of the IAEA Regulations for the Safe Transport of Radioactive Material 16
J.H. Mairs
The IAEA's radioactive waste management programme 20
D.E. Saire
Safety standards for radioactive waste management 25
E. Warnecke
LSA/SCO provisions in IAEA Safety Series No. 6, Regulations for the Safe Transport of Radioactive Material 31
D.R. Hopkins

MEMBER STATE EXPERIENCE AND RECOMMENDATIONS FOR INTERNATIONAL TRANSPORT REGULATIONS (Session 2)

Contaminated objects in actual transport 43
M. Grenier, A. Laumond
Transport categories for radioactive waste 46
E.P. Goldfinch
Preliminary investigations on radiological criteria and requirements for the transport of LSA and SCO materials - GRS/IPSN/NRPM collaboration 52
F. Lange
Multiple containment for LSA and SCO wastes 54
M.H. Burgess
Application of the transport system concept to the transport of LSA waste 57
J. Lombard, P. Appleton, H. Libon, H. Sannen, T. Schneider
System certification for radioactive waste with application to LSA/SCO 62
R.E. Luna, J.D. Whitlow, D. Lillian, L.H. Harmon, F.P. Falci, D.R. Hopkins

MEMBER STATE EXPERIENCE WITH NATIONAL REGULATIONS (Session 3)

Transportation issues facing the international community 69
C. Haughney, E. Easton, C. Chappell, N. Osgood, R. Cunningham
Regulatory aspects of the transport of irradiating and alpha waste in France 74
C. Devillers, M. Grenier, J. Lombard, F. Mathieu
Regulatory aspects in the transport and disposal of low and intermediate level radioactive wastes from fuel cycle operations 78
T.N. Krishnamurthi
Regulations for the safe transport of radioactive materials in the Philippines 85
O.L. Amparo

WASTE TRANSPORT AND HANDLING (Session 4)

The planned integrated transport system for the deep repository in the United Kingdom 91
I.L.S. Gray
GNS — Transport system and handling practices for the transport of low and intermediate level wastes 98
G. Gestermann

LLW transport by IP-2 packaging 103
K. Tanaka
Transport of radioactive waste in Germany — A survey 108
U. Alter
Transport of radioactive waste in Dukovany nuclear power plant 112
J. Kulovaný
A plan of radioactive waste management in China 116
X. Wang
Handling and transportation of low level waste in Indonesia 119
W. Suyatno, S. Yatim
Problems related to the transport of low and intermediate level radioactive waste in the Republic of Croatia 123
S. Kučar-Dragičević
The transport of low and intermediate level waste from Spanish nuclear facilities 129
J.L. González Gómez, C.E. Marchal

WASTE GENERATION VOLUMES, CHARACTERISTICS, AND DISPOSAL REQUIREMENTS (Session 5)

Transport and disposal requirements for some selected waste shipments to the Konrad repository 135
F. Nitsche, F.W. Collin
Inventory and characteristics of current and projected low-level radioactive materials and waste in the United States of America 140
A. Bisaria, R.G. Bugos, R.B. Pope, R. Salmon, S.N. Storch, P.B. Lester
Radioactive waste assay to verify the fulfillment of waste acceptance requirements 143
L. Bondar, R. Dierckx, S. Nonneman, B. Pedersen, P. Schillebeeckx
Characteristics, conditioning and acceptance requirements on radioactive waste to be disposed of 149
P.W. Brennecke
The acceptance, before transport, of the waste radioactive packages on the disposal facilities of ANDRA 156
P. Lecoq
Initial estimates of samples and residues requiring transport arising from the United States Department of Energy's analytical services program 159
R.B. Pope, A. Bisaria, R.D. Michelhaugh, M.J. Conroy
Recycled scrap metal and soils/debris with low activity contents 162
A.W. Carriker

RISK ASSESSMENT (Session 6)

Probabilistic safety assessment for the transport of radioactive waste to a UK repository at Sellafield 171
P.R. Appleton
Safety analysis of the transportation of radioactive waste to the Konrad waste disposal site 177
F. Lange, H.J. Fett, D. Gründler, G. Schwarz
Risk assessment associated with the transportation of low and intermediate level radioactive waste to the Centre de L'Aube disposal facility, France 183
D. Raffestin, V. Tort, P. Manen, T. Schneider, J. Lombard
The RADTRAN 4 computational system for transportation risk assessment: A prototype for the information superhighway 187
K.S. Neuhauser

TRANSPORT AND WASTE PACKAGES (Session 7)

Optional multilayer cask for transportation of spent sealed neutron sources 193
E.E. Ahmed, F.A. Rahman
Status of the beneficial uses shipping system cask (BUSS) 200
H.R.. Yoshimura, R.G. Eakes, D.R. Bronowski
Packaging of radioactive waste in the United Kingdom for disposal at the deep repository . . 204
S.V. Barlow
Packagings for irradiating waste: Industrial experience and evolution of the Regulations 210
P. Malesys, A. Laumond, P. Dybeck
The Swiss concept: Container concept and the transport of L/ILW to the final repository . . . 214
J. Migenda
Multipurpose storage/transport/disposal packages for DOE nuclear low level wastes:
An emerging need and a regulatory challenge 218
M.P. Keane, D. Lillian, F.P. Falci, K.B. Sorenson, G. Hohnstreiter
Transport of low level wastes from COGEMA reprocessing plants 223
B. Desnoyers, C. Ringot
Design features of shipping containers for low and intermediate level wastes 228
K. Singh, R. Bhattacharya, K.N.S. Nair, K.K. Prasad
Development and evaluation of containers for radioactive waste disposal 233
B. Droste, P. Zeisler, H. Völzke, R. Rödel
Some aspects regarding the qualifications tests of packages used for transport and
storage of radioactive waste (low activity) in INR Pitesti 239
G. Vieru
Packaging design and qualification: The experience of the Centro de Desenvolvimento da
Tecnologia Nuclear/Comissão Nacional de Energia Nuclear 244
R.P. Mourão, S.T.W. Miaw
An industry standard for industrial packages 249
R.D. Cheshire, J. Higson

List of Participants 252

SUMMARY OF THE SEMINAR

A Seminar on Developments in Radioactive Waste Transport was held in Vienna from 21 to 25 February 1994. More than 80 experts from 28 Member States and international organizations attended. The Seminar was limited to low and intermediate level radioactive waste and spent sealed sources and did not include vitrified highlevel waste or irradiated or spent nuclear fuel.

Papers presented during the seven scientific sessions covered the following areas: IAEA activities in radioactive waste transport; Member State experience and recommendations for international transport regulations; Member State experience with national transport regulations; waste transport and handling; waste generation volumes, characteristics, and disposal requirements; risk assessment; and transport and waste packages. The IAEA is heavily involved in the revision of Safety Series No. 6 with completion planned for 1996. The proposed amendments to Safety Series No. 6 related to the transport of low specific activity materials and surface contaminated objects that will be tabled at the next Revision Panel meeting were also presented and discussed.

Currently radioactive waste is being transported, but the number and volume of shipments is expected to increase and the characteristics of some of the waste will change in the future, especially with decommissioning and facility remediation activities. Many examples of packages already in use for the transport of radioactive waste were presented. Because of the existence of the current generation of packages, a need was expressed for regulatory stability between what currently exists in Safety Series No. 6 and what is being proposed for the 1996 Edition. However, because of the changing nature of waste management activities, the materials that need to be transported will change and the regulations should try as best as they are capable to anticipate the changes. The development of the concept of transport systems for the 1996 Edition of Safety Series No. 6 was seen as a potentially useful method for handling wastes that result from decommissioning and remediation activities and for other unanticipated waste streams.

The proposed amendments to the current regulations for LSA/SCO materials were generally well received with two exceptions. First, the current requirement to control external exposure by a material dose rate limit was preferred instead of the proposed amendment of a package total activity limit. A total activity limit would reduce the capacity of the packages and would therefore result in the shipment of more packages. Second, the proposed amendment for Type A package requirements for LSA-II, LSA-III and SCO-II was not favoured. The adoption of this amendment was seen as resulting in an even wider difference than already exists between packages designed to IAEA requirements and those designed to ISO and UN requirements, especially for LSA-II liquids.

There was also support for compatibility between the requirements for the transport package and the waste package. However, it was recognized that the packages are designed for different environments and scenarios and total harmony would be difficult to achieve.

IAEA ACTIVITY IN RADIOACTIVE WASTE TRANSPORT

(Session 1)

Chairman

T. POLLOG
IAEA

THE TRANSPORT SAFETY PROGRAMME

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Abstract

The transport safety programme is one of the smaller technical subprogrammes in the Radiation Safety Section of the Division of Nuclear Safety, in terms of both regular budget and professional staff allocations. The overall aim of the programme is to promote the safe movement of radioactive material worldwide. The specific objectives are the development, review and maintenance of the Regulations for the Safe Transport of Radioactive Material, Safety Series No. 6, and its supporting documents Safety Series Nos 7, 37 and 80 and the assistance to Member States and international organizations in the proper implementation of the Regulations. The degree of implementation of the Regulations is high as was confirmed by a recent survey in the Member States. A preliminary overview of the results will be presented. The survey clearly showed that more assistance to Member States in the implementation of the Regulations is required. It is envisaged that resources be made available to strengthen this part of the programme. One of the important issues that emerged during the ongoing review/revision process is the transport of low-specific activity (LSA) material and surface contaminated objects (SCO). Many of the radioactive waste materials fall in one of these categories. The subject has gained substance because it is expected that in the next decade radioactive waste could become available in so far unprecedented quantities and volumes due to decontamination and decommissioning of nuclear facilities.

INTRODUCTION

The International Atomic Energy Agency's Regulations for the Safe Transport of Radioactive Material [1] recently celebrated their 30th anniversary. They continue to serve as the regulatory basis for both international and domestic transport in most of the IAEA's Member States. In the more than 30 years of their existence five comprehensive revisions were published to keep the Regulations abreast of major scientific and/or technological developments. As recommended by SAGSTRAM, the Advisory Group to the Director General on matters related to transport of radioactive material, comprehensive revision has evolved into a formal process of review and revision of the Regulations, involving a 10-year cycle, the current one culminating in a new edition in 1996. More information on this Continuous Review and Revision Process will be presented in the second paper on this topic.

THE IAEA PROGRAMME

The transport safety programme is one of the smaller technical subprogrammes in the Radiation Safety Section of the Division of Nuclear Safety, in terms of both regular budget and professional staff allocations.

A summary of the regular budget figures of the last two years and the estimates for 1994 for the Radiation Safety Section are summarized in Table I [2].

TABLE I. PROGRAMME AND BUDGET RADIATION SAFETY SECTION

No.	Subprogramme	Budget year		
		1992	1993	1994
H.1	Basic Radiation Safety Policy	702 000	687 000	737 000
H.2	Occupational Radiation Safety	977 000	709 000	888 000
H.3	Radiation Protection of the Public and the Environment	615 000	599 000	650 000
H.4	Safe Transport of Radioactive Material	392 000	538 000	569 000
H.5	Emergency Planning and Preparedness	442 000	478 000	499 000
H.6	Safety of Radiation Sources	635 000	517 000	549 000
H.7	Radiation Safety Services	404 000	410 000	434 000
H.8	Radiological Consequences of the Chernobyl Accident	- -	107 000	115 000
Programme H - Radiation Safety		4 167 000	4 045 000	4 436 000

The overall aim of the programme is to promote the safe movement of radioactive material worldwide. The specific objectives of the IAEA concerning the safe transport of radioactive material can be distinguished in three main areas:

1. *The maintenance of the Regulations*, which includes the development, the review and the updating of Safety Series No. 6 and its supporting documents;
2. *The implementation of the Regulations*, which includes assistance to Member States and co-operation with other international organizations in the proper implementation of the Regulations; and
3. *The establishment of Co-ordinated Research Programmes* which support both the maintenance of the Regulations and their implementation.

The international and most of the national transport of radioactive material is governed by the Regulations. The Regulations serve as the regulatory basis for all international mode-specific transport agreements as outlined in Table II.

BASIC CONCEPTS OF THE REGULATIONS

Either through those modal agreements or by direct incorporation or referencing, it is ensured that the Regulations are implemented worldwide in essentially the same way. The initial concept of the Regulations envisaged that they would not only be applied uniformly throughout the world but also that they would be multi-modal, i.e., that they would be

TABLE II INTERNATIONAL TRANSPORT ORGANIZATIONS IN THE SUBJECT AREA

Mode of transport	International/ regional organization	Name of Agreement/Regulations	Scope
Air	ICAO[1]	Technical Instructions for the Safe Transport of Dangerous Goods by Air	Worldwide
Air	IATA[2]	Dangerous Goods Regulations	Worldwide
Sea	IMO[3]	International Maritime Dangerous Goods Code	Worldwide
Road	ECE[4]	European Agreement concerning the International Carriage of Dangerous Goods by Road	Regional
Rail	OCTI[5]	International Regulations concerning the Carriage of Dangerous Goods by Rail	Regional
Inland Waterways	ECE[6]	European Agreement for the International Carriage of Dangerous Goods by Inland Waterways	Regional
Post	UPU[7]	Acts of the Universal Postal Union	Worldwide

[1]ICAO International Civil Aviation Organization
[2]IATA International Air Transport Organization
[3]IMO International Maritime Organization
[4]ECE UN Economic Commission for Europe
[5]OCTI Central Office for International Transport by rail
[6]ECE UN Economic Commission for Europe
[7]UPU Universal Postal Union

basically independent of the mode of transport or the particular conveyance carrying the radioactive material. In general, the IAEA has maintained the multi-modal nature of its Regulations.

According to the IAEA philosophy reliance on the package design is the main instrument to ensure safety during the transport of radioactive material. Consequently, package designs are required to meet certain performance standards which are graded according to the risk posed by the radioactive contents: the higher the risk the more stringent the design requirements. So far, the most robust packages are those which have been designed to withstand accidents (Type B packages). To demonstrate the ability to meet the design requirements, tests have been specified which simulate different conditions encountered during transport (routine conditions, normal conditions and accident conditions). The tests that have been laid down to represent accident conditions do not aim to simulate specific accidents or accident scenarios, but rather to produce the same kind and amount of damage that would result from real accidents. In principle, the tests take account of the different accident environments for each of the modes of transport. The validity of these concepts can be derived from the excellent safety record associated with transport of radioactive material. During more than 30 years annually tens of millions of packages containing widely varying amounts of radioactive material have been transported all over the world by all modes of transport. No accidents have occurred which entailed significant releases of the radioactive contents or resulted in significant exposures of the transport workers or the members of the public. This is an accomplishment that should not be overlooked in the development of new regulations governing the transport of these materials.

IMPLEMENTATION

The mechanism of implementation through mode-specific international agreements, which, contrary to the IAEA Regulations have a binding character to Member States, ensures that the degree of national implementation of the Regulations is high. This was again confirmed by a recent survey in the Member States. In this survey a questionnaire was circulated to all IAEA Member States with the request to provide the IAEA with information on the status of the implementation of the latest edition of the Regulations. Although the high response by Member States and the high degree of implementation of the transport regulations is gratifying, there is no reason for complacency. A significant number of the responding Member States have indicated the need for assistance in one or another form by the IAEA.

Additionally, a preliminary assessment of the results of the survey shows an increasing tendency for mode-specific and national variations to the Regulations, which could give rise to some concern because it will eventually affect the expeditious movement of the radioactive cargo.

A matter for even greater concern form those countries that did *not* respond to the questionnaire. Their number amounts to about 50. The geographical distribution of these countries can be construed to consist of two main clusters. The first cluster is located in the eastern part of Europe plus the new countries emerging from the former Soviet Union, the second cluster is formed by the African countries. It is known that in the first cluster of countries transport of radioactive material occurs on a regular scale. Less is known, however, about the regulatory framework governing these transports and whether the Regulations are being implemented. Even less information is available on the situation in the African region. It is certain that radioactive material is being used in many countries and it seems logical to assume that use of radioactive sources or material will entail transport as well. From the viewpoint of the IAEA it is unsatisfactory not to know whether there is any involvement by a national competent authority in these shipments, whether packaging requirements have been met and if so, whether and which version of the Regulations applies.

Partly as a response to this survey the IAEA intends to initiate a more active programme aimed at assisting Member States to properly implement the Regulations. Already efforts have been undertaken to increase the frequency of training courses. The frequency for training courses which is currently envisaged as feasible with the available resources amounts to one annually, with alternating regional and interregional courses. Also the publication of two important Safety Series documents is envisaged in 1994. One document, on compliance assurance gives guidance on the structure and the tasks of competent authorities and consequently addresses the national competent authorites directly. The other document, on quality assurance, gives guidance to all users of the Regulations, since the 1985 Edition of the Regulations requires that all activities related to the transport of radioactive material should be covered by a quality assurance programme. It is particularly important for small companies staffed with personnel who have little or no training in quality assurance.

Finally, in the 1995/1996 programme and budget a further strengthening of the technical co-operation activities is foreseen by setting up a mechanism for holding peer reviews in Member States on the implementation of the Regulations on a voluntary basis. Similar services in other areas are already in existence at the IAEA.

INFORMATION SERVICES

The exchange and dissemination of information is one of the statutory functions of the IAEA. In the area of transport of radioactive material it aids national competent authorities in Member States in assuring regulatory compliance. The information services currently available are the following:

- Training;
- Co-ordinated research;
- Information and guidance material;
- Databases.

Training

It is essential that staff involved in transport operations are adequately trained. For transport safety the need for three levels of training were identified which differ in scope and depth. The IAEA provides training for the first and the second level, including senior staff at competent authorities and managers responsible for transport at shipping companies. As note before, the IAEA has recently increased its training activities in the subject area. It intends to maintain a scheme of one training course a year, by alternating regional and interregional training courses. In support of these activities the IAEA has published the training curriculum and lecture notes used at an interregional course held at Bristol in 1987 as a template for future courses [3]. Translations of the course material into Spanish and Russian are planned to be published in 1994.

Co-ordinated research

The IAEA promotes research in Member States through Co-ordinated Research Programmes (CRPs). These CRPs should either support the development of the Regulations or assist the Member States in the implementation of them. At present there are two CRPs running while two others are in the process of being set up. In addition to that the IAEA aims to exercise a wider role in fostering the exchange of information by publishing abstracts of research recently concluded and in progress in the area of radioactive material transport. Publication of the second edition of the Transport Safety Research Abstracts is due in the first semester of 1994 [4]. This edition is established with INIS, the International Nuclear Information System, to take advantage of their experience and protocol for data processing.

Information and guidance material

In 1993 the information brochure "Transport of Radioactive Material" was published and distributed widely to regulatory and other government offices, and to the nuclear industry in Member States. It is targeted at policy makers and senior officials and is available in four official languages of the IAEA.

In support of the training manual the development of visual aids material (film on transport safety, transparancies, slides, scale models, posters, etc.) is envisaged, with the aim to make it available as a training kit to Member States for national courses.

Databases

Information is collected in the areas of national competent authorities, package design certificates, events in radioactive material transport, research and development, shipments, and exposure data. The main purposes of the data collection activities include:

- To serve as regulatory aids to the national competent authorities responsible in the Member States for the transport of radioactive material, both internationally and nationally;
- To foster the exchange of information among competent authorities and international modal organizations;
- To support the continuous review and revision process of the transport Regulations and their supporting documents; and
- To assist in answering public concerns.

TRANSPORT OF RADIOACTIVE WASTE

There may be an increasing need to transport new types of materials or materials occurring in unprecedented volumes and quantities. As many nuclear power stations and research reactors approach the end of their economic life, wastes from decommissioning or decontamination of nuclear facilities could arise in massive amounts, depending on national waste management policies. The current concept of transporting radioactive material, i.e. in a packaged form, and consequently with a minimum of operational controls might prove to be inadequate.

It is of paramount importance that the new Regulations do not inhibit the movement of those materials but remain flexible enough to accommodate unconventional cargo. A systems approach might be considered as a better alternative. To exchange views and experience between waste management experts and transport safety experts on this matter is one of the objectives of this seminar.

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PROGRESS TOWARDS THE 1996 EDITION OF THE IAEA REGULATIONS FOR THE SAFE TRANSPORT OF RADIOACTIVE MATERIAL

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Abstract

A well established review and revision process is now in place whereby the IAEA Regulations for the Safe Transport of Radioactive Material undergo a comprehensive revision every ten years. The next Edition is scheduled for publication in 1996. Within this process, a large group of experts, known as the Revision Panel, provides instructions on the drafting of the revised Regulations and their supporting documents. The purpose of this paper is to outline the review and revision process and to give a brief, general account of the progress made, especially focussing on those issues that are of interest to transporters of waste. This paper will deal in the main with those issues that have been processed by the Panel, but will also refer to some topics, relevant to the Seminar, that will reach the Panel in the future.

1 INTRODUCTION

To keep the IAEA's Transport Regulations[1] abreast of recent scientific and technological developments, a major review of the Regulations takes place at intervals of approximately ten years. This time interval allows international organizations, such as the International Maritime Organisation and the International Civil Aviation Organisation, as well as Member States to schedule rule making activities regarding the revised Regulations while maintaining an acceptable level of regulatory stability. The next Edition of the Regulations is due to be published in 1996.

The purpose of this paper is to give a brief, general account of the progress made in the review and revision process, especially focussing on those issues that are of interest to transporters of waste. However, the paper sets aside the matters related to Low Specific Activity (LSA) material and Surface Contaminated Objects (SCO) which are presented in a separate paper.

Within the frame of revising the Regulations, a large group of experts, known as the Revision Panel, instructs the Secretariat on the drafting of the revised Regulations and their supporting documents[2,3,4]. Four meetings of the Panel have been planned for the comprehensive review of the As Amended 1990 version of the Regulations. To date, this Panel has met twice and has instructed the Secretariat to produce the first draft of the 1996 Edition of the Regulations It should be noted that this entire document is open to amendment at future meetings of the Panel. Therefore, the draft should be neither quoted nor referenced beyond the scope of revising the Regulations

Only fully developed proposals are tabled at meetings of the Panel. In the case of major changes to the Regulations, this requires the preparation of detailed proposals for amendment that have been debated previously. Consequently, there are a number of issues which have been deferred to later meetings of the Panel, pending a review by other groups in the meantime. This paper will deal in the main with those issues that have been processed by the Panel, but will also refer to some topics relevant to the Seminar that will reach the Panel in the future.

2 AN OUTLINE OF THE REVISION PROCESS

This section attempts to explain how changes to the Regulations are made.

An aim of the review and revision process is to ensure that identified problems and proposed amendments to the Regulations are raised at an early stage and dealt with systematically. Indeed, in 1986 the Director General's Standing Advisory Group on the Safe Transport of Radioactive Material (SAGSTRAM) laid down a detailed set of procedures to promote such an approach. This set of procedures became known as the Continuous Review and Revision Process.

Following the publication of comprehensively revised Regulations, competent authorities may become aware of problems with the Regulations and the supporting documents. Sometimes the competent authorities are informed of such difficulties by others using the Regulations, i.e. package designers or transport operators. At intervals of two years during the Continuous Review stage, the Agency requests Member States to provide proposed amendments to the Regulations and to identify any unresolved problems with the Regulations. Often, competent authorities act as a focal point within Member States for submitting returns to the Agency. Proposed amendments must be fully developed, including a clear statement concerning the deficiency in the current version of the Regulations (or its supporting documents), a justification for the amendment, an indication of the urgency for reform, and a full draft for both the changes to the Regulations and the supporting documents. Less input is expected in the case of identified problems. A basic statement of the problem with a discussion of how the problem might be studied and resolved through the IAEA can be sufficient.

The input received by the Agency is presented to the Review Panel that processes any minor amendments and changes of detail using the procedure established by SAGSTRAM. This procedure leads to the publication of Supplements or As Amended versions of the Regulations. The major changes and unresolved problems are referred to SAGSTRAM for advice on further action and scheduling within the Agency's work programme on safe transport. In the event of the system working well, these topics will re-surface at the Revision Panel at a mature stage of development and resolution.

The Continuous Review of the 1985 Edition of the Regulations culminated in the publication of the As Amended 1990 version which contains only minor amendments and changes of detail from the original 1985 Edition. In 1994, the Agency is well advanced in the second phase, the Revision Process, which will lead to a comprehensively revised set of Regulations in 1996.

3 DRAFT AMENDMENTS TO THE REGULATIONS AFFECTING THE TRANSPORT OF WASTE

This part of the paper discusses some of the likely amendments that can be expected to impinge on the transport of low- and intermediate level radioactive wastes. An area of general importance is the new recommendations on radiological protection which affect a number of general provisions in the Regulations, as well as the content limits for different types of package, the allowable release rates from damaged packages and also the issue of exemption from the Regulations for small quantities of material or material with very low mass activity concentrations. Waste can arise in large quantities with a consequent demand for large, heavy packages. These large packages may be subject only to the tests simulating normal conditions of transport which were originally devised with much smaller packages in mind, i e. small sources or radiopharmaceuticals. An amendment to relieve the severity of the drop test to demonstrate the ability of large packages to withstand normal conditions of transport is being considered Measures to reduce costs while maintaining an appropriate level of safety should be of interest to all transport operators. In this regard, there is a proposal to include intermediate bulk containers within the Regulations. Currently, many packages designed for the transport of low and intermediate wastes do not require competent authority approval of the design. The question of whether they should attracts considerable debate within the Revision Process. Lastly, a sub-set of waste arisings will be fissile material and it is interesting to note the proposal to create a criticality safety index that is separate to the transport index for radiation protection.

3.1 Incorporating the revised basic safety standards

One of the major topics being considered in the Revision Process is the incorporation of the new Basic Safety Standards. The Standards[5] are being revised to reflect the consensus that can be drawn from the latest recommendations of the International Commission on Radiological Protection[6]. The revision of the Regulations is an activity that is taking place in parallel with the review of the Basic Safety Standards. However, it is important for the Standards to be published first because the Regulations call upon the Standards as a general provision for radiological protection. The anticipated completion of the Standards later in 1994 is timely in view of the plans to produce the 1996 Edition of the Regulations.

3.1.1 General Provisions

The Revision Panel has accepted, with just a few open points remaining, new text for the General Provisions on Radiation Protection. Some important changes have been recommended, including the need to establish Radiation Protection Programmes (RPP's) for the transport of radioactive material. RPP's will serve to emphasize the importance of the General Provisions which provide the justification for maintaining the current regulatory limits for radiation levels around packages and conveyances. Dose assessment programmes for occupational exposures arising from transport operations will be prescribed on the basis of likely annual doses. Workplace monitoring is required for exposures expected to be in the range of 1-6 mSv/y. Individual monitoring is required for exposures likely to exceed 6 mSv/y. To assist operators in estimating the exposure of their workforce, advice has been provided that correlates exposure with the number of packages handled and the radiation level at one metre from the packages.

3.1.2 Exclusion and exemption

The third meeting of the Revision Panel will be asked to consider the important topics of exclusion and exemption. It is hoped that the Standards, especially the appendix containing the exemption values, will have been finalized before June 1994 when the final meeting of the Technical Committee on the impact of the revised Standards on the Regulations will be held. A detailed set of proposed amendments should emerge from this Committee that can be tabled at the Revision Panel meeting. Since the appearance of the exemption values currently contained in the draft Standards, no Agency meeting has discussed exemption in the context of transport. However, consultant groups that met previously have stressed the benefit of harmonization between regulations. In particular, noting the importance of consistency between the Standards and the Regulations Nevertheless, adopting the values for exemption presented in the draft Standards will be a major change to the Regulations. The current Regulations define radioactive material as any material having a specific activity greater than 70 kBq/kg. The draft Standards choose a radionuclide specific approach which leads to derived exemption values ranging several orders of magnitude, spanning 70 kBq/kg in the case of activity concentration. The draft Standards also present exemption values for activity quantities (Bq). In conclusion, the impact of introducing the exemption values in the draft Standards into the Regulations will only become known when the values, together with the method of their impementation, are given a wider review. Clearly there will be a greater impact on some practices than others. The potential increase in the number of radioactive shipments caused by more restrictive exemption values for some radionuclides (i.e. alpha emitters and ^{60}Co) will be offset by the more relaxed values for other radionuclides (i.e. soft beta emitters).

3.1.3 A_1/A_2 Values and the Q System

Another topic that will be debated by the Technical Committee in June before reaching the Revision Panel is the Q System used to derive A_1 and A_2 values in the Regulations. The fundamental assumptions in the Q System constrain the detriment to an individual in the event of serious damage to a single Type A package by restricting the dose to the order of 50 mSv. The A_1 and A_2 values are activity quantities, calculated for each radionuclide, that basically determine the limit on contents for different package types. The impact of the new Standards on the Q System is limited in extent following the decision to place the Q System in the domain of potential exposures. Potential exposures are not expected to be delivered with certainty, and can result from an accident or events of a probabilistic nature. Since potential exposures are not subject to the dose limits applying to normal exposures (20 mSv a^{-1}, in general), the reference dose of 50 mSv can continue to be used in the context of the Q System. However, a group of specialists are calculating revised A_1 and A_2 values that are based on complete spectral emissions from radionuclides and also taking into account new radiation weighting factors, new tissue weighting factors and the latest metabolic models incorporated into the new Standards. This work will lead to proposals for new A_1 and A_2 values which will in turn stimulate a debate on the merits and drawbacks of introducing changes to the Regulations. Maintaining the scientific rigour of the Q System is important for presenting the Regulations as being well defined and abreast of current thinking in radiological protection. But, the Q System contains some grossly simplifying assumptions in terms of release rates from damaged packages and intake factors for which the uncertainties are large compared with the fluctuations caused by the factors being analysed by the group of specialists. A balance will have to be found between the costs of introducing new A_1 and A_2 values, which will be difficult to assess, and the unknown costs of allowing the Q System to lag behind current recommendations.

3.2 Drop test configuration for normal conditions of transport (limiting secondary impacts)

The first Revision Panel considered a proposal to limit the effects of secondary impacts from the free drop tests of large packages undergoing the tests representing normal conditions of transport. The meeting agreed to the proposal in principle, but was unable to determine the means of achieving the objective. An example of the problem arises with the testing of some designs of freight container, whereby the most damaging drop orientation may be an inverted corner drop. Under such an orientation, it is possible for other parts of a large package, less prone to damage, to undergo a greater free drop distance than prescribed by the Regulations that might result in the design failing the tests for withstanding normal conditions of transport. It has been argued that the scenario of an inverted corner drop for a freight container is an accident rather than an incident that could be associated with normal conditions of transport. The second Revision Panel decided that there is a need to control the package orientation to normal situations, at least in the case of large packages. For normal orientation it is assumed that the axes of packages do not deviate from a vertical or horizontal plane by more than about 10°. Although this angle is still an open point, the meeting agreed to make an amendment such that the height of the centroid of the specimen shall not exceed 120% of the specified free drop distance.

3.3 Intermediate bulk containers

The Revision Panel has agreed to include intermediate bulk containers (IBC's) within the Regulations. As with tanks and freight containers, they are an allowed packaging type in the UN Orange Book[7] (see Chapter 16). It was noted that IBC's offer the potential for cost savings over the use of conventional packaging with suitable safety properties for the shipment of radioactive material. An IBC is a rigid, semi-rigid or flexible packaging that has a capacity of not more than 3 m^3; is designed for mechanical handling and is resistant to the stresses produced in handling and transport, as determined by performance tests.

3.4 Competent authority approval for designs of Type A, IP-2 and IP-3 packages

At this stage the Revision Panel is unable to resolve the question of whether competent authority approval is appropriate for designs of Type A, IP-2 and IP-3 packages. There was agreement that compliance with the Regulations must be assured for all packages, but the best method for achieving this goal caused the meeting to be divided between the following options.

- competent authority design approval
- certification by manufacturers, coupled with quality and compliance assurance measures
- registration of designs.

The matter has been deferred to a Technical Committee meeting on compliance assurance that is now planned to take place in 1995.

3.5 Creating two package indexes

Several proposals to create two separate package indexes, one for radiation protection (based on the radiation level at one metre) and one for criticality safety (based on the allowable number of packages that can be transported together) are being considered by the Revision Panel. The Revision Panel provisionally accepted the idea subject to a review of the detailed regulatory amendments that will be needed to give effect to the decision. Subsequently, a consultants group has met to discuss the proposals and their report which includes the detailed text changes lends further support to the idea. The consultants group observed that the draft text simplifies the Regulations and does not change concepts such as the form of the package label. Indeed, the proposed changes introduce clarity with anticipated benefits for training, operations and administration

4 DRAFT AMENDMENTS TO THE REGULATIONS AFFECTING THE TRANSPORT OF OTHER MATERIAL

Briefly discussed in this section are three topics which may not be of direct relevance to transporters of waste, but which have demanded a great deal of attention in the revision process. The proposals to introduce a new package type into the Regulations and to include specific provisions for uranium hexafluoride are probably the biggest changes to Regulations envisaged for the 1996 Edition. The matter of reprocessed uranium is of considerable commercial importance.

4.1 Type C packages

Agreement was reached, at the Revision Panel, on the principle of requiring a more robustly designed package for certain high-activity shipments transported by air. This proposed new package type is called a Type C package. Many of the requirements for Type C packages recommended in IAEA-TECDOC-702[8] were accepted into the first draft. However, solutions to the some contentious issues remain elusive. These contentious issues include establishing;

- conveyance limits for radioactive material in non-Type C packages (i.e. pre-empting the sub-division of the payload into separate packages so as to fall below the package content threshold requiring Type C packages);
- 'super special form' for material that is not dispersible even under extra-severe accident conditions (i.e. ^{60}Co sources can be solid metal and Type B packages may afford adequate protection, even for very large sources); and
- additional requirements for fissile material travelling by air in non-Type C packages (i.e. the package content threshold requiring Type C packages is unrelated to criticality safety so perhaps all fissile quantities travelling by air need additional protection).

Beyond these issues there is recognition that NUREG-0360[9] is more stringent than the requirements drafted in the Regulations. NUREG-0360, inspired by political motives, seeks almost absolute protection irrespective of the probability of occurrence of any accident. If the arguments behind NUREG-360 prevail, and some Member States withhold support for the technical consensus on Type C package requirements, there may be little chance of adoption by the other Member States. A Technical Committee is planned for August 1994 that will seek a breakthrough to these issues in time for the third Revision Panel.

4.2 Packaging requirements for UF_6

The Revision Panel has accepted most of the regulatory provisions for the transport of UF_6, as presented in IAEA-TECDOC-608[10], Interim Guidance on the Safe Transport of Uranium Hexafluoride. It was recommended that all of the requirements would be placed into a new Section VIII of Safety Series No. 6. This avoids complicating the other Sections of the publication with requirements applicable to a single material. It also avoids any legal difficulties associated with annexes to regulations. The decision to draft Regulations for a specific material reflects the importance of UF_6 within the nuclear fuel cycle, the very large quantities being shipped and the peculiar physical and chemical properties of the material.

The Revision Panel dealt with a number of difficult items concerning UF_6. Namely, cylinder pressure test requirements, the specification of the thermal test, criticality safety, and the need for competent authority design approval. The Revision Panel will need to re-visit the issue of the thermal test specification on completion of the Agency's Coordinated Research Programme that is addressing this point. The resource implications of competent authority approval for designs of UF_6 packages is being reviewed by the competent authority of the United States upon whom the main burden could fall.

4.3 Provisions for reprocessed uranium

Increasingly reprocessed uranium is being used in nuclear fuel fabrication. Compared to unirradiated uranium, the slightly different isotopic composition of reprocessed uranium, together with trace impurities, give rise to radiation levels and radionuclide inventories that are generally higher. As a result, the packages used for the transport of unirradiated uranium are not necessarily suitable for the transport of reprocessed uranium compounds. A revision to the definition of unirradiated uranium for the purposes of the Regulations and an allowance to take account of chemical form in deriving A_1 and A_2 values will relieve some of the regulatory constraints placed on the transport of reprocessed uranium. However, in taking account of the chemical form, due allowance must be given to any changes in chemical form that may result from accident conditions.

5 THE NEXT STEPS

The Revision Process advanced significantly at the second meeting of the Revision Panel. It provided the groundwork for developing the first draft of the 1996 Edition of Safety Series No 6. While not all issues have been resolved, most of them do have identified courses of action to reach closure by the end of this revision process It is envisaged that the Revision Panel will convene on two more occasions to complete the preparation of the 1996 Edition of Safety Series No. 6 and its supporting documents. The third

Revision Panel is due to meet in Vienna, 10-14 October 1994, with the fourth meeting expected in 1995

Experience shows that major issues are very difficult to resolve in a single meeting. This is due, in part, to the wide diversity of experience among Member States in implementing the regulations It has therefore been an objective of the revision process to have all major issues addressed by at least one Revision Panel before the final meeting in 1995

The first draft of the 1996 Edition of the Regulations has been distributed for information to members of the Revision Panel and other parties expressing an interest in furthering the work of the revision process. The Secretariat intends to produce the second draft after receiving instruction from the third Revision Panel It is hoped that the second draft will reflect sufficient technical consensus to merit sending the document to Member States for comment.

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THE IAEA'S RADIOACTIVE WASTE MANAGEMENT PROGRAMME

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Abstract

The IAEA's radioactive waste management programme is presented The paper illustrates key activities in the various elements that integrate into the programme Particular attention is given to the areas of information exchange, development of international standards, technical assistance, advisory services and special project activities.

I. INTRODUCTION

The International Atomic Energy Agency was formed to " accelerate and enlarge the contribution of atomic energy " in fostering the peace, health and prosperity of its Member States Today, with 118 Member States, these words which have been taken from the formulation charter of the Agency as given in the statute signed on the 26 October 1956, remain as valid as ever as the IAEA pursues its programme objectives.

In fostering the use of atomic energy for nuclear power and nuclear energy applications, the Agency has recognized the important linkage between such promotional activities and the need to establish safe systems for the handling, treatment, conditioning, transportation and disposal of radioactive waste that result from the use of atomic energy This linkage is important to recognize very early in the use of the atom to ensure radiological safety for occupational workers and the public and to avoid accidents or unnecessary releases of radioactivity to the environment This linkage is also of course vital for shipments of radioactive waste, as safe systems following guidelines which have international consensus remains a firm requirement for the nuclear community

The IAEA as an international organization has the role of fostering co-operation and co-ordination of activities that can best be handled at the international level. Within the area of radioactive waste management, the role of the Agency has been well established and activities which integrated into the IAEA's waste management programme can be characterized into seven main elements as provided below

1 Collect, review and publish up to date technical information in the form of Technical Reports Series and Technical Documents

2 Establish, reach consensus and publish Safety Standards, Guides and Practices for the safe management and disposal of wastes

3. Provide a forum for the dissemination and exchange of information at international conferences, symposia and seminars.

4. Sponsor and co-ordinate research and development through co-ordinated research programmes.

5. Provide technical assistance projects and training opportunities for developing Member States.

6. Provide advisory and peer review services.

7. Special Projects.

The allocation of resources to these elements is continually under review as the Agency's waste management programme is dynamic in nature, following the changing needs of Member States. With guidance from the International Radioactive Waste Management Advisory Committee (INWAC), a committee established by the Director General to give advice to the Agency in the formulation of its waste management programme, the activities of the programme are directed to focus on the diverse and different status of Member States' nuclear energy programmes. To illustrate the scope and nature of the Agency's waste management programme, a short summary of major activities, that have been recently undertaken in the seven elements listed above, follows.

II. PUBLICATION OF TECHNICAL INFORMATION

The level of activity in the area of dissemination of information can be reviewed by summarizing the number of publications over the past five years and indicating what area of waste management they apply to. Over the period 1989-1993 a number of interesting observations can be made:

- Covering the field of radioactive waste management, including decontamination, decommissioning and environmental restoration, a total of 76 technical publications was distributed to Member States. This averages to about 15 documents per year.

- Document production rates for the key waste management categories over the period were:

  - General: 11
  - Handling, Treatment, Conditioning and Storage of Radioactive Waste: 29
  - Radioactive Waste Disposal: 11
  - Radiological and Environmental Effects of Waste Disposal: 12
  - Decontamination and Decommissioning of Nuclear Installations: 13

Reviewing the above data it can be seen that a significant effort was made in the dissemination of the latest state-of-the-art technical information to Member States. Considering that the majority of the Agency's Member States generate waste only from nuclear applications and have not reached the point of active involvement in waste disposal, it is no surprise that the waste handling, treatment, conditioning and storage category received heavy emphasis. It is of course difficult to judge how successful the Agency's activity in the dissemination of technical information in the field of radioactive waste management has been. However, it is interesting to note that several publications have to be reprinted because of demand. While it is not possible to provide details of the documents published, additional information can be obtained from the IAEA pamphlet "International Atomic Energy Publications - Nuclear Power, Nuclear Fuel Cycle". Before moving on to the next element I would like however to direct the attention of the participants of this meeting to recent publications which have strong links with the transportation field; namely,

- Containers for packaging of solid low and intermediate level radioactive waste [1], and

- Interfaces between transport and geologic disposal systems for high level radioactive waste and spent nuclear fuel [2].

III. INTERNATIONAL STANDARDS

The Agency has been asked by Member States to demonstrate that there is international agreement in the Standards, Principles and Criteria for the safe management and disposal of radioactive waste. To implement this request the RADioactive WAste Safety Standards (RADWASS) programme [3] was initiated in 1990 to:

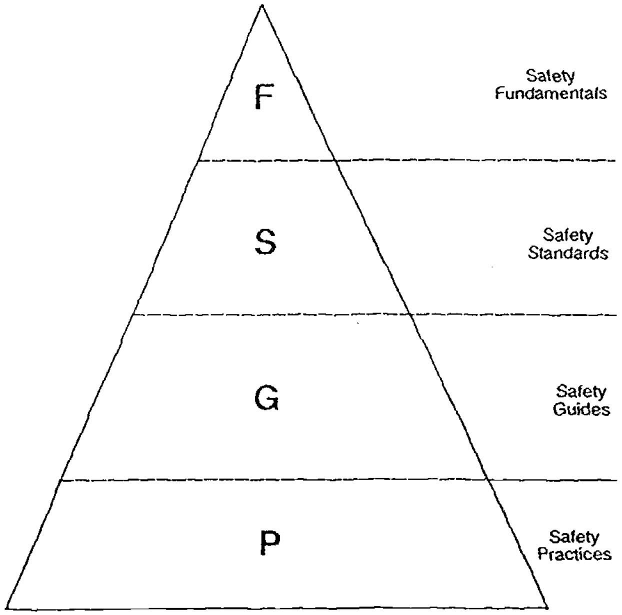
- document existing international consensus in the approaches and methodologies for the safe management of radioactive waste;

- create a mechanism to establish consensus where it does not exist, and

- provide Member States with a comprehensive series of internationally agreed upon documents to complement national standards and criteria.

Since the RADWASS programme will be presented in detail in a later paper, I will only provide some information of the structure and content of the programme.

Within RADWASS a set of safety documents will be produced following the hierarchical structure shown in Figure 1. The highest level is comprised of a single Safety Fundamentals document which provides the basic Safety Principles and Criteria that must be incorporated into a national waste management programme. The lower levels, in descending order, comprise of Safety Standards, Safety Guides and Safety Practices. The RADWASS programme is divided into six subject areas, with each subject area headed by a Safety Standards (Fig.1). Implementation of the RADWASS



Subject areas:

1. Planning for a national radioactive waste management system
2. Pre-disposal management
3. Near surface disposal
4. Deep geological disposal
5. Management of radioactive waste from mining and milling ore
6. Decommissioning

FIG. 1. RADWASS structure and publication areas.

programme began in 1991 and, at present, it is planned that 55 Safety documents will be published. It is anticipated that several of the top level priority documents will be published by the end of 1994 and that the entire programme will be completed by about 2001 [4]. RADWASS is a very ambitious programme and its success depends on the Agency having the full co-operation and support of the international waste management community.

IV. INTERNATIONAL MEETINGS

Another mechanism of information exchange used by the IAEA are international meetings. As is the case for this Seminar, the IAEA often organizes such meetings. Recent examples in waste management include:

- International Symposium on Geologic Disposal of Spent Fuel, High Level and Alpha Bearing Waste, October 1992, Antwerp, Belgium;
- Interregional Seminar on Ageing, Decommissioning and/or Major Refurbishment of Research Reactors, May 1992, Bangkok, Thailand;
- International Seminar on the Storage and Disposal of Low Level Radioactive Waste, October 1991, Paris; and
- International Symposium on Safety Assessment of Radioactive Waste Repositories, October 1989, Paris.

The next international seminar planned for waste management will be held in China in October of this year on the subject of "Radioactive Waste Management Practices and Issues in Developing Countries". The IAEA also co-operates with other organizations that are arranging international meetings. Our co-operation usually includes active participation on technical programme committees, session chairing and sponsoring the attendance of a significant number of scientists from developing countries. Recently, the Agency was involved in Waste Management '93 (Tucson, Arizona), SAFEWASTE '93 (Avignon) and the Fourth International Conference on Nuclear Waste Management and Environmental Remediation held in Prague in September 1993.

V. FOSTERING R & D

Agency activities in supporting R&D in radioactive waste management is centered on the establishment of co-ordinated research programmes (CRP). A CRP is implemented when a number of Member States identify a subject of common interest for performing research and sharing results and experience. The research is conducted at laboratories or institutes within each participating country and the results of the work is reported periodically at Research Co-ordination Meetings attended by investigators holding research agreements or contracts with the IAEA. Current CRPs underway or planned, that may be of interest to participants attending this meeting, are listed below:

- Performance of high level waste forms and packages under repository conditions;
- Waste treatment and immobilization technologies involving inorganic sorbents;
- Methods for extrapolating short term test periods to time periods required for waste isolation;
- Decontamination technology;
- Validation of models for the transfer of radionuclides in the environment;
- Safety assessment of near-surface radioactive waste disposal facilities.

The final research report from investigators participating in a CRP is usually published by the Agency in the form of a Technical Document. For example, results of a CRP on decontamination and decommissioning of nuclear facilities, conducted over the period 1989-1993, was reported in TECDOC-716 published in 1993 [5].

VI. TECHNICAL ASSISTANCE

A large number of the Agency's Member States are developing countries and require technical assistance in the development of their national waste management programme. The kind of assistance provided is in the form of technical projects and training. In the area of waste management, the Agency is currently funding 43 technical assistance projects. These projects cover a wide range of assistance from providing the legislative framework for waste management to direct assistance in constructing waste management facilities. A current high interest effort is a model project entitled "Upgrading Waste Management Infrastructure". This project has as its objective "...to demonstrate to Member States acceptable levels of operational and safety requirements in the management of nuclear waste by upgrading waste management infrastructures in selected developing Member States." The "acceptable level" will be based on the IAEA's RADWASS Fundamentals and Standards and the specific national waste management needs of the country. The project is organized in phases; during the first phase (1994-95) country profiles will be established and criteria will be developed for "acceptable waste management infrastructure" based on the different types and levels of radioactive waste generated by each country. In the second phase (1996-97) a standard TC assistance package, developed to fit the need of the country, as shown in Fig.2, will be implemented in select countries. Later phases will assess the effectiveness of the assistance packages, revise them as necessary, and apply the revised packages to other candidate Member States. It is expected that this project will serve as the model for future Agency efforts in offering technical assistance to Member States.

As part of the Agency's Technical Assistance programme increased emphasis has been placed on providing training opportunities to scientists in developing Member States. A series of radioactive waste management regional and interregional training courses have

CLASS	COUNTRY NEEDS BASED ON NUCLEAR APPLICATION USE
A	Single Use of Radioisotopes
B	Multiple Use of Radioisotopes
C	Multiple Use with Nuclear Research Reactors
M	One of the above classes plus uranium mining/milling wastes

FIG.2. Standard Technical Assistance package developed for each country class.

been developed on such topics as management of waste from nuclear applications, management of waste from nuclear power plants, decontamination and decommissioning of non-power nuclear facilities, management of spent radiation sources, quality assurance and control in radioactive waste management and safety assessment methodology for near surface radioactive waste disposal facilities. The next training course offered by the Agency in radioactive waste management is a regional course on Waste Management Techniques to be held in Cairo, Egypt in May 1994. Currently, an interregional training course on safety assessment methodologies for near-surface waste disposal facilities is in progress in Argonne, Illinois, USA.

VII. ADVISORY SERVICES

Within its waste management programme, the IAEA offers advisory services as a key element of the assistance provided to Member States. Two types of advisory services are offered. One was established for providing direct assistance to developing Member States (WAMAP) and one was organized to offer international peer review services to Member States with mature waste management programmes (WATRP).

The Waste Management Advisory Programme (WAMAP) was organized in 1987 to facilitate Agency efforts to provide direct assistance to developing Member States as they plan and implement national radioactive waste management programmes. It is an Interregional Technical Co-operation project funded by the Departments of Technical Co-operation and Nuclear Energy and Safety with technical management under the auspices of the Waste Management Section. The objective of WAMAP is to provide a technical assistance mechanism which offers international expertise to solve waste management problems/issues faced by developing countries. Since the programme's inception, a total of 38 developing countries have been visited by WAMAP missions. To provide some understanding of the extent of the WAMAP programme Figure 3 shows a map which illustrates countries visited by WAMAP.

A major contribution of WAMAP is that it evokes an awareness in the countries about the need for the safe management of radioactive waste and assists in confidence building, so that countries can reach such reliance in this area. The WAMAP programme by pointing out deficiencies in Member States and recommending follow-up actions will eventually lead to a country establishing a well planned national waste management programme which will reduce radiological exposure, prevent potential accidents and protect the environment.

In response to the need for international peer review services, the Agency established the Waste Management Assessment and Technical Review Programme (WATRP) in 1989. WATRP formalized an ad-hoc service of the Agency that had been offered since the mid 1980s. WATRP is being offered to Member States with a mature waste management programme as a mechanism for independent international peer reviews of national plans, policies or projects. It may be seen as a way for Member States to establish technical credibility and enhance public confidence and acceptability of waste management plans and/or systems. A WATRP peer review can be either "micro" or

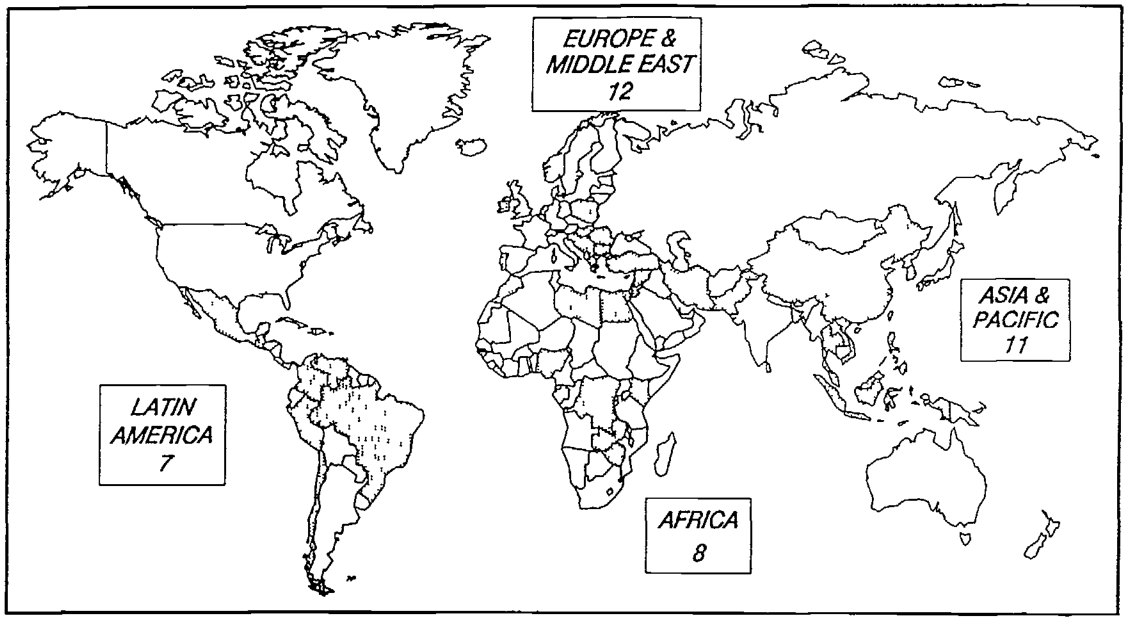


Fig. 3. WAMAP Missions as of end of 1993.

"macro" in nature, depending on the terms of reference established by the organization requesting this service. For example, the WATRP review of the Finnish waste management programme (conducted in August 1993) was macro in scope as the WATRP team reviewed and commented on Finland's national waste management programme. The WATRP review requested by the Republic of Korea was "micro" in nature as it was limited to an assessment of the siting criteria that they planned to use to site a low-intermediate waste repository.

With increasing public concern over waste management plans and activities, the Agency looks at the WATRP service as a way to help install both confidence and credibility in national waste management programmes.

VIII. SPECIAL PROJECTS

The IAEA has a number of special projects in the area of waste management, that offer direct assistance to Member States. One such project is the Waste Processing and Storage Facility (WPSF) design package for Member States with waste arising only from nuclear applications. This Architecture-Engineering (A-E) design package offers a reference design of a facility that can safely process and store waste typically generated from isotope applications and nuclear research centres. Along the same lines is the special project on "spent radiation sources". This project provides direct assistence to Member States in handling, conditioning and disposal of spent radiation sources. One of the key elements of this project is the availability to Member States of on-site advisory teams to actually handle and condition spent radiation sources.

Another special project under consideration is the concept of regional repositories for the disposal of radioactive waste. It is very clear to the international community that such a concept has a very strong appeal from the technical, safety and economic standpoints. However, political acceptance and the general public "outcry" of "not in my backyard" have prevented any serious international efforts from developing on this concept. The Agency has initiated a small effort on the regional repository concept and will start by attempting to clearly identify the large technical and economic benefits that could be received by the country that agrees to be the host for such a repository. A lengthy and challenging path to reaching the objective of a regional repository is foreseen, and the IAEA is convinced that the safety, technical and economic benefits that would be derived justify the efforts involved.

A new special project is the International Arctic Seas Assessment Project (IASAP) recently initiated by the Agency to evaluate the health and environmental risks posed by radioactive waste dumping in the Arctic Seas. This programme will evaluate the impact on the environment from a series of dumpings in the Kara and Barents Sea and evaluate the feasibility of possible remedial actions. Recognizing the political sensitivity of this subject to many countries, this project aims to produce an independent and objective assessment of the situation through the involvement of leading international laboratories and scientists by 1996.

IX. CONCLUSIONS

In concluding, it should be mentioned that the management of radioactive waste is one of the critical issues surrounding the use and growth of nuclear energy. Member States must continually assess both the technical and public acceptance aspects of the issues involved, and when necessary, adjust plans and programmes accordingly. Recognizing this, the Agency's waste management programme must be flexible to respond to the needs of Member States with activities that are both beneficial and timely.

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SAFETY STANDARDS FOR RADIOACTIVE WASTE MANAGEMENT

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Abstract

The International Atomic Energy Agency (IAEA) has established the Radioactive Waste Safety Standards (RADWASS) programme upon request by its Member States to provide evidence that radioactive waste can be managed safely. The RADWASS programme consists of a series of fifty-five international consensus documents covering all parts of radioactive waste management, i.e. the subject areas:

- Planning
- Pre-disposal
- Near surface disposal
- Geological disposal
- U/Th mining and milling waste
- Decommissioning

A single Safety Fundamentals document will set out the basic safety principles for radioactive waste management. Each subject area is headed by a Safety Standard. Twenty-eight Safety Guides and twenty Safety Practices will provide further details for the implementation of safety requirements stated in the Safety Standards.

The programme was started in 1991 and is being carried out in three phases (Phase I: 1991-1994; Phase II: 1995-1998; Phase III: post 1998). Phase I includes twelve documents comprising the Safety Fundamentals, four Safety Standards, five Safety Guides and two Safety Practices. The Safety Fundamentals and the Safety Standards are planned to be submitted to the Board of Governors for review and approval in 1994. Four of the Safety Guides have been or will soon be submitted for publication and the fifth will be finalized by the end of 1994. One Safety Practice on "Application of Exemption Principles" was published at the end of 1992 and the second Safety Practice of Phase I is planned to be finalized in 1994.

The thirty-seventh regular session of the General Conference in the 361st plenary meeting adopted the resolution "Strengthening Nuclear Safety" through the early conclusion of a Nuclear Safety Convention. It calls for "Measures to Strengthen International Co-operation in Matters Relating to Nuclear Safety and Radiological Protection" and requests the Director General inter alia to initiate preparations for a convention on the safety of waste management as soon as the ongoing process of developing the RADWASS Safety Fundamentals has resulted in broad international agreement. Approval of the document by the Board of Governors will be an important step toward convening the waste management Safety Convention.

It is intended to finalize work on Phase I (1991-1994) documents by the end of 1994. Phase II envisages the initial preparation of thirteen documents comprising one Safety Standard, eleven Safety Guides and one Safety Practice. It is planned to start some of these activities in 1994.

1 Introduction

Radioactive waste is generated from the production of nuclear energy and from the use of radioactive materials in industrial applications, research and medicine The importance of safe management of radioactive waste for the protection of human health and the environment has long been recognized and considerable experience has been gained in this field Thus it is desirable to establish and promote in a coherent and comprehensive manner the basic safety philosophy for radioactive waste management and the steps necessary to assure its implementation

The International Atomic Energy Agency (IAEA) has been regularly requested by its Member States (MS) to provide evidence that radioactive waste can be managed safely and to help demonstrate a harmonization of approaches at the international level by providing safety documents

In response, IAEA has established a special series of safety documents devoted to radioactive waste management These documents are being elaborated within the Radioactive Waste Safety Standards (RADWASS) programme which covers all aspects of radioactive waste management The purpose of the RADWASS programme is to (i) document existing international consensus in the approaches and methodologies for safe radioactive waste management, (ii) create a mechanism to establish consensus where it does not exist and (iii) provide Member States with a comprehensive series of internationally agreed upon documents to complement national standards and criteria

2. RADWASS Programme Structure

RADWASS is organized in the hierarchical structure following the general framework of IAEA Safety Series documents and will be published as advisory documents under Safety Series No 111 The top level publication is a single Safety Fundamentals document which provides the basic safety objectives and fundamental principles to be followed in national waste management programmes

Documents below the Safety Fundamentals level, i e Safety Standards, Safety Guides and Safety Practices, will be organized into six subject areas

- Planning,
- Pre-disposal,
- Near surface disposal,
- Geological disposal,
- U/Th mining and milling waste and
- Decommissioning

Each subject area is headed by a Safety Standard The Safety Guides and Safety Practices in the individual subject area will provide further details of implementing safety requirements stated in the Safety Standards

Standing Technical Committees (STCs) have been established for each of the subject areas to review the respective documents The STCs will contribute to a consistent approach in the development of RADWASS documents and provide the national expertise of participating Member States

The whole RADWASS programme is overseen by the International Radioactive Waste Management Advisory Committee (INWAC) INWAC consists of senior experts from MS, who in their overall function to advise IAEA in its radioactive waste management programme, provide guidance to the RADWASS Programme INWAC provides advice on establishing the RADWASS publication plan and the scheduling of publications They review and approve the Terms of Reference for each of the RADWASS documents and review and approve the Safety Fundamentals and Safety Standards The close and intensive co operation among senior experts from IAEA MS is an important element in the elaboration of RADWASS documents

3 Publication Plan

The RADWASS programme was developed by IAEA in consultation with senior experts and upon advice from expert groups Initial work to structure the RADWASS programme began in early 1990 and full approval to execute the programme was given by the IAEA Board of Governors in September 1990

The RADWASS programme was established in 1991 to provide a series of twenty-four international consensus documents on the safe management of radioactive waste The initial programme included one Safety Fundamentals document, six Safety Standards and seventeen Safety Guides Safety Practices were planned to be added as necessary

At the time the RADWASS programme was first established it was already envisaged that a formal review of the programme would be undertaken in 1993 to define publication production rates and the resources needed for the post-1994 period INWAC held this planned review of the RADWASS programme on 22-25 March 1993 This meeting resulted in the completion and extension of the RADWASS programme to include 55 documents (Annex 1) In particular, Safety Practices not previously included were defined for all six subject areas of the RADWASS programme Additionally, eleven Safety Guides were added, covering topics such as licensing, quality assurance, safety assessments, definitions and environmental restoration

No modifications were made on the level of the Safety Fundamentals and the Safety Standards, except in the subject area "Decommissioning" "Environmental Restoration" will also be covered in this subject area, making it necessary to expand the respective Safety Standard at a later time

Priorities were assigned to individual documents as outlined in Annex 1, with high ranking documents receiving first priority Despite an increase in document production rates it was still necessary to add a Phase III (post 1998) to the ongoing Phase I (1991-1994) and the planned Phase II (1995 1998)

4 RADWASS Operation

A standardized process is applied to the development of individual RADWASS documents (Fig 1) If necessary additional steps may be added

A very elaborate process is applied to the preparation of the Safety Fundamentals document and the Safety Standards, thus reflecting their importance and high

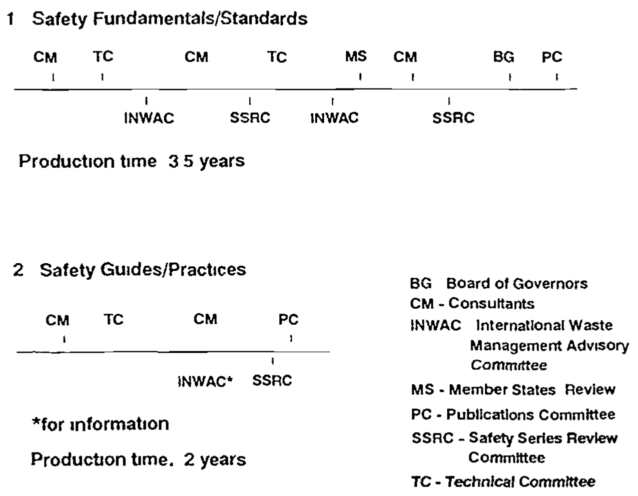


Fig. 1. Process for the preparation of RADWASS documents

hierarchical level Before these documents are submitted to the IAEA Board of Governors (BG) for approval and to the IAEA Publications Committee (PC) for final editing, they undergo three Consultants Meetings (CM), two STCs, two INWAC reviews and a review by Member States Additionally, the IAEA internal Safety Series Review Committee (SSRC) has to check all Safety Series publications in order to ensure consistency and compatibility It is necessary to apply this elaborate process in order to exchange and harmonize views of Member States and to find consensus on the individual documents

The preparation of Safety Guides and Safety Practices is not as complex Before these documents are submitted to PC, they undergo two CMs, one STC and a review by SSRC The documents will be submitted to INWAC for information and approved by the Director General of the IAEA

5. Programme Status

Good progress has been made in the elaboration of RADWASS Phase I documents Detailed information on programme activities and current status is presented in Annex 2

In December 1992 the first RADWASS document "Application of Exemption Principles to the Recycle and Reuse of Materials from Nuclear Facilities" was published. This Safety Practice assesses various scenarios for exposures of humans to radionuclides from such materials and presents the results of these assessments

In the MS review of the Safety Fundamentals there was good agreement on the main features of the document Consultants' Meetings were held in late 1993 and early 1994 to consider MS comments The revised draft is expected to be ready for submission to the BG in 1994

The Safety Standard "National Radioactive Waste Management System' has progressed to the point that submission to MS for review should occur in the first half of 1994 Since this Safety Standard provides input into the other Safety Standards, namely "Pre-disposal Management of Radioactive Waste", "Near Surface Disposal of Radioactive Waste", and "Decommissioning of Nuclear Facilities", these documents, after appropriate revision, are also expected to be submitted for MS review in 1994

Two Safety Guides, namely "Classification of Radioactive Waste and Siting of Geological Disposal Facilities have been submitted for publication The Safety Guide on "Siting of Near Surface Disposal Facilities" has received SSRC approval. The Safety Guide "Recommended Clearance Levels for Radionuclides in Solid Material" is under SSRC review. Preparation of the Safety Guide "Pre-disposal Management of Low and Intermediate Level Waste from Medicine, Industry and Research" began in 1993 The document is planned to be finalized by the end of 1994.

- The Safety Practice "Application of Exemption Principles to Materials Resulting from the Use of Radionuclides in Medicine Industry and Research" has been reviewed in CMs, an Advisory Group Meeting (AGM) and a TCM The document is now planned to be submitted to SSRC

6. Convention on the Safety of Waste Management

The thirty-seventh regular session of the General Conference in October of 1993 adopted the resolution "Strengthening Nuclear Safety through the Early Conclusion of a Nuclear Safety Convention. It requests the Director General inter alia to initiate preparations for a Convention on the safety of waste management as soon as the ongoing process of developing the waste management Safety Fundamentals has resulted in broad international agreement

Such a Convention will result in a stand alone document and will have a legally binding character for signatory states It has to be initiated and prepared with great care This applies to its timing as well as to its contents. Further guidance is expected in these questions from MS At present it seems to be agreeable to initiate work on a waste management Convention after the RADWASS Safety Fundamentals and possibly also the Safety Standard on the national waste management system have been approved by the Board of Governors A "bridging process" will be able to identify those elements of the RADWASS documents that should be used for the formulation of a waste management Convention Further impetus for such a Convention could be expected from the International Seminar on "Requirements for the Safe Management of Radioactive Waste" organized by the IAEA and scheduled for 27 August 01 September 1995 This Seminar will provide a forum for the discussion of results from Phase I of the RADWASS programme and of the waste management experience in MS Conclusions from such discussions could be provided as input in drafting the waste management Convention

7. Safety Principles and Requirements

Safe management of radioactive waste involves the application of technology and resources in an integrated and regulated manner so that occupational and public exposure to ionizing radiation is controlled and the environment protected in accordance with national regulations and international consensus documents To meet this overall objective, the following internationally agreed upon safety principles, defined in the most recent draft RADWASS Safety Fundamentals entitled "The Principles of Radioactive Waste Management", need to be applied

Principle 1. **Protection of human health**

Radioactive waste shall be managed in a way to secure an acceptable level of protection of human health

Principle 2. **Protection of the environment**

Radioactive waste shall be managed in a way that provides protection of the environment.

Principle 3. **Protection beyond national borders**

Radioactive waste shall be managed in such a way as to assure that possible effects on human health and the environment beyond national borders will not be greater than what is acceptable within the country of origin.

Principle 4: **Protection of future generations**

Radioactive waste shall be managed in a way that predicted impacts on the health of future generations do not exceed relevant levels that are acceptable today.

Principle 5: **Burdens on future generations**

Radioactive waste shall be managed in a way that will not impose undue burdens on future generations.

Principle 6. **Legal framework**

Radioactive waste shall be managed within an appropriate legal framework including clear allocation of responsibilities and provision for independent regulatory functions

Principle 7: **Control of radioactive waste generation**

Generation of radioactive waste shall be kept to the minimum practicable

Principle 8. **Radioactive waste generation and management interdependencies**

Interdependencies among all steps in radioactive waste generation and management shall be appropriately taken into account.

Principle 9 **Safety of facilities**

Safety of facilities for radioactive waste management shall be appropriately assured during their lifetime.

In order to achieve the safety objective of these principles, a national radioactive waste management system must be established Such a system must specify the objectives and requirements of a national strategy for radioactive waste management and the responsibilities of the parties involved It must also describe other essential features, e g licencing processes and safety and environmental assessments. The elements of such a national radioactive waste management system are summarized in the most recent draft Safety Standard "A National System for Radioactive Waste Management" which assigns the following ten responsibilities to the State, the regulatory body or the operators

Responsibilities of the State:	1	Establish and implement a legal framework
	2	Establish a regulatory body
	3	Define responsibilities of waste generators and operators
	4	Provide for adequate resources
Responsibilities of the Regulatory Body	5	Apply and enforce legal requirements
	6	Implement the licencing process
	7	Advise the government
Responsibilities of the Operators	8	Identify an acceptable destination for the radioactive waste
	9	Safely manage the radioactive waste
	10	Comply with legal requirements

Achievement of the safety principles outlined in "The Principles of Radioactive Waste Management" also requires the definition of technical safety requirements for each individual subject area in radioactive waste management. These requirements are being formulated or will be formulated in the RADWASS Safety Standards for the respective subject areas (pre-disposal, near surface disposal, geological disposal, U/Th mining and milling waste and decommissioning)

8. Summary

The preparation of documents within the RADWASS programme has made good progress A first publication was released in 1992 Most of the eleven RADWASS Phase I documents will be submitted for publication in 1994

The RADWASS programme was reviewed by INWAC in 1993 It was supplemented as appropriate and Phase II was defined A Phase III will be necessary to accomplish the extended programme The basic safety principles in radioactive waste management have been drafted in the Safety Fundamentals and the responsibilities in a national radioactive waste management system were outlined in the respective Safety Standard

A resolution to prepare a Convention on the safety of waste management has been adopted by the General Conference in 1993 First considerations on how to approach such a broad international document have been made

DATE 25 June 1993

RADWASS PUBLICATION PLAN

SAFETY SERIES NO. 111

SAFETY FUNDAMENTALS

111-F Principles of Radioactive Waste Management	1

SAFETY STANDARDS

1 PLANNING	2 PRE-DISPOSAL	3 NEAR-SURFACE DISPOSAL	4 GEOLOGICAL DISPOSAL	5 U/Th MINING AND MILLING	6 DECOMMISSIONING AND ENVIRONMENTAL RESTORATION
111 S 1 **1** Establishing a national radioactive waste management system	111-S-2 **1** Pre disposal management of radioactive waste	111-S-3 **1** Near surface disposal of radioactive waste	111-S-4 **2a** Geological disposal of radioactive waste	111-S-5 **2a** Management of waste from mining and milling of uranium and thorium ores	111-S 6 **2b/3** * Decommissioning of nuclear facilities (and environmental restoration)

SAFETY GUIDES

1 PLANNING	2 PRE-DISPOSAL	3.NEAR-SURFACE DISPOSAL	4 GEOLOGICAL DISPOSAL	5 U/Th MINING AND MILLING	6 DECOMMISSIONING
111-G-1 1 **1** Classification of radioactive waste	111-G-2.1 **2a** Collection and treatment of low and intermediate level waste from nuclear facilities	111-G-3.1 **1** Siting of near surface disposal facilities	111-G-4.1 **1** Siting of geological disposal facilities	111-G-5.1 **2b** Siting, design, construction and operation of facilities for the management of waste from mining and milling of U/Th ores	111-G-6 1 **2a** Decommissioning of nuclear power and large research reactors
111-G-1.2 **2a** Planning and implementation of national radioactive waste management programmes	111-G-2 2 **1** Pre disposal management of radioactive waste from medicine, industry and research	111-G-3 2 **2a** Design, construction, operation and closure of near surface repositories	111-G-4.2 **3** Design, construction, operation and closure of geological repositories	11-G-5 2 **2b** Decommissioning of surface facilities and closeout of mines, waste rock and mill tailings from mining and milling of U/Th ores	111-G-6.2 **2a** Decommissioning of medical, industrial and small research facilities
111-G-1.3 **2a** Licensing of radioactive waste management facilities	111-G-2 3 **2a** Conditioning and storage of low and intermediate level waste from nuclear facilities	111-G-3 3 **2a** Safety assessment for near surface disposal	111-G-4.3 **2b** Safety assessment for geological disposal	111-G-5 3 **3** Safety assessment for the management of waste from mining and milling of U/Th ores	111-G-6 3 **2b** Decommissioning of nuclear fuel cycle facilities
111-G-1.4 **2b** Quality assurance for the safe management of radioactive waste	111-G-2 4 **2b** Treatment, conditioning and storage of high level reprocessing waste				111-G-6.4 **2a** Safety assessment for the decommissioning of nuclear facilities
111-G 1 5 **1** Exemption from Regulatory Control Recommended unconditional clearance levels for solid materials	111-G-2 5 **2b** Preparation of spent fuel for disposal				111-G-6 5 **2b** Environmental restoration of previously used or accidentally contaminated areas
111 G 1 6 **3** Derivation of discharge limits for waste management facilities	111 G-2 6 **2b** Safety assessment for pre disposal waste management facilities				111-G 6 6 **3** Recommended cleanup levels for contaminated land areas
111-G-1 7 **2a** Radioactive waste management glossary					

SAFETY PRACTICES

1 PLANNING	2 PRE DISPOSAL	3 NEAR-SURFACE DISPOSAL	4 GEOLOGICAL DISPOSAL	5 U/Th MINING AND MILLING	6 DECOMMISSIONING
111-P-1.1 **1** Application of exemption principles to the recycling and reuse of materials from nuclear facilities	111-P-2.1 **3** Off-gas treatment and air ventilation systems at nuclear facilities	111-P-3/4.1 **3** Validation and verification of models for long-term radioactive safety assessment of waste disposal facilities		111-P-5.1 **3** Procedures for closeout of mines, waste rock and mill tailings	111-P-6.1 **3** Techniques to achieve and maintain safe storage of nuclear facilities

* Amendment of this SS to include environmental restoration

1 RADWASS phase 1

2a RADWASS phase II (1995-96)

2b RADWASS phase II (1997-98)

3 RADWASS phase III (post 1998)

▬ The initially planned publications plan (with amendments) can be found above the bold line and extensions as of March 1993 can be found below

111-P-1 2 **1** Application of exemption principles from the use of radionuclides in medicine, industry and research	111-P-2 2 **3** Characterization of raw waste	111-P-3/4 2 **3** Procedures for closure of radioactive waste disposal facilities		111-P-5.2 **3** Operational and post operational monitoring, surveillance and maintenance of facilities for the management of waste from mining and milling of U/Th ores	111-P-6 2 **3** Procedures and techniques for the decommissioning of nuclear facilities
111-P-1.3 **3** Data collection and record keeping in radioactive waste management	111-P-2.3 **3** Control of waste conditioning processes	111-P-3.3 **2b** Waste acceptance requirements for near surface disposal of radioactive waste	111-P-4.3 **3** Waste acceptance requirements for geological disposal of radioactive waste		111-P-6 3 **2a** Methods for deriving cleanup levels for contaminated land areas
	111-P-2.4 **3** Testing of radioactive waste packages	111-P-3.4 **3** Selection of scenarios for safety assessment of near surface disposal facilities	111-P-4.4 **3** Selection of scenarios for safety assessment of geological disposal facilities		111-P-6.4 **3** Monitoring for compliance with cleanup levels
		111-P-3.5 **3** Systems for operational and post-closure monitoring and surveillance of near surface disposal facilities			

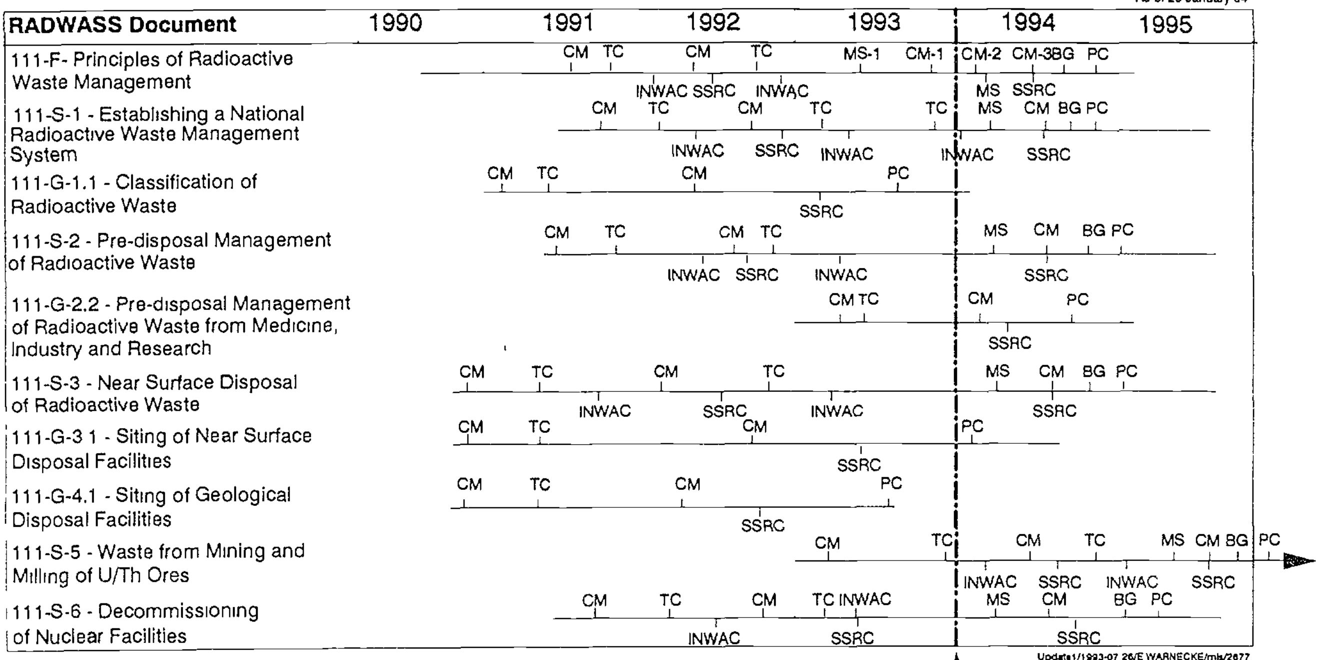
1 RADWASS phase 1
2a RADWASS phase II (1995-96)
2b RADWASS phase II (1997-98)
3 RADWASS phase III (post 1998)

The initially planned publications plan (with amendments) can be found above the bold line and extensions as of March 1993 can be found below

Annex 2

RADWASS Operation (Phase I 1991-1994) - Updated Plan

As of 28 January 94



BG Board of Governors
CM Consultants Meeting
INWAC International Waste Management Advisory Committee
MS Member States Review
PC Publications Committee
TC Technical Committee
SSRC Safety Series Review Committee

End of Current Reporting Period

LSA/SCO PROVISIONS IN IAEA SAFETY SERIES No. 6 REGULATIONS FOR THE SAFE TRANSPORT OF RADIOACTIVE MATERIAL*

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Abstract

The first edition of the IAEA Transport Regulations was published in May 1961, just four years after the Agency was formed, and two years after the United Nations asked it to develop transport regulations. The Low Specific Activity radioactive material provisions in that first edition were based on existing U.S. Interstate Commerce Commission regulations and the International Convention Concerning the Carriage of Goods by Rail. Thirty years later much of the original provisions are still in force, with additions related to transport of contaminated objects and solidified wastes. A number of significant changes have occurred through four major revisions of the regulations, some good, in my opinion, and some bad. Somehow, however, we seem to be able to let our experience with the regulations filter out the bad and preserve the good. The result is that, considering the decisions of the most recent Technical Committee meeting in October 1993, the IAEA Transport Regulations should be, for the most part, efficient, effective, and defendable. The major decision of the recent Technical Committee to extend the concept of Surface Contaminated Objects to include objects which are themselves radioactive, recognizes the need for efficient transport of wastes from decommissioning facilities. The decision to require Type A packages for transport of LSA-II, LSA-III, and CO-II materials is a major step toward simplification. The major decision to retain the existing Type B package and all its criteria for transportation of LSA/CO material avoids the complication of having yet another Type B package. The major decision to make the consignor responsible for imposing vehicle limits on shipments of LSA/CO, by either shipping under exclusive use or providing quantity information to the carrier in terms of a multiple of A_2, makes that segment of the regulations defendable.

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*This work was performed at International Energy Consultants, Inc. under subcontract 31462401 with Argonne National Laboratory acting under Prime Contract W-31-1-109-ENG-38 with the US Department of Energy.

1. HISTORY

The first edition of the IAEA Regulations for the Safe Transport of Radioactive Material was published in May 1961 by a private printer in Vienna, Austria whose name was Paul Gerin. The IAEA had only been in existence since July 1957. Its' first General Conference had just been held in the old Kaerntnerring offices in October 1957. In July, 1959, the United Nations Economic and Social Council expressed the desire "that the Agency be entrusted with the drafting of recommendations on the transport of radioactive substances." The various national and international regulations that existed at that time were mostly based on regulations of the United States Interstate Commerce Commission. These U.S. regulations essentially aimed to facilitate the movement of radioactive ores and concentrates and packages containing relatively small amounts of radioisotopes for medical and industrial use. The transportation of larger amounts of radioactive or fissile material was not allowed under the regulations and therefore required special permits. These special permits were issued in the U.S. by the Bureau of Explosives, a segment of the Association of American Railroads, a private organization. Such a regulatory structure, where the special permits are issued by part of the regulated industry, would not and should not be tolerated today.

The initial regulations on safe transportation were the result of two IAEA revision panels, each of which met in 1959 and again in 1960, one dealing with the transport of radioisotopes and radioactive ores and residues of low specific activity, the other with the transport of large radioactive sources and fissile materials. The revision panel dealing with the transport of radioisotopes and radioactive ores and residues of low specific activity already had considerable experience with such transport from the U.S. regulations and the International Convention Concerning The Carriage of Goods by Rail. What they needed, however, was a basis for defining "low specific activity material" so as to cater to the increasing need to transport intermediate products from the processing of natural uranium or thorium and various kinds of low activity building rubble, scrap materials, and process wastes arising in the nuclear industry.[6] From this consideration came the all too familiar concept of "inherently safe" meaning that "it is inconceivable that under any circumstances arising in transport, a person could suffer an intake to the body of a sufficient mass of material to give rise to a significant internal radiation hazard..."[6]

For the first edition of the IAEA regulations in 1961, the Revision Panel accepted that "no person will take into and retain in the body more than 1mg of radioactive material." A person exposed to an extremely dirty atmosphere containing 10mg of dust per cubic meter of air would need to inhale that atmosphere for about 6 minutes for an intake of 1mg. The Panel applied this assumption in specifying concentration limits of 1.0 Ci/g for sludges and solids and 1.0 Ci/ml for liquids. These limits were reduced by a factor of 10 when a high toxicity Group I radionuclide was present. LSA materials in 1961 were defined as follows:

14. TRANSPORT OF RADIOACTIVE MATERIALS OF LOW SPECIFIC ACTIVITY [1]

14.1. Shipments of radioactive materials which belong to one of the categories specified in paragraphs 14.1.1. and 14.1.2., and in paragraphs 14.1.3 and 14.1.4., subject to the proviso contained in paragraph 14.1.5., and which comply with the requirements set forth in this section, shall be exempted from the packaging and labeling requirements of sections 5 (packaging), 6 (quantity), and 7 (external radiation limits), and sub-section 8.1 (labeling and marking).

14.1.1. Unirradiated uranium, containing 0.72% U-235 by weight, or less, and unirradiated thorium, in a non-friable, massive-solid form or contained in an inert metal cover or other substantial coating such that the surface of the uranium or thorium is not exposed*, provided that:

(a) the radioactive materials are packed in a manner which will prevent the ingress of moisture and movement of the material within the package or vehicle; and

(b) beryllium, graphite (pile-grade) or heavy water are not included in the package containing the radioactive materials.

14.1.2. Ores and concentrates (of ores) of natural uranium and natural thorium;

14.1.3. Intermediate products, i.e. in-process materials in gaseous, liquid, sludge or solid form arising from the processing of natural uranium and thorium before enrichment or irradiation of the uranium or thorium, but not including refined isotopes of radium.

14.1.4. Low-activity materials, i.e. residues from the processing of natural uranium and thorium; wastes such as building rubble, metal, wood and fabric scrap, glassware, paper and cardboard; reactor and process plant wastes in liquid or solid form; sludges and ashes from incinerators containing radioactive materials; or other materials, provided that, in such low-activity materials, the estimated maximum radioactivity content for radioactive materials in Group I does not exceed:

0.1 uCi/g in the case of radioactive material in sludge or solid form; or

0.1 uCi/ml in the case of radioactive material in liquid form;

where radioactive materials in Group I are not present, the limits shall be, respectively, 1.0 uCi/g and 1.0 uCi/ml.

14.1.5. In the case of intermediate products and low-activity materials, or of ores and concentrates (of ores) of natural uranium and natural thorium, shipments shall be made in such form and quantity that the estimated total radioactive content of any one container, vehicle or compartment does not exceed:

100mCi of any material in Group I;
1Ci of any material in Group II;
20Ci of any material in Group III; or
for any combination of radioactive materials involving more than one toxicity group:
(total activity in mCi of GroupI)X10+
(total activity in Ci of Group II)+
(total activity in Ci of Group III)X1/20
shall be equal to or less than 1Ci.

14.2. The radioactive materials listed in sub-section 14.1. shall be packed in strong, leak-proof packages or loaded in vehicles or compartments specially designed to ensure that there will be no leakage under conditions normally incident to transport.

* For instance, aluminum-clad fuel elements.

Although the "inherently safe" concept does not require vehicle load limits expressed in terms of activity, it was considered prudent to limit contamination spread and, as a consequence, the numbers of people exposed. Accordingly, the 1961 regulations imposed "load limits" for LSA material in Groups I, II, and III of 100mCi, 1 Ci, and 20 Ci, respectively, in "any one package, vehicle, or compartment." So- for those of us who have been frustrated because we could never determine the origin or reason for today's vehicle quantity limits, let me repeat it. The Revision Panel participants for the 1961 regulations thought it prudent to limit contamination spread in order to limit the number of people exposed.

Some changes were made in LSA modeling and limits when the first revision of the IAEA regulations was issued in 1964. First,the breathing rate of the exposed person was raised from $1m^3$/hr to $2m^3$/hr on the grounds that the person would likely be physically exerting himself at the transport accident scene. That same heavily dusty atmosphere of $10mg/m^3$ would result in inhalation of 10mg of dust in 30 minutes, the same model we use today. As a result of this model and an expansion of the radionuclide groupings, the range of specific activities allowed in the general LSA category in the 1964 regulations is the following:

A-2.8 LOW SPECIFIC ACTIVITY MATERIAL [7]

Low specific activity material shall mean any of the following:

(d) Material in which the activity is uniformly distributed and in which the estimated concentration per gram does not exceed:

(i) 0.0001 mCi for Group I radionuclides; or
(ii) 0.005 mCi for Group II radionuclides; or
(iii) 0.3 mCi for Group III and IV radionuclides;

In order to facilitate the transport of LSA material within any of three broad categories, the 1964 regulations specified both concentration limits and full load limits as follows:

(a) In industrial-type packaging subject to all the standard controls for labeling, surface contamination and package control.
(b) In industrial packaging in amounts up to specified quantity limits and under full load conditions exempt from the standard labeling and surface contamination provisions.
(c) In bulk containers in amounts up to specified quantity limits and under full load conditions.

The Revision Panel for the 1964, IAEA regulations specified 5 Ci/liter as the limit up to which tritium oxide in aqueous solution (tritiated water) may be transported as LSA material.

The 1967 edition of the Transport Regulations should not be remembered for big changes in the LSA definitions, but for stability and clarity. Following is the LSA definition from the 1967 regulations, which included the general prescription based on the inhalation and retention of 10mg of dust, and the surface-contaminated, non-radioactive objects with limits unchanged from the previous regulations:

A-2.8. LOW SPECIFIC ACTIVITY MATERIAL [3]

Low specific activity material shall mean any of the following:

(a) Uranium or thorium ores and physical or chemical concentrates of those ores;

(b) Unirradiated natural or depleted uranium or unirradiated natural thorium;

(c) Tritium oxide in aqueous solutions provided the concentration does not exceed 5.0 mCi/ml;

(d) Material in which the activity is uniformly distributed and in which the estimated concentration per gram does not exceed:

(i) 0.0001 mCi for Group I radionuclides; or
(ii) 0.005 mCi for Group II radionuclides; or
(iii) 0.3 mCi for Groups III and IV radionuclides;and

(e) Objects of non-radioactive material externally contaminated with radioactive material, provided that:

(i) The radioactive material is in a non-readily dispersible form and the surface contamination does not exceed:

0.0001 mCi/cm for alpha emitters of Group I: or
0.001 mCi/cm for other radionuclides, when averaged over 1 m ; and

(iii) The objects are suitably wrapped or enclosed.

Just as the Revision Panel for the 1967 regulations should be known for stabilizing and clarifying the Transport Regulations, the Revision Panel for the 1973 revision should be remembered as complicating the regulations. At the time of the revision panel, rapid expansion of the nuclear industry was still expected, as was continued increase in bulk transportation of LSA material. The grouping system for arranging radionuclides according to their radiotoxicity gave way to the A system, and everyone agreed that the limit of 10 A Ci/g was a simpler, clearer way to limit materials in the LSA category. The Revision Panel apparently could not accept this universal agreement, and looked for ways to recomplicate the regulations. They found two ways. To specify radioactive materials to which the new limit of 10 A Ci/g could apply, the panel took account of the fact that the following two assumptions were implicit in its derivation:[6]

(a) The activity must be uniformly distributed throughout the material.

(b) The specific activity cannot be increased by any mechanism that could conceivably arise during transport.

On the basis of these assumptions, after noting that the original LSA materials, namely ores and concentrates of natural uranium and thorium, and tritiated water, all satisfy the assumptions,the Panel replaced the relatively simple 1967 LSA definition with the following:

LOW SPECIFIC ACTIVITY [4]

121. Low specific activity material (LSA) shall mean any of the following:

(a) Uranium or thorium ores and physical or chemical concentrates of those ores.

(b) Unirradiated natural or depleted uranium or unirradiated natural thorium.

(c) Tritium oxide in aqueous solutions, provided the concentration does not exceed 10 Ci/litre.

(d) Materials in which the activity, under normal transport conditions, is, and remains, uniformly distributed and in which the average estimated specific activity does not exceed $10^{-4} A_2$/g.

(e) Materials in which the activity is uniformly distributed and which, if reduced to the minimum volume under conditions likely to be encountered in transport, such as dissolution in water with subsequent recrystallization, precipitation, evaporation, combustion, abrasion, etc., would have an average estimated specific activity of no more than $10^{-4} A_2$/g.

(f) Objects of non-radioactive material contaminated with radioactive material, provided the non-fixed surface contamination does not exceed ten times the values given in Table XI and the contaminated object or the contamination on the object, if reduced to the minimum volume under conditions likely to be encountered in transport such as dissolution in water with subsequent recrystallization, precipitation, evaporation, combustion, abrasion, etc., would have an average estimated specific activity of no more than $10^{-4} A_2$/g.

(g) Objects of non-radioactive material contaminated with radioactive material, provided that the radioactive contamination is in a non-readily dispersible form and the level of contamination averaged over 1 m^2 (or the area of the surface if this is less than 1 m^2) does not exceed:

1 uCi/cm^2 for beta and gamma emitters and the low toxicity alpha emitters indicated in Table XI;

0.1 uCi/cm^2 for other alpha emitters.

In addition, as though the revision panel for the 1973 edition of the regulations had not complicated the regulations enough with its specification for "uniform distribution" which has never been satisfactorily resolved, and mechanisms for concentration during transportation, the panel invented a new concept called low level solid radioactive material (LLS). This material would be subject to packing in "strong industrial packages", which, in addition to meeting the general design prescriptions for all packagings and packages, are required to meet the package performance tests specified by the United Nations for the transport of dangerous goods. The 1973 regulations allow solid radioactive waste to be transported as LLS material provided that:[6]

(a) The activity under normal transport conditions is and remains distributed throughout a solid or a collection of solid objects or is and remains uniformly distributed in a solid compact binding agent and does not exceed an average of $2x10^{-3} A_2$ Ci/g and that even under loss of outer packaging the activity loss per package per week resulting from its total immersion in water does not exceed 0.1 A_2Ci; or

(b) For objects of non-radioactive material contaminated with radioactive material, the objects have been cleaned to remove loose contamination and the level of contamination in a non-readily dispersible form averaged over 1 m^2 (or over the surface area if less) does not exceed 2 uCi/cm^2 for high toxicity alpha emitters and 20 uCi/cm^2 for all other radionuclides.

You will probably recognize the provisions of paragraph (a) as those which eventually became LSA-III. The provisions of paragraph (b) eventually became the current provisions of Surface Contaminated Objects (SCO-II). The definition of Low Level Solids lasted through the 1973 Revised Edition as it was amended in 1979, but it was eliminated in the 1985 revision, with its provisions transferred elsewhere.

Mr. Alan Fairbairn, a major contributor to the early versions of the IAEA Transport Regulations, and a prolific writer of regulatory history, offers the following explanation of the limits appearing in the definition of Low Level Solids:[6]

> The leaching rate of 0.1 A_2/week was derived by considering a block of active material in its packaging (usually a reinforced concrete outer vessel with or without an outer steel drum) to have been exposed to the weather and to have leaked in rain sufficiently to have virtually surrounded the active material block with a film of water for one week. It was then assumed that as a result of a mishap during handling some of this liquid escapes and 10^{-3} of its activity content is taken into the

body of a bystander. Since the outer packaging is required to be of good quality meeting the appropriate UN industrial packaging specifications, it was considered to be good enough to limit the escape of liquid to 1/100th as opposed to 1/1000th for a Type A package. Then since the total body intake must be limited to 10^{-6} A_2 to maintain consistency with the safety built into Type A packages, the activity leached from the package immersed in water for a period of one week must be less than 0.1 A_2. This leaching limit, together with the upper limit for estimated average activity content of $2x10^{-3}$ A_2/g, means that for a full-size package (e.g. 200-litre volume weighing 500 kg) the binding of the activity in the active material block must be sufficiently good to limit its loss by water leaching to 0.01% of the total contents per week.

The increase by a factor of 20 for the maximum permissible surface contamination in "non-readily dispersible form" on objects of non-radioactive material of the level specified in the LSA prescriptions represents surface beta dose rates of the order of 2000 rem/h and radiation levels at 30 cm from the surface of some 700 rem/h. These contaminated objects are potentially very hazardous, hence the importance of the packaging specification and the requirements for labeling and shipment under full load conditions. It is important to appreciate that the contamination on the other surface of the packaging must comply with the levels specified in the regulations, which are a factor of $2x10^{3}$ less than the contamination levels specified under this LLS prescription.

The big issues for the Revision Panels associated with the 1985 revision were formal acceptance of the Q-system modeling and new quantity limits for Type A packages, and, in the LSA/SCO area, a limit on unshielded radiation levels to limit the consequences of transportation accidents involving LSA/SCO materials. As a participant in that revision process, I should not be too critical of the resultant regulations, although I cannot understand how we failed to recognize the importance of having the SCO definition include a contaminated solid radioactive object.

In any event, the 1985 LSA definition shown below is relatively consistent with its earlier versions and is fairly free of unimportant detail. The SCO definition, not shown, is in need of additional provisions and needs to be consolidated, as well. I believe the recommendations of the recent Technical Committee will improve that definition considerably.

LOW SPECIFIC ACTIVITY MATERIAL [5]

131. Low specific activity (LSA) material shall mean radioactive material which by its nature has a limited specific activity, or radioactive material for which limits of estimated average specific activity apply. External shielding materials surrounding the LSA material shall not be considered in determining the estimated average specific activity.

LSA Material shall be in one of three groups:

(a) LSA-I

(i) Ores containing naturally occurring radionuclides (e.g. uranium, thorium), and uranium or thorium concentrates of such ores;

(ii) Solid unirradiated natural uranium or depleted uranium or natural thorium or their solid or liquid compounds or mixtures; or

(iii) Radioactive material, other than fissile material, for which the A_2 value is unlimited.

(b) LSA-II

(i) Water with tritium concentration of 0.8 TBq/L (20 Ci/L); or

(ii) Other material in which the activity is distributed throughout and the estimated average specific activity does not exceed 10^{-4} A_2/g for solids and gases, and 10^{-5} A_2/g for liquids.

(c) LSA-III
Solids (e.g. consolidated wastes, activated materials) in which:

(i) The radioactive material is distributed throughout a solid or a collection of solid objects, or is essentially uniformly distributed in a solid compact binding agent (such as concrete, bitumen, ceramic,etc.);

(ii) The radioactive material is relatively insoluble, or it is intrinsically contained in a relatively insoluble matrix, so that, even under loss of packaging, the loss of radioactive material per package by leaching when placed in water for seven days would not exceed 0.1 A_2; and

(iii) The estimated average specific activity of the solid, excluding any shielding material, does not exceed $2x10^{-3}$ A_2/g.

2. CURRENT REVISION CYCLE

The road to the 1996 revision of IAEA Safety Series No.6 began in June 1991 with the first revision panel, under the chairmanship of Bengt Petterson of Sweden. Issues regarding LSA/SCO were considered under a working group led by Ken Shaw of the United Kingdom. A second revision panel met in May 1993 under the chairmanship of Richard Rawl of the United States. Mr. Shaw again led the working group on Radiation Protection, but the working group was instructed to refer any LSA/SCO issues to the scheduled Technical Committee meeting in October 1993.

Of the LSA/SCO issues considered at the first Revision Panel in 1991, only two were resolved. It was agreed to modify and expand an existing category in LSA-I to include "Ores containing naturally occurring radionuclides (e.g. uranium, thorium), ores which have been processed to partially remove the naturally occurring radionuclides, and uranium or thorium concentrates of such ores". Rejected was a second type of waste (e.g. contaminated soil) which is a product of decommissioning activities. The Panel decided that this second category was uncertain as to its properties, and would be better considered and approved under the "special arrangement" provisions. The final issue resolved was whether any critical properties of LSA material would disqualify the material from the LSA category by having the material compacted. The Panel decided that no important properties would be affected.

Other LSA/SCO issues which were considered but not resolved were:

- Classification of solutions with specific activity $>10^{-5} A_2/g$
- Limits for LSA gases
- Radiation limit for unshielded contents of IP packages
- Multiple of A_2 as radiation measurement
- Unit of SCO contamination limits
- Inconsistencies in LSA/SCO definitions
- Specify LSA degree of uniform distribution
- Specify "essentially uniform distribution" for LSA-II
- Phyico-chemical form of material
- Leach test for LSA-III materials, and
- "Solidified intermediate material"

These issues were almost all referred for resolution to the future planned activity to develop a LSA/SCO model which is frequently referred to as the Q-system analog because it would be similar to the Q-system developed in the early 1980's by Eddie Goldfinch and Hugh Macdonald. We have recently discovered that the Q-system analog will not be completed in time to be used for the 1996 regulations, although parts of it may be completed in time to be of partial use.

As noted earlier, issues related to LSA/SCO were not addressed in the second revision panel in May 1993 since it was recognized that these would be specifically addressed by a Technical Committee in October 1993. For both Revision Panels completed thus far, LSA/SCO issues have been referred on to future activities for resolution. Now is appears that the third Revision Panel, in October 1994, will carry the load of whether to support the Technical Committee's decisions.

For the preparation of appropriate analyses and other material for consideration by the Technical Committee in October 1993, the IAEA Secretariat convened three Consultant Services Meetings (CSM). The first CSM, in October 1991, seemed to be a regular CSM with three participants and a number of specific issues which had been submitted by Member States for consideration in the development of the 1996 Revision. The consultants added their views on the issues whether or not they had already been considered and resolved.

The second CSM, in November 1992, also had specific issues to consider, some technical and some organizational. In addition, they also had a document drafting a suggested extension of the Q-system to apply to LSA/SCO. The basis for the suggested changes was to provide consistency between the Q-system and the LSA/SCO categories. A number of radical changes were proposed in the interest of being consistent.

The third CSM, in May 1993, was tasked to review and consider the proposed extension of the Q-system to LSA/SCO and to review the Member State comments which had been solicited on the report of the second CSM. Finally, the third CSM was to prepare a report, including any necessary changes, to the Technical Committee scheduled to meet in October 1993, regarding the proposed extension of the Q-system to LSA/SCO.

The Terms of Reference for the Technical Committee Meeting in October 1993 were to review the issues that have been identified and the work that has been done so far, and to make recommendations on LSA/SCO provisions in the Transport Regulations. All open issues on LSA/SCO that could be identified were presented to this TCM. The decisions of the TCM, including those which require changes to the regulations and those which do not, are as follows:

DECISIONS OF THE 1993 TECHNICAL COMMITTEE MEETING

1. The terminology "Surface Contaminated Object (SCO)" is changed to "Contaminated Object (CO)". The terminology "Low Specific Activity (LSA-I, LSA-II, and LSA-III) remains the same.

2. Extension of SCO-I to include a radioactive object contaminated by radioactive material. A package limit of.002A,was imposed to control external radiation levels. A vehicle limit of 1$A_2$was imposed on unpackaged CO-I dispersible by fire. Unpackaged CO-I contents must be shipped exclusive use. Radiation level at any point 10cm from unpackaged CO-I not exceed <0.1mSv/h.

3. LSA-II, LSA-III, and CO-II are to be shipped in packages which satisfy the performance requirements for a Type A package, but may also be shipped in tank containers, tanks, and freight containers. The conveyance limits found in Table VI of the current regulations should be retained as they apply to LSA-II, LSA-III, and CO-II packages (the limits are changed for LSA-I and SCO-I materials). Any LSA-II, LSA-III, or CO-II packages containing more than an A quantity must be shipped exclusive use unless the quantity in the package is listed as a multiple of A in the Transport Documents. The total activity in these packages is not to exceed 2A, to protect against post-accident excessive radiation levels.

4. A new kind of Type B package, with other than existing package criteria, is not necessary for shipment of LSA/CO material which either cannot satisfy the unshielded external radiation criterion, or which has properties between LSA-II and LSA-III but which satisfies neither. It was determined that the consequence reducing properties of the LSA/CO material can be considered in the design of a Type B package to alleviate some pertinent package features which are quite costly, such as the extreme leak-tightness necessary for other more toxic materials.

5. In reconsidering the issue of including slightly contaminated material, such as contaminated earth, and processed ores from which naturally occurring radionuclides have been partially removed, the Technical Committee rejected the proposal. This same issue had been previously considered by a CSM in March 1991 and fully accepted with a reduction in required specific activity, by an AG in June 1991 and accepted only with respect to the processed ores, and by a TC in June 1992 and accepted only with respect to the processed ores. This TC rejected the entire proposal "for consistency with our requirement that LSA-I concerns only unlimited A_2 materials."

6. To simplify the definition of contaminated objects to take account of the new limits for that category, the restrictions on fixed contamination and contamination on inaccessible surfaces in the current definitions of SCO-I and SCO-II, are eliminated.

7. Paragraph 425 (c) concerning transfer of contamination from the surfaces of unpackaged SCO-I to the conveyance is deleted.

8. Thorium concentrates are to remain in LSA-I.

9. The proposal to increase package activity limits when there are multiple layers of containment for LSA/SCO wastes was not supported.

10. The concept that there should be two individual dose criteria for developing standards for routine conditions of transport, one when failure is due to normal conditions of transport, the other when failure is due to accident conditions of transport, was rejected.

11. The Technical Committee rejected any change to existing marking and labeling requirements for LSA-I/SCO-I shipments.

12. The Technical Committee retained the requirement for exclusive use shipments, as necessary, for the shipment of both packaged and un-packaged material. CS-23 had recommended that the distinction between exclusive and non-exclusive use be eliminated.

13. A proposed downgrading of the packaging requirement for liquid LSA-I in non-exclusive use,from IP-2 to IP-1 was rejected because no compensating requirements were proposed.

14. The existing individual dose criteria for all kinds of LSA/CO shipments are to be retained, at least until the Q-system analog work is complete.

15. The existing modeling which relates the individual dose criteria to the regulations is to be retained except for the substitute use of multiples of A contents limitations instead of unshielded radiation level limitations for controlling direct radiation doses in the event shielding is lost from the package in an accident. The Q-system analog work may change the modeling. The CO-I conveyance limit is .002A, for total radioactive material. The conveyance limit is 1A_2 for radioactive materials dispersible by fire.

16. Material specifications are to remain unchanged. A clarification for LSA-III that powders are not included in its definition has been recommended.

17. Package specifications will remain the same except that the IP-2 and IP-3 packages will disappear as Type A packages will be required for LSA-II, LSA-III, and CO-II materials. Type B package specifications will remain the same, even when applied to LSA/CO contents.

18. No changes to marking and labeling requirements were recommended.

19. Controls during transport for LSA/CO were tightened in the recommendations of the Technical Committee, to assure that the conveyance activity limits imposed by paragraph 427 (including Table VI) of Safety Series No.6 are adhered to. Exclusive use transport was recommended for shipments of unpackaged CO-1, and for LSA-II, LSA-III, and CO-II shipments.

20. The Technical Committee recommended that the leach test for LSA-III materials should be retained.

21. Administrative requirements for packages containing LSA-II, LSA-III, CO-II will be the same as those for Type A packages.

3. RECOMMENDATIONS FOR CHANGE

As a follow-up to the Technical Committee Meeting where many decisions were made, IAEA convened yet another Consultants Services Meeting (CS-92) to take those decisions and turn them into recommendations to the next Revision Panel in October 1994. This requires development of proposed wording changes to Safety Series Nos.6, 7, and 37.

While developing proposed wording changes, the consultants noted some difficulties, in a few cases, with the decisions of the TCM as they were presented in the Working Group reports. These difficulties were identified as the views of the consultants, and they were written down and will be forwarded to the next Revision Panel. Following is a draft listing of the recommendations from the Consultant Services Group for changes to Safety Series No.6, including those few cases where difficulties were identified by CS-92:

1. In order to extend the SCO rules to include a radioactive object contaminated on its surface by radioactive material, the following revisions are necessary.

   . A revision of the definition of Surface Contaminated Object in paragraph 146 would read as follows:

     146. Contaminated Object (CO) shall mean a solid, unconditioned object or item which has radioactive material distributed on its surfaces. CO shall be in one of two groups:

     (a) CO-I: An unconditioned solid object or item on which the non-fixed contamination on the surface averaged over 300 cm^2 (or the area of the surface if less that 300 cm^2) does not exceed 4Bq/cm^2 (10^{-4} μCi/cm^2) for beta and gamma emitters and low toxicity alpha emitters,or 0.4 Bq/cm^2($10^{-5}$$\mu$Ci/$cm^2$) for all other alpha emitters.

     (b) CO-II:An unconditioned solid object or item on which non-fixed contamination on the surface exceeds the applicable limits specified for CO-I in (a) above and on which the non-fixed contamination on the surface averaged over 300 cm^2 (or the area of the surface if less that 300 cm^2) does not exceed 400 Bq/cm^2($10^{-2}$$\mu$Ci/$cm^2$) for beta and gamma emitters and low toxicity alpha emitters, or 40 Bq/cm^2($10^{-3}$$\mu$Ci/$cm^2$) for all other alpha emitters.

   . There are 18 cases where paragraphs and Tables refer to SCO that need to be changed to CO.

   . A revision to paragraph 425 (conditions for unpackaged LSA and SCO) such that paragraphs 425(b) and 425(c) read as follows:

     425 LSA material and CO in groups LSA-I and CO-I may be transported unpackaged under the following conditions:

     (a) (unchanged)

     (b) Each conveyance shall be under exclusive use and, when transporting unpackaged CO-I, the conveyance limit for activity dispersible by fire shall be 1A_2; and

     (c) The radiation level at 10cm from any point on the external surface of any unpackaged CO-I object or item shall not exceed 0.1mSv/h (10 mrem/h).

   . Add a footnote under Table VI as follows:

     Add a footnote indicator on the CO which must replace the present SCO, such as CO*. Add a footnote under the Table to say "Additionally, in case of unpackaged CO-I, paragraph 425(b) has to be applied".

   . CS-92 found no justification to refer to unconditioned solid objects in the definition of Contaminated Object, and recommended that any reference to unconditioned be deleted.

2. The TCM decided that accident induced external exposure for LSA-II, LSA-III, and CO-II packages should be limited by the package contents being restricted to 2A,rather than the present limitation of 10mSv/h at 3 meters from the unshielded material, object or collection of objects. The TCM also decided that, for CO-I, the total activity per

package should be limited to $0.002A_1$. Therefore, the following revisions are necessary.

- Paragraph 422 is amended to read:

  422. The quantity of CO-I in a single package shall be limited to $0.002A_1$. The quantity of LSA-II, LSA-III and CO-II in a single package shall be limited to $2A_1$.

- Although the Technical Committee recommended changing the provisions of paragraph 422 to apply an external radiation limit by limiting the quantity of radioactive material in a single package to $2A_1$, the Chairman of the Working Group (WG), in presenting the WG report to Plenary, pointed out that there was no clear concensus in the WG for the limits of 1)10mSv/h at 3 meters, 2)100mSv/h at 1 meter, or 3)$2A_1$ activity limit. The Consultants Group felt that imposing the $2A_1$ activity limit would reduce the capacity of packages and lead to additional numbers of packages shipped. Without justification, this would impose increased costs and exposures. CS-92, therefore, considered that the present paragraph 422 should not be changed.

3. The Technical Committee (TC) decided that packages for LSA-II, LSA-III, and CO-II shall meet all requirements for a Type A package as specified in paragraph 527. Additionally, the TC decided that these materials may be transported in tank containers, tanks, and freight containers as permitted in paragraphs 523-525. This means that package categories IP-2 and IP-3 would be deleted. As a consequence of this decision, LSA-I liquids carried under non-exclusive use must also be transported in Type A packages because of Table V. The following revisions are necessary.

- Delete paragraphs 521-526 inclusive. Replace with paragraphs 521-524 as follows:

  521. Tank Containers may also be used to transport LSA-II, LSA-III, and CO-II provided that: ...

  522. Tanks, other than tank containers, may also be used to transport LSA-II, LSA-III, and CO-II and for transporting LSA-I liquids .....

  523. Freight Containers may also be used for transporting LSA-II, LSA-III, and CO-II provided that ......

  524. Intermediate Bulk Containers may also be used to transport LSA-II, LSA-III, and CO-II provided that .....
  (replace "IP-3 packagings" in subparagraph b) with "Type A packagings").

- Change paragraphs 439 and 422 as follows:

  439. a) An Industrial Package Type-1 shall be legibly and durably marked with "Type IP-1".

  b) (No changes)

  422. The quantity of CO-I in a single package shall be limited to $0.002A_1$. The quantity of LSA-II, LSA-III, and CO-II in a single package shall be limited to $2A_1$.

- Change Table V to replace all Type IP-2 and Type IP-3 by Type A.

- Delete paragraphs 136(b)(ii) and 136(b)(iii)

- Replace paragraph 136(c) by the following:

  136. c) Type A Package is a packaging, tank, or freight container containing an activity up to the following:

  i) A_1 if special form radioactive material;

  ii) A_2 if normal form radioactive material;

  iii) $2A_1$ if LSA-I liquid transported not under exclusive use; and

  iv) $2A_1$ if LSA-II, LSA-III, or CO-II.

- CS-92 recommends that the change to Type A requirements for LSA-II, LSA-III, and CO-II should not be adopted. IP-2 and IP-3 packagings should be combined into one packaging group to be known at IP-2. This would avoid the imposition of onerous requirements for packages of LSA-II and LSA-III liquids (paragraphs 542 and 543) which have not been justified. The Technical Committees decision to apply Type A requirements for LSA-II, LSA-III, and CO-II packages while allowing alternative designs following UN and ISO standards, e.g. a 9 meter drop test for packages with liquids as against no drop test for a tank container designed to UN recommendations, is unwarranted.

4. The Technical Committee decided that the consignor should assist the carrier in controlling the total activity of LSA-II, LSA-III, or CO-II in a conveyance, as required by paragraph 427, by making the shipment exclusive use if the quantity in any package exceeds the A_2 quantity. However, the TC also recognized that if the multiple of A_2 was listed in the Transport Documents, the carrier would be able to comply with paragraph 427 (including Table VI) and exclusive use would not be required. Following is the single required change.

. A new paragraph to be included as follows:

A consignment of LSA-II, LSA-III, CO-I, or CO-II in which any package contains more than an A_2 quantity shall be shipped under exclusive use unless the quantity of activity in each package is also listed as a multiple of A_2 in the Transport Documents.

5. The Technical Committee decided that powders should not be categorized as LSA-III since the intent of the higher allowed specific activity by a factor of 20 is given because of the indispersibility of the material. Following is the single required change.

. Amend paragraph 132(c), the definition of LSA-III, as follows:

132.(c) Solids, excluding powders, (e.g. consolidated wastes, activated materials) in which: ...... (no further changes)

4. WHERE DOES THIS LEAVE US

Finally, I have looked at the decisions of the Technical Committee and compared them to what needs to be done to make our LSA/SCO regulations workable and defendable. I'd like to share these thoughts with you.

. We need a Q-system analog. When the original Q-system was being considered for acceptance, I thought it needed further development and tried to bring about its defeat. Now I see it as a monument to what can be done with patience and understanding. I now often refer to the Q-system and its components. It has us all looking in the same direction for answers to our questions. We need a Q-system analog in the LSA/CO area, but it cannot be done overnight or in two 3-day consultants meetings. The reports produced by CS-75 and CS-23 will be valuable input to the task ahead, but I stand with those of you who chose to pass up the opportunity for a quick Q-system analog, and chose instead the longer, mature development. The rejection of proposals which are not quite ready for acceptance is what separates us from the politicians, who can't quite reject anything.

. We need some work on individual dose criteria, and modeling, especially in the excepted, LSA-I and CO-I areas. I'm content to let it happen with the development of the Q-system analog, as I know it will.

. We need to develop a rationale for the "essentially uniform distribution" and "distributed throughout" requirements in the definitions of LSA material. There is little agreement among the definitions, the advisory material, and actual shipments being made. The early participants who initiated the LSA category have written of the importance of uniform distribution in the modeling of LSA shipments, but so far we have failed to emulate their attention to important detail. The averaging of specific activity in LSA material depends entirely on uniform distribution. And yet we use the averaging and look away when anyone mentions uniform distribution.

. We have accomplished much in this revision cycle in the area of LSA/SCO. The Technical Committee meeting was, by most standards, a huge success. When we needed to change, we changed. When the right answer was not yet available, we hung on to what we have. We will know better after the next Revision Panel in October how well we have done, but for my eyes the view is much improved. The original framers of these regulations would be proud of the ways in which they have been improved.

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MEMBER STATE EXPERIENCE AND RECOMMENDATIONS FOR INTERNATIONAL TRANSPORT REGULATIONS

(Session 2)

Chairman

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CONTAMINATED OBJECTS IN ACTUAL TRANSPORT

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Abstract

Contaminated objects may in principle be anything. In reality, they can be actually anything.

Their definition criteria given in transport regulations are the following :

1) Radiotoxicity of contaminating isotopes
2) Surface specific activity
3) Fixed or non fixed character of the contamination
4) Surface accessibility.

On the basis of these criteria, two levels of contaminated objects are defined, under certain limits of surface activity, and two levels of packaging (IP1 and IP2).

We examined these criteria in relation with their practical implementation. It appears that some of them are not so easy to consider as it seems, in front of the extreme diversity of situations.

The diversity of contaminated objects shows itself along numerous axes of which the more obvious are :

a) their more or less complex structure
b) their "solidity"
c) their "multiplicity"
d) their dimension

These aspects will intervene, in connection with regulatory criteria, in the definition of their packaging for transport.

The destination of the contaminated objects plays also a part. When they are not intended to be recovered and reused they become wastes and they are normally incorporated in a matrix or compacted. At that time, they are no longer to be considered as contamined objects as least from transport vewpoint, here again the relevant criteria for classification is massic specific activity and they have to be classified as LSA material.

When such objects are supposed to be decontaminated at the end of their voyage and reused, they can no longer be transported as LSA material and their surface may become "accessible".

The new limits given by the SS6 (1985) to the "accessible" non fixed contamination, made it necessary in the general case to carry them in Type A packages.

That is a very new situation when compared in the old one, where no distinction was made between fixe and non fixed, accessible and non accessible contamination.

I. INTRODUCTION

The IAEA Regulations for the Safe Transport of Radioactive Material (Safety Series No. 6, 1985 Edition - As amended 1990) consider contaminated objects from a new point of view.

They parted them explicitely from LSA materials and brought new parameters in their definition as non fixed or accessible character of the contamination, giving to them some stringent criteria.

We consider in this paper the entailed difficulties and the solutions which can be envisaged.

II. THE STRUCTURE OF OBJECTS

The part of surface accessibility, and even its definition needs from our vewpoint, some examination. We think about this matter in a group of several users of the nuclear French industry.

The word accessibility means in a first approach, accessibility to measurements and more specially measurement by sweeping the surface. But that is not completely connected with the hazard presented by the material on the surface. In reality, when the substance is accessible to measurements by sweeping, it can escape during the carriage even in normal conditions.

The reciproque is not true. Many practical cases show possibility for the radioactive material to get out from the contamined surface, while it is impossible to have any material human access to this surface. That is typically the case, e.g. : for narrow open tubes, for narrow pins around the packages for spent fuels, and for many complex devices.

On the contrary, some of these devices can be contaminated, like for instance vapour generators, where it could be relatively easy to contain the contamination inside by shutting the tubes.

That is the reason why we propose to replace the words : "accessible surface" by "unclosed surface" and "inaccessible surface" by "closed surface".

This is important, not only for formalist reasons but for the following

As far as the contamination is contained, and at least in certain limits contained in normal conditions of transport, there are no fundamental reasons to consider contaminated objects where the radioactive material on the surface is forced to remain by shutting or fixing it in a different way that any packaged radioactive material

III. THE "SOLIDITY" OF OBJECTS

During power plant operations and maintenance numerous low contamined wastes are generated such as Vinyl (which represent 60 % of the total volume), gloves, dusters, papers, pieces of cardboard, etc

Inside a metallic drum of 200 l these "technological" wastes are compacted with a 25 tons compacter in order to limit their volume before to be transported to the ANDRA sites disposals

15 000 metallic drums are transported par year inside ISO freight containers

It is indeed very unsuitable to measure the surface specific activity of each "piece" of wastes and because of the PWR plants contamination is only coming from β or γ emetters it shall "cost" a lot in radiation exposure of workers who verify the contamination

So, we consider these compacted wastes in this metallic drum of 200 l not as O C S but as a material in which the activity is distributed "throughout" As they are very low activity wastes (maximum 400 MBq or 10 μCi per drum), each package shall be classified as Industrial Package type 2 (IP-2) containing LSA II material and transported in an overpack

In the power plant, workers measure the surface radiation level of each metallic drum and convert into total activity inside using a "transfer fonction'

This method ensure an improvement of the dose limitation for consignors

IV. THE "MULTIPLICITY" OF OBJECTS

a) Multiple, interchangeable objects

Some contaminated objects, where the radioactive material is able to escape are nevertheless difficult to consider and classify as such they are indeed small, numerous and interchangeable That means difficult to tell one from the other Their set is generally considered as a unique lot of low specific activity material, the significant parameter being then the massic average specific activity

Measuring the surface activity of one or several of these objects doesn't present any warrant of time and spatial constancy

Therefore it is more logical to consider them as low specific activity material That could be the case for nails, screws, bolts, etc

b) Non-contamined objects

For tool-outfit which have to return in the public domain", it is impossible to measure the non contamination threshold of 0,4 Bq/cm² during power plant operations , the assessment is indeed quite uncertain because of the background noise which often overtake 4 Bq/cm²

The Regulations (S S 6) don't specify if this non contamination threshold means the presence of non fixed or/and fixed radioactive substance on the surface

It is obvious, for the numerous small tools which are used by workers for maintenance, that the contamination is non-fixed, because the movement of these tools is almost permanent and they are cleaned as far and often as possible

Then, if the non-fixed contamination on the tool's surface doesn't exceed 0,4 Bq/cm² (the rubber shall be verify in an area without background noise) and if the worker also verify the surface radiation level of the tool on the maintenance's premises is less than the background noise, the tool is non-contaminated

c) Very-low contaminated objects

Some very low contaminated tools can slightly overtake the non contamination threshold of the SCO I (4 Bq/cm²) or of the SCO II (400 Bq/cm²)

These overtakes regard a very few points of the same tool Also, we can ask the following question can we overtake the SCO limits if the total activity inside the packaging doesn't reach the "excepted" package limit activity i e 10^{-3} A2 ?

A similar question was put on the floor for LSA materials, it's an other story but we could have a similar answer as least when A2 has a limited value

- **If the non fixed contamination is between 0,4 Bq/cm² and 4 Bq/cm² (SCOI)**
  When the total activity inside the packaging is higher than 10^{-3} A2, it has to be transported like a IP I package
  But if the total activity inside the packaging is less than 10^{-3} A2 and if the surface radiation level of the package is less than 5μSv/h, the transport worker can transport it like a "excepted" package after verifying the external non fixed contamination of the package in a proper area (without background noise)

- **If the non fixed contamination is between 4 Bq/cm² and 400 Bq/cm² (SCO II)**

The package has to be transported like a IP2 package as it is requested by the Regulations.

But, as above, if the total activity inside the packaging is less than 10^{-3} A2 and if the surface radiation level of the package is less than 5μSv/h, the transport worker can transport it like a "excepted" package after verifying the external non-fixed contamination of the package in a proper area (without background noise).

- **If the non fixed contamination is "slightly" higher than 400 Bq/cm²**

The large tools used for maintenance into the reactor building or into the fuel building (they are submerged in the water of the pools) are fastly rinsed and cleaned to limit radiation exposures to workers.

The measurements made on these tools show that the non fixed contamination in some points of some tools is above the SCO 2 threshold (400 Bq/cm²). However the level doesn't exceed 1000 Bq/cm² maximum.

The Regulations request normally a Type A package. As we already say these tools are very large. They must be transported in freight containers type "open-top" because they have to be loaded on the top in the power plants.

It is quite impossible of complying with the requirements for Type A package especially the drop test and the reduction of ambient pressure to 25 kPa.

The assessments of the total activity inside these freight containers give a value less than 10^{-3} A2, in all cases. As above, if the surface radiation level of the freight container is less than 5 μSv/h it can be transported like a "excepted" package.

But if the surface radiation level of the freight container is higher than 5 μSv/h, could it be acceptable to transport these ISO containers like IP2 package in accordance to the § 523 of the SS6 Regulations instead of Type A package as requested ?

It is proposed that the Industrial Packages be canceled in the next IAEA Regulations. Only Type A package would be retain for transport in normal conditions. But the last Technical Committee has "saved" the § 523 above : does that mean that ISO containers should be used as Type A package ?

This question could have a very important industrial impact (fifty lorries each day in France) and has to be seriously examined for example considering the general radiological exposure to workers.

V. THE DIMENSIONS

In maintenance and decommissionning many objects may have to be carried which have no longer anything to do with packages purposely built for transport of dangerous goods. There are not intended for being transported, but they have to.

That is a somewhat specific and new situation, where the leading factors are no longer the compliance with the Regulations, but features which we have to deal with and to carry anyway.

The dimension is not taken in account in regulations however it has obviously a direct impact on total carried activity. The same surface specific does not obviously give the same dangerous character to a contaminated hammer or to the contaminated lid of a nuclear reactor.

We could quote here, the case of contaminated lids of nuclear reactors or the one of vapour generators : it would be very unprofitable and very prejudicial in term of exposure of the workers to decontaminate them near the reactor they come from.

Beside there could be in some extend an activation of these objects, even if it is often very difficult to tell the difference between activation and fixed contamination at least of the level of transport.

For such big objects Type A packages are already very difficult to design and manufacture, Type B packages which in certain cases would be Regulatory needed are almost impossible to conceive as far as the objects themselves present dimensions and weights which are at the limit of what is transportable.

In this context the use of Special Arrangement is difficult to avoid and the concept Transport System could in this case be developped in very useful manner.

CONCLUSION

There are many cases in the consideration of contamined objects, we tried to underline certain aspects of their multiplicity and complexity. There are certainly many other ones.

It seems that, this effort in accuracy made in Regulations SS6 1985 (amended in 1990) were certainly positive, but the implementation of these new Regulations show that many difficulties remain.

If the consideration of closed and unclosed surfaces and also fixed and non-fixed contamination is certainly a way to facilitate the classification of contaminated objects in transport, the limits of non contamination in the new Regulations entail many pratical difficulties, the thresholds are indeed rather low and it is sometimes difficult, time consuming and prejudicial from the exposure viewpoint to reach them.

A possible solution could be, when the total activity is low enough, that the criteria of classification be this total activity instead of the surface contamination only.

We have also considered the real problem of objects of which form and dimension do not adapt themselves very well to standardized packagings, even IP2. A regulatory way for carrying them should be preferable to the systematic use of special arrangements.

TRANSPORT CATEGORIES FOR RADIOACTIVE WASTE

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Abstract

This paper reviews past and present regulatory requirements for the transport of waste materials in other than Type A and Type B packages It identifies three groups of materials analogous to Excepted packages, Type A packages and Type B packages into which the various waste forms can be fitted, so that they may be transported to the same levels of safety as their counterparts The paper makes proposals for materials which are intrinsically safe without packaging other than for administrative convenience and for wastes to be transported to the same levels of safety as Type A packages It is proposed that waste forms to be transported to the same level of safety as Type B packages cannot be prescribed in advance without the need for Competent Authority approval for each specific form or combination of waste form and packaging Finally it is proposed to revert to simple packaging requirements, equivalent to the earlier industrial and strong industrial packages The former have no quantitative performance requirements and the latter have requirements identical to Type A packages

1 INTRODUCTION

One of the most topical subjects under review within the IAEA Regulations for the Safe Transport of Radioactive Materials[1] is that of transport of low level radioactive waste A wide range of consultancy groups and technical committees have been or will be arranged by the Agency This Seminar should contribute very significantly to understanding the problems and their solutions, and, hopefully, assist in the current review and revision process for the regulations leading to the issue of the next edition around 1996 The author of this paper has been fortunate enough to have been involved with the development of thoughts and ideas, including participation in the two most recent consultancy groups and technical committee The ideas and proposals in this paper therefore reflect the input of many people at various stages of the development The author gratefully acknowledges the stimulus from discussions at the above meetings It is hoped that the final recommendations reflect the best of the ideas wherever they came from It is not practical to acknowledge ideas individually

2 HISTORY

Early editions of the regulations recognised the practical need to transport radioactive waste materials in packages other than Type A or Type B For such materials either the specific activity or the physical form or both give a degree of inherent safety not taken into account in the factors limiting the contents in Type A packages or the leakage from Type B packages A number of fundamental radiological principles have been applied historically in developing the IAEA Regulations for the Safe Transport of Radioactive Materials Some have remained unchanged since the publication of the first edition in 1961 Others have matured or been adapted to meet changing international radiation protection recommendations A considerable degree of rationalisation of the justification of the radiological standards in the transport regulations has been introduced in recent years by the development of the Q system[2]

1 1 1961 regulations

The 1961 regulations allowed shipments of unirradiated uranium or thorium ores and concentrates of these materials and residues from the processing of these materials to be exempted from the basic packaging and labelling requirements of the regulations, subject to exclusion of moisture, beryllium and pile grade graphite, and subject to quantitative radioactive contents limits in the vehicle, container or compartment It was required to be packed in strong, leak-proof packages or into vehicles or compartments thereof so that there would be no leakage during normal transport and no contamination of the conveyance, with loading and unloading under the direct supervision of the consignor or consignee Other low activity materials were allowed, with the same container, vehicle or compartment limits as above, but without the necessity for <u>strong</u> industrial packages The consignor and consignee were responsible for the radiological control of the handlers At that time radionuclides were grouped into only three groups in the regulations and the contents in both Type A and Type B packages were limited to specified quantities for each of the three groups

1 2 1967 regulations

At the time of the 1967 regulations radionuclides were divided into seven groups Contents for Type A and Type B packages were still both limited quantitatively but higher activities were allowed as large radioactive sources with the packages having to withstand more hostile environmental insults The range of allowed low specific activity materials was extended to cover externally contaminated non-radioactive items and tritiated water Strong industrial packagings were required for the uranium and thorium materials, but liquids and gases were excluded Further relaxations were given for transport by exclusive use vehicles (then called Full Load)

1 3 1973 regulations

The 1973 regulations introduced the concept of A_1 and A_2 values for individual radionuclides and applied these to the quantitative description of low specific activity materials The limit for low specific activity was basically 10^{4} A_2 per gram, essentially the same criteria being applied as for radionuclides listed as unlimited in the A_2 table, in other words limited by the radiological consequences of inhaling no more that 10 mg of the material Non-radioactive contaminated items were still allowed In addition categories of low level solid materials were introduced These allowed for the transport of consolidated non-leachable materials at specific activities 20 times higher and non radioactive items contaminated to levels also 20 times higher, than for the corresponding low specific activity materials The accompanying advisory material drew attention to the fact that these latter contamination levels gave rise to surface dose rates of 20 Sv per hour and radiation levels of 7 Sv per hour at 30 cm The contamination had to be non-dispersible but no comment was made about its accessibility Of the seven categories of Low Specific Activity material five could be carried in bulk as full load or alternatively in industrial packages and two in strong industrial packages The Low Level Solid materials had to be carried as Full Load in strong industrial packages

1 4 1985 regulations

The 1985 Edition of the regulations and the amended version in 1990 introduced a reorganisation of the fundamental types of material into two groupings namely three categories of Low Specific Activity and two categories of Surface Contaminated Objects LSA-I was essentially the same as the earlier uranium/thorium categories but materials with unlimited A_2 were included LSA-II limited materials to 10^{-4} A_2 per gram with an arbitrary reduction factor of 10 for liquids Tritiated water up to 0 8 TBq per litre (20 Ci per litre) was included LSA-III was relatively insoluble consolidated solid wastes with a specific activity 20 times higher than LSA-II This factor was a simple carry over from the Low Level Solid category of the 1973 regulations The two Surface Contaminated Object categories derived directly from one each of the LSA and LLS categories of the earlier regulations However subdivisions allowed for various combinations of loose and fixed contamination in accessible and non-accessible locations The origin of the levels of contamination, namely the risk from high radiation levels at or near the surfaces, was ignored Interpretation of these requirements has been somewhat difficult The greatest complication introduced was that of three types of industrial packagings, namely IP-I, IP-II and IP-III, the first corresponding roughly to the earlier industrial packages, the second to strong industrial packages and the third to Type A However the distinction between IP-II and IP-III was minimal Allocation of each of the waste categories to a packaging type appears to have been somewhat pragmatic and complicated by arbitrary relaxations for the exclusive use of conveyances Conveyance activity limits were set dependent upon the type of material and its combustibility Radiation level limits on the unshielded contents were introduced to take account of the consequences of accidents leading to loss of shielding

1 5 Regulations review process

In the deliberations on the suggested amendments to the 1985 regulations, leading to the amended version in 1990, certain suggestions were regarded as major changes and could not be dealt with within the revision process until the major review, now in process, which will give rise to the 1996 edition, in other words the ten yearly review The problems identified related to the pragmatic factors embodied in the relationships between material types and corresponding packaging requirements, the use of freight containers as packagings and the inbuilt requirement to limit contents from an external radiation point of view on the basis of the dose rate from the unshielded contents This did not allow designers to incorporate shielding which could be shown to withstand the most severe accidents without impairment

2 UNDERLYING PRINCIPLES WITHIN THE REGULATIONS

The regulations embody two main principles Firstly all limitations where relevant, are now cast in terms of A_1 and A_2 Idealistically these parameters may be considered as yardsticks of risk for each radionuclide for external and internal dose routes, respectively This may seem a little simplistic but it is not far from true, at least for pure radionuclides Secondly, and historically, packagings and transport controls relate to three perceived levels of hazard, namely those arising from routine transport operations those arising from the incidents likely under normal conditions of transport and those arising following severe accidents Acceptable radiological consequences are related to packagings required to withstand each type of situation, respectively These can be considered as three groupings, one required to withstand only routine conditions of transport, one required to withstand incidents associated with normal conditions of transport and one required to withstand severe accidents These will be referred to hereinafter as Group I, Group II and Group III However, as an over-riding feature the radiological consequences of severe accidents must be able to be shown to be acceptable whatever group the material being transported falls into

2 1 Q system

The foundation to the current regulations, the Q system, limits contents in Type A packages (Group II) by modelling radiological consequences following severe accidents, to a limiting dose of no more than 50 mSv These packages must be shown to give rise to quite minimal radiological consequences following exposure to the conditions relevant to the respective Group, and so the requirements are quite restrictive, namely no loss of contents and only a small increase in the external dose rate The A_2 values are used in quantifying the allowable leakage from Type B packages (Group III), following tests to simulate the effects of very severe accidents The limiting consequences are again set at 50 mSv At the other end of the scale we have Excepted Packages (Group I), for which contents are much more severely limited than for Type A packages At this stage in the development of the regulations, the radiological modelling leading to contents control and dose control is not consistent for Excepted Packages and materials (Group I) and control is exercised by factors which have a significant degree of pragmatism. The basic content limit for Group I is 10^{-3} of that for Type A (Group II) but because of the assumptions as to the behaviour of Excepted Packages, which are assumed to be completely destroyed and the whole of the contents dispersed, the accepted external dose is 0 05 mSv, whereas the accepted internal dose is 50 mSv

3 CATEGORISATION OF WASTE MATERIALS

In this paper the underlying logic of three Groups of materials is carried over to waste materials Proposals are developed for the radiological modelling for materials divided into the three groups, as above However, as a consequence, the lack of logic in Group I referred to above will be exposed It would make sense, but is not the purpose of this paper, to revisit the requirements for Excepted Packages and materials, in order to unify the radiological logic throughout the regulations Suffice it to say that it is immediately obvious that some established levels could be increased and some could be decreased History has shown, though, that there is always significant reluctance to either increase or decrease levels which have been seen to be sacrosanct for many years

3 1 Radiological modelling criteria

In each group of materials it is necessary to propose and defend acceptable radiological modelling criteria, consistent with the Q system models The basic proposals are

3 1 1 Group I (withstand routine conditions only)

Routine conditions	occupational dose limited by package external dose rates the same as for the present excepted packages
Normal conditions	0 5 mSv
Severe accident	50 mSv

3 1 2 Group II (withstand normal conditions of transport)

Routine conditions -	occupational dose limited by package external dose rates the same as for the present Type A packages
Normal conditions -	occupational dose limited by package external dose rates the same as for the present Type A packages
Severe accident	50 mSv

3 1 3 Group III (withstand severe accidents)

Routine conditions -	occupational dose limited by package external dose rates the same as for the present Type B packages
Normal conditions -	occupational dose limited by package external dose rates the same as for the present Type B packages
Severe accident	50 mSv

It is apparent that for severe accidents acceptable consequences are the same for all groups However, the severity of the incident which is regarded as an accident and would be allowed to cause these consequences is quite different For Group I, required to withstand virtually no insult, even an incident corresponding to normal conditions of transport may be expected to cause major or complete package failure The allowable internal dose consequences proposed here are reduced by a factor of 10 compared with those accepted for a major accident i e 5 mSv instead of 50 mSv, because of the high frequency of use of packages in this category

It can be argued that the total risk from transport operations is inversely proportional to the perceived risk in each category because the total risk is determined by multiplying the number of movements by the radiological consequences The apparent risk from the transport of Excepted Packages is higher than that from Type A packages, which in turn is higher than that from Type B packages This is because the underlying assumption for Excepted Packages that the whole of the contents become dispersible compared with between 10^{-2} and 10^{-4} for Type A packages (Q system model) is pessimistic, except for severe accidents

3 2 **Assumptions and requirements**

It is now necessary to make assumptions or place requirements as to the behaviour of packages within the respective groups following incidents at the required level of severity, consistent with the Q system assumptions Thus it is assumed that for

Group I the entire shielding is lost, the entire contents are released and 10% of the contents become dispersible following an incident to simulate normal conditions of transport It is further assumed that the entire shielding is lost and the whole of the contents become dispersible following severe accidents Severe accidents, of course, include fires These minimal requirements are effectively the same as having no packaging and allow the potential for material to be unpackaged other than for administrative convenience

Group II assumptions are ready made in the case of Type A packages For waste materials it again should be assumed that shielding is lost after a severe accident, but that this should not be so following 'normal conditions' incidents. The properties of the materials will dictate the internal dose consequences However the Q system assumption of a limiting physical intake of 10 mg of the material (viz as used for unlimited A_2 nuclides) can be well utilised

Group III requirements are ready made in the form of Type B package requirements, namely minimal loss of shielding following normal conditions incidents and shielding retention to limit the external dose rates following severe accidents For Type B packages leakage is limited to 10^{-3} A_2 in a week but this is derived from an acceptable post accident dose of 50 mSv For Group III wastes the same post accident dose criteria are appropriate, although not necessarily applied as a leakage rate in the case of internal dose consideration

4 TYPES OF WASTE

Let us now consider the types of waste that can be fitted into each of the groups and consequently the requirements which must be placed on these wastes or the assumptions which may be made about their behaviour following specific incident or accident tests

4 1 Group I

Into this group, natural ores containing uranium and thorium, concentrates of such ores, solid unirradiated natural or depleted uranium or natural thorium, or their solid, liquid compounds or mixtures, fall quite readily, as in the present regulations The present LSA-I also includes radioactive material, other than fissile material, for which the A_2 value is unlimited The present SCO-I should also be considered within this Group

4 1 1 *Internal dose*

The assumption which is used in the Q system to determine which radionuclides have unlimited A_2 is that a person cannot inhale more than about 10 mg in a dusty atmosphere without physical rejection Thus any radionuclide for which 10 mg contains less than the annual limit of intake (ALI) would have an unlimited A_2, because the resultant committed effective dose equivalent would be less than 50 mSv Whilst in a major accident, for which a dose of 50 mSv is not unacceptable, it can be conceived that the material could be dispersed sufficiently that a person could inhale the fine dust, this is highly unlikely following the type of incident which occurs during normal transport operations In addition these radionuclides are rarely transported as pure radionuclides, although sometimes the natural element may be so transported (e g uranium and thorium as above) Thus it seems fully justifiable to include these 'unlimited' materials within Group I In short, therefore, the constituents of the present LSA-I fall naturally and completely into Group I It is suggested that these materials be named Low Specific Activity (LSA), and need no packaging for radiation protection purposes

Consider now objects or items which may be contaminated and include induced activity Such materials arise in every day operations at nuclear power stations and during decommissioning Some materials may arise from hospitals although these are most unlikely to be activated It is quite impossible to distinguish radiation arising from activation from that due to fixed contamination Thus it is necessary to impose requirements both on the level of non-fixed contamination on accessible surfaces, on the radiation level arising from fixed contamination on accessible surfaces, on contamination in inaccessible places and on the radiation level due to activation If no requirements are placed upon the packaging the levels of loose contamination on accessible surfaces should be limited to that used in workplaces, namely 4 Bq per cm^2, averaged over 300 cm^2 for beta and gamma nuclides and 0 4 Bq per cm^2 for alpha emitting nuclides on external or readily accessible areas These values were originally derived for long term occupational exposure and use of the same values can be justified on the grounds that minimal controls would be necessary in the event of dispersal of the contents

So far as physical or chemical description of the Group I material is concerned there need be no limitations except that the material should be solid The total quantity of activity in a combustible form should be limited to A_2, because of the potential for dispersion in a fire

It is suggested that these materials be named Low Level Solid (LLS-I)

4 1 2 *External dose*

The radiation levels at 1m from natural ores or concentrates, etc , are sufficiently low that a dose of 0 5 mSv would not be reached after a 30 minute exposure if all containment or shielding is lost (maximum 0 2 mSv for uranium or thorium ores) For the LLS materials an external dose of 0 5 mSv would be reached after a 30 minute exposure as used in the Q system, if a contents limit of 10^{-2} A_1 is imposed This can be compared with a basic limit of $10^{-3} A_2$ for excepted materials, which would give a projected dose of 0 05 mSv

4 2 Group II

Group II materials should achieve the same levels of safety as Type A packages, both following incidents representing normal conditions of transport and accidents It is necessary to consider both internal and external dose routes and to attempt to take account of categories of materials in the existing regulations, namely material of specific activity higher than Group I and contaminated radioactive materials at levels higher than in Group I The activity is distributed throughout the material, such as solid activated objects, solid conditioned wastes, aqueous and non-aqueous wastes, ion exchange resins etc These materials may be of low density or compacted to high density They may be or may contain combustible materials Each has the potential to give both internal and external radiation dose following either a normal conditions of transport incident or an accident

4 2 1 *Internal dose*

In Type A packages the total contents are limited in accordance with the Q system No account can be taken of the physical or chemical properties of the material other than the relaxation for special form material Thus even if the material is known to be in a much less dispersible form than assumed in the Q system modelling for content dispersal and intake after an accident, no advantage can be taken of this However there is sound justification to transport such material provided it has properties or can be controlled such that the internal and external doses calculated for Type A packages would not be exceeded Thus Group II materials should be able to be transported in packages at an equivalent level of safety to Type A packages without necessarily meeting the specific Type A content restrictions The existing regulations, and in particular the LSA II category, provide a suitable mechanism in materials where the intake would be limited to 10 mg and for which that 10 mg contains no more activity than 1 ALI, namely where the specific activity is less than 10^{-4} A_2 per gram The packaging should be such that this intake could only be possible in a severe accident and thus following tests to simulate normal conditions incidents it should be shown that there is no loss of contents

This requirement is exactly analogous to the Type A requirement The difference is that the potential intake is assumed to be no more than 10 mg, rather than no more than an assumed proportion of the contents Thus for this material there is no need to impose an upper limit on the contents for the purpose of controlling internal dose For non aqueous liquids there is no logical reason to impose any further concentration restriction but for aqueous solutions the specific activity of the solute should be limited to 10^{-4} A_2 per gram It is suggested that these materials be named Intermediate Specific Activity (ISA)

For material which is essentially non-combustible, it is possible to consider transporting such materials at a higher specific activity at the same levels of safety In this context non-combustible is taken to mean materials with a flash point less than $55^{o}C$ Care should be taken to avoid materials which are not themselves combustible but form combustible gases in contact with moisture or water Specific guidance on the combustibility of many materials can be found in the International Chemical Safety Cards issued jointly by the International Programme on Chemical Safety and the Commission of the European Communities(3) Thus for conditioned non combustible and hence non-dispersible wastes it is not unreasonable to argue that the allowable specific activity should be increased by a factor of 10 to 10^{-3} A_2 per gram, since the mechanism of dispersion by fire does not exist This relaxation only applies, of course, to solid materials Conditioning refers to processes designed to immobilise the material, such as are currently described in the present LSA III category, albeit at a higher specific activity These materials would generally be classed as intermediate level waste and it is suggested that this category of waste (Group II) should be included in the ISA category, and could be named ISA (NC), to represent non-combustible intermediate specific activity waste

Turning now to contaminated or activated materials, where the activity cannot be expected to be uniformly distributed, or necessarily non-dispersible, it is difficult to see how any significant relaxation against the fundamental Type A package criteria can be justified The radioactivity may be in the form of non-fixed contamination, fixed contamination and activation Fixed contamination or activation products may become dispersible under fire conditions However, if the contents are essentially non-combustible the proportion of the fixed contamination likely to be dispersed is quite small Thus it is proposed that this category should be limited to contents no more than A_2 of combustible material to limit the potential for internal dose It is suggested that these materials should be named LLS-II

For tritium there is no need to change the requirements from those in the present regulations for LSA-II tritium No total package content limit is necessary

4 2 2 *External dose*

In both of the above cases, content control is necessary to limit external dose following an accident In the case of Type A packages the control is exercised by assuming that all shielding is lost, including the effects of any self shielding, except in the case of beta emitting radionuclides The materials likely to be transported as Group II wastes are likely to be more bulky than Type A package contents However it is difficult to quantitatively justify a factor of relaxation of more than 2 because of the diverse nature of the material carried It is thus proposed to apply a contents limit of $2A_1$ It should, however, be required that, analogous to Type A packages, the surface dose rate on the surface of the package does not increase by more than 20% following the normal conditions of transport tests

4 3 Group III

It would be highly desirable to be able to propose and justify materials of higher specific activity and higher contamination or irradiation levels to be placed in Group III These would approximate to the existing LSA-III and SCO-II In the present regulations these two types of materials embody factors of 20 increase in the criteria for LSA-II and SCO I, without radiological justification Indeed it would probably be easier to make a case that the factor of 20 is unacceptably high than that it is acceptable The simplistic argument that it already exists in the regulations and should not be changed is not sensible There is now no substantiation for the factor of 20 As pointed out earlier the radiation dose from unshielded beta emitting objects is unacceptably high The present requirement to limit contents so that the radiation level at 3m from the unshielded material or object or collection of objects is difficult to interpret and is not conservative in the case of a collection of objects with a high content of beta emitting radionuclides

Any attempt to make proposals with radiological justification based on dosimetric modelling would require material descriptions of a very prescriptive nature in order to justify the radiological modelling It would be possible, for example to prescribe a material with known dispersal characteristics, but the prescription could not be generic it would have to be based upon the known characteristics of the material Indeed it has been suggested that it may be possible to develop an analogue for the Q system by identifying and characterising a wide range of waste materials The dosimetric modelling would have to be supported by extensive experimental work and could possibly identify generic types of wastes with known dispersal characteristics, from which allowable contents limitations could be set An example may be bitumen, for which the behaviour in a fire may be well known However the behaviour of bitumen could well be modified by the addition of waste materials It would be better to justify the safety on a case by case basis, allowing demonstration of compliance with the internal and external dose criteria to be allowed on the materials themselves or in combination with the packaging In other words the designer or consignor needs to seek competent authority approval in each case Thus, effectively there can be no relaxation on the requirement for Type B approval, unless or until work on a Q system analogue has been completed Since the approval would be for the package and not the packaging, the well established methods of demonstrating safety can be used There need be no implication that all of the safety be built into the packaging The behaviour of the contents (e g combustibility) can be taken into account Indeed, it is possible that thermal testing could be done on the contents and impact testing on the packaging, or vice versa, for some types of consignments

5 PACKAGING AND CONVEYANCE

For Group I materials, for which the justification above has shown no need for packaging integrity, any packaging used will be used for administrative purpose only It is suggested that such packaging can be simply described as industrial This may include freight containers without performance testing although it goes without saying that the integrity should be adequate to contain the material under conditions of routine transport, not so much from the point of view of safety, but public relations Although packaging is not required for safety reasons for LSA materials, where it is transported unpackaged this must be under exclusive use, for clear reasons In the case of LLS I, for which the allowable loose contamination levels on surfaces or readily accessible areas is in fact no greater than would be allowed on the outside of the packaging, the requirement to limit the contents to A_2 dictates that some form of packaging is necessary to identify the contents to which the limit applies This may be in the form of strong sealed polythene or other containment

For Group II materials, packaging of the same integrity as Type A packages is required The only difference between such packaging carrying LSA and LLS I materials instead of Type A contents is that contents limitations can take some account of the inherent safety properties of the materials Performance testing requirements should, however, be identical to Type A packages To avoid confusion these packages could be named Strong Industrial packages This type of packaging may include freight containers or tankers, provided that the contents conform with the requirements above In a serious accident it can be expected that integrity will be lost and no justification can be found to relax the contents limitations because of the physical size and capacity of the containers Performance against the tests to represent normal conditions of transport, viz no loss of contents, and an increase in radiation level limited to 20% is still necessary However it does not seem reasonable that incidents likely to be encountered in normal conditions of transport be represented either by a drop in the attitude of transport, or by a drop perpendicular to this It is therefore suggested that the necessary test prescription be a horizontal impact onto a solid target at a velocity of 2 km per hour

For Group III materials no packaging prescription is required to meet the recommendations of this paper Instead demonstration of compliance with all the Type B requirements is necessary, albeit that account can be taken of the physical properties of the contents in demonstrating compliance to the Competent Authority

The present Regulations allow some relaxation for transport by Exclusive Use and incorporate conveyance contents limits It is a matter of conjecture whether exclusive use may reduce the probability of accidents but it cannot be argued that the potential severity can be reduced Thus no quantitative advantage can be gained in terms of contents limits and no relaxations can be recommended However, common sense dictates that where materials are transported unpackaged, this must be under Exclusive Use Similarly, since the recommendations in this paper demonstrate the intrinsic safety of the packages, there is no need to apply conveyance limits specific to the transport of waste materials Whatever conveyance limits are necessary for consignments of Type A packages then the same reasoning would be applicable to wastes

6 CONCLUSIONS

This paper gives proposals for the categorisation of radioactive wastes into three groups (Group I, Group II and Group III) to be transported at levels of safety equivalent to Excepted packages, Type A packages and Type B packages, respectively The corresponding packagings would be industrial (or no packaging), strong industrial (identical to Type A) and Type B No relaxations for transport by Exclusive Use are recommended The proposals are summarised in Table 1 The proposals show great similarity with existing or past regulatory requirements but quantitative justification is provided

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[3] International Chemical Safety Cards, EUR 14410/1, ISBN 92 826 3697 6 (UNEP/ILO/WHO) (1993)

Table 1 Summary of contents, proposed names, packaging types and package limits

Contents	Proposed name	Type of package	Package limit
Group I (equivalent to Excepted Packages)			
Natural ores, concentrates, solid or liquid natural or depleted uranium or thorium compounds and mixtures	LSA	Industrial (or no packaging)*	n/a
Radioactive material, other than fissile material, for which the A_2 is unlimited	LSA	Industrial (or no packaging)*	n/a
Contaminated irradiated or unirradiated solid materials, with loose contamination up to 4 Bq per cm^2 beta/gamma or 0.4 Bq per cm^2 alpha	LLS-I	Industrial	A_2 and $10^{-2}A_1$
Group II (equivalent to Type A packages)			
Solids or liquids with the activity distributed essentially uniformly and up to 10^{-4} A_2 per gram, and for aqueous solutions no more than 10^{-4} A_2 per gram of solute, and tritium up to 0 8 TBq per litre	ISA	Strong industrial (identical requirements to Type A)	A_2 combustible, and $2A_1$ total (except tritium)
Conditioned non-combustible solids up to $10^{-3}A_2$ per gram	ISA (NC)		
Contaminated or irradiated solid objects	LLS-II		
Group III (equivalent to Type B)			
All materials in excess of ISA, ISA (NC) and LLS-II	n/a	Type B	Competent Authority approval

* Exclusive Use conveyance if unpackaged

PRELIMINARY INVESTIGATIONS ON RADIOLOGICAL CRITERIA AND REQUIREMENTS FOR THE TRANSPORT OF LSA AND SCO MATERIALS - GRS/IPSN/NRPB COLLABORATION

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Abstract

In the frame of a CEC study contract which started in November 1993 a joint collaboration of GRS (Germany), IPSN (France) and NRPB (United Kingdom) is aiming to develop radiological criteria and requirements for the transport of LSA and SCO materials. The objectives of this feasibility study is to group materials according to release behavior under accident conditions, to specify material requirements accordingly and to derive package content limits. The structuring of the study and preliminary considerations will be presented.

1 INTRODUCTION

Within the present revision of the IAEA Transport Regulations one of the important issues is a review of the requirements concerning material properties, activity and activity concentration limits of LSA and SCO materials. It has been recognized that the present radiological basis for the derivation of requirements should be reviewed and very likely be further developed and extended.

With the aim to contribute to this important objective a CEC sponsored collaboration between GRS (Germany), IPSN (France) and NRPB (United Kingdom) is presently conducting a feasibility study. It is hoped that the results of the study are a starting point for further developments in this field. The one year contract started beginning of November 1993 and is structured in the following way:

2 OBJECTIVE

The aim of the study is to define 4-5 material groups to replace or supplement LSA II, LSA III, SCO I and SCO II. Materials will be grouped in such a way that within one group the release behavior under mechanical or thermal impact is comparable. Potential radiation exposures after an accident should be limited to 50 mSv. From this requirement radionuclide-specific content limits are derived. As in the Q-system for Type A-packages several exposure pathways are considered.

3 COMPILATION OF MATERIALS TO BE SHIPPED AS LSA/SCO

In order to derive new material categories information about the kind of LSA/SCO materials produced and needing shipment in the future will be compiled and tentatively grouped. Grouping of materials is restricted to solid materials in the present stage. General notions in doing this are criteria such as dispersibility after mechanical impact, combustibility of the radioactive material or of a compact binding agent, low or high melting point. Special attention will be given to the necessity of SCO. Who is really using it? Is the practice acceptable in this way? What problems exist concerning accessible and inaccessible surfaces, fixed or non-fixed contamination, distinction between SCO and LSA? Is shipping of SCO materials without packaging really needed? What is expected in the future to result from decommissioning?

4 GROUPING INTO MATERIAL CATEGORIES

The quantity to express release behavior of materials and/or packages is probably release fraction of radioactive contents. As one possibility to proceed it is envisaged to define a number of categories with e.g. release fractions of $< 10^{-6}$; 10^{-6} to 10^{-5}; 10^{-5} to 10^{-4}; 10^{-4} to 10^{-3}; 10^{-3} to 10^{-2}; 10^{-2} to 10^{-1}. Separately for mechanical and thermal impact to each of these release fraction ranges materials are attributed. In a next step a number of 3 to 4 different material categories are defined with specified ranges of release fractions after mechanical and after thermal impact.

5 RADIOLOGICAL BASIS TO DERIVE CONTENT LIMITS

One open question is how content limits will be ultimately expressed - as radionuclide-specific absolute activity limits within a package or as activity concentration (specific activity) limits? One could envisage to make a distinction between materials which are conditioned and which are not conditioned into a compact binding agent:

- if the material is not conditioned into a compact binding agent the total activity in a package is limited and in addition a limit on specific activity is introduced to avoid a conflict with Type A-packages,
- if material is conditioned into a compact binding agent the total activity is limited and some additional requirement concerning material properties is introduced.

A radiological analogue to the Q-system will be developed. The following exposure pathways are probably relevant:

- direct γ radiation close to the package after an accident,
- inhalation pathway; this can be easily modelled using generic values of time-integrated ground-level air concentration values χ which are different for mechanical and for thermal release, e.g. $\chi = 5 \cdot 10^{-3}\ s \cdot m^{-3}$ for near ground-level release and $\chi = 5 \cdot 10^{-5}\ s \cdot m^{-3}$ for elevated effective release height from a fire,
- groundshine pathway from deposited γ emitters; the deposition level can be estimated from the ground-level time-integrated air concentration by multiplying with a proper deposition velocity v_g. Since also particles with larger aerodynamic equivalent diameters can be generated and released in accidents with mechanical impact a reasonable choice could be $v_g = 2 \cdot 10^{-2}$ m/s; the period, e.g. 1 year, 50 years over which groundshine exposure is integrated and whether processes leading to a reduction of ground contamination are considered will be a matter of discussion; this exposure pathway is in a sense some measure for potential clean-up problems after an accident.

Potential radiation exposures after a release of radioactive material would be expressed as effective dose of an adult. The procedure to calculate potential doses from different pathways following atmospheric dispersion and deposition is straightforward by using precalculated values of radionuclide-specific doses per unit concentration. It is probably sufficient, as in the Q-system, to consider different pathways independently, the most restrictive will then determine the activity or activity concentration limit of a radionuclide.

The working program of this study and the progress of work achieved at the time of the IAEA seminar will be presented.

MULTIPLE CONTAINMENT FOR LSA AND SCO WASTES

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Abstract

Current limits on packages and conveyances for LSA or SCO contents take no account of secondary containment provided by metal drums, for example. Radioactive wastes are frequently packed in drums or other forms of container for ease of handling. These containers may be qualified as Industrial Packages. An outer container can be used and may also qualify as an Industrial Package although dual qualification is unusual and not necessary.

Clearly such double containment reduces the release of the contents in extreme accidents as well as in the so-called Normal Conditions of Transport. This may conveniently be represented as factors to be applied to the container or conveyance limits defined in IAEA Safety Series No 6 (Table VI of Reference 1). The following factors are proposed for LSA-II with double containment, one of which is qualified as an IP-2 and the other is defined below.

	Form of Secondary Containment	Multiplier
(a)	Closed Steel Drum/Container	1000
(b)	Breached Steel Drum/Container or Intact Plastic Drum/Container	100
(c)	Cardboard or Fibre Drum/Container	10
(d)	Plastic Wrapping or no Containment	1

1. INTRODUCTION

Radioactive wastes are generally transported in the form of Low Specific Activity (LSA) materials or Surface Contaminated Objects (SCO). This paper is concerned only with solid LSA-II and SCO materials, i.e. not bulk powders, although the principles may be extended to liquid and gaseous forms.

LSA comprises bulk materials with the radioactivity spread more or less uniformly throughout the volume. It ranges from bulk untreated wastes in the form of contaminated clothing and wrapping materials, through large volumes of contaminated earth or rubble to immobilised forms where the active items are grouted with cement. LSA can therefore be transported as single items in the form of cement blocks or as volumes of loose material.

SCO may take many forms but usually comprises single items such as pipes or metal items individually identified. Small contaminated items are generally consolidated and carried in the form of LSA.

Prescriptions in the IAEA Transport Regulations [1] are under review to bring them within the Q System of risk classification [2]. The current Regulations are based on the assumption that the transport risks for LSA and SCO are acceptable because of the low **concentration** of activity. Limits on conveyance capacity are therefore many times the A2 limits derived for Type A packages. A2 values tabulated in [1] are the largest quantities of each isotope which can be carried in packages not required to survive severe accidents. The contents of a Type A package may therefore by allowed to escape with acceptable risks to transport workers and the public, provided the risk is recognised and prompt action taken to minimise the hazard.

LSA and SCO are transported in Industrial Packages where packaging is necessary. (LSA-I need not be packaged). The packagings require approval based on tests for "Normal Conditions of Transport". These represent extreme conditions of transport and handling which fall short of severe accidents. A proposed definition [3] of Normal Conditions extends testing only to create damage which is not obvious or which would not prevent the package continuing its journey.

It is unusual to carry bulk wastes loose in large Industrial Packages although some LLW is carried in closed vessels such as garbage skips, tested and approved to IP-2 standards. It is much more common to pre-pack the wastes in more easily handled units, often metal drums of about 200 litres capacity. In some cases these drums are approved to IP-2 standards and are transported in over-packs such as ISO Freight Containers which are not necessarily approved transport containers. In other cases, these drums are not approved as Industrial Packages, but are transported in over-packs approved to IP-2 standards. These large IP-2 containers are generally modified ISO Freight Containers of various sizes.

The modifications range from minor changes, providing improved standards of fabrication and Quality Assurance, to strengthened designs with double-sealed closures which can be tested to demonstrate leak-tightness before despatch. Some are used as one-way transport and disposal packagings. These are used in the UK to carry super-compacted waste drums to Drigg for in-situ grouting and shallow burial in engineered trenches.

In each case, a substantial inner or outer metal container provides additional protection from dispersal of the radioactive contents, even if the Industrial Package (IP-2) is destroyed in a severe accident. While the protection will not be perfect, it will substantially reduce the hazard to those close to the accident. This reduced risk should be recognised in the Transport Regulations. It may be used to justify retention the existing LSA capacity of IP-2 packages, if this is likely to be reduced by the revisions in the 1996 Edition of the IAEA Regulations, or to relax some of the existing limits on the contents of packages or conveyances.

Since the revisions of requirements and or definitions for LSA and SCO are not yet fixed or published, these proposals will, of necessity, be based on the current 1985/1990 Edition of the IAEA Transport Regulations [1].

2. RADIOACTIVE HAZARDS - THE Q-SYSTEM

The Q-System of assessing hazards [2] results in values of A2 for each isotope and methods of combining these for materials comprising more than one isotope. The A2 values listed in [1] are used to define contents limits for all but Type B packages. The limit may be of the form of A2 multiples (e.g. one A2 is the limit for Type A packages) or of the form of concentrations expressed as multiples of A2/g (e.g. the concentration limit of 10^{-4}A2/g for combustible LSA-II).

The Q-System recognises means of dose uptake to those in the vicinity of damaged packages. These are:

(a) Direct gamma radiation from the contents.

(b) Direct beta radiation from the contents.

(c) Inhalation of airborne activity.

(d) Absorption of contamination by physical contact or ingestion.

(e) Radiation from immersion in airborne activity.

In each case, the contribution is assessed on the assumption that a severe impact (or crush) has breached the containment and shielding and that a subsequent fire has created an aerosol of active components. The input to this assessment includes the specific activity, its form (alpha, beta, gamma or neutron) and energy, and the potential release fraction of the isotopes.

By its nature, Low Specific Activity material does not generally provide a severe direct radiation problem. Most of the risk is from ingestion of active aerosols and contamination. An individual is pessimistically assumed to ingest 10 mg of the mixture; see Para E131.6 of Reference 2. The potential for spread of activity is obviously of importance in the assessment of risk so it is precisely in this area that the existence of other containment boundaries will help to alleviate the risk by reducing the probability of total release. In fact, even an imperfect secondary containment can reduce the release of active species by several orders of magnitude.

The risk from radioactive dose to those present comprises the product of the probability of an incident leading to a release, the proportion of the contents contributing to the hazard and the probability of death from this contribution. This can be reformulated to give an annual risk from a transport operation:

$$\text{Risk} = f \,.\, t \,.\, A \,.\, R \,.\, C \,.\, D \,.\, M$$

where
f is frequency of an operation (journeys per year)
t is the journey time (hours at risk)
A is the severe accident probability (accidents per hour)
R is the proportion of release
C is the total radioactive inventory
D is the dose to those present (man.Sv assuming release of contents)
M is the probability of death (per Sievert)

The Q-System effectively represents all of the above factors, apart from C, in the form of the A2 parameter. Limits of package capacity or conveyance load are therefore proportional to C whenever they are limited in terms of A2, either by capacity or concentration. Clearly the risk is proportional to the inventory of the package (or the conveyance load).

The secondary containment considered here will affect the parameter R, the release proportion. The dose factor D allows for both external radiation and committed internal dose. The assumed 10 mg ingestion by an individual was used to derive concentration limits for LSA.

3. PROPOSALS

A method of acknowledging the beneficial effects of multiple containment should be written into the 1996 Edition of the Regulations. Experience used to assess risks from movements of radioactive material on nuclear sites in the UK can be applied to develop safety arguments justifying the alleviation of off-site transport risks. The release probability is reduced by factors derived for a variety of typical packaging methods and applied generically. These, or similar factors, should be incorporated in the revised Regulations in the form of multipliers on the A2 limits for Industrial Packages.

Secondary (but not tertiary or further) containment should be recognised by allowing limits on Industrial Package contents, in terms of A2 (or A2/g), to be increased by factors appropriate to the form of containment. The proposed factors applicable to solid LSA-II are as follows:

	Form of Secondary Containment	Multiplier
(a)	Closed Steel Drum/Container	1000
(b)	Breached Steel Drum/Container or Intact Plastic Drum/Container	100
(c)	Cardboard or Fibre Drum/Container	10
(d)	Plastic Wrapping or no Containment	1

A closed drum should be interpreted as one sealed by a positive mechanical system but not leak-tested. An outer ISO Freight Container can be treated as a closed metal container because of the substantial physical (impact and fire) protection provided for the inner Industrial Packages. The definition of breached steel drums include those crushed by compaction operations.

Similarly, a cardboard or fibre drum should be closed positively, possibly by adhesive tape, but need not be leak-tested. Leak-testing should not be necessary for any Industrial Package or lesser containment.

Only one of the factors can be applied irrespective of the present of tertiary or further containment. The factors are justified in the following way.

4. JUSTIFICATION

The current version of the Q system assumes a release fraction for Type A packages involved in a severe accident in the range 10^{-3} to 10^{-2} and an uptake by those involved in the accident in the range 10^{-4} to 10^{-3}. These combine to give an intake of 10^{-6} of the package contents. The fractional release is based on a limited number of accidents involving type A packages. The following are some of the more readily available data;

(a) A series of experiments carried out by Mishima are reported in Reference 5. In these experiments, typical low level waste (i.e. paper, plastic, cardboard, rubber) contaminated with uranium and contained in plastic bags and a cardboard box was burned. The measured fractional airborne was between $1x10^{-5}$ and $2x10^{-4}$. In a similar set of experiments carried out by Sutter et al (Reference 6), a fractional release of between $2x10^{-4}$ and $5x10^{-4}$ was measured, depending on whether the contaminant had been a powder or a solution. The main weakness of these experiments is that they do not give any indication of the release fractions for other more volatile contaminants.

(b) A further series of experiments by Mishima is reported in Reference 7. These specifically refer to transport accidents involving crushing of liquid-filled equipment by sudden impact. The measured release fraction is reported as $5x10^{-3}$ for droplets of respirable size.

(c) Experiments were carried out at Harwell Laboratory in which sealed metal drums were crushed to simulate a transport accident. Release fractions of 10^{-5} were reported. This release fraction is comparable with release fractions quoted in Reference 8 for the release from internally pressurised cans.

(d) The BNFL accident consequences database quotes "decontamination factors" for two types of containment:

100 for non-gas tight containers, such as drums and
10 for damaged packages

These fractional releases are based mainly on operational experience at BNFL plant.

From (a) it is clear that for uranium-contaminated waste contained in cardboard, the release fraction can be between 1/20 and 1/1000 of the range assumed under the Q system. To allow for higher releases of more volatile contaminants, it is conservatively assumed that a cardboard containment would release 10% of the material assumed under the Q system. (Note: that in the situation of a fire severe enough to damage both the inner and outer container, entrainment of the release material in a hot buoyant plume is likely to give an increased dispersion and a lower intake than that assumed under the Q system, probably by several orders of magnitude). Applying the decontamination factors for other forms of containment gives rise to the factors quoted above.

The current limits on package or conveyance contents of LSA-II assume an individual ingests 10 mg of the contents mixture. If this assumption is to be retained in future, the secondary containment will allow larger inventories because of the demonstrable effect of additional barriers reducing releases in the form or aerosols.

Several assumptions have been made in deriving the above factors. These are:

(i) No more than 10% of a solid or powder can be released in the form of an aerosol. In fact, release rates measured during the combustion experiments cited are considerably less than this.

(ii) Cardboard or fibre drums will not influence the release rate. This is also pessimistic as it is unlikely that all of such inner containers will be destroyed in an accident. This does, however, allow for a single inner container being destroyed.

(iii) A closed metal container will resist much accidental damage and will continue to protect the contents even if damaged. Such a container cannot be destroyed in an impact or fire although the lid (or other closure) could be removed. The assumption that such containers allow a 1% release in the form of an aerosol (0.1% of the solid contents with the 10% factor in (i) above) includes a probability of failure and of subsequent release of contents. This is considered pessimistic even for combustible material as demonstrated by the references cited below.

(iv) A breached metal container will be less efficient than a closed metal container but it will release much less than a cardboard or fibre drum. It is therefore allocated a release fraction of 10% which reduces to 1% when combined with the solid matter factor in (i) above.

(v) Plastic wrapped or uncontained material is considered to be only singly contained, so a reduced release factor is not appropriate. Since the material could be in loose powder form, the solid material release factor in (i) above is also inappropriate.

5. CONCLUSIONS AND RECOMMENDATION

The 1996 Edition of the IAEA Transport Regulations should incorporate a method for increasing the contents limits of Industrial Packagings where a secondary containment system will reduce the dispersal in accident conditions.

Where a package (IP-II or IP-III) or conveyance is limited in contents by the A2 parameter (either directly in the form of A2 or as a concentration A2/g) these limits may be increased by the factors tabulated above.

A secondary internal packaging system should have a minimum dimension of 100 mm (as for a package for radioactive materials specified in the IAEA Regulations). This should be considered as the minimum unit for leakage from the approved Industrial Package in Normal Conditions of Transport as proposed by a companion paper by J Higson [4].

These proposals should be examined by an Expert Group and, if endorsed, submitted to the IAEA for incorporation in the 1996 Edition of the Regulations. Suitable paragraphs for Safety Series Nos. 6, 7 and 37 should be prepared on an appropriate timescale to allow such regulatory changes to be introduced at the same time as the Advisory Material.

ACKNOWLEDGEMENT

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APPLICATION OF THE TRANSPORT SYSTEM CONCEPT TO THE TRANSPORT OF LSA WASTE

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Abstract

Transport Regulations rely essentially on the packaging and do not take into account the contribution to safety which may be made by other features of the operation. In some situations, mainly for routine transports not fully complying with the Regulations, it would be beneficial to envisage the possibility of using a package which does not meet all the type B requirements, complemented by additional safety measures put in place to compensate for these shortfalls. The "Transport System" concept will take into account the contributions to safety from these additional measures. It will ensure that the proposed system is at least as safe as a reference operation complying fully with the Regulations. If this equivalent safety level can be properly demonstrated, the Competent Authority will provide a "Transport System Approval" for well defined shipments over a specific period. Two examples are presented, in the first case, a thermally insulated ISO container is envisaged for the transport of drums containing combustible LSA material having a total activity per conveyance up to 600 A2. In the second one, two dedicated trucks transporting conditioned waste in drums has been shielded so as to comply with the regulatory dose rate limits. These examples show the benefits of the TS concept. Nevertheless, the full requirements of the Regulations should be implemented as far as reasonably practicable, and the TS concept should be applied only to particular difficulties and is not suitable to all situations. Therefore, some general restrictions (applicable to every TS) have to be set by IAEA. Depending on the case, complementary ones may be required by the CA. Bearing in mind possible restrictions presented in this paper, the TS concept will be useful to solve some of the current problems of the transport of waste without needing a fundamental change in the Regulations

1) INTRODUCTION

The transport of radioactive material is an activity which, by nature, is not limited to the border of a country and from the beginning the need for international recommendations was recognised The current national regulations follow the recommendations of IAEA presented in the Safety Series n°6 "Regulations for the Safe Transport of Radioactive Material" [1]

One of the basic tenets of these recommendations is that the Safety mostly relies on the package used and does not take into account the contribution to safety which may be made by other features of the transport operation such as the conveyance Different types of packages are specified depending upon the nature and activity of the radioactive material to be transported The main advantage of this basic tenet is that the same package can be used whatever the mode of transport is It also minimises the responsibilities required for the carrier and allows consignments to be transported with minimal special handling controls

About every 10 years, these recommendations are modified in order to take account of technical developments, new transport operations or lessons learned from the application of the current Regulations

The last edition of the Safety Series n°6 (published in 1985 and to be in force 5 to 7 years after) introduced new supplementary constraints mainly for the transport of waste classified as Low Specific Activity (LSA) or Surface Contaminated Objects (SCO), specially for the transport of LSA combustible materials or high radiation level waste

Like every modification in the Regulations, it raises some new problems mainly for the transports complying with the previous edition of the Regulations but not the new one Sometimes the use of a type B package is then required instead of a type A or an Industrial package In order to solve these problems, according to the current situation, four alternatives can be envisaged

1 - **to reduce the quantity transported** in the current packages in order to comply with these new constraints This solution increases the number of transports and is not always suitable for financial and also safety reasons, the expected number of accidents or incidents being proportional to the number of transports

2 - **to use a type B package** for which no quantity limit is required For low level waste (LLW) this expensive solution is sometimes difficult to justify when the potential risk is not really important Furthermore it may require the development a new type B package, which could be a long process or it may raise other problems if the waste disposal facility has to recondition the waste in order to reuse the type B package

3 - **to ask for a "special arrangement"** to the Competent Authority providing appropriate compensatory measures Depending on the problem and on the Competent Authority, this procedure can be obtained for a set of similar transports, but is obviously not intended for frequent transports on a routine basis

4 - **to propose new modifications of the IAEA recommendations** This requires time and during this long and uncertain process one of the three previous solutions has to be adopted

2) THE TRANSPORT SYSTEM CONCEPT A FIFTH ALTERNATIVE

In some particular situations it could be beneficial to adopt none of these four alternatives but to envisage a fifth one the possibility of using a package which does not meet all the type B requirements, complemented by additional safety measures put in place to compensate for these shortfalls This solution is called here **"Transport System"**

In a dedicated study, the Transport System concept will take into account the contributions to the overall safety level from these additional measures On the basis of this study, the Competent Authority will then judge if the proposed Transport System is adequate to ensure that the overall level of safety of the proposed system is at least equivalent to that which would be achieved if all the applicable requirements were met (as it is requested for a special arrangement in the § A-211 1 of the Safety Series n°37, [2]) If this is the case, a kind of special arrangement, called here **"Transport System Approval"** will be provided by the Competent Authority for this well defined Transport System over a specific period or for a set of consignments

The Competent Authority will have to define an appropriate set of rules to apply, based on a framework approved by the IAEA These rules will deal with the application field and the safety comparison procedure For example, the Competent Authority could have to decide if a proposed Transport System dealing with the transport of indispersible waste in drums presenting high dose rates is suitable or not, and check if the proposed compensatory measures are safe enough The requirements should be discussed between the Competent Authority and the applicant from an early stage

The justification of the Transport System, like the justification of a special arrangement, can range from considered judgement or measures, to probabilistic risk assessment In the latter case, it is more difficult to justify the Transport System and the Competent Authority has to decide if this approach is acceptable or not, and may wish to develop appropriate guidance material

Nevertheless, the Transport System Approval procedure will apply in a limited number of situations, mainly

1 - when the Competent Authority is faced with a request concerning frequent transports on a routine basis and wants to be sure that appropriate corrective measures are taken to limit the consequences of all conditions of transport for both public and workers,

2 - for interim situations, for example before the availability of a new type B package, where special supplementary measures are requested in the meantime by the Competent Authority on the existing packages, or for interim situations where the proposed solution is consistent with proposed future Regulations but not the existing ones

3 - for particular situations, where the present IAEA Regulations are difficult to implement or are in conflict with other Regulations (see section 3 2)

In all cases, the prime consideration will be to meet the full packaging requirements of the Regulations as far as reasonably practicable

3) APPLICATION EXAMPLES

The last edition of the Safety Series n°6 [1] introduced new supplementary constraints mainly for the transport of waste classified as LSA or SCO Different routine transports of these waste currently made in industrial packages were therefore impossible and the limitation of the content the use of a type B package or a special arrangement were then required

For two particular situations one in France and one in Belgium the possibility of using a package which does not meet all the Type B requirements has been envisaged within the framework of the "Transport System concept"

3 1 The shipment of combustible alpha waste

The transport of LSA combustible material is affected in the revised regulation by the new limitation to 100 A2 of the total activity per conveyance This is a strong constraint for the transport (to storage or disposal facilities) of low concentrated alpha waste originating from various laboratories of the Commissanat à l'Energie Atomique (CEA) Complying with this limit will significantly reduce the quantity transported by conveyance and therefore multiply the number of journeys, which is not advisable for financial as well as safety reasons Before the availability of a dedicated type B package for these transports, it has been proposed in France to adapt an ISO container to cope with the fire problem

A specific ISO 20' container, thermally insulated, transporting 60 drums of 200 litres has been proposed for the transport of material having a total activity per conveyance restricted to 600 A2 (instead of 100 A2) The thermal protection is so designed that, in the event of a fire leading to engulfing 800°C flames for half an hour, the temperature reached at the hottest point of a drum will be less than the temperature leading to a radioactive release from the material inside the drum, or the opening of the drum

A specific study [3], sponsored by CEC, has been performed to compare the doses and risks associated with the Transport System option to those associated with a reference option complying fully with the current Regulation The first option (so called "DV 77") involves 20 journeys of 60 drums in the thermally improved container The second one ("ISO") involves 100 journeys restricted to 12 drums (to cope with the 100 A2 constraint) in a classical ISO container The occupational routine doses and the risks of potential radiological consequences associated with accidental situations have been assessed for both cases

The analysis has shown that the routine doses were equivalent and that the Transport System option ("DV 77") presented a probability of an accident occurring 5 times lower and consequences 5 times higher for collision and 2 to 5 times higher for the different types of fire (see figure 1) The corresponding maximum individual dose in the worst accidental situation, being close to 50 mSv, it was judged that the proposed Transport System option should be acceptable Nevertheless, because the development of a type B package (so called "GEMINI") is nearing completion, this solution will not now be used for activities up to 600 A2 per conveyance

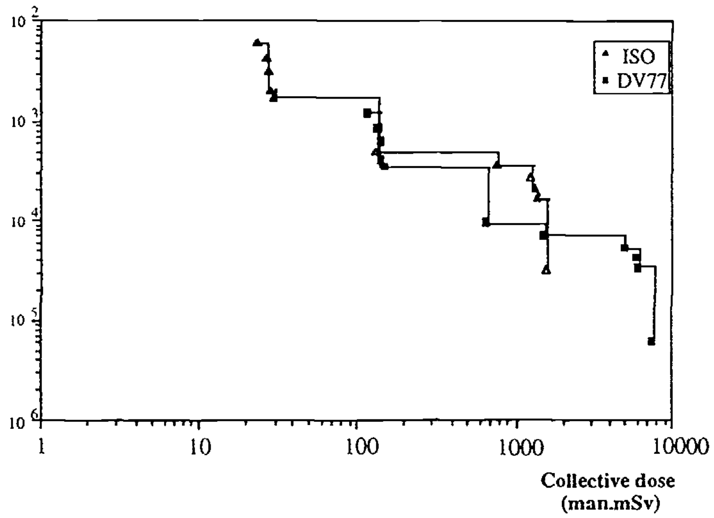

Figure 1 Farmer curves associated with the two options

3.2 The shipment of conditioned waste

Since sea dumping of low level radioactive waste was abandoned in 1982, the Belgium Agency in charge of the radioactive waste called NIRAS/ONDRAF has set up a new radioactive waste management program This program mainly consists of reducing the volume as far as possible and standardising the radioactive waste packages

One of the consequences of the reduction of the volume is the increase of the radiation levels produced by that waste which raises problems for the transport If one is supposed to use the standardised reinforced steel drums containing about 400 l of conditioned waste as primary packages for transport and storage, even if the content is classified as LSA III, the radiation dose rate at the outer surface of that package may mount up to more than 10 mSv/h As such, these industrial packages have to be carried under special arrangement

If one wants to avoid that kind of special arrangement for these routine transports, one has to add some shielding either inside or outside the drum Adding shielding inside the drum limits the useful content (if one uses concrete for example) or is a waste of useful material (if one uses lead for example)

The better alternative is to add at the outside of the drum some shielding which is an integral part of a conveyance or a larger packaging The drums are therefore regarded as inner receptacles of a larger packaging designed in such a way as to limit the radiation level to less than 10 mSv/h at its outer surface Taking into account that this larger packaging when mounted on a trailer may also be considered as a conveyance, the radiation level must be further reduced to less than 2 mSv/h at its outer surface, to 0 1 mSv/h at a distance of 2 m and to 0 02 mSv/h in the driver's cabin

Taking account of these considerations, two dedicated shielded trucks have been developed by Transnubel for these transports The first one (TNB 0167) is designed for the shipment of 14 drums with maximum surface radiation level of 50 mSv/h or 7 drums of 300 mSv/h To reduce doses to workers, this container is equipped with an integral gantry crane with a capacity of 2 5 t to lift the drums and to place them in the inner transport rack The crane is fully remotely operated and controlled by TV cameras The second one (TNB 0178) is designed for the shipment of 20 standard drums with an average mass of 1 t and up to a mean surface radiation level of 5 mSv/h Loading and unloading are performed by means of a remotely operated crane, which lifts and lowers the drum vertically without any manual intervention Opening and closure of the conveyance and fixing of the drums on the conveyance is also done remotely

The Belgium Competent Authority has provided a special shipment approval on the basis of the Safety report presented by Transnubel for these two shielded trucks Within the framework of the joint IPSN/SRD/Transnubel study for the CEC DGXVII [3] this example has also be analysed in order to prove that this solution complies with the "Transport System concept"

3 3 Other examples

Several other examples, very close to the application of Transport System Approval, have been presented during the 1992 PATRAM at Yokohama

The first one [4], concerns the Swedish sea transportation system of the waste to be stored in the Central interim storage facility for spent nuclear fuel or the final repository for radioactive operational waste

Others [5] concern various US transports

- the use of the ATMX railcar for the movement of TRU waste material from Rocky Flats to temporary storage in Idaho,
- the carriage of up to 1000 TI (Transport Index) in radio pharmaceuticals in a single conveyance,
- the approval for the US DOE and US EPA (in separate programs) to transport mill tailings in bulk loads without detailed identification of the nuclide content of each load

All these examples indicate that, for these particular situations dealing with routine transports not complying fully with the current Regulations, there is a real need for a "Transport System Approval"

4) POSSIBLE RESTRICTIONS OF THE USE OF THE TRANSPORT SYSTEM

The full requirements of the IAEA Regulations should be implemented as far as reasonably practicable, and the Transport System should be applied only to overcome particular difficulties In addition, the Transport System concept is not suitable for application to all situations, and possible restrictions are described in the following paragraphs Some restrictions have to be set by IAEA and are applicable to every Transport System Complementary other ones may also be required by the relevant Competent Authority

The General restrictions (Gi), to be set by IAEA, should be as follows

G1 - transports by land and sea, under exclusive use,

G2 - real compensatory measures have to compensate for the gap between the Regulatory Requirements and the proposed solution (for example, transports of combustible LSA materials having an activity greater than 100 A2 per conveyance can be considered only if supplementary thermal protection is put in place),

G3 - the number of variations from the regulatory requirements should be limited (to 1 or 2),

G4 - upper bounds (one or two orders of magnitude) for alternative numerical values (100 A2, dose rates,) should be defined in order to limit the radiological consequences, should an accident happen,

G5 - multilateral approval will be required for shipments through more than one country,

G6 - the Transport System should ensure doses are kept ALARA and below dose limits

The possible supplementary restrictions provided by the Competent Authority (CAi) will mainly depend of the specific situation They should be as follows

CA1 - Transport System can be limited to the transport of material which broadly falls into the current LSA or SCO categories (or others), but which does not fully meet the current IAEA requirements concerning this type of material,

CA2 - Transport System concerns only shipments which are well defined (the consigning and receiving sites and routes should be defined, and the consignor or consignee should control all aspects of the carrier operations),

CA3 - "hardware" rather than operational compensatory measures are preferred and the compensatory measures should, as far as practicable, be "built-in" to the transport system,

CA4 - the Transport System analysis demonstrating that the proposed operation is at least as safe as the reference operation complying fully with the current Regulations, should be based on a quantified procedure, measures or tests if possible,

CA5 - more restrictive upper bounds than those defined in G4 can be introduced,

CA6 - only qualified Transport Companies with Quality Assurance Transport approval can operate these transports

Some of these restrictions have already been discussed at IAEA, as is described in the following section These restrictions and the distinction between the General (Gi) and the other ones (CAi) have to be accepted and finalised at this level, which will require some time It is therefore not clear that the Transport System Concept could be introduced for the next edition of Safety Series n°6, forecast for 1996

5) DISCUSSION OF THE TRANSPORT SYSTEM CONCEPT AT IAEA

Discussions at IAEA of the Transport System started in 1991 Up to the end of 1993, two IAEA meetings had briefly considered this issues

1) The Technical Committee Meeting on the development of a research programme supporting the revision process of the regulations for the safe transport of radioactive material (TCM 764), 8-12 April 1991 (see [6])

From this first discussion, it was obvious that the concept was rather difficult to understand, particularly the difference between Transport System and Special Arrangement, and no decision was taken Nevertheless a first definition of the Transport system concept was introduced *" Specially Dedicated Transport System (SDTS) means the design of a transport system to fulfil the needs of routine transport operations among defined places by using dedicated vehicles or special use vessels together with particular operational or design conditions intended to provide a high level of global safety to the RAM transport The whole system operates under the responsibility either of the consignor or the consignee and is operated by qualified personnel In some cases the SDTS has inherent or operational measures that improve particular safety aspects and that subjected to multilateral Competent Authority approval can allow the use of alternative provisions to certain specific requirements of the current IAEA Regulations "*

During the TCM, this approach was considered to be suitable for the transport of such radioactive material that broadly falls into the current LSA/SCO categories of material, to cope with the four main problems linked to the homogeneity, the dose rate restriction of 10 mSv/h at 3 metres, the 100 A2 conveyance limit for combustible material and the leach test

2) the Technical Committee Meeting on Issues Related to Low Specific Activity Materials and Surface Contaminated Objects (see [7])

Before this meeting, the Belgium and French delegations asked IAEA for the opportunity to present briefly the results of the joint contract for the CEC [8], which is mainly related to the transport of LSA or SCO Two Working Papers were presented by H Sannen and J Lombard (WP9 and WP8) to illustrate our points of view The following discussions were very interesting and it was obvious that real progress had been made from the 1991 meeting The concept was better understood and almost everybody recognised a need for a Transport System Approval

Working Group 2 of this TCM was actioned to prepare a text for the next SAGSTRAM This text is as follows

"TRANSPORT SYSTEMS CONCEPT"

The "Transport System Concept" is considered to be applicable for regular shipments of larger amounts of material of a specified type, which cannot appropriately be transported under the rigid scheme of the present regulations This system could be defined as a set of packages, conveyances, handling equipment, operational procedures, etc A transport system would be designed to optimise handling and transport of well defined packages The system may also reduce radiation exposure to transport workers and the general public It would not need to be limited to low specific activity materials or contaminated objects or be limited to specific modes of transport as recommended in WP8

It is clear that currently a competent authority can approve a "transport system" by giving it a "special arrangement" approval This gives the competent authority the flexibility in approving systems that are safe and effective This approval can be issued on an individual shipment basis or for multiple shipments

Current regulations consider certain types of material and provide recommendations how these types of materials should be transported Where situations, which meet the safety objectives of the regulations, can be clearly identified, a framework for those kinds of shipments should be provided The explanatory and advisory material should include information to facilitate the use of the "transport system" concept

Regulations covering the "transport system" concept would assist users and competent authorities in different countries and would help in multilateral approval of international shipments Currently the concept of "transport system" is applied in different countries under national approval

The working group recommends that the Agency considers the inclusion of the "transport system" concept into the 1996 Regulations in the appropriate manner

This text does not correspond perfectly to our position, mainly because the distinction between special arrangement and Transport System Approval is not really clear Nevertheless, the time devoted during this TCM on this subject was very limited and the main point for us was that the concept was better accepted Some of the restrictions described in the section 4 were presented during this meeting but the distinction between the General restrictions (**Gi**), and the possible supplementary ones (**CAi**) was not discussed

6) CONCLUSION

The current IAEA Regulations allow transports which do not fully comply with the Regulations under a special arrangement A special arrangement should be devoted to transports limited in time and is not relevant for transports on a routine basis Nevertheless, in some particular situations a need for this kind of specific routine transport may exist, and so there is a need for a complementary procedure where the Competent Authority wants to be sure that these transports will be made in safe conditions The Transport System Approval would provide this complementary procedure to address the problem

As M PETTERSSON said during the 1991 IAEA Technical Committee Meeting [6], "Special Arrangement carried a stigma and an unfavourable public perception" This was also recognised during the following 1993 Technical Committee Meeting [7] It is therefore advisable to restrict this procedure to a limited number of transports

As was shown by the examples, although the problem could be viewed as a domestic issue, there is a clear benefit to Member States in seeking international agreement on the concept of Transport System Bearing in mind the possible restrictions presented in section 4, the Transport System concept will be useful to solve some of the current problems of the transport of waste without needing a fundamental change in the Regulations

ACKNOWLEDGEMENTS

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SYSTEM CERTIFICATION FOR RADIOACTIVE WASTE WITH APPLICATION TO LSA/SCO

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Abstract

This paper projects there will be shipments of radioactive material and waste, as the nuclear industry moves into maturity, which may not be well accommodated within the current transportation regulations. Although these shipments could be made under Special Arrangement approvals, the regulatory system would be better served if more formal requirements and criteria were included in the regulations and the shipments were considered in accordance with the regulations. A Special Arrangement approval is now defined as authorizing transportation of a shipment which does not satisfy all the applicable requirements of the regulations. This paper proposes the Transport System approach to regulating these types of shipments, where operational restrictions or other packaging provisions could compensate for the absence or inadequacy of packaging or other associated requirements. These shipments would require Competent Authority approval, and acceptance criteria would be included in terms of limits on probability, consequences and risk. The process would be limited to those types of shipments where the package system does not work well. The advantages of including Transport System approval within the regulations include reduction in the time required to obtain an approval, greater efficiency of decontamination and decommissioning operations, and assurance of an equivalent level of safety.

As the nuclear industry moves into maturity, there will be a constant stream of projects in the building, operating, dismantlement, and disposal stages. The need will arise to transport radioactive materials which, because of their size or special characteristics, do not easily fit into the categories of the transportation regulations or into the available packagings meeting these regulations. Yet many of the materials in question will need to be transported for storage disposal, processing for disposal, or other activities.

Most waste shipments in the future, as with most waste shipments now, can and will be made in packages satisfying regulatory standards. Most waste shipments will satisfy the definitions of Low Specific Activity (LSA) Materials or Surface Contaminated Objects (SCO), and will be made in Type A or Industrial Packages (IP) in routine conditions of transport. The radioactive content of these packages is limited so even in the case of a severe accident causing the Type A or Industrial Package to fail, the radiologicial consequences will not be large. Competent Authorities and members of the general public seem to accept the safety level provided by the regulatory standards and Competent Authority control of Type B package designs and quality assurance provisions. The radioactive material transportation safety record over the last 30 years supports the adequacy of the current regulations and procedures.

In the near future, however, waste shipments which may not be well accommodated within the current transportation regulations will need to be made. Transportation of the seemingly endless quantities of radioactively contaminated piping from decommissioned nuclear facilities and large pieces of contaminated equipment or building decommissioning waste presents new regulatory challenges.

If the radioactive material involved fits within SCO limits or can easily be decontaminated to meet these limits, the material could be shipped under current regulations and those likely to be adopted in the near future. However, regulations for SCO shipments have a low limit on total radioactivity in a single package, and if not packaged, they have an even lower limit on the dispersible radioactivity on a single conveyance. For shipments not meeting SCO limits, the immediate answer may in the Special Arrangement provisions.

There are two problems with the Special Arrangement approach to waste and decommissioning shipments. First, Special Arrangement shipments are, by definition, shipments which do "not satisfy all the applicable requirements of these Regulations...." These shipments will not be readily accepted by the public, and will not be popular with Competent Authorities. The second problem is related to the criterion for approving a Special Arrangement shipment... "the overall level of safety in transport and in-transit storage is at least equivalent to that which would be provided if all the applicable requirements had been met." Proving a particular Special Arrangement shipment meets the criterion could be very difficult. There is guidance in IAEA Safety Series No.37, including the warning approval of a Special Arrangement is "subject always to the discretion of the competent authorities concerned..." Potential shippers might be understandably nervous when applying for an inherently unpopular Special Arrangement approval when there are no established criteria for acceptance. An approval for an international shipment, where multiple Competent Authorities must agree the overall level of safety is equivalent to an imaginary shipment meeting all applicable requirements, could present a complex regulatory exercise.

The concept of Special Arrangement is a valuable relief valve in the regulations, but should apply to a very specific set of circumstances. Special Arrangement should be limited to single event shipments clearly outside the scope of the regulations but where there will be no significant impact even if an accident occurs. For multiple shipments, not meeting regulatory requirements, more rigorous evaluation of the probabilistic features of the operations and the risks are needed. This is the basic concept behind Transport System Certification. A controlled tradeoff among risks, costs, and benefits is made explicit together with the quality and compliance steps of the Transport System approval process visualized by the authors.

The establishment of formal criteria based on radiological controls on the Transport System could make approval more efficient and reduce uncertainty associated with Special Arrangement approval. This Transport System approval process would need to include criteria for when such an application would be accepted for consideration, e.g. when traditional packaging is impossible or when personnel or general public exposures would be dramatically reduced. There would also need to be criteria and guidance for approval of a Transport System proposal, including guidance to regulators and shippers for how to evaluate whether an "equivalent level of safety" is being preserved. A quantitative demonstration an equivalent level of safety is being provided should be based on probability and consequence considerations. An established Transport System approval process would improve Competent Authorities level of comfort in approving sound proposals, and would likely achieve public acceptance.

A primary issue in establishing criteria for when a Transport System application would be accepted for consideration is how to prohibit, or at least to discourage, the Transport System option for shipments which could reasonably be accommodated by traditional packagings. Applying for a Transport System approval simply for convenience of the consignor should be discouraged. This paper is not a proposal to establish the Transport System option as an equal alternative to the use of a package; it is a proposal to establish criteria, within the regulations, for approval of a Transport System when traditional packagings are not suitable for the shipment. This includes cases where significant cost savings could be achieved.

What cases are there when traditional packagings will not work? The best examples are radioactive waste where a contaminated object is too large to fit into something reasonably meeting regulatory requirements for a package and yet the radiation levels or contamination levels are too high to ship the object unpackaged. When operational controls can be put into place to compensate for a complete absence of packaging or for the use of packaging not satisfying all the regulatory requirements, a Transport System approval is appropriate. Some general examples of real cases might be long lengths of contaminated piping from processing facilities, chunks of concrete from nuclear reactor mounts, or neutron shielding walls, and other structural types of wastes which are too large to fit in reasonable packages.

To clarify, a specific example may be instructive. Shipment of a decommissioned nuclear reactor pressure vessel is an example where a Transport System approval might be appropriate. An argument could always be made against a Transport System approval. In the example just given, one could argue the pressure vessel could be cut up into pieces which would be small enough to be put into packagings meeting regulatory requirements. However, before a decision to cut up a large contaminated object into smaller pieces is made, the question "at what cost in terms of additional radiation exposure and money" would have to be addressed. Refer to paragraph 202 under subsection "Radiation Protection" of IAEA Safety Series No.6, Regulations for the Safe Transportation of Radioactive Material (As Amended 1990) where the "as low as reasonably achievable, economic and social factors being taken into account" (ALARA) principle of minimizing radiation exposures is applied to transportation operations. There is the strong implication additional radiation exposure, accepted simply to use a package when operational restrictions would supply the same level of safety during transportation, is not justified. Likewise, significant increased radiation exposure associated with further decontamination so limits for shipping unpackaged SCO waste objects are satisfied, would not be justified if operational restrictions would achieve the same level of safety. In essence, an analysis of a request for Transport System approval is much like an optimization analysis "where consideration must be given to optimization of (1) requirements related to package design and test requirements including quantity and external radiation level limitations and (2) operational requirements for the implementation of, and compliance with, the Agency's Regulations."*

In summary, a request for approval of a Transport System would not be acceptable unless it could be effectively shown the radioactive material could not be reasonably packaged according to the regulations. Consideration of efforts to alter or condition the radioactive material to allow it to be packaged or shipped unpackaged under the regulations must take into account issues such as the additional radiation exposure (ALARA concerns) to alter or condition the material and the additional risks of making more shipments.

One can visualize the concept of Transport System approval shown by Figure 1. Five factors must be considered for Transport System approval:

- Incident-Free Transportation Risk
- Transportation Accident Risk
- Shipment Preparation Dose
- Economic and Social Cost
- Operational Restrictions

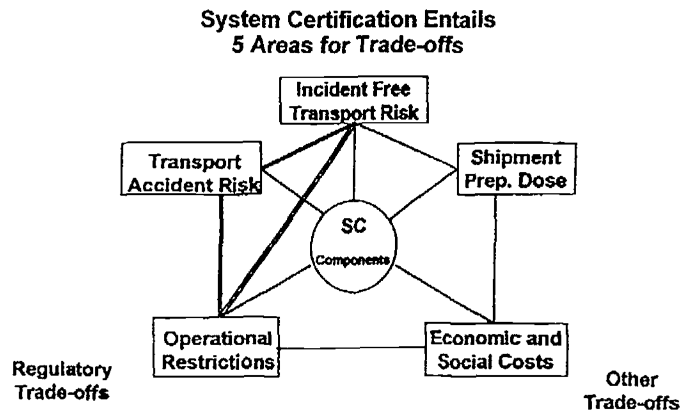

Figure 1

All these factors are interconnected; what minimizes one may increase another as the following example will illustrate. An important feature of this model of the problem is there are circumstances outside the regulatory boundary of the transport process impacting the decision on Transport System Approval. This is an unusual situation for a Competent Authority applying the regulations. A second feature is there is no common measure by which the components can be

* From the Foreword to Safety Series No.6 (1985 Edition).

measured. Individual or collective dose to the public is not treated the same as dose received by radiation workers (even though the units may be the same) nor is expected dose from accidents treated the same. Similar remarks hold for economic and social costs and benefits.

Concerning criteria for approval of a Transport System proposal, the primary issue is how to demonstrate an equivalent level of safety is being preserved by a Transport System. Risk assessment can provide a quantitative demonstration an equivalent level of safety is being provided. The IAEA regulations currently are not risk based, and delays in reaching agreement on how to use risk to demonstrate an equivalent level of safety could be prolonged.

So let us initially propose simple conservative criteria for approval of a Transport System which controls risk, probability, and consequence. By limiting any increased accident related individual radiation dose to 150mSv, a factor of 3 higher than the current criterion for maximum individual dose from an accident, the exposure would stay within the range of non-detectible medical change to the body. If this increase in consequences were accompanied by a corresponding decrease in the probability of the shipment being involved in the accident, there would be no change in the risk, and the level of safety would be preserved. There is no possibility of short term radiation effects from an individual dose of 150mSv, and with no change in risk, there would be no change in the potential for long term effects, since those effects are a function of population dose (a measure of risk).

In addition to the accident risks, possible changes in the radiological consequences resulting from normal exposure during the transportation operations should be considered in a Transport System application.

Suppose, as an example, we have a fleet of 100 nuclear reactor waste packages for shipping nuclear reactor process wastes, e.g. resins, filters and sludges which have been qualified as IP-3 packages. When these packages were designed and constructed, the regulations prescribed by IAEA included a 10mSv per hour unshielded radiation level limit at 3 meters to protect against direct radiation exposure from shielding loss in a transportation accident. Assume the radiation level limit in the regulations will be replaced by a $2A_2$ contents limit, as has been proposed and accepted by the recent Technical Committee Meeting in October 1993. Assume also an analysis shows the allowable content of these casks will be reduced by a factor of 3 by new limits, causing an increase in the number of shipments by a factor of 3. Assume also that this relates to an increase in radiation exposure of 10 person-Sv per year in increased processing and packing costs, an increase in cost of ten million dollars per year for the additional shipment costs, and an increase in the length of the shipping campaign from 3 to 9 years.

These reactor process wastes can certainly be shipped in a qualified package, but the cost of forcing them into the existing packages is very high in terms of radiation exposure, shipment costs, and shipping campaign delays. Because of these costs, the shipment method qualifies to be considered under the Transport System criteria. The proposed Transport System is to use the design capacity of the casks, filling them with $6A_2$ contents to eliminate the need for extra shipments and the extra radiation exposure, costs, and delays.

If we accept the arguments of the advocates of the $2A_2$ controls, a quantity limit of $2A_2$ in a single package of LSA material limits doses to any individual at an accident scene to 50mSv. The $6A_2$ content of the packages in this example would then allow individuals at the scene to receive radiation exposures up to 150mSv. This individual exposure does not produce any changes in the human body which are detectible by the medical profession and, as far as is known, produces no short term health effects. The population dose potentially associated with this accident is 3 times that of the accident involving only the partially full packages, so a compensating factor of 3 is needed to maintain the risk at the level associated with the $2A_2$ package. This would show that the overall level of safety is at least equivalent to that which would be provided if all the applicable requirements of the package system had been met. Table 1 shows a comparison of the key factors in determining normal and accident impacts for this example with and without System Certification.

There are a large number of operational restrictions which can be placed on this type of shipment as long as the shipment is made by exclusive use. Many are aimed at reducing the probability of the shipment being involved in a severe road accident, such as limits on speed, type of roads traveled, time of day and day of week traveled. There are restrictions aimed at minimizing damage once the vehicle is involved in an accident, such as an accompanying vehicle containing fire fighting equipment or tarps to isolate a damaged package. There are routing restrictions to avoid potentially serious environments such as high bridges, mountainous roads, refineries, or railroad crossings. And there are routing restrictions to minimize radiation exposures once a package is damaged, such as avoiding densely populated areas, controlling the time of day and day of week that the shipment travels, and mode of transport.

It is not clear compensating values can be placed on these operational restrictions without knowing more about the packages, the environment through which the shipment will

Table 1: Relative Comparison of Factors in System Certification Example

	Without System Certification	With System Certification
Units Shipped	6N	6N
Units/Shpt	2	6
Activity/Unit	2A1	2A1
Pkg Ext Dose Rate	D	3D
Shpt Duration	X	X
Accident Rate (No/h)	R	R
Incident Free Risk	X6ND/2=3XND	X6N3D/6=3XND
Accident Risk	XR4A16N/2=12XRA1N	XR12A16N/6=12XRA1N
Acc Ind Dose	2A1	6A1
Prep Dose (d is single pkg dose)	3d	d
Prep Cost (C is cost to prepare a single 2A1 unit)	3C	C

pass, or both. The example, therefore, ends here, after showing an approach to evaluating equivalent level of safety using limits on consequence, probability and risk.

There are many details to work out on when the Transport System concept should be considered applicable and when a proposal should be accepted. This paper provides a conservative way to begin and would allow us to gain some experience in accepting and rejecting Transport System proposals. We expect there will be some sharing of proposals among Competent Authorities and some development of statistics to support the compensating value of operational restrictions.

In the coming years, the need to transport radioactive materials which do not easily fit into the current categories of transport regulations or into reasonably available packagings will arise more frequently. Also, there is growing concern and scrutiny, by the public, of all activities involving radioactive material. Formal criteria and a formal process for Special Arrangement approval based on Transport System safety would have many beneficial effects. These benefits include a reduction in the time required to obtain an approval, an increase in the efficiency of decontaminating and decommissioning nuclear facilities, and assurance of an equivalent level of safety. Wouldn't it be better to develop criteria for when and how these Transport System proposals are evaluated within the regulations rather than try to accommodate them on an ad hoc basis under the Special Arrangement system?

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MEMBER STATE EXPERIENCE WITH NATIONAL REGULATIONS

(Session 3)

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TRANSPORTATION ISSUES FACING THE INTERNATIONAL COMMUNITY

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Abstract

This paper presents the concerns of the technical staff of the U.S. Nuclear Regulatory Commission on several proposals to revise the IAEA Regulations for the Safe Transport of Radioactive Material and the supporting documents.

1. INTRODUCTION

The International Atomic Energy Agency (IAEA) first published international regulations on the safe transport of radioactive material in 1961, and has revised these regulations, from time to time, as needs and experience indicated. Most member states having significant nuclear programs have adopted IAEA transport regulations as a basis for their national regulations and for application to international transportation. The packaging standards embodied in the IAEA transport regulations have provided a high level of protection for the public, and have made a significant contribution to the excellent transportation safety record achieved by IAEA member states.

Nevertheless, the public remains quite concerned about the transport of nuclear materials, particularly spent fuel, on their highways and railways and through their communities. These shipments bring large segments of the public in closer contact with large quantities of radioactive material than most other nuclear activities. As a consequence, major studies and tests have been conducted in the United States and other countries, to demonstrate that large margins of safety exist under IAEA regulations. Now, however, we see several proposals to revise IAEA regulations, that seem to be moving toward lower costs and reduced safety margins (i.e., less rigorous package standards). It is likely that the NRC staff would be unable to justify or support movement in this direction.

This paper presents the views of NRC technical staff on several important issues before the IAEA. These include: the transport of low-specific-activity material (LSA); the properties of materials used for shipping casks; plutonium air transport; and transport system certification. The views presented in the paper are based on the NRC's experience as the lead agency in developing and implementing Type B and fissile material package standards for the US. However, it should be emphasized that the views presented do not necessarily represent the official or final US position on these issues. The official US position will be determined by the U.S. Department of Transportation, which serves as the National Competent Authority for the United States.

2. LOW SPECIFIC ACTIVITY MATERIAL

The shipment of LSA material is an important issue both within the United States and within the international community. We believe that there needs to be a fresh look at the rules governing shipments of LSA material. Some types of LSA materials being shipped today were not contemplated in the initial development of LSA package requirements. The transport of large volume LSA packages containing contaminated resins from nuclear reactors, for example, raises several technical questions about existing package performance criteria. The NRC staff believes that the system used to regulate the shipment of LSA materials in the United States needs to be re-examined in light of these new materials being shipped as LSA material.

The total quantity of radioactivity which can be shipped in a non-accident resistant LSA package is limited by the dose rate from the unshielded material. The dose rate limit does not fully address potential problems in that it: a) does not prevent tons of highly dispersible contaminated resins from being shipped in a single package; b) is not an effective method to control total activity content of the package; c) bears little, if any, relationship to the radiological risk associated with large volume, highly dispersible materials in the event of a package rupture; and d) does not take into account the costs of intervention (as defined by ICRP) in the event of package rupture, particularly if it should occur on a major highway transport system (such as the accident involving fresh reactor fuel which occurred on a major US interstate highway near Springfield, Massachusetts on December 16, 1991). Intervention costs might include, for example, closing of a major highway system and the safety problems that could create, evacuation of contaminated areas and associated costs, elevated public concern and the cost of decontamination.

The potential for large intervention costs exists because of the large volume of material, and total activity level that can be shipped in a single LSA package.

For example, an LSA package used to ship contaminated reactor resins can contain up to six cubic meters of resin, and may have a total activity of greater than a terabequerel of cobalt-60. In contrast, non-LSA Type A packages typically contain much smaller quantities of material, generally only tens of cubic centimeters, and are limited, in the case of cobalt-60, to an activity of 0.4 terabequerel. Because of the larger volumes and activities, cleanup efforts in the event of a severe accident could be significantly more difficult and costly for an LSA package than for a Type A package. In fact, the magnitude of the cleanup efforts needed could lead to significant secondary safety problems and dislocations should the accident occur on a major highway. In addition, the larger volumes of material in an LSA package could become more widely dispersed than the contents of a Type A package, exposing a greater segment of the public. For these reasons, NRC staff believes that current transport regulations, which allow large volumes of contaminated reactor resins to be shipped as LSA material, may not be consistent with the level of protection afforded by other non-accident resistant, i.e, Type A, packages.

In short, NRC staff believes that a new approach is needed in regulating the shipment of LSA material - an approach that requires a much more in-depth review of the totality of accident consequences, available options and rigorous application of the optimization principle. The in-depth review should include a rigorous regulatory analysis of the potential costs versus benefits for the various options considered, as well as any necessary backfit analysis that may be required.[1] The review is a necessary first step at arriving at a defensible position, as to what kinds of shipments are appropriate for the current type of LSA packages, and which ones need to be shipped in an accident resistant Type B package.

While on the subject of LSA packaging, it should be noted that there are also technical problems with dose rate limit as a means to control the quantity of radioactivity in LSA packages. Dose rate measurements from a shielded package are often too low to permit a meaningful or reliable extrapolation to an unshielded configuration. This is particularly true for radionuclides with lower gamma energies. For example, a typical resin package, which has substantial lead and steel shielding, would provide an attenuation factor of seven orders of magnitude for cesium-137. Dose rate calculations can introduce additional errors, for example, incorrect buildup factors or exposure-to-dose conversion factors. Nonuniform source concentration and irregular source geometry also complicate calculations of unshielded dose rates. Applying the IAEA limit may also result in confusion as to whether the unshielded dose rate measurements should include secondary containers such as steel drums or liners.

[1] NRC regulations in 10 CFR Part 50 require a backfit analysis to determine if proposed regulatory changes have a significant impact on the operational requirements for the Commission's licensed nuclear power reactors.

In summary, our collective challenge is to develop within the IAEA safety framework, an overall strategy for regulating the shipment of LSA material that considers both the consequences and intervention costs of a severe accident.

3. MATERIAL PROPERTIES OF SHIPPING CONTAINERS

IAEA and US transportation regulations provide a high degree of confidence that transportation packages will survive the conditions expected in most transportation accidents. One of the factors that has contributed to this high degree of confidence in the United States, is that NRC has applied stringent criteria to materials used for transportation packages. This practice is justified by the potentially demanding loads and conditions to which transport packages may be subjected. It is also desirable to have large margins in material behavior, for transportation packages, because these packages are used in environments where public access cannot be controlled, and where the potential consequences of a package failure are great.

NRC has performed safety studies of spent fuel shipping casks under accident conditions that are beyond those encompassed by its regulations.[1] These studies have shown that NRC-certified casks provide a high degree of safety, even for accidents that exceed the performance requirements in US regulations. A major reason is that NRC-certified transportation casks are constructed of materials that behave in a ductile, plastic manner, when subjected to high levels of stress and strain. If overloaded, the degree of failure would likely be limited, and characterized by arrested cracks and localized leaking.

In contrast, brittle failure can be characterized as a sudden fracture and potentially total rupture. Based on this potential for catastrophic failure, NRC has not approved brittle materials as structural components of shipping packages or casks. These materials include nodular cast iron, depleted uranium, borated stainless steel, and borated aluminum.

The use of one material, nodular cast iron, for spent fuel transport casks, has been a matter of controversy in the United States between the NRC staff and industry. It has been used in Europe for transportation casks for several years. Casks constructed from this material could possibly pass US requirements in 10 CFR Part 71,[2] depending on the size of potential internal material flaws and the effectiveness of cask impact limiters in controlling stresses. However, NRC staff believes that the experience and data available for this material do not provide the assurance of sufficient margin, given the uncertainties of potential loads, uncertainties of the existence of flaws and the temperature to which the package may be subjected. Also, nodular cast iron casks do not have the large tolerance for overload that is inherent in present casks.

The material properties of nodular cast iron (and other non-ductile materials) are sensitive to the fabrication process and are difficult to reproduce. As a result, a high level of quality assurance is required to adequately control the fabrication process. Nodular cast iron is not authorized by the American Society of Mechanical Engineering (ASME) code for use in nuclear vessels, nor for use in non-nuclear vessels that contain lethal substances.

The NRC staff believes that the IAEA's recent intent to publish guidance on brittle fracture criteria, which would allow use of brittle materials in spent fuel casks, runs counter to good safety practice and is not in the public interest. The criteria being considered by the IAEA appears to represent a substantial reduction in the safety margins provided by previous IAEA regulations, and as practiced by NRC. Use of the IAEA brittle fracture criteria would introduce the possibility of catastrophic failure of casks involved in transportation accidents. In addition, the environmental and safety studies that have been conducted in the United States and elsewhere, to demonstrate the safety of spent fuel shipping containers to the public, are based upon ductile behavior of cask materials at high levels of stress and strain.[3] As a consequence, existing studies cannot be used to support the safety of non-ductile materials in transport casks.

Technical criteria for preventing brittle failure in transportation casks have been developed and published in NRC regulatory guides.[4,5] These guides have undergone extensive peer review and evaluation in the United States. NRC staff is currently reviewing the appropriateness of adopting these fracture criteria in 10 CFR Part 71 to exclude specifically, by regulation, non-ductile materials for use as structural components in transportation casks. The NRC staff would urge the IAEA to carefully consider the implications of its brittle fracture criteria, not only on package safety, but also on the public's perception of, and confidence in, transportation safety.

4. SYSTEM CERTIFICATION

A fundamental principle underlying both IAEA and NRC transportation regulations is that the shipping package provides the primary means of protection for the public. The shipping package must be demonstrated to provide adequate containment, shielding, and criticality control. By requiring that shipping packages meet these conditions for a rigorous set of normal and accident conditions, the public is assured that the package will protect against a wide range of possible accidents. Other factors, such as restrictions on the way packages are transported, or required operating conditions, have been used in some package approvals, but have always been secondary in importance to package integrity.

A change that IAEA currently is considering for its 1996 regulations is system certification. System certification would permit packages to be approved that do not meet current performance standards, provided that other controls are placed on the shipment of these packages. To approve a package under system certification, it would be necessary for a package designer to show that controls placed on package shipment provide a comparable degree of safety as a package meeting the required performance criteria. Application for package approval, under system certification, would be based on a risk analysis of the package, imposed operation controls, and individual shipment plans.

It is not clear that system certification is needed or justified. There are already provisions in IAEA, and in most national regulations, that permit exemptions, under specific instances, for packages that do not meet required performance standards. For example, special arrangements are permitted in Safety Series 6, in paragraphs 141, 211, 720, and 727. Similarly, exemptions from US package regulations can be approved under 49 CFR 107, Part B (the U.S. Department of Transportation) or 10 CFR 71.7 (NRC). The existing IAEA and US provisions are broad enough to deal with any situation that might arise, on a case-by-case basis.

Although an argument could be made that either package certification or system certification would provide a very low level of risk (risk = probability times consequence), it seems much more prudent to rely on package standards. Package design standards are well understood. They have been the subject of numerous environmental and risk studies that show that the standards provide a wide margin of safety against the consequences of most probable accidents. Once a package has been certified as meeting regulatory standards, there is a very high degree of certainty that the packages will performed as expected. In relying on package standards, one can assume that the probability of a severe accident (i.e., one involving conditions within the performance envelope of existing standards) is unity. In system certification, the hope is that the controls in the system prevent the package from experiencing the forces of a severe accident or otherwise mitigate the consequences of a package failure.

For system certification, it would be necessary to establish an appropriate limit on risk, and then perform a probabilistic safety assessment (PSA) involving all components of the system to demonstrate that it meets the risk criteria. Examples of special operating controls or special conditions that are often mentioned include: routing through less populated areas, limiting transit to nighttime hours, satellite tracking of shipments, prepositioning of highly trained response personnel, reduced speeds of transit, and traffic control. All of the operating controls or special conditions mentioned seek to control risk by reducing the probability of an accident occurring. The risks of relying on special operational controls are not well quantified and may be dependant on many variables that are difficult to determine. It is doubtful that a PSA could be performed with a high degree of confidence, if at all.

Simply stated, there is little assurance that performance standards can be developed to analyze system safety standards, which will be as rigorous and as well understood as those now applied to cask design. Before proceeding further, there should be a careful review of ICRP 60 and 64, as well as the IAEA-INSAG report (currently in draft), as they relate to potential exposure and the status and applicability of PSA to these situations.

5. PLUTONIUM AIR TRANSPORT

One area where the public has expressed significant concern is the shipment of plutonium, especially by air. The public has demanded, oftentimes through its legislative representatives, a very high standard of safety for plutonium air transport packages. The risk that the public is willing to accept from plutonium shipments is perhaps lower than for any other hazardous substance. The NRC staff is concerned about the discrepancy between the proposed IAEA Type C standards and existing US standards for plutonium air transport, and the effect this discrepancy might have.

The United States has had very conservative safety standards for the air transport of plutonium since 1975, when U.S. Public Law 94-79 (the Scheuer Amendment) was enacted. This law prohibited NRC from licensing the export, import, or domestic shipment of plutonium by air until a crashproof package, able to withstand the crash and explosion of a high-flying aircraft, was designed and certified to the U.S. Congress. The NRC staff believes that the new Type C standards under consideration for the 1996 Edition of Safety Series 6 (which would apply to plutonium air shipments) cannot be shown to be as stringent as the standards required under the Scheuer Amendment. Accordingly, plutonium air transport packages designed to the proposed Type C standards would not be acceptable for import, export, or domestic shipments, within the United States.

There are several good technical reasons why the IAEA should consider adopting criteria based on the more stringent US standards. First, two package designs, PAT-1 and PAT-2, have already been developed, tested and certified as meeting the performance standards of US law. The certifications by the NRC are based upon the criteria in NUREG-0360, "Qualification Criteria to Certify a Package for Air Transport of Plutonium."[6] In addition, other countries have reportedly designed packages that meet the US criteria. Second, the criteria in NUREG-0360 have been reviewed and endorsed by two important peer groups - the U.S. National Academy of Sciences and NRC's Advisory Committee on Reactor Safeguards.

The US peer reviews have concluded that the qualification criteria in NUREG-0360 would assure that package survival will approach certainty in aircraft accidents occurring during take-off, landing, or ground operations, and would provide a high degree of protection against accidents that occur in other phases of flight, (e.g., mid-air collisions). Other studies have predicted that packages built to these criteria, which require a package to withstand an impact test at 130 m/sec on to an essentially unyielding surface, would survive over 99 percent of aircraft accidents.[7] In contrast, the same studies predict that packages built to the proposed Type C standards, which require an impact test at 85 m/s onto an unyielding surface, would survive anywhere from 67 to 98 percent of possible aircraft accidents (ref.7).

Finally, it should be noted that the trend, at least in the United States, has been towards even more stringent standards. In 1987, the U.S. Congress passed U.S. Public Law 100-203 (the Murkowski Amendment). This law prohibits transport of plutonium through US airspace unless the NRC certifies to Congress that the package can withstand both an aircraft crash test, and a drop test from the aircraft's maximum cruising altitude. The law specifies that tests used to demonstrate the package's survivability be based on stresses that could be experienced in a worst-case aircraft accident.

6. CONCLUSION

Under IAEA and US regulations, the safety of shipping radioactive material is primarily dependent on the integrity of the shipping package. The NRC staff believes that proposals before the IAEA in several key areas, brittle fracture criteria, system certification, shipment of LSA material, and plutonium air transport, could lead to a reduction in safety margins for radioactive material packages standards.

The safety margins of current package standards are supported by numerous technical studies, and we believe that these standards have a great degree of public confidence. Further, the safety margins are based on the use of non-brittle materials for shipping casks. The NRC staff would urge the IAEA to carefully consider the implications of publishing its brittle fracture criteria, not only on package safety, but also on the public's perception of, and confidence in, transportation safety.

NRC staff does not believe that system certification, which would rely on a sophisticated PSA, would provide the same margin of safety as package performance standards. A PSA relies on special operational controls that are not well quantified and may be dependant on many variables that are difficult or impossible to determine.

Finally, the NRC staff believes that an overall strategy for regulating the shipment of LSA material needs to be developed that considers both the radiological consequences and intervention costs of a severe accident.

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REGULATORY ASPECTS OF THE TRANSPORT OF IRRADIATING AND ALPHA WASTE IN FRANCE

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Abstract

The introduction of the 10 mSv/h at 3 m limit for LSA unshielded material makes impossible the transport as LSA material of the most irradiating wastes from EdF PWR's operations At the present time, the concerned EdF's waste blocks are allowed to be transported as LSA III material under special arrangement A new package design, equivalent to a type B package, will be available for their transport before the end of the year 1995 It consists in a reusable steel cylinder over packing each block Compliance of this package model with transport safety requirements will be demonstrated by taking into account the non dispersability, as LSA III material, of the irradiating waste

The transport of LSA combustible material is affected in the revised regulation by the new limitation to 100 A2 of the total activity per conveyance Complying with this limit would strongly reduce the quantity transported by conveyance and therefore multiply the number of journeys, which is not desirable A two step approach has been accepted by the French Competent Authority for the transport of these waste

a) A specific ISO 20' container, thermally insulated, can be used under special arrangement for the transport of LSA combustible material having a total activity per conveyance higher than 100 A2 Furthermore additional safety measures have to be implemented for these transports

b) After the end of the year 1995, a type B package must be used for activity contents per conveyance higher than 100 A2 A specific 20' ISO container, complying with type B requirements, is being developed for that purpose The total plutonium mass transported per conveyance will be limited to 400 g for criticality and physical protection considerations

An interpretation of the general LSA requirements formulated in the current IAEA Regulations is presented with respect to the homogeneity of the radioactive material, and the definition of the unshielded material

1) INTRODUCTION

The aim of this paper is to present the regulatory aspects or consequences of recent developments concerning the transport of low and intermediate wastes in France They are mainly due to the impact on LSA material transport activities of the revision of the French Regulation for the transport of dangerous goods [1], according to the IAEA Regulations for the Safe Transport of Radioactive Material [2], 1985,1988 and 1990 editions

2) IRRADIATING WASTE

Safety Series n°6 [2] indicates in its § 422 that the quantity of LSA material in a single industrial package shall be so restricted that the external radiation level at 3 m from the unshielded material or object or collection of objects does not exceed 10 mSv/h

The introduction of this new limit makes impossible the transport as LSA material of the most irradiating waste from EdF PWR's operation such as primary loop filters or ion-exchanger resins Therefore a type B package should be used

These wastes, conditioned in cement or polymer, correspond roughly to 5% of the total volume of waste sent by EdF to the Agence Nationale pour la gestion des Déchets RAdioactifs (ANDRA) waste disposal facility

At the present time, due to power plant operation and waste storage constraints on site, the existing EdF's waste blocks are allowed to be transported as LSA III material under special arrangement for a maximum total activity of 5 TBq, corresponding to a maximum dose rate up to 4 Sv/h at the surface of the unshielded material (or up to 100 mSv/h at 3 m from the unshielded material) Additional safety measures were required for these transports such as the choice of certified carrier road transports (about 30% of these transports)

To cope with the technical requirements of the ANDRA waste disposal facility, in addition to leaching tests these blocks have been submitted to 1 2 m drop tests, as well as a 800°C, 30 minutes fire test These tests guarantee a good behaviour for leaching, fire and potential mechanical constraints For more severe accidental conditions, such as those envisaged by the IAEA Regulations for qualifying type B packages, the consequences of atmospheric dispersion are a priori limited due to the fact that the material is LSA Nevertheless, these tests are not sufficient to prove that the radiation shielding is adequate to withstand the IAEA type B tests

Before the end of the year 1995, a new package design equivalent to a type B package, will be available for their transport It consists in a reusable steel cylinder over packing each block (see figure 1) It can be handled by the existing means at

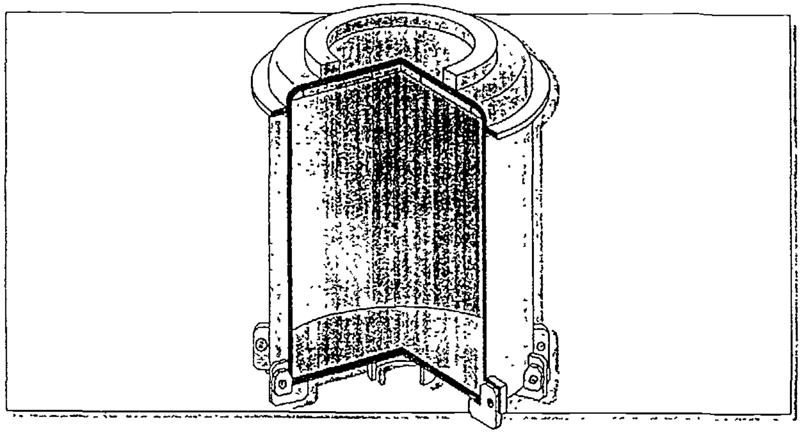


Figure 1 : OVERPACK FO IRRADIATED WASTE

EdF power plants and at the ANDRA waste disposal facility. It will ensure the biological protection in case of accidental situations considered for the qualification of type B packages.

Compliance of this package model with transport safety requirements will be demonstrated by taking into account the non dispersability, as LSA III material, of the irradiating waste. That is to say, the designer would not have to demonstrate compliance with regulatory release rate criteria both in normal and accidental situations. This exception would insure consistency with Safety Series n°6 requirements applying to packages used for the transport of non irradiating LSA wastes. Therefore attention will mainly be devoted to the compliance with external dose rate criteria after the type B tests (the 10 mSv/h at 1 m criterion).

3) ALPHA WASTE

The transport of combustible radioactive materials with low specific activity (LSA II or LSA III), in industrial packages, is affected in the 1985 issue of the IAEA recommendations [2] regarding the safety of radioactive material transport by a limitation at 100 A2 of the total activity per conveyance (road vehicle, rail car...). The transport of combustible surface contaminated objects (SCO) comes under the same limitation.

3.1 Problems raised by the transport of non immobilised waste with alpha emitter low specific activity :

Solid waste containing small quantities of alpha emitters per mass unit may be put into the following categories (e. g. plutonium 239):

- LSA II, if their alpha emitter specific activity is less than 2.10^{-5} TBq/kg (0.5 Ci per ton),
- LSA III, if the alpha specific activity is less than 4.10^{-4} TBq/kg (10 Ci per ton), provided that the requirements concerning leaching properties are met.

Accordingly, drums, blocks or boxes containing non immobilised waste with small amounts of alpha emitters per mass unit, arising essentially from CEA laboratories and likely to be accepted in surface waste disposal sites, may be put into the LSA II category, the upper limit of the specific activity for this category, corresponding roughly, after immobilisation at the waste disposal site, to the limit of 4.10^{-6} TBq/kg (0.1 Ci per ton) set for the alpha specific activity of waste packages being accepted in such sites. If the waste are considered as combustible, the limitation to 100 A2 of the total activity per conveyance would require a reduction of the mass of waste transported in the form of industrial packages (in the present case 20' or 40' containers) as low as one ton if the activity of all the waste were at the upper limit of the LSA II category. An accurate inventory of the total alpha activity to be shipped, and realistic estimates of the alpha emitters concentrations in the waste might attenuate the effect of this limitation. However, the limitation of the total activity per means of transport concems the transport practices in shipments of Low Level Waste from CEA laboratories to surface waste disposal facilities.

The effect of limiting to 100 A2 the total activity per conveyance applicable to combustible wastes, is still more pronounced for drums, blocks or boxes of solid wastes which are not acceptable in surface waste disposal centres because of their excessive alpha emitter content, which have to be sent to interim storage sites, and which can be classified as LSA III. Indeed to be able to transport such wastes in the form of industrial packages, the mass of waste transported by conveyance would have to be reduced to a value between 50 and 1000 kg.

It should be pointed out that the classification as "non-combustible" material of solid non immobilised wastes transported in drums, blocks or boxes is delicate, as soon as the waste are put in an organic envelop. Most of contaminated non immobilised radioactive waste shall therefore be considered as "combustible".

On the other hand, it should be borne in mind that wastes made of well geometrically defined independent pieces, for instance those coming from the dismantling of nuclear installations, can be put into the category of surface contaminated objects (SCO). The limitation of the total activity to 100 A2 also applies to this category, whether the wastes are combustible or not, if industrial packages are to be used. It appears clearly that the intention of 1985 issue of the IAEA regulations is to prevent the transportation of too large amounts of LSA or SCO materials in industrial packages, not designed to withstand accident conditions, as soon as a subsequent fraction of the activity contained in the conveyance is likely to be dispersed in the event of an accident.

This situation did not escape to the attention of the French transport safety authorities Well in advance of the issue of both international and national transport regulations, complementary safety measures were taken for this category of transportation both to prevent accident and to limit the consequences of a possible accident in particular as concerns the risk of fire In addition, each transport campaign is the subject of a specific safety examination and of inspection of the loading conditions

The entry into force of the 100 A2 conveyance limit has major practical repercussions

it could lead to subdividing the material into batches of activity less than 100 A2 and accordingly increasing the number of shipments, so that industrial packages may still be used (in which case, large containers could no longer be employed) , obviously, this solution is penalising on an economical level and the benefits as regards safety are low,

- or, what is more likely, it imposes a requirement to move directly from industrial packages (IP) to far stronger type B packages for which there is no limitation of the total activity transported , the number of transport to be carried out is an incitement to develop a large-size package, but at the cost of difficult technological problems

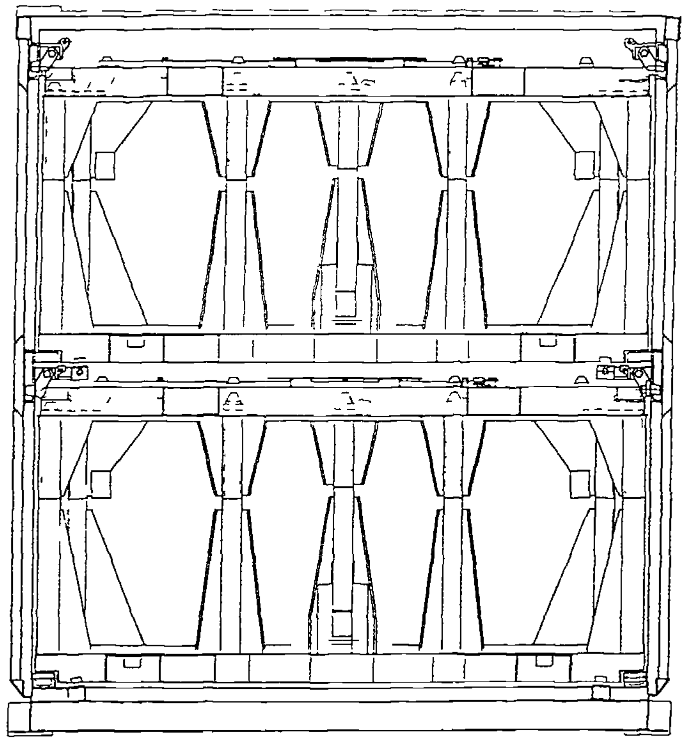
<u>3 2 Envisaged solutions</u>

A two step approach has been accepted by the French Competent Authority for the transport of these waste conditioned in 100 or 200 litres drums

a) A specific ISO 20' container, so-called DV77 (see figure 2), thermally insulated, can be used under special arrangement for the transport of LSA combustible material having a total activity per conveyance up to 600 A2 The maximum individual dose associated with the most severe accident involving this quantity is about 50 mSv, (see [3]), figure considered as acceptable This ISO 20' container can transport 60 drums of 200 litres or 150 drums of 100 litres

Furthermore additional safety measures have to be implemented for these transports such as escort car and speed limitation

b) For activity contents per conveyance higher than 100 A2, after the end of the year 1995, a type B package must be used A specific 20' ISO container so called TN GEMINI (see figure 3) complying with type B requirements is being developed for that purpose It will allow to carry transuranic waste in 40 drums of 200 litres or 60 drums of 100 litres , 5 m^3 or 10 m^3 metallic containers or fibber reinforced concrete cylinders may also be accommodated The total plutonium mass transported per conveyance will be limited to 400 g for criticality and physical protection considerations



<u>Figure 2 THE DV 77</u>

4) INTERPRETATION OF THE LSA IAEA REQUIREMENTS

An interpretation of the general LSA requirements formulated in the current IAEA Regulations [2] has been made with respect to the homogeneity of the radioactive material, and the definition of the unshielded material

The applicants mentioned their difficulties in the interpretation of the new rules in this field [2], [4], [5]

A group of the French main applicants was gathered under the aegis of IPSN

Two main parameters were considered the massic specific activity and the distribution of the activity in the material Homogeneity is desirable in order to avoid the diversion of the low specific activity concept However the practice requires to

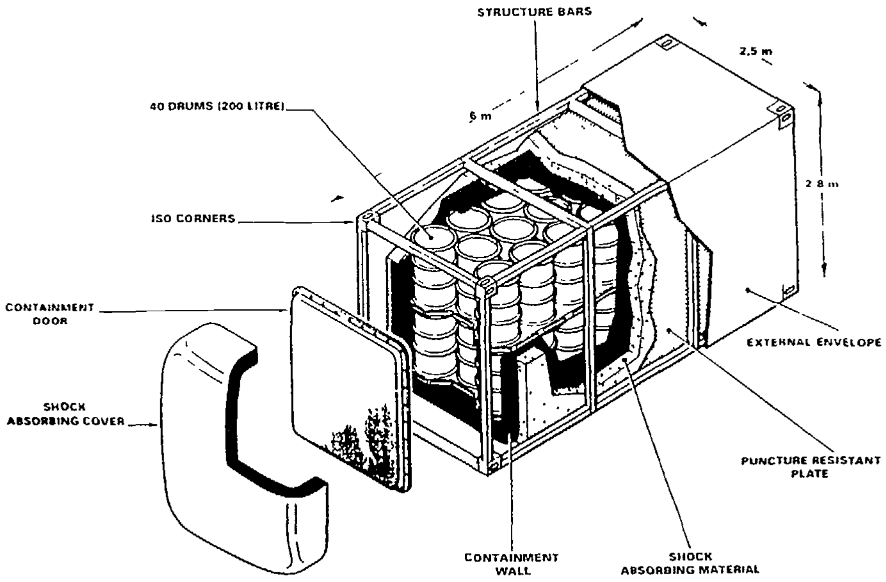


Figure 3 : THE TN GEMINI

limit and define this homogeneity in a way as simple as possible. Therefore Safety Series n°37 [5] proposes a factor 10 in the LSA II case, 3 in the LSA III one, between the global average activity and that of each elementary volume. A fire for example involving only the most active part of non homogeneous waste would have by definition consequences of limited dimensions. On the contrary, a fire involving the whole material would bring us back to the average specific activity.

The elementary volumes were chosen essentially for practical reasons 0.2 m^3 being the volume of the most frequent LSA drums, this figure was selected in the LSA II case by the group in accordance with Safety Series n°37 Half this volume was considered as reasonable for LSA III, supposed to be more homogeneously distributed.

Besides it seams logical that each isolated volume of 0.2 m^3 having an average specific activity of 10^{-4} A2/g is supposed as acceptable as an inserted elementary one having an activity 10 times higher.

The proposed outcome is the following :

In order to verify the homogeneity of distribution of this activity in the material, the maximum volume of the wastes in the packaging is assessed and is divided in N elementary volumes smaller than or equal to 0,2 m^3 for LSA II material, smaller than or equal to 0,1 m^3 for LSA III material

It is then verified that the specific activity contained in each of these elementary volumes is lower than 10^{-3} A2/g for LSA II material, lower than 6.10^{-3} A2/g for LSA III material.

Furthermore, the waste producers have to make sure that the activity is not gathered in one or a few of these elementary volumes. Every volume of waste smaller than 0,2 m^3 for LSA II material (respectively 0,1 m^3 for LSA III material) is considered as meeting the rules of homogeneity, when complying with specific activity conditions.

Furtermore, the assessment of the dose rate at 3 m from the surface of the material has to be made for the volume of the waste, without considering radiological protection.

5) CONCLUSION

The adaptation to the new rules of transport of radioactive waste was not obvious in France and it is still in development. But the target will be hit and the safety of these transport will be substantially improved.

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REGULATORY ASPECTS IN THE TRANSPORT AND DISPOSAL OF LOW AND INTERMEDIATE LEVEL RADIOACTIVE WASTES FROM FUEL CYCLE OPERATIONS

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Abstract

REGULATORY ASPECTS IN THE TRANSPORT AND DISPOSAL OF LOW AND INTERMEDIATE LEVEL RADIOACTIVE WASTES FROM FUEL CYCLE OPERATIONS

Nuclear Fuel cycle facilities contribute significantly to the generation of radioactive wastes. As per the Atomic Energy Act, 1962, processing and management of radioactive wastes in India is the responsibility of the Department of Atomic Energy , in compliance with the rules, regulations and guidelines prescribed by the competent authority, namely the Atomic Energy Regulatory Board. In the Indian programme of waste management, each nuclear site has its own waste management facilities catering to the needs of the nuclear installations at the site. By this scheme, the transport of radioactive wastes is avoided in the public domain except for spent fuel shipments to the reprocessing plants. All waste movements take place inside the controlled areas of the nuclear site. Radiation Protection Rules, Radiation Surveillance Procedures for Safe transport of Radioactive materials and the AERB code for Safety in the Transport of Radioactive Materials provide the necessary regulatory controls for safety in transportation. The safety code is further supplemented with technical and administrative procedures, issued in form of regulatory guides. The paper describes in detail the regulatory control aspects in transport. The type and form of radioactive wastes generated in nuclear fuel cycle operations, their conditioning and the type of packages that are employed for the transport of the waste within the nuclear sites are explained. An adequate and competent infrastructure has been built to cater for the diverse waste management requirements including transport at nuclear installations in India.

1. INTRODUCTION

Nuclear power generation and associated nuclear fuel cycle activities from mining to fuel fabrication and reprocessing of irradiated fuel contribute to radioactive waste generation in significant quantities. Decommissioning of nuclear facilities will also result in considerable amounts of waste materials that are radioactive or contaminated with radionuclides to a widely varying extent. The facilities and plants connected with the above operations are under the direct control of the Department of Atomic Energy (DAE) of the Government of India. The other installations, not under the control of DAE, such as research institutions, hospitals, industrial radiography companies and other users of radioisotopes also contribute to radioactive waste generation. However, it is the nuclear fuel cycle, which is of major concern from the point of view of protection of the environment and is addressed in this paper. DAE has established waste management facilities at all nuclear sites to cater to the requirements of the nuclear installations at the sites.

2. REGULATORY ASPECTS OF WASTE MANAGEMENT

Radioactive waste management in India must comply with the provisions of the Atomic Energy (Safe Disposal of Radioactive Wastes) Rules, 1987 issued under the Atomic Energy Act, 1962, Environmental (Protection) Act, 1986 and Environmental (Protection) Rules, 1987. The waste disposal rules cover all aspects ranging from the processes resulting in generation of radioactive wastes to conditioning, storage and disposal of such wastes. Provisions have also been made for hospitals and research laboratories using small quantities of radioisotopes.

Chairman, Atomic Energy Regulatory Board (AERB) has been notified by the Government of India as the Competent Authority for the enforcement of the provisions of Atomic Energy (Safe Disposal of Radioactive Wastes) Rules 1987. An authorisation is to be obtained from the Competent Authority by each person or installation for disposal of radioactive wastes or for their transfer to an authorised waste management agency., It is the responsibility of the authorised person or installation to ensure compliance with the terms and conditions of the authorisation, maintain records of waste disposed, ensure compliance with the Radiation Protection Rules, 1971 and carry out personnel monitoring and environmental surveillance on a continued basis. He should also appoint with the approval of the Competent Authority a Radiological Safety Officer to carry out radiation safety surveillance. The Competent Authority is authorised to carry out inspection of the installation, to issue directives as deemed necessary in the interests of radiological protection and to suspend or cancel the authorisation given to a person if the latter has failed to comply with the terms and conditions of the authorisation.

Chairman, AERB has been notified as the statutory authority to carry out certain duties envisaged under Environmental (Protection) Act and Rules in respect of the installations set up for the furtherance of the objectives of the Atomic Energy Act, 1962. These duties include (a) power of inspection of the installation and (b) power to take samples of air, water, soil or any other substances from any factory or premises for purposes of analysis. The Ministry of Environment and Forests has recognised the Environmental Survey Laboratories set up by the

Health Physics Division of the Bhabha Atomic Research Centre (BARC) at nuclear sites to carry out environmental protection surveillance for the installations of the Department of Atomic Energy.

3. RADIOACTIVE WASTE TRANSPORT IN INDIA

In the Indian programme of radioactive waste management, waste management facilities are co-located with nuclear installations at each nuclear site. No off-site storage, transport or disposal is practiced as yet. Each nuclear site has a liquid waste treatment plant and a solid waste management facility. A near-surface repository for low and intermediate level solid waste is also set up at each site. In addition, an interim storage facility for high level solid wastes is also planned at each site, where plant for immobilisation of high level liquid wastes is located. In the Tarapur site, where a twin unit BWRs and a reprocessing plant for power reactor fuel are in operation, an interim storage facility based on natural convective air cooling with induced draft is already operational. In general, siting considerations for locating nuclear power plants and / or reprocessing plants take into account establishment of the required waste management facilities also. By the above scheme, transport of radioactive wastes are avoided in the public domain, except for irradiated fuel shipments to reprocessing plant. All waste movements take place within the controlled areas inside nuclear sites. The only radioactive waste materials that are transported in the public domain are decayed sealed sources from the users and occasionally contaminated objects sent to BARC for final disposal.

The Indian programme currently envisages investigation of candidate sites for a final repository for the high level vitrified waste products and alpha bearing wastes. This facility is planned to be set up on a centralised basis and will be common for many nuclear sites. When such a facility is established, transport of high level immobilised wastes in the public domain will arise and will be subject to regulatory control.

The type and form of radioactive wastes that are required to be transported within the controlled areas of the nuclear sites depend on the nature of operations that are carried out at the waste management facilities. In the following paragraphs, a brief summary of the principles and processes followed for treatment of low and intermediate level liquid and solid wastes are described. High level waste management is excluded from the scope of this paper as well as that of the seminar.

4. WASTE MANAGEMENT PRINCIPLES

The basic philosophy in the management of all radioactive wastes has been (i) dilution and dispersal of low active wastes, (ii) delay / decay and dispersal of waste containing short-lived radionuclides and (iii) concentration and containment of highly active waste containing long lived radionuclides.

The broad outlines of the waste management policy are :

(1) Any discharge of radioactive liquid or gaseous wastes to the environment should be as low as reasonably achievable, economic and social factors being take into consideration;

(2) Solid wastes resulting from conditioning of waste concentrates or liquid wastes generated in the operation of reactors and research facilities are to be stored in near surface repositories, specially engineered for the purpose. Conditioned low and intermediate level wastes along with trace quantities of alpha contamination from fuel reprocessing facilities are also permitted for storage in such facilities;

(3) High level liquid wastes from reprocessing plants are initially stored for an interim period in high integrity stainless steel tanks underground. These wastes will be vitrified and the solidified products will be provided interim storage in near surface engineered storage facilities with appropriate cooling and surveillance provisions. The interim storage will allow for decay and hence reduction in heat generation;

(4) High level vitrified and cooled waste products and alpha wastes will be disposed off in a suitable deep geological formation, serving as a centralised repository.

5. WASTE CATEGORISATION

For the purpose of waste handling and treatment, radioactive wastes are categorised generally as :

Low Level Waste (LLW)	-	Waste because of its low radioactivity content, does not require shielding during normal handling and transportation.
Intermediate Level Waste (ILW)	-	Waste because of its radioactivity content, requires shielding but needs no provision for heat dissipation during handling and transportation.
High Level Waste (HLW)	-	Waste with high radioactive content, mostly from fuel reprocessing, requires shielding during handling and transportation and needs cooling due to decay heat.

Alpha Waste - Waste with high concentration of alpha activity and low decay heat, requires remote handling in air-tight enclosures.

6. TREATMENT OF LOW AND INTERMEDIATE LEVEL WASTES

Present practices in respect of low and intermediate level radioactive waste management are summarised below :

(1) Uranium mining and milling wastes

The barren liquor produced in the uranium recovery process is neutralised with lime and treated with $BaCl_2$ to precipitate radium and other daughter products of uranium. The precipitate is mixed with the tailings slurry and is disposed off into a tailings pond, which is normally a natural depression that ensures settling. The waste transportation from the mill to the tailings pond is by means of pumped transfer.

(2) Low and Intermediate Level Liquid Waste

The low level liquid radioactive wastes are treated by chemical, ion-exchange or evaporation methods. The sludge containing most of the radioactivity is separated and sent for solidification, storage and disposal. The treated effluent is discharged after further dilution and monitored to meet the specified regulatory discharge limits. With the continuing trend to restrict the discharges of radioactivity to the environment to as low as possible, a solar evaporation facility is being operated at the Rajasthan Atomic Power Station. The concentrate reduced in volume several fold, is solidified in suitable matrices for long term storage. Cement-concrete is found to be a suitable matrix for incorporation of waste concentrates with low activity. Bitumen is used for incorporation of concentrates from evaporation with intermediate levels of activity. Polymer matrices have also been used to immobilise spent ion exchange resins and higher level active waste concentrates.

(3) Low and Intermediate Level Solid Waste

Radioactive solid wastes are generated in nuclear fuel cycle operations in a wide variety. Primary solid wastes consist of contaminated articles and products resulting from active chemical processes, viz., tissue materials, glassware, plastics and protective rubber wares in the contaminated category and ion exchange resins, filter sludge, incinerator ash etc. in the latter category. While low level wastes are produced in large volumes in the front end of the fuel cycle, intermediate levels of activity in solid waste are mainly from reactors and fuel reprocessing plants and isotope production laboratories.

Low active and combustible waste which constitute a major portion of the solid waste, are sent for incineration in specially designed incinerators, provided with gas cleaning systems for volume reduction. Other wastes are baled to reduce the volume prior to disposal.

Permanent storage or disposal of radioactive solid wastes is carried out only at controlled sites, evaluated for the purpose. In India, as a matter of policy, each nuclear site is planned to have a near-surface repository for low and intermediate level solid waste. The facilities in the repository include earthen trenches, RCC trenches, tile holes, RCC vaults etc. The reinforced concrete trenches located underground with waterproofing on the external surface and bitumen based painting on the inner surface provide good containment for low level solid wastes. For storage or disposal of higher level active wastes, deep circular underground tile holes are used. These are steel lined RCC pipes extending to 4-5 metres down and having water proofed external surface.

7. REGULATORY ASPECTS OF RADIOACTIVE WASTE TRANSPORT

Transport of radioactive waste is regulated under the same provisions that are applied to the movement of radioactive materials in the public domain for use in industry, hospitals and research laboratories. Radioactive material transportation in India is regulated under the provisions of the Atomic Energy Act, 1962, which empowers the Central Government to frame rules and lay down safety standards for ensuring an acceptable level of safety for transport workers and members of the public.

Radiation Protection Rules, 1971 issued under the Atomic Energy Act, the Radiation Surveillance Procedures for Safe Transport of Radioactive Materials issued under the above Rules and the AERB code for Safety in Transport of Radioactive Materials provide detailed regulatory control for safety during transportation. For the purpose of enforcement of the transport regulations, Chairman, AERB is notified as the Competent Authority by the Central Government. The regulatory provisions in the code are essentially based on the IAEA Regulations for Safe Transport of Radioactive Materials, Safety Series No.6 (1985 edition, as amended in 1990). IAEA has also issued supplementary information relevant to the interpretation of the regulations viz.,

(1)	An explanatory material	- Safety Series 7	[1]
(2)	An advisory material	- Safety Series 37	[2]
(3)	Schedules of requirements	- Safety Series 80	[3]
(4)	Guidance to Competent Authorities	- TECDOC 413	[4]

The IAEA regulations serve as the basis for shipments of radioactive materials in most countries, thus ensuring complete harmony among various national and international regulations.

The AERB safety code on transport of radioactive materials prescribe safety standards governing the package design and operational aspects. These standards are supplemented with technical and administrative procedures in a number of documents issued by AERB as guides. These are listed below with a brief description of their scope :

(1) **Safety Guide on Compliance Assurance Programme for the Safe Transport of Radioactive Materials (1991)**

The guide outlines the procedures for review and assessment of package designs, special form of radioactive materials, shipments and inspection activities.

(2) **Safety Guide on Standards of Safety in Transport of Radioactive Material (1991)**

The guide outlines the basic radiation safety standards governing the transport safety requirements stipulated in the code. It must be noted that the new ICRP recommendations with lower dose limits and introduction of the concept of dose constraints will have an impact on the transport regulations. In India, the effective dose limit for radiation workers is being reduced in a phased manner from 1991 onwards and the dose limit for the year 1993 was set at 30 mSv by AERB. For members of the public, the dose limit is 1 mSv / year from exposure from all radiation sources.

(3) **Safety Guide on Procedure for Forwarding, Transport, Handling and Storage of Radioactive Consignments (1991).**

The procedures to be followed by consignors of radioactive material of various descriptions for different modes of transport are detailed in this guide. Compliance with the procedures stipulated in the guide will help in avoiding transport incidents and improve safety in transportation in the public domain.

(4) **Safety Guide on Quality Assurance in Transport of Radioactive Materials (under issue)**

Essential features of quality assurance programme in the design, manufacture, use, maintenance, transport, documentation and storage-in-transit are explained in this guide for compliance. The guide is based on IAEA Safety Series No. 37 (as amended 1990) and the experience of AERB in regulating the safe transport of radioactive materials in the country.

(5) **Safety Code on Emergency Response Planning and Preparedness for Transport Accidents involving Radioactive Material (1990)**

The code specifies the requirements for establishment of emergency provisions, identifies emergency response organisations and prescribes response measures. The code is intended to be the basis for emergency response action plans to be drawn up by organisations / individual consignors of radioactive materials.

8. TRANSPORT PACKAGES FOR RADIOACTIVE WASTE

Four types of packaging are specified in the transport regulations viz., Excepted, Industrial, Type A and Type B. Excepted packages are permitted to contain only relatively small amounts of radioactivity. Hence, other types of packages are being used for the transport of radioactive waste. Industrial and Type A packages are being used to transport low level and intermediate level waste and the regulatory requirements in respect of radiation levels, specific activity and total radioactivity content are satisfied. Type B packages are used for the transport of intermediate and high level waste and irradiated fuel. Type B packages are designed to retain adequate shielding and containment under severe accident conditions. Prior approval of the Competent Authority i.e., Chairman, AERB is required for the design of Type B packages and their shipment. All Type A packages are required to be actually subjected to the prescribed tests. For Type B packages, theoretical assessments of package response to test conditions are examined prior to approval.

Radioactive waste classification is usually linked to safety aspects of their management. A quantitative approach to classification, recommended by IAEA, is adopted by AERB, and is given below. This classification is used for purposes of obtaining regulatory clearance from AERB by waste generating / management facilities.

Liquid Wastes

Category	**Activity Level Bq / m3 (beta-gamma)**	**Treatment / Disposal**
I	$< 3.7 \times 10^4$	Direct discharge to environment
II	3.7×10^4 to 3.7×10^7	Treatment, dilution and discharge
III	3.7×10^7 to 3.7×10^9	Treatment required
IV	3.7×10^9 to 3.7×10^{14}	Shielding, treatment, solidification, storage
V	$> 3.7 \times 10^{14}$	Cooling, shielding, immobilisation, storage

Solid Waste

Category	Gamma dose rate (mGy/h)	Type of waste	Type of container
I	0 - 2	Paper trash, concrete, cotton mops, rubberwear	Simple container, No shielding
II	2 - 20	Contaminated equipment, filters, hardwares	Simple container, nominal shielding
III	Above 20	Process concentrates, sludges, spent resin, highly contaminated equipments, spent sources	Specially designed container with shielding and cooling provision
IV	Alpha waste	Zircaloy hulls, spent resin from reprocessing plants, alpha waste from laboratories	High integrity container with special precaution for alpha emitters

9. TRANSPORT OF LIQUID WASTE

Low level liquid wastes are normally transported from the place of generation to the treatment plant in the site by engineered pipelines or by tankering under special cases.

(1) Transport by pipelines :

Design and operation of transfer pipelines should take into account quality assurance aspects during design, erection and operation, secondary containment, in-service inspection and maintenance, detection of line failure and management and provisions for cleaning and decontamination. Low level effluent pipelines could be above ground for purposes of inspection and prompt corrective actions. For transport of medium level liquid waste, underground pipelines with necessary shielding is a better option. The pipeline should have double containment to prevent release of activity into the ground in the event of pipe failure. Concrete trenches, if employed, should be given waterproofing to prevent ingress of subsoil water.

(2) Transport by Tankering

Transport by tanker within the establishment, where the liquid waste is generated, stored and treated does not fall under the purview of the provisions contained in the transport code. However, the design and operation of the tanker should take into account quality assurance in design, fabrication and operation, adequate shielding, safe containment of leaks, provision for decontamination, in-service inspection for assessment of integrity and safe anchorage to vehicles. The tank should not be filled to capacity and filling level depends on the gradient of the route to be followed. Tankers should preferably be free of contamination on external surface. If levels are in excess of 0.4 Bq/sq.cm for beta-gamma and 0.04 Bq/sq.cm for alpha, the tanker is treated as contaminated. The tankers are periodically decontaminated to minimise the residual radiation field to less than 2 mGy/h on the surfaces. At BARC, Trombay the maximum activity that can be transported in a tanker is at present maintained at 1.5×10^{11} Bq. Another stream of waste being tankered is degraded solvents such as TBP and Kerosene from the reprocessing plant. Because of the high gamma dose rate observed for this waste, transportation is carried out in high integrity shielded containers.

10. TRANSPORT OF SOLID WASTE

Since solid wastes are of widely varying radioactivity levels, segregation into separate castegories are practiced for appropriate disposal of such wastes. The categorisation, as described in section 8, is in line with IAEA recommendations.

For collection purposes, the wastes are tagged white, yellow or red at the source of generation.

White type	-	all non-active wastes and suspect-active materials
Yellow type	-	Active wastes with surface dose rates less than 2 mGy/h or containing less than 3.7×10^4 Bq alpha emitting or hazardous long lived isotopes or fissile materials per standard package.
Red type	-	All radioactive wastes with surface radiation levels greater than 2 mGy/h or containing more than 3.7×10^4 Bq of alpha emitting or other hazardous long lived isotopes per standard package.

All packages should be sealed properly, labelled with proper tags indicating the dose rate and activity content. Packages should preferably be free from external surface contamination but should be within the allowable limits. Waste collection forms duly filled in by the users and certified by the Health Physicist should accompany all waste consignments. Waste consignments above certain radiation/activity levels use shielded casks or other special containers. They are transported with an escort vehicle and accompanied by a Health Physicist and a senior officer from the Waste Management Plant.

Tables 1 and 2 give typical details regarding the nature of solid waste generated and the type of transport containers/casks in use at the Tarapur site, where a nuclear power station , a fuel reprocessing plant and associated waste management facilities are located. Transportation is carried out in approved containers/casks within the controlled area of the nuclear site.

11. CONCLUSION

In the Indian programme of radioactive waste management, each nuclear site has its own waste management facilities, catering to many nuclear installations at the site. The need for

TABLE 1

Details of Solid Waste Transport and Disposal at Tarapur Site

Type of waste (category)	Source of generation	Radiation level (Gy/h)	Type of packaging	Transport container	Mode of disposal
Assorted Waste Paper,PVC,bags, glass,protective wares (I)	NPP FRP WMF	≤ 0 002	Double PVC bag/canvas bag	MS Box with lid	Incinera-tion / baling, Disposal in ET/RCT
Spent liquid filters (II)	FRP NPP	upto 0 1	Polythene sheets and MS drum	MS box or Shielded box if > 2 mGy/h	Disposal in RCT
Spent ventilation filter(II)	FRP	upto 0 1	Polythene sheets	MS box or Shielded cask on trailor if > 2 mGy/h	Disposal in RCT
Chemical sludge(II)	WMF	0 02 - 0 2	Sludge transfer cask	Carried by trailor	Fixed in cement, Disposal in RCT or TH

TABLE 1 (contd.)

Type of waste (category)	Source of generation	Radiation level (Gy/h)	Type of packaging	Transport container	Mode of disposal
Spent resins, vermiculite, filter sludge (II)	NPP FRP WMF	upto 1	MS drums	Concrete cask by trailor	Disposal in RCT
Aluminium/ Zircaloy Hull (III)	FRP	1 - 100	MS drums	Shielded cask on trailor	Disposal in TH
Decommissioned equipment (II and III)	FRP NPP	So far handled upto 400	Cut into pieces and packed in MS drums	Shielded cask on trailor	Disposal in RCT or TH
Spent radiation sources (III)	Non-DAE users	Variable	Source in sealed capsules	Lead cask in vehicle	Disposal in TH (at BARC)

Abbreviations

ET - Earthen Trench
RCT - Reinforced Concrete Trench
TH - Tile Hole
NPP - Nuclear Power Plant
FRP - Fuel Reprocessing Plant
WMF - Waste Management Facility

transport in public domain of radioactive waste from nuclear fuel cycle operations has not arisen as yet. Transport is thus restricted to controlled areas within the nuclear sites. The provisions of AERB safety code on transport of radioactive materials do not apply to transport within the nuclear establishment. However, for the on-site transport of the waste, the packaging and transport specifications of the code are strictly followed to provide adequate containment to prevent dispersal of the waste and to keep the radiation exposure of site personnel and transport workers to as low as reasonably achievable. It is also ensured that adequate capability exists at site to handle an emergency situation arising from transport accidents. Thus, a competent infrastructure has been built to cater to the diverse waste management and transport requirements needed for nuclear fuel cycle facilities.

TABLE 2

Transport Cask Details at SWMF / Tarapur

Type of waste		Cask description	Shield thickness	Weight (tons)	Dimensions (mm)	Suitable upto (Gy/h)
1	Filter sludge from liquid filter (NPP)	Concrete cask	400mm concrete	7	1430 ϕ x 1850	0 6
2	NPP fuel pool components (fuel plugs, LPRM)	Lead cask vertical cyclinder	150mm lead	13 2 lid 2 5	1280 ϕ x 1650	20 to 200
3	Chemical sludge	Lead cask horizontal cylinder	100mm lead	17	1250 ϕ x 1740	2
4	Zircaloy Hull from FRP	Lead cask with Bottom entry and exit	200mm	18 5	750 ϕ x 4250	100
5	FRP off-gas filter	Lead cask cylindrical with separate top lid	35mm	7 2	2140 ϕ x 1290	0.1

SWMF - Solid Waste Management Facility

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REGULATIONS FOR THE SAFE TRANSPORT OF RADIOACTIVE MATERIALS IN THE PHILIPPINES

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Abstract

The Philippine Nuclear Research Institute (PNRI) is the government agency designated to be the Competent Authority responsible for regulation and control of the uses of radioactive material to protect the public and the environment from the hazards of ionizing radiation. To implement this responsibility, PNRI is authorized to issue licenses, promulgate rules and regulations, and enforce compliance through inspections and audits of licensed materials and facilities.

An important enforcement area is the safe transport of radioactive materials. This activity is performed in conjunction with other authorities in government involved in commercial transport. Radioactive material transport includes the transport of radioactive wastes from the licensee's facility to the waste management facility. PNRI manages and operates the only low-level radioactive waste management facility in the country. The facility is designed for interim storage of radioactive wastes. Stored wastes will be retrieved and transferred to a permanent disposal site when the government has determined and established a permanent repository. This permanent repository will be capable of handling radioactive wastes arising from the industrial and medical uses of radioisotopes, and from the nuclear fuel cycle when nuclear power is available.

PNRI regulations for safe transport of radioactive materials are according to IAEA guidelines. All radioactive wastes generated in the country come from users in industry, medicine and research. These are generally low-level hospital wastes and spent sealed sources. A pre-transport criteria issued by PNRI requires waste generators to segregate and package the wastes according to approved procedures before loading it for transport to the PNRI rad-waste facility. This process facilitates compliance with IAEA packaging requirements as well as the requirements of other agencies in government who are responsible for commercial transport.

PNRI regulations, including the regulations for transport of radioactive materials, are currently being reviewed for amendment and revision to consider lessons learned and current standards and practices.

THE REGULATORY AUTHORITY

The Philippine Nuclear Research Institute, hereinafter referred to as *PNRI*, is the government agency responsible for the licensing and regulation of the use of radioactive materials in the Philippines. In 1958, the **Philippine Science Act** was established by the Philippine Congress when it passed **Republic Act No. 2067.** This law created the **Philippine Atomic Energy Commission (PAEC)**, which was given the authority to promote the peaceful uses of atomic energy in the country. An amendment to this law in 1963 extended this authority to cover the licensing and regulation of radioactive materials. In 1968, the Philippine Congress passed and approved another law, **Republic Act No. 5207**, which was known as the **Atomic Energy Regulatory and Liability Act.** This new law defined the provisions for the licensing and regulation of atomic energy materials, to include nuclear fuel and radioisotopes and their by-products, and atomic energy facilities, to include nuclear research and power reactors. The responsibility to enforce this law was assigned to PAEC. In 1986, a Presidential decree re-organized the PAEC and re-named it PNRI. This re-organization retained with PNRI all the functions, duties and responsibilities of PAEC particularly the authority to license and regulate. The **Nuclear Regulations, Licensing and Safeguards Division, (NRLSD),** a unit under PNRI, direct the implementation of this regulatory and licensing authority. Approximately **235** radioactive material licenses are being administered by PNRI. **Table 1** shows the distribution of these licenses by type of use.

TABLE NO. 1 LICENSES BY TYPE OF USE

TYPE OF LICENSE	USE AND APPLICATION	NUMBER
Commercial	Import/sale/ distribution	28
Medical	Nuclear Medicine/ Radiopharm.	39
Medical	Teletherapy	10
Medical	Brachytherapy	6
Medical	Individual user	5
Research	Education	24
Industrial	Radiography	25
Industrial	Nuclear Gauges	98
	TOTAL	235

In 1976, the government started a nuclear power project aimed at operating the 620 Megawatt Bataan Nuclear Power Plant. During the construction of the plant, one of the important activities was the search for a final repository site for radioactive wastes arising from the nuclear fuel cycle and from other uses of radioactive materials. The Philippine Atomic Energy Commission, led a multi-agency group in this activity and conducted site selection studies and surveys to determine those potential disposal sites of radioactive wastes. A change in government in 1986 resulted in a decision to mothball the Bataan nuclear power project because of alleged charges that the plant, as constructed, was unsafe and riddled with defects. This decision also derailed all other activities related to the Bataan Nuclear Plant including the search for a final radioactive waste repository site. This issue about the nuclear plant has not yet been settled. The present government which was installed in 1992 has however directed the inclusion of the nuclear power option as an alternative energy source in future energy planning endeavors even as the Bataan Nuclear power plant issue remains unresolved. This directive also revived the radioactive waste repository studies, which PNRI is currently preparing for.

PNRI has a 3MW TRIGA research reactor and a 30,000 curie Co-60 source operated by a number of research units within its premises. The TRIGA reactor has been under repair and inoperable for a number of years.

PNRI manages the only radioactive waste conditioning and storage facility in the country. The facility is located inside the premises of PNRI within the heart of Quezon City the capital city of the Philippines. The **Radiation Protection Section**, (RPS), is the PNRI unit that operates and maintains this facility and conducts such services as waste management, radiation control and personnel monitoring.

TRANSPORT REGULATIONS AND CRITERIA

The national policy on the protection of the public against the hazards of radiation is adequately expressed in the law, Republic Act No 5207, which provides for the regulation and control of the uses of radioactive material. In Republic Act No 2067, PAEC promulgated the following regulations, CPR **Part 2, "Licensing of Radioactive Materials, CPR Part 3, "Standards for the Protection Against Radiation ", and CPR Part 4, "Regulations for the Safe Transport of Radioactive Materials in the Philippines".** The regulations for the transport of radioactive materials including radioactive wastes are according to the guidelines of **IAEA Safety Series Nos. 6 and 37**, while the regulations on radiation protection are according to **IAEA Safety Series No. 9.** Presently, CPR Part 4 is being revised and amended to address current practices and applicable developments in the transport industry and lessons learned from radioactive material transport experiences in the country. This process is expected to be completed by year's end. In Republic Act No 5207, the regulations promulgated were mainly those that were needed for the licensing of the Bataan Nuclear Power Plant.

Besides the requirements of CPR **Part 4**, PNRI issued the **Guidelines for the Acceptance of Low-Level Radioactive Wastes from Off-Site waste Generators.** These guidelines establish the criteria for the segregation, collection and packaging of radioactive waste before it is delivered to the waste treatment facility. The guidelines cover spent sealed sources, solid wastes and liquid wastes. It requires that the size and configuration of the waste packages for transport should be standardized and properly segregated at the place of origin to permit the appropriate application of waste processing methods and procedures at the facility. Compliance with these criteria will considerably extend the space available for the interim storage of conditioned wastes at the facility.

In response to international concern to protect the environment from uncontrolled dumping of radioactive wastes, the Philippine Congress passed a law, **Republic Act No. 6969, The Toxic Substances and Hazardous and Nuclear Wastes Control Act of 1990,** that will prohibit the entry, even in transit, of nuclear wastes and their disposal into Philippine territorial limits. This law refers only to nuclear wastes that are reportedly transported and disposed indiscriminately in oceans and seabeds, which could pose adverse contamination of Philippine waters or in areas within the country without the knowledge of the appropriate regulatory authorities.

Table 2 shows the list of Philippine laws and regulations that are relevant in safe transport of radioactive materials.

WASTE CONDITIONING AND STORAGE FACILITY

The **PNRI Radwaste Management Facility (RMF)** is located within the premises of PNRI. The facility was established in the early seventies. Conditioned wastes are being stored in trenches inside metal drums. A waste treatment section of the facility performs in-drum compaction of solid wastes and cementation of liquid wastes. PNRI receives and conditions these wastes for interim storage in the facility. When a final repository site has been located and established in the country, all the stored wastes in the facility will be retrieved and transferred to that site for disposal. The **RMF** is adequately monitored and controlled to prevent radioactive contamination to the surrounding urban areas.

TRANSPORT MECHANISM

Before radioactive waste is packaged for transfer to the facility, the waste generator or licensee is required to submit a written request addressed to the Director of PNRI. The request should provide all the information required in the waste acceptance criteria and should be submitted to PNRI for every request

TABLE 2. PHILIPPINE LAWS AND REGULATIONS ON SAFE TRANSPORT OF RADIOACTIVE MATERIALS

BASIC LAW	TITLE OF THE LAW	REGULATIONS/CRITERIA PROMULGATED	REGULATORY PROVISIONS
REPUBLIC ACT NO 2067 (1958)	--Phil Science Act of 1958, creating the Phil Atomic Energy Comm.	--Standards for the Protection Against Radiation	-safety standards for protection against ionizing radiation arising from any use of radioactive material, in accordance with IAEA safety Series No 9
		--Licensing of Radioactive Materials	-requirements for the licensing of radioactive materials and general requirements for handling and transport
		--Safe Transport of Radioactive Materials	-regulations in accordance with IAEA Safety Series No 6
		--Guidelines for the Acceptance of Low-level Waste from Off-site Waste Generators	-pre-transport requirements for segregation, collection and packaging of waste at origin
REPUBLIC ACT NO 5207 (1968)	-- Atomic Energy Regulatory and Liability Act of 1968, granting the Phil Atomic Energy Comm. the authority to license and regulate	--Licensing of Atomic Energy Facilities	-rules for construction and operation of nuclear power plants, including general requirements for transport of nuclear fuel and wastes
		--Physical Protection of Atomic Energy Facilities and Materials	-physical protection of of nuclear materials in-transit, shipping and transport
REPUBLIC ACT NO 6969 (1990)	--Toxic Substances and Hazardous and Nuclear Wastes Control Act of 1990	--Implementing Rules for the Control of Toxic Substances and Hazardous and Nuclear Wastes	-rules prohibiting the entry, even in transit, as well as the storage and disposal of unregulated nuclear waste into the country

from waste generators for radwaste management services. The licensee, as the waste generator, also acts as the collector and transporter of the waste to the facility. Before such waste is loaded on the transport vehicle, the licensee is required to secure a **Certificate of Transport (CT)** from the **NRLSD**, according to CPR **Part 4.** The CT indicates the **radiation levels of the waste** container and the approval of the container for transport to PNRI. The CT also shows the concurrence of the **Radiological Health and Safety Officer (RHSO)** of the licensee attesting to the radiation levels indicated in the package. The services of freight forwarders are often utilized when licensees do not have the necessary transport facilities.

PROBLEMS ENCOUNTERED

Being a country made up of several islands, direct transport of waste by land is not always possible. Domestic inter island commercial vessels are normally utilized where bulky or large packages are transported. These vessels do not always have trained people nor designated areas to handle and store the radioactive waste safely for transport to PNRI. Air and overland transports are still considered the most reliable means of conveyance.

Radioactive materials discovered in remote mountainous areas were sometimes reported to PNRI. These were apparently unregulated nuclear density and level gauges used in abandoned mines. The reported location of such materials would pose some security problems to PNRI personnel assigned to verify and retrieve the material such that transport measures could not be pushed through effectively.

During transport of radioactive waste to the PNRI RMF, these vehicles pass through heavy traffic and densely populated areas. The occurrence of vehicular accidents involving the waste carrier could not be discounted.

FUTURE ACTIONS

PNRI is in the process of amending its existing regulations on radiation protection and for the safe transport of radioactive materials, **CPR Part 3 and CPR Part 4**, respectively. In this activity, PNRI shall consider local conditions and current standards and practices in the transport industry, and lessons learned from implementing current regulations.

In compliance with the President's directive to consider a renewed nuclear power program, PNRI is currently pursuing studies to determine and identify sites that will serve as permanent and final repositories of radioactive wastes generated for radioactive materials users and nuclear power facilities that may be built in the future. The determination and establishment of the site will eventually transfer all radwaste management activities from the PNRI premises. This development augurs well to the development of an effective and comprehensive radioactive waste transport framework. PNRI is heading an inter-agency group composed of government agencies involved in energy, environmental protection and public information, that looks into this matter.

WASTE TRANSPORT AND HANDLING

(Session 4)

Chairman

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United Kingdom

THE PLANNED INTEGRATED TRANSPORT SYSTEM FOR THE DEEP REPOSITORY IN THE UNITED KINGDOM

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Abstract

UK Nirex Ltd is responsible for developing a deep repository for the safe disposal of intermediate and low level radioactive wastes (ILW and LLW), and is concentrating its investigations on Sellafield as a potential location. A key part of the repository development programme is a transport system to deliver packaged wastes from sites elsewhere in the UK. The transport system must be able to handle a range of standard waste packages, and all transport through the public domain must comply with the IAEA Transport Regulations. Two design concepts have been developed for re-usable shielded transport containers for ILW, which are predicted to withstand accidents at least as severe as the IAEA Type B test conditions. Assessment, testing and further development of both concepts continues, with a view to selecting one for quantity production. Nirex is working closely with various organisations to establish the optimum transport routes for a potential repository at Sellafield. The current policy is that rail transport shall be used wherever practical for the transport of waste to the repository, although some road transport may also be required; the company has assessed a range of options. A Probabilistic Safety Assessment of the proposed transport operations has predicted that the radiological risks are expected to be extremely low, reflecting the adequacy of the packaging concepts. In addition, Nirex has identified a suitable transport emergency plan to deal with any unforeseen events.

1.INTRODUCTION

UK Nirex Ltd (Nirex) was created by the major organisations in the UK nuclear industry, with Government agreement, and is responsible for developing a deep repository for the safe disposal of intermediate and low level radioactive wastes (ILW and LLW) arising in the UK.

Nirex is concentrating its investigations on Sellafield in West Cumbria as a potential location for the repository. Although a significant proportion of the ILW and LLW which is destined for the repository will arise at the adjacent BNFL Sellafield site, the remainder of the waste will arise from sites elsewhere across the UK (Figure 1). Therefore a key part of the Nirex repository programme is the development of a transport system to convey packaged wastes to the repository.

Transport has a considerable influence on many other aspects of the repository development programme, including the specification of waste packages and the design of the repository itself. Transport has also been important in repository site selection. Beyond the

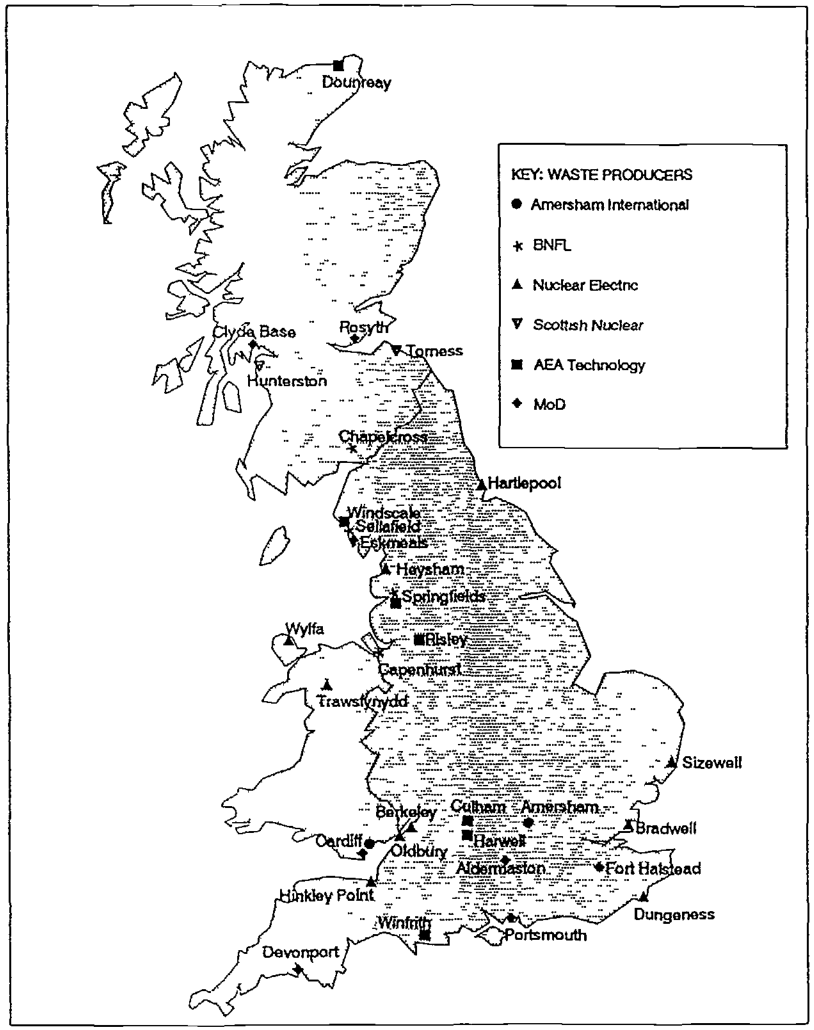

Figure 1

Waste producing sites in the UK.

immediate localities of the waste producing sites and the repository, transport of wastes will be that activity associated with disposal of waste at the repository which most obviously affects the public. Therefore it is essential to develop a transport system in such a way that it is safe and acceptable, and to demonstrate this.

The transport of construction materials is already taking place as part of the site investigation programme, and transport of wastes to the repository is expected to take place in the first half of the next century. Hence the transport system currently being planned must be sufficiently flexible to deal with changing circumstances over the next 60 years or more.

This paper discusses:

- The requirements for a transport system (Section 2)
- Their effects upon the design of waste packages and transport containers (Sections 3 and 4)
- The proposed waste transport network linking other UK sites to Sellafield (Section 5)
- Transport safety and emergency plans (Section 6)

The summary and conclusions are given in Section 7.

2. REQUIREMENTS FOR A TRANSPORT SYSTEM

The transport system must be able to handle the range of standard waste packages which Nirex is developing in conjunction with the waste producers [1]. In addition, transport through the public domain must take place according to the Transport Regulations laid down by the International Atomic Energy Agency (IAEA) [2]. Some of the standard waste packages - particularly those intended for LLW - can be designed to conform to the IAEA Transport Regulations as well as to all the other relevant requirements, and may thus be transported without any additional packaging. However, most of the waste packages for ILW require additional radiation shielding, containment and/or mechanical protection for the purposes of transport. Nirex is therefore developing a range of re-usable shielded transport containers (RSTCs) for packaged ILW.

In addition to the IAEA Transport Regulations, the transport of radioactive wastes must conform with general UK legislation and regulations governing rail and road transport. This involves considerations such as the size limitations imposed by the railway loading gauges, and road traffic legislation governing the laden weights of vehicles.

The current policy of Nirex is that rail transport will be used wherever practicable for transport of waste to the repository. However, the integrated transport system being developed now must have sufficient flexibility to allow the use of whatever transport modes are the most appropriate during the operational lifetime of the repository. To this end, detailed assessments have been carried out for both rail and road transport. (Sea transport could be used for part of the routes from some sites, but is not currently favoured.)

The following transport assessments have been carried out:

- Route selection, logistics and economics
- Operational arrangements
- Need for new or improved transport links to the repository
- Environmental impact of construction of transport facilities
- Environmental impact of operation of transport system
- Safety and emergency arrangements.

3. NIREX STANDARD TRANSPORT PACKAGES

In order to achieve the necessary throughput of waste into the repository, it is essential to standardise on a limited range of waste packages. On the other hand, the available range of containers must be suitable for packaging the wide variety of wastes arising. To meet both of these needs, Nirex has developed standard containers for ILW and LLW in cooperation with the waste producers. Each standard container is justified by an identified need to package particular types of waste.

There are important distinctions between a waste container, a waste package, a transport container and a transport package. The waste is placed inside a **waste container** to form a **waste package** suitable for disposal. The term **transport package** is defined by the IAEA Transport Regulations [2] as the complete assembly of the radioactive material plus its outer packaging, and it is the transport package as a whole that must comply with the regulations. When appropriately filled, some waste packages will also qualify as transport packages in their own right, because they provide sufficient radiation shielding (if needed), robustness and integrity of containment for their particular radioactive contents. But other combinations of waste and waste container will require an additional **transport container** in order to form a suitable transport package.

Table 1 summarises the range of Nirex standard transport packages and containers. The 2m and 4m boxes [1] are based on the outlines of ISO-standard commercial transport containers and are categorised by the IAEA Transport Regulations as 'industrial packages, Type 2' (IP-2). The re-usable shielded transport containers (RSTCs) are designed to accept packaged ILW, and together with their contents will form IAEA 'Type B' packages. RSTCs will be manufactured in a range of nominal shielding thicknesses, so that wastes can be transported in a cost-effective manner using a near-optimum thickness of shielding.

Currently two different RSTC design concepts designated 'L' and 'N' (Figures 2, 3, 4 and 5) are being developed in parallel, with a view to selecting one design for final testing and manufacture. The following section described the development and testing of these RSTC design concepts.

4. RE-USABLE SHIELDED TRANSPORT CONTAINERS

4.1 Design Considerations

Each RSTC will carry four 500 litre drums of immobilised ILW, or alternatively a single $3m^3$ drum or box which occupies the same space. The transport containers are therefore approximately cuboidal in shape. Although the family of RSTCs comprises four

Table 1

Summary of Nirex Standard Transport Packages

Package	IAEA Type	Construction	External Dimensions (mm)	Empty Weight (t)	Max Payload (t)	Max Gross Weight (t)	Container: disposable or re-usable
ILW (70 mm) Concept L	B	Steel	2230x2230 x1989	16	12	28	Re-usable
ILW (146mm) Concept L	B	Steel	2480x2480 x2234	26	12	38	Re-usable
ILW (210mm) COncept L	B	Steel	2410x2410 x2191	35	12	47	Re-usable
ILW (285mm) Concept L	B	Steel	2530x2530 x2341	48	12	60	Re-usable
ILW (70mm) Concept N	B	Steel	2530x2530 x1910	15	12	27	Re-usable
ILW (145mm) Concept N	B	Steel	2530x2530 x2157	26	12	38	Re-usable
ILW (210mm) COncept N	B	Steel	2404x2404 x2005	36	12	48	Re-usable
ILW (285mm) Concept N	B	Steel	2530x2530 x2205	50	12	62	Re-usable
ILW 4m Box	IP-2	Reinforced Concrete	4013x2438 x2200	24	26[1]	50[1]	Disposable
LLW 4m Box	IP-2	Steel or reinforced concrete	4013x2438 x2200	3	23	26	Disposable
LLW 2m Box	IP-2	Steel or reinforced concrete	1969x2438 x2200	2	24	26	Disposable

Notes All dimensions and weights are nominal

[1] It may be possible to increase the maximum payload and gross weight of this package to 41t and 65t respectively

different wall thicknesses, all containers will have the same internal dimensions. The containers will weigh from around 15t for the thinnest-walled up to about 50t for the thickest. The additional weight of the contents will range from 5t to 10t, depending on the nature of the waste.

Compliance with IAEA standards for Type B packages means, among other things, that the total transport package must be designed to withstand normal transport conditions and minor mishaps, and also to withstand transport accident conditions including impact, fire and water immersion, while sustaining no significant loss of either shielding or containment.

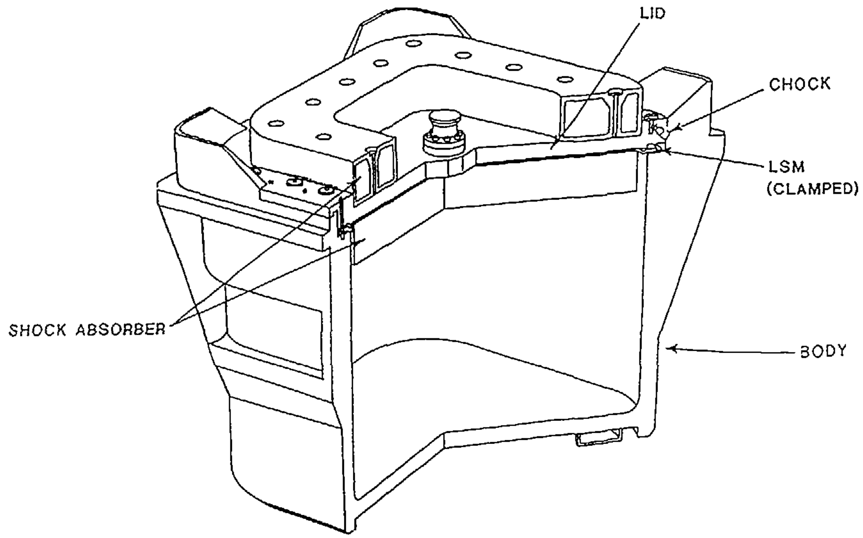

Figure 2

Concept L re-usable shielded transport container - 70 mm Wall thickness.

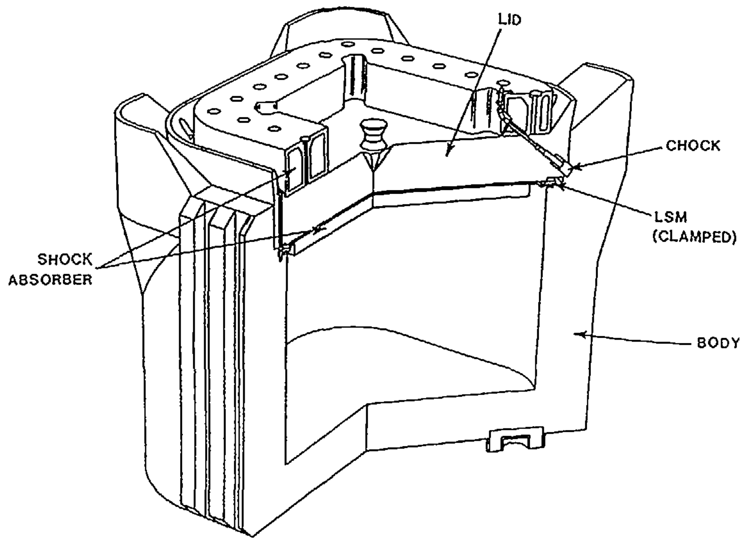

Figure 3

Concept L re-usable shielded transport container - 285 mm Wall thickness.

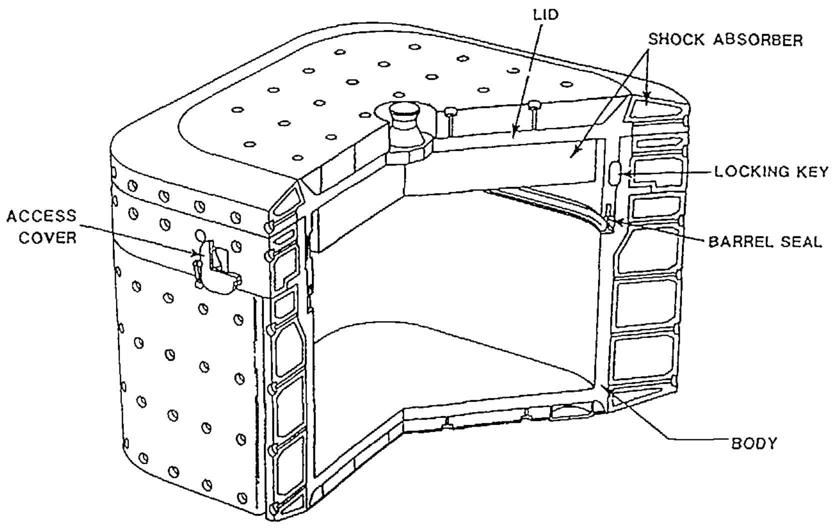

Figure 4

Concept N re-usable shielded transport container - 70 mm Wall thickness.

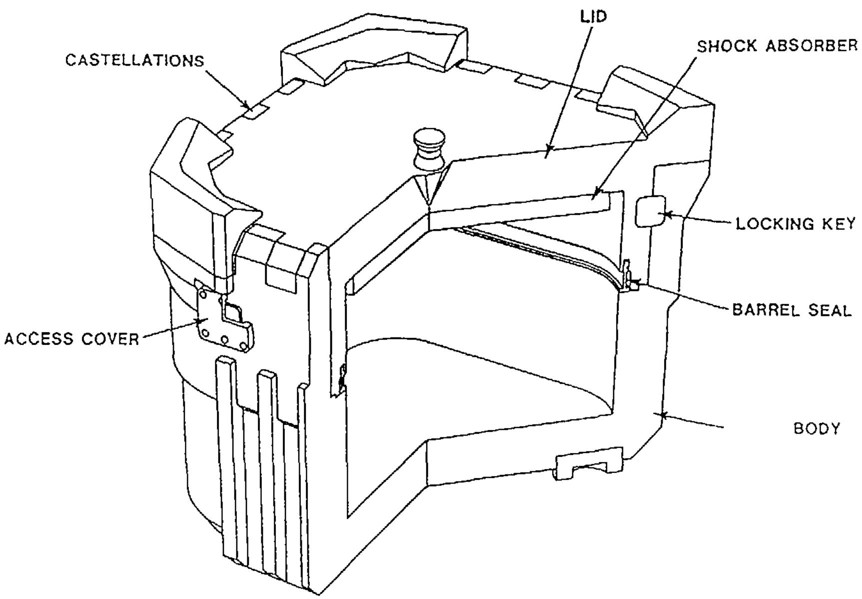

Figure 5

Concept N re-usable shielded transport container - 285 mm Wall thickness.

The design of the RSTCs is based upon extensive UK experience with the transport of irradiated nuclear fuel from gas-cooled nuclear power stations. The flasks used for this purpose are of the same cuboidal shape as envisaged for the RSTCs. However, the design requirements for the Nirex transport containers are different from those for fuel flasks, in two important respects:

- The mass of the contents is much greater than in the irradiated fuel flasks. This increases the stresses in any impact accident.
- The throughput of RSTCs at the repository will be greater than that of fuel flasks at a reprocessing plant, and the operations will be carried out without water shielding. Thus all processes such as the unfastening and removal of lids must be quick and simple, and preferably capable of remote operation.

The impact and fire accident scenarios have had a marked influence on the container designs. The main impact requirement is to maintain shielding and containment following a 9m drop test on to an unyielding target in any impact attitude [2]. The main design requirement associated with the 800°C, 30 minutes fire test [2] is to protect the O-ring seals, because the elastomeric material would degrade if the temperature exceeds certain limits for a significant length of time.

4.2 Design Concepts

The presently envisaged range of nominal wall thicknesses for RSTCs is 70mm, 145mm, 210mm and 285mm. These thicknesses are based on steel construction, and have been derived from consideration of the types and quantities of the conditioned waste streams requiring transport to the repository [3]. The descriptions which follow are specific to the 70mm and 285mm variants of each design concept; details of the 145mm and 210mm variants may be inferred by interpolation.

Concept L

The Concept L containers are shown in Figure 2 and 3. The design is similar to the existing irradiated fuel flasks, with a conventional top lid which is recessed into the body for added strength. The main impact protection is provided by massive steel shock absorbers at the top corners of the body. In the 70mm version, a separate wooden shock absorber is bolted to the outer surface of the lid to give additional protection against a direct top-down impact.

The lid is connected to the body by 24 radial chocks which engage in a continuous V-shaped groove around the top of the container body. Sealing is achieved by a separate Lid Sealing Member (LSM), a semi-flexible diaphragm which carries two concentric O-rings on the underside of its rim. Once the lid has been placed in position, the O-rings are clamped against the mating surface on the container body. The LSM is attached to the lid for ease in routine handling, but the attachments are designed to break away under impact leaving the LSM free to move and flex independently. Inside the container, a layer of crushable aluminium honeycomb is provided underneath the LSM to help absorb the kinetic energy of the contents in top-down impacts.

The thermal behaviour of the container is controlled by an intumescent coating over the majority of the surface. In the event of a fire, this coating is formulated to char and swell to several times its original thickness, providing a very efficient layer of thermal insulation. The lids of the 70mm and 145mm containers are not coated because the wooden shock absorber will itself provide a good thermal barrier.

Concept N

The Concept N containers are shown in Figures 4 and 5. It differs from Concept L by using a lid with a deep skirt which inserts inside the container body. The seal is well protected inside a vertical groove running around the bottom of the skirt. The two O-rings are attached to a rail on the container body, and when the lid in lowered into place the rail inserts into the groove in the skirt to form a seal. This 'barrel seal' requires no additional clamping and designed to be tolerant of vertical and lateral relative movements between the lid and the body. The lid is secured to the body by four large corner chocks at mid-height.

Impact and thermal protection in the 70mm and 145mm variants of Concept N is provided by bolted-on wooden shock absorbers, but this is not possible for the 210mm and 285mm thicknesses because of overall size constraints. These heavier containers rely on similar integral shock absorbers to those in Concept L, with an intumescent coating for thermal protection.

4.3 Assessment and Testing

The impact and the fire performance of the Concept L and Concept N containers have been assessed using finite element methods of calculation. These assessments indicate that either design concept will pass the regulatory impact and fire tests for Type B packages.

One third scale models of the 70mm and 285 mm variants of the Concept L and N containers have been manufactured and are currently undergoing a programme of drop testing to determine the performance of the containers during the regulatory impact tests. To date, both concepts have demonstrated an acceptable performance during drop tests.

The seal arrangements for Concept L and Concept N have also undergone further testing and development [4], particularly of their capability to remain leak-tight under mechanical deflections that might be produced by a severe impact (by comparison, the distortions induced by a fire accident would be much smaller). Experimental rigs were developed to simulate the Concept L and Concept N double O-ring seals, and to provide controlled radial and axial deflections from the normal geometry. Promising elastomer materials were identified and leakage tests were carried out at temperatures ranging from -40°C to +200°C, in some cases using O-rings which had been irradiated to simulate the dose that might be received during their operational lifetime. One material was identified which performed adequately across the entire temperature range. The indications were that the Concept L seal would be easier to manufacture and assemble in operational use than Concept N.

4.4 Material of Manufacture

Nirex is examining the feasibility of manufacturing these containers by means of casting instead of the more usual forging process, as this would bring advantages of lower cost and shorter manufacturing time. A programme of work has been carried out [5] to generate data to enable a decision to be made as to whether cast steel or ductile cast iron (DCI) could be used as the material of construction of the transport containers and to enable a choice of preferred material to be made for a subsequent programme of full-scale analysis and testing. Castings have been producied in both cast steel and DCI and subjected to a test programme involving non-destructive testing to locate and characterise flaws, mechancial property tests, including dynamic fracture toughness tests, welding and cladding tests and drop tests. Assessments of the results from this work is continuing.

5. THE TRANSPORT NETWORK

This section describes the proposed transport network linking UK waste producing sites (Figure 1) to Sellafield.

5.1 Possible Modes of Waste Transport

Radioactive waste could be transported from the waste producing sites to the repository in one of three ways:

- By rail from a siding located within the site boundary
- By road to an off-site railhead, and thence by rail
- By road, direct to the repository.

The capacity of most existing on-site sidings or off-site railheads serving waste producing sites is limited, so it may be appropriate to assemble the wagons into longer trains at various marshalling points before their onward journey to the repository. Extended or completely new sidings or railhead facilities may be required for some sites.

The maximum gross weights of road vehicles are relevant in considering the most appropriate means of transporting the waste. From 1999 onwards the maximum laden weight of a conventional 5-axle articulated Heavy Goods Vehicle (HGV) will be 40t. Taking into account the typical unladen weights of HGVs, the maximum weight of a package that can be transported within the 40t limit will be 25-27t, though the discussion which follows is not particularly sensitive to the exact value.

Packages too heavy to be carried on the UK public highways by ordinary HGVs will have to be transported as 'Abnormal Loads' on special vehicles. Within the proposed transport system, the movement of Abnormal Loads on the public highways is likely to be limited mainly to the short journeys to off-site railheads. At most of the sites involved, this would be a continuation of well-established transport practices for fuel flasks.

For assessment purposes, two particular scenarios have been considered:

- Rail-only: all packages would be transported by rail
- Road/rail: all packages which can be transported on conventional HGVs to be carried by road; all other packages (or where road transport is inappropriate) to be transported by rail.

5.2 Numbers of Packages

When appropriately filled with waste, the Nirex standard packages described in Section 3 can be divided into three groups according to their weight and radioactive contents: 'heavy ILW', 'light ILW' and 'light LLW'.

A 'light' transport package can be defined for these purposes as a load capable of being carried by road on a standard HGV, ie. its weight will have to be less than 25-27t. Any package of greater weight is classified as 'heavy' and would qualify as an Abnormal Load for road transport.

The waste packages will orginate at several different locations within the UK as shown in Figure 1. It is not possible at present to precisely define exactly what waste will be disposed of in the repository or when it will be despatched. Utilising the information presented in 1991 UK Radioactive Waste Inventory [4] and taking a total repository waste capacity of between 400,000 m^3 (made up of 300,000 m^3 of ILW and 100,000 m^3 of LLW) and two million m^3 (made up of 600,000 m^3 of ILW and 1.4 million m^3 of LLW), it is estimated that if all packages arrive at the repository by rail (the rail-only scenario defined above) there would be an average of between 100 and 300 trains per year, depending upon the train length and total volume to be disposed. Alternatively, if 'heavy' packages were transported by rail and all 'light' packages by road (the road/rail scenario) this would give an annual average of between 50 and 200 trains plus up to 2000 HGVs, ie. about 40 HGVs per week.

5.3 Potential Transport Routes

Nirex is consulting with British Rail, the UK Department of Transport and the local government authority (Cumbria Country Council) to establish the optimum transport routes for a potential repository at Sellafield.

Rail routes

The waste producing sites are widely dispersed throughout the UK (Figure 1) and in order to operate rail transport in the most efficient manner it is likely that marshalling points would be used. At these locations, short trains from individual sites would be combined into full trains for the onward journey to the repository. Similar arrangements already exist for the movement of fuel flasks, mostly from the same sites.

At Sellafield, a new rail spur to the repository from the Cumbrian Coast line is envisaged. A link to the BNFL internal rail system will be used for on-site transfers of waste packages.

Road routes

The routes to be used for the road transport of waste comprise the following elements:

- Route from the site to the motorway network
- Route via the motorway network to the repository

or alternatively -

- Route from the site to its off-site railhead.

The actual routes chosen from each site will take account of road conditions when the repository becomes operational. Closer to the repository, the existing traffic routes and their capacities have been extensively surveyed; the additional volume of traffic due to the repository will not be sufficient to require any substantial alterations to the existing road infrastructure. At the repository itself, suitable access roads will be constructed where necessary.

5.4 Transport Operations

Precise details of the management of the overall transport operation will be determined nearer to the opening of the repository. The prime function of the transport system management will be to liaise with the repository operations staff and the waste producers, to ensure the required flow of waste to the repository.

All activities relevant to the transport of radioactive waste to the repository will be conducted in accordance with a quality management system, which will include appropriate QA programmes. Each waste producer will be required to operate a QA programme which interfaces with the QA programme of Nirex.

6. TRANSPORT SAFETY

Nirex intends to ensure high safety standards in all transport operations associated with the deep repository.

6.1 Routine Operational Safety

Nirex intends to manage the non-radiological aspects of transport safety in the same way as they are ordinarily managed for freight transport by road and rail. Particular attention will be paid to routine operational safety by implementing procedures to ensure high levels of vehicle maintenance, worker training and Quality Assurance in all aspects of transport.

All transport operations must comply with the IAEA Transport Regulations [2] and the relevant legislation concerning the radiological aspects of transport. A number of procedures will be required to ensure that during transport of radioactive wastes the radiological risks to the general public and transport workers are as low as reasonably practicable. These measures will include monitoring of transport packages, leak testing and restrictions on access to working areas, labelling, identification arrangements etc.

6.2 Risk Assessments

In addition to meeting all regulatory requirements, Nirex wishes to be assured that the risks associated with transport will be very low, that the risk targets in the Nirex Radiological Protection Policy [6] will be achieved, and that everything reasonably practicable will be done to minimise risk. Therefore a Probabilistic Safety Assessment for transport associated with a repository at Sellafield has been carried out, which considers both radiological and non- radiological aspects of transport; the results of this are presented in [7].

These show that the radiological risks associated with the proposed waste transport system are expected to be extremely low, reflecting the adequacy of the packaging concepts.

6.3 Emergency Arrangements

In addition to the protection provided by the packaging requirements of the IAEA Transport Regulations [2], the Regulations also call for appropriate emergency arrangements in the event of an accident involving the transport of radioactive material.

Following a detailed appraisal of existing schemes, it was concluded that the most favourable option would be to make use of the existing UK Nuclear Industry Road Emergency Response Plan (NIREP) [8] with an extension to cover rail transport.

7. SUMMARY AND CONCLUSIONS

1. Nirex is concentrating its investigations on Sellafield as a potential location for a deep repository. A key part of the company's repository development programme is a transport system to convey packaged wastes to the repository.

2. The transport system must be able to handle the range of standard waste packages which Nirex is developing. In addition, transport through the public domain must take place according to the IAEA Transport Regulations and must comply with other UK legislation.

3. Two design concepts have been developed for re-usable shielded transport containers for ILW, which are predicted to withstand accidents at least as severe as the IAEA Type B test conditions: a 9m drop on to an unyielding target in any impact attitude, followed by an 800°C, 30 minute engulfing fire test. Assessment, testing and further development of both concepts continues, with a view to selecting one for final qualification testing and quantity production.

5. Nirex policy is that rail transport shall be used wherever practical for the transport of waste to the repository, although at least an element of road transport may also be required; the company has assessed a range of options including the transport of all light packages by road.

6. Nirex is consulting with British Rail, the Department of Transport and the local government authority to establish the optimum transport routes for a potential repository at Sellafield.

7. Nirex intends to ensure high safety standards in all transport operations associated with the deep repository, and has commissioned a Probabilistic Safety Assessment for transport associated with a repository at Sellafield. The radiological risks are expected to be extremely low, reflecting the adequacy of the packaging concepts.

8. In addition, a preferred option for a suitable transport emergency plan to deal with any unforeseen events has been identified.

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GNS - TRANSPORT SYSTEM AND HANDLING PRACTICES FOR THE TRANSPORT OF LOW AND INTERMEDIATE LEVEL WASTES

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Abstract

During the past years GNS has developed and installed several new transport systems for the transport of low and intermediate level waste. There are:

1. "open-all" containers for transportation of drums, containers and Type B (U) casks

2. 20' container qualified as IP II, for the transport of scrap and miscellaneous waste

3. tank containers for the transport of liquids

These containers are already being used internationally and have proven functionability. This transport system and the handling practices are described in detail in the report.

1. Introduction

The company GNS is a subsidiary of the German electric utilities. Besides activities in the field of high level waste, GNS is also responsible for the conditioning of radioactive waste, interim storage of this waste and of its transport to interim storage. The necessary transports are performed by German Federal Railways (Deutsche Bundesbahn).

In addition to the equipment for the conditioning of waste, which are mostly mobile facilities and allowing direct conditioning work in nuclear power plants, GNS has developed transport systems in recent years which were tested and are in general use. The main systems which now are approved by the competent authorities and which manage the whole transport volume, are presented below:

2. Transport Systems

For the realization of the transports, GNS decided to carry them out by means of standardized packages used nationally and internationally as far as possible. The 20' container had been chosen as transport facility for which the required transport auxiliaries and equipment are available by road, railway and ship.

The 3 main systems are:

- "open-all" - container for transportation of drums, "KONRAD" container, and MOSAIK-casks (Type B (U) - casks)
- containers qualified as IP II for the transportation of scrap and miscellaneous waste
- tank container for the transport of liquids

2.1 "Open-all" - container

The "open-all" - container is a combination between a platform and a 20' standard container. The container had been developed as overpack for nuclear transports and has the following basic characteristics:

On a stable base frame, the hood consisting of side walls and a roof can be moved. For this the two front supporting pillars, which are designed as swivel pillars can be turned down. Through folding the pillars, the bar between platform and cover is released and the hood is lifted up out of a labyrinth seal.

The folded swivel pillars serve as supporting arm and handling of the hood while moving the hood. The opening and closing of the cover can be done by one person within 30 sec.. The container is provided with an easily decontaminable paint.

The main data of the "open-all" - container are:

- outer dimensions:	length 6058 mm
	width 2438 mm
	height 2591 mm

- inner dimensions:	length 5880 mm width 2270 mm height 2220 mm
- tare weight:	approx. 3600 kg
- max. permissible payload:	24000 kg

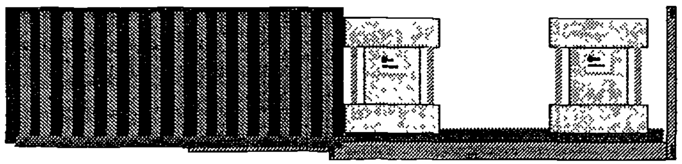
For the transport of different packages, various adapter frames can be fixed on the vase plate and secured by means of a high-speed locking system.

Besides fixing the load, the adapters, also partially take over load securing and accident control. In this way, for instance the shock absorbers, which are necessary for a Type B (U) transport of MOSAIK-casks, are integrated into the adapter frame.

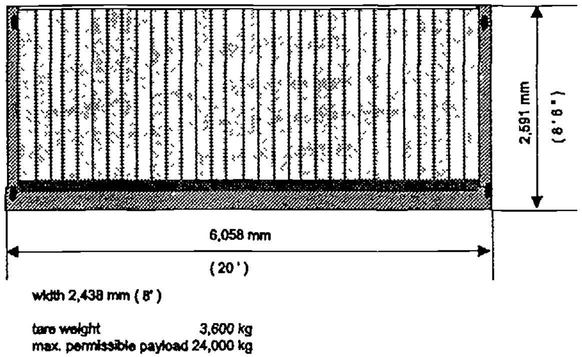
2.1.1 Permissible content

The "open-all" - container is licensed for commercial traffic and serves as overpack for single shipping goods during the performance of radioactive transports according to the regulations for dangerous goods.

- 200-l-drums by using special palettes
- MOSAIK-casks as Type A package
- MOSAIK-casks as Type B (U) package
- KONRAD-container Type 1 to Type 6



Opened "OPEN -ALL CONTAINER" with MOSAIK II B (U) and shock absorbers



Closed "OPEN-ALL CONTAINER"

2.2 IP-II-Container

The container is an all-round closed 20'-ISO- piece good container in an all-steel design with a double door at the front wall. The floor of the container is tub-shaped formed and made of stainless steel. It goes up to a height of approx. 380 mm on the side walls and front wall and forms a threshold at the door.

The transition to the walls is designed free of joints. At each side wall in the container there are 24 fixing rings in distances of approx. 400 mm melted with the frame work and fixed in two lines in heights of 425 mm and 1480 mm above the floor for load securing by means of fixing straps. There are further 4 lash eyes arranged vertically at the side walls near to the door for fixing the door net which secures contents in direction of the door not able to be strapped.

The container has a decontaminable paint inside and outside with exception of the stainless steel tub.

The main data of the IP-II-Containers are:

- outer dimensions:	length 6058 mm width 2438 mm height 2591 mm

- inner: length 5850 mm
  width 2230 mm
  height 2340 mm
- tare weight: 5460 kg

- max. permissible payload: 24000 kg

The IP-II-container will be used
- as box container or
- as open-top container

Besides the double door the whole roof can be taken off from the open-top container is for easy loading and reloading by means of a crane.

Roof frame and door wings are equipped with a surrounding rubber seal. For securing non-strapping contents towards the container roof, a roof tarpaulin with integrated straps is intended for this purpose which is fixed at the 12 fixing straps of the upper line of both side walls.

2.2.1 Permissible Content

- Materials with lower specific activity (LSA II and LSA III) according to German regulations No. 700 of the appendix to the decree for dangerous goods (railway) and No. 2700 of the appendix A to the decree for dangerous goods (road) in solid form without additional dangerous characteristics.

- for LSA II / LSA III materials for which IP-II-packages are needed at GNS include:

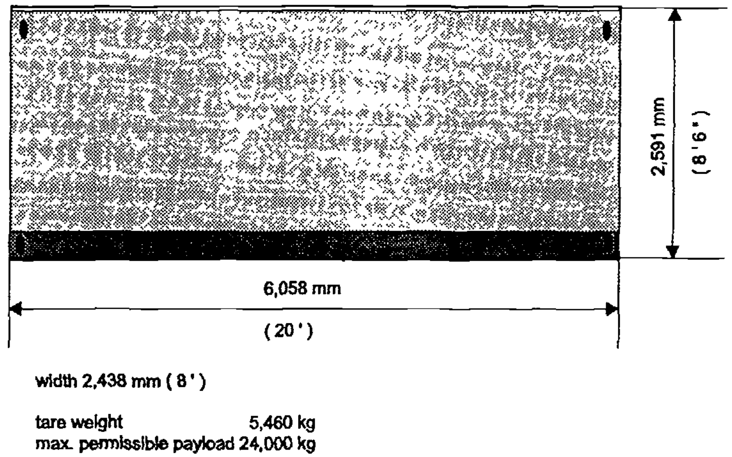
- compactable mixed waste
- incinerable mixed waste
- compacted mixed waste (pellets)
- rubble
- ashes/cinders
- active carbon

For these kinds of waste, the primary packages are mostly plastic bags, pre-compacted balls, 180-l-drums, etc.

- Surface contaminated objects according to the regulations No. 700 of the appendix to the decree for dangerous goods railway and No. 2700 of the appendix A to the decree for dangerous goods without additional dangerous characteristics.

In practice the following goods are transported:

- contaminated scrap and rubble
- contaminated power plants parts e. g. motors, control panels etc.
- contaminated tools e. g. crane traverses, towropes etc.
- conditioning facilities e. g. super compaction facilities, resins filling facilities, drying facilities



"IP-2-CONTAINER"

2.3 Tank container

For the execution of transports with liquid radioactive materials GNS has also developed containers which can safe transports of liquids.

To guarantee a smooth transport as far as possible i. e. a combined transport in which a transfer from road onto rails takes place or inversly, the form of a tank container was chosen for transport of liquids also.

The tank consists of an inner cask with a double jacket and leakage monitoring. This inner cask is surrounded by an outer cask. The space between inner and outer cask is filled with lead. The wall construction has a capacity of 5 m³ at 93 % loading.

The tank vehicle is equipped with filling and emptying equipment, of which the pipes lead out of the tank above and are conducted into a separate instrument box. Inside the tank a mixer for stirring of the loading is installed.

On the front side, a control panel is fitted for the connection of the power supply and for the control of the mixer. The control panel also contains the indicators for the filling level measuring probe and overfilling safety device. Further main data of the tank container with 5 m³ tank volume are:

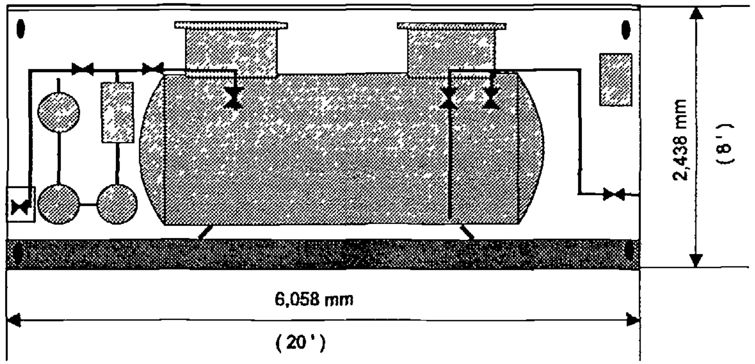
- dimensions:	length 6058 mm
	width 2500 mm
	height 2438 mm
- net weight:	23900 kg
- max. permissible total weight:	30480 kg

2.3.1 Permissible content

The tank containers used by GNS are approved for transports of watery solutions and evaporator concentrates with the typical compositions of the concentrates which occur in pressurized water reactors and boiling water reactors.

The transport of decont waters and corrosive solutions is also approved. The specific activities of the transported liquids can reach values of approx. 2x 10^{11} Bq / Co - 60 for the 5 m³ tank container presented here which stick to the permissible dose rates for the transports.

According to the same construction principle, however with small shielding, GNS has further containers which are able to hold up to 10 m³.



width 2,500 mm (8 ' 2.4 ")

tare weight 24,900 kg
max. permissible payload 5,580 kg

"TANK-CONTAINER TC 5"

3. Execution of the transport

A short overview about the described available transport facilities and their application in the year 1993 is shown below.

3.1 Available transport equipment

At the end of 1993 GNS has the following transport equipment:

- 20' containers IP-2 / IP-3 171 pieces

The still available 20' containers without IP 2/3 license are continuously exchanged for qualified ones.

- 20' "open-all" - containers 47 pieces

- 20' tank containers 3 pieces

3.2 Transports carried out in 1993

The following transports with different materials were executed in 1993 by using the mentioned containers.

Executed transports
01.01.1993 - 31.12.1993

	Number of Transports	Number of Container
Solid radioactive waste	227	393
Contaminated scrap	56	75
Liquid radioactive waste	63	54
Contaminated facilities / machines	87	105
Empty packages contaminated on the inside	95	88
Total	1.263	715

4. Conclusion

The modernization and rearrangement of the available transport systems at GNS over the last 6 years turned out to be the right way.

With the "open-all"-container, a versatile transport and handling equipment had been put into operation, which has become the required standard for its users as it makes possible an easy loading and reloading.

The use of approved IP-II-container especially for transports of non-conditioned waste, brings additional safety through the load-securing equipment inside the container. Safety has also been increased for transports of non-homogeneous radioactive materials.

By using containers for liquid transports, the transport of liquid waste to external conditioning facilities could be ensured. This is an important aspect for the overall waste management of German nuclear power plants. Through the design of transport equipment with internationally standardized outer dimensions a simplification of the transfer as well as rapid handling by means of uniform tools had been achieved. This makes it alsopossible to reduce of the radiation exposure of the service staff in the end.

LLW TRANSPORT BY IP-2 PACKAGING

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Abstract

As the Japanese nuclear power plants are located on the sea coast, optimal system of the LLW transport consists of sea and land modes. A special ship "Seiei Maru" was built to transport the LLW from nuclear power palnts to the LLW Burial Center in Rokkasho–mura, Aomori Prefecture and dedicated trucks were prepared to transfer the LLW from the receiving wharf to the Burial Center. Containers were developed to efficiently transport LLW drums and were designed and tested to meet the IP–2 packaging requirements. 3,000 units of such containers have been used since 1992 and the safe transport of LLW has been demonstrated by means of the IP–2 packagings.

1. Introduction

In Japan 46 nuclear power plants are in operation at 17 sites and about 30 per cent of electric power is supplied by these plants.

Low level wastes (LLW) generating at the nuclear power plants are packaged in 200 litre drums and temporarily stored in the on–site storehouse. The number of these drums accumulated so far amounts to approximately 480 thousands.

In December 1992 the LLW Burial Center was established at Rokkasho–mura, Aomori Prefecture, as one of the nuclear fuel cycle facilities to receive the LLW drums for shallow land disposal.

The capacity of the LLW Burial Center is 200,000 drums for the first stage ofconstruction and the object of burial is at present low level wastes which have been solidified homogeneously by cement and asphalt.

The transport of these wastes from power plants to the LLW Burial Center is performed by the Nuclear Fuel Transport Co., Ltd. (NFT), and the mode of the transport is a combination of sea and land transports.

The packagings used are special enclosed containers which are able to contain 8 drums. They were developed by NFT and were confirmed to meet all the requirements for the IP–2 packaging by various tests which had been performed during the R&D stage.

The sea transport is serviced by a special dedicated ship which is able to carry 384 packages per voyage and the land transport is performed by trucks which carry 2 packages per truck.

More than one year has already passed since the first shipment was made and safe and trouble free operation has been performed.

2. Outline of the transport system

The LLW drums stored at the power plants are contained in the IP–2 packagings after they are confirmed to meet the burial requirements by Wastes Confirmation Inspection. Then they are transferred by trucks to the dedicated port or near–by commercial port of the nuclear power plant.

All the Japanese nuclear power plants are located on the seashore and mostly provided with their own ports for 3,000–ton–class ships. Therefore sea transport is convenient for the LLW shipment in Japan (Fig.1, Fig.2).

The special dedicated ship "Seiei Maru", which was constructed to transport LLW has the dead weight of approximately 3,000 tons and is provided with concrete shield against radiation. In 7 cargo holds of the ship the cell guides are installed to meet the size of the packages so that they are surely stowed in the cargo holds (Fig.3).

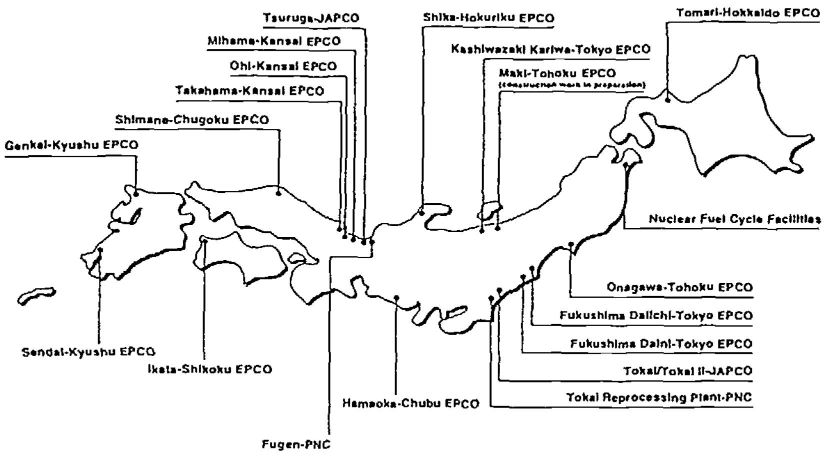

Fig.1 Location of Nuclear Power Plants in Japan

The Seiei Maru is installed with the on–board bridge crane which is used for loading and unloading of packages at the ports of nuclear power plants.

The Seiei Maru leaving the port of a power plant arrives in 2 to 5 days at the Mutu–Ogawara Port (M.O. Port) which is located at Rokkasho–mura, Aomori Prefecture.

The M.O. Port is a public port under the control of Aomori Prefecture. An NFT' s bridge crane of 25–ton lifting capacity is installed at the wharf of the port (Fig.4) In order to open the wharf for the public use the crane is moved to the parking lot on a curved track when it is not used for LLW operations.

The wharf crane as well as on–board crane has a remote and semi–automatic control system to reduce the radiation exposure of the operators.

The packages in the ship holds are lifted up by the wharf crane, two at a time, and loaded to the exclusive use trucks which are standing by at the wharf.

The dedicated trucks are standard 11–ton trucks of which beds are partially remodelled and the tie–down and the releasing operations of the packages can possibly be done remotely at the driver' s seat (Fig.5).

The distance of the land transport is about 9 km, most of which is on–site road of the Japan Nuclear Fuel Limited (JNFL) who owns the LLW Burial Center. However, as the trucks should partly cross the public road, it is necessary before shipment to

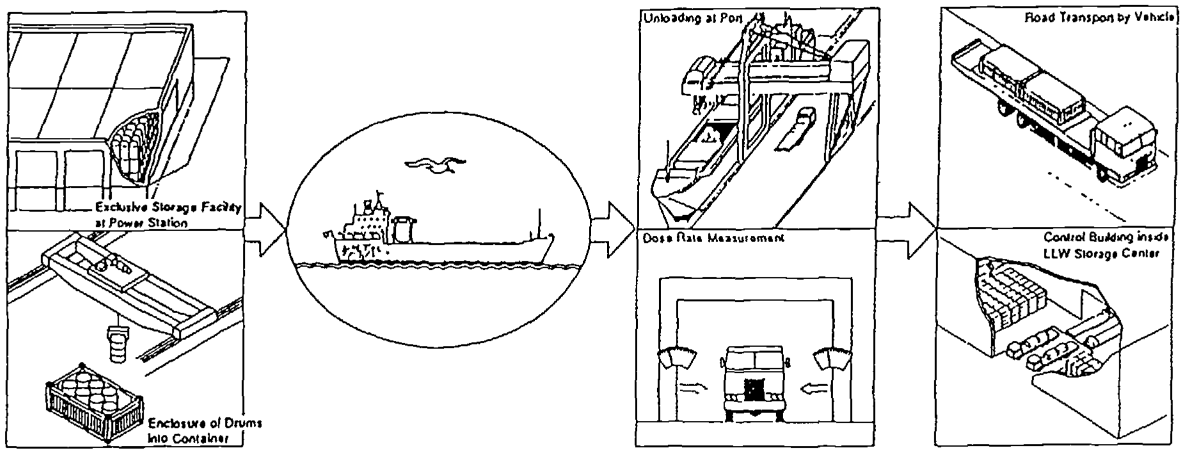

Fig.2 LLW Transport System in Japan

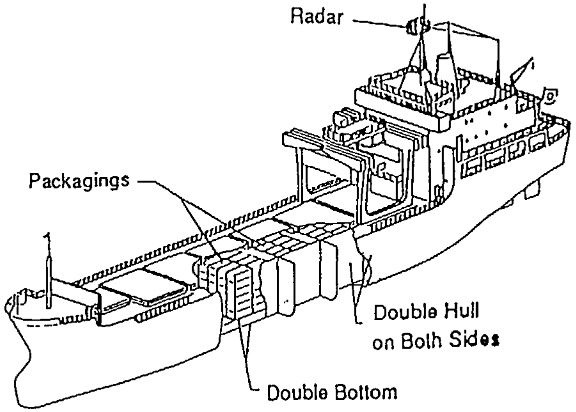

Fig.3 The Special Dedicated Ship "Seiei Maru"

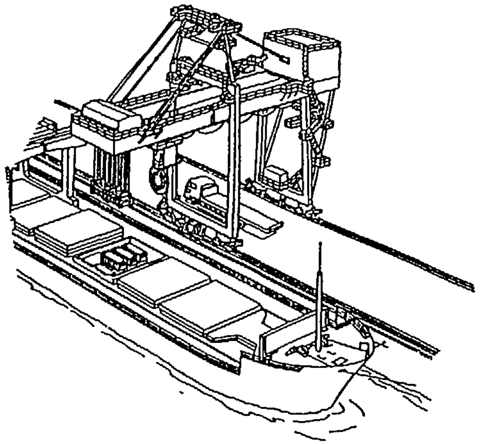

Fig.4 Bridge Crane at the Receiving Wharf

measure the radiation dose rates and confirm that they are below the regulatory limits, i.e. 2 mSv/h at the surface of the truck and 0.1 mSv/h at 1 meter therefrom.

To automatically measure the dose rates and reduce the exposure, a device which is called "Gate Monitor" was developed and the measurement at 21 points is able to be completed in a very short period of time, namely, in approximately 4 minutes (Fig.6)

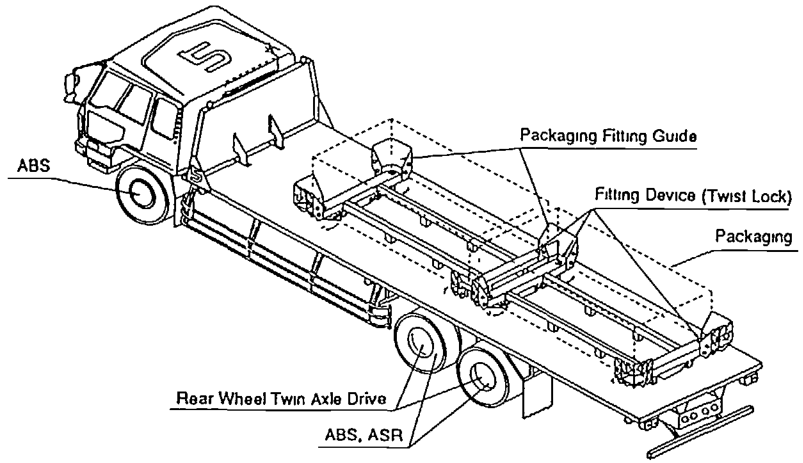

Fig.5 The LLW Transport Truck

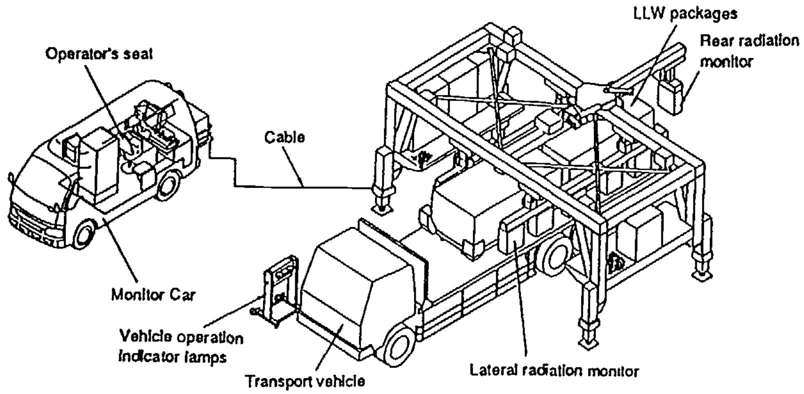

Fig.6 Gate Monitor

The trucks arriving at the LLW Burial Center are released of their packages which are then delivered to the JNFL. The packages are opened by the JNFL, the drums are taken out and the empty packagings are returned to the NFT.

These empty packagings are temporarily stored in the NFT owned Packaging Control Center, where they are cleaned, checked and repaired and delivered to the power plants whenever necessary (Fig.7)

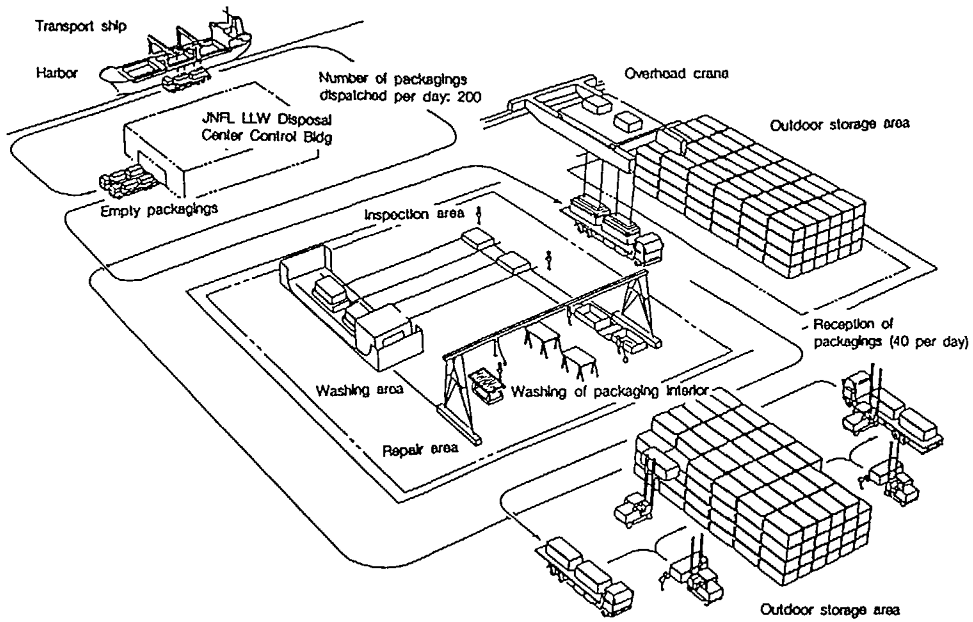

Fig.7 Packaging Control Center

3. Development of the packaging

The mass transport of the LLW was to be made in Japan for the first time and especially sea transport was not frequently done in the world when it was planned.

Therefore in developing the packagings, thorough deliberation and testings were repeated in view of compliance assurance and ease of handling.

It took about 6 years to complete the R&D and the fabrication of 3,000 units of the IP–2 packagings. The development was proceeded as follows:

Deliberation on the basic specification

At the first stage of the deliberation it was not clear whether the drum itself or the container should be considered the packaging and it was argued whether a simple open rack or an enclosed container should be used.

The enclosed container was finally selected on the judgement that it would be desirable to enclose the drums in a container from both containment performance and public acceptance point of view.

It was considered that the container should not be of standard type but should be the specific one which would be best suited to the transport system.

Regarding the capacity of the container verious number of drums were comparatively examined; 4 drums (2 drums x 2 rows), 8 drums (2 drums x 4 rows), 16 drums (4 drums x 4 rows), 18 drums (3 drums x 3 rows x 2 tiers), 24 drums (3 drums x 8 rows) and so on and 8–drum container was selected in consideration of ease of handling at the warehouse and possibility of loading on the standard 11–ton truck (Fig.8).

Detail design

In 1991, Japanese transport regulations were amended to comply with the IAEA transport regulations of 1985 edition and the category of the Type IP–2 Packages was introduced. Therefore the detail design of the container was carried out to provide

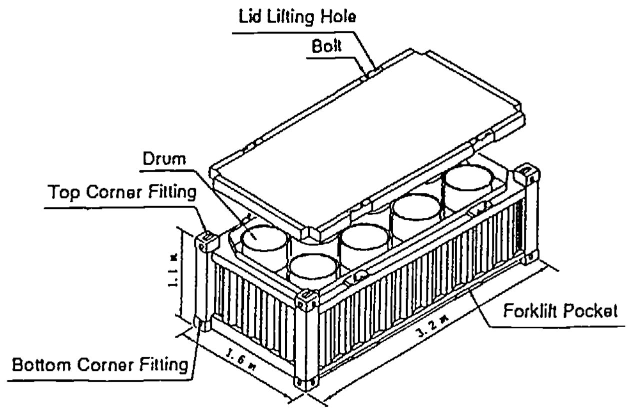

Fig.8 LLW Package

structural strength and leaktightness which satisfy the requirements of the Type IP–2 package. The compliance was confirmed by subjecting the models to stacking test and drop test (Fig.9)

Determination of number of packages to be manufactured

In determining the number of packages to be manufactured, a simulation analysis was performed in consideration of number of days required for packaging operations, inspections and receiving operations, and on the assumption of an LLW shipping plan that 25,000 LLW drums would be annually transported from the nuclear power plants.

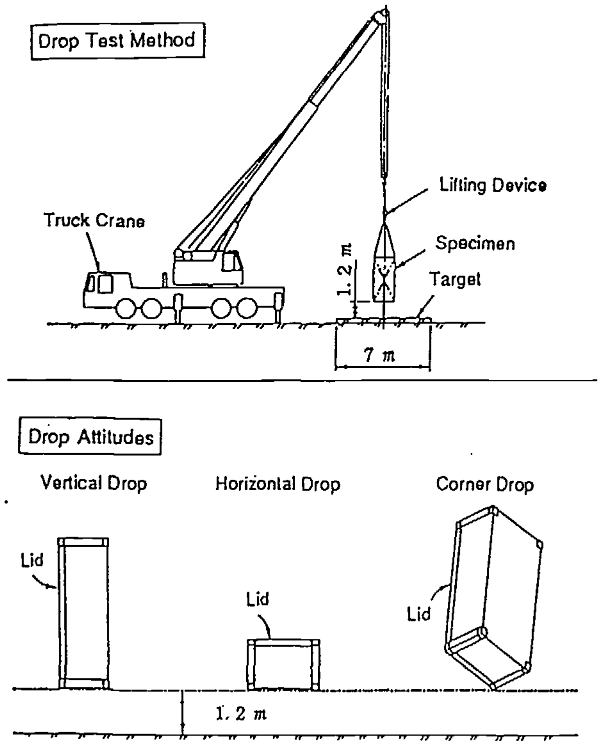

Fig.9 Testing of LLW Package

The number of packagings thus obtained with some spares resulted in the required number of 3,000.

Manufacture of the IP–2 packagings

The manufacture of 3,000 packagings was decided to be done by a local manufacure, Aomori Hoei Kogyo Company, with the objectives of promoting local industry. As the company had never fabricated containers, investment for new production lines was made and technical cooperation was provided by one of the major container manufactures, which resulted in monthly production of 200 containers.

4. Transport experience

The transport record of LLW in Japan is shown in Table 1. Some 20,000 LLW drums have been transported since the first shipment started in December 1992.

Table 1 Transport Record of LLW in Japan (as of 1.12.1993)

Period of Time transported	Name of Nuclear Power Plant	Name of Electric Power Company	Number of Packages transported
12/3 ~ 9, 1992	Tokai II	JAPCO*	185
2/3 ~10, 1993	Fukushima 1	TOKYO	335
3/8 ~12, 1993	Hamaoka	CHUBU	240
4/12~20, 1993	Mihama	KANSAI	250
5/17~21, 1993	Fukushima 1	TOKYO	240
6/16~21, 1993	Turuga	JAPCO*	163
6/24~28, 1993	Ohi	KANSAI	125
7/21~27, 1993	Shimane	CHUGOKU	200
8/18~23, 1993	Ikata	SHIKOKU	63
10/4 ~ 8, 1993	Hamaoka	CHUBU	140
10/18~26, 1993	Mihama	KANSAI	250
11/24~30, 1993	Genkai	KYUSHU	75
12/1 ~10, 1993	Fukushima 1	TOKYO	336
Total 13 Transports			2,602 (20,816 drums)

* Japan Atomic Power Company

There has so far been no transport accident and all the IP–2 packagings stay quite sound and are repeatedly used.

Radioactivity of the package contents sometimes exceeded A_2 but there has been no problem in view of specific activity limit of $A_2 \times 10^{-4}$/g.

5.Conclusion

The transport of LLW has been successfully performed in Japan by means of containers which meet the requirements of the IP–2 packagings and the concept of the IP package is thought to be quite appropriate for mass transport of LLW.

TRANSPORT OF RADIOACTIVE WASTE IN GERMANY - A SURVEY

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Abstract

The transport of radioactive waste is centralized and coordinated by the German Railway Company (Deutsche Bahn AG, DB) in Germany. The conditioning of radioactive waste is now centralized and carried out by the Gesellschaft für Nuklear Service (GNS). The German Railway Company, DB, is totally and exclusively resposible for the transport, the GNS is totally and exclusively responsible for the conditioning of radioactive waste.

The German Railway transports all radioactive waste from nuclear power plants, conditioning facilities and the existing intermediate storage facilities in Germany. In 1992 nearly 177 shipments of radioactive waste were carried out, 1991 the total amount was 179 shipments.

A brief description of the transport procedures, the use of different waste packages for radioactive waste with negligible heat generation and the transport routes within Germany will be given. For this purpose the inspection authorities in Germany have used a new documentation system, a special computer program for waste flow trakking and quality assurance and compliance assurance, developed by the electrical power companies in Germany.

Final Repositories in Germany

The first final radioactive waste repository in Germany was the former salt mine "Asse" near Braunschweig/Wolfenbüttel. Disposal of radioactive waste was started in 1967 but only for 11 years up to 1978. During this time nearly 120 000 m^3 of low- and medium-radioactive waste were disposed with an activity content of 1 250 TBq beta/gamma-activity and nearly 88 TBq α-activity.

In the former German Democratic Republic (GDR) a disused salt-mine was chosen for the disposal of low-radioactive waste /1/ situated in Morsleben near to Helmstedt at the former German-German border. The final disposal started in 1978. Low level radioactive waste from the nuclear power plants in Greifswald and Rheinsberg, from the research and development facility in Rossendorf (Saxonia) and from different users of low radioactive material in the former GDR were disposed in Morsleben from 1978 to 1991. Due to the decision of the court in Magdeburg the facility was closed between February 1991 and the beginning of 1994. The Morsleben final waste disposal is back in operation since 13. January 1994. Low level radioactive waste from the shut-down nuclear power reactors in Greifswald and Rheinsberg are disposed in the facility now.

In the western part of Germany the disused iron ore mine Konrad near Braunschweig/Salzgitter is planned to be the final disposal for radioactive waste with negligible heat genaration. The capacity is scheduled to be up to 600 000 m^3 of radioactive waste with a maximal beta/gamma-activity of nearly 5 000 000 TBq and an α-activity of maximal 150 000 TBq.

An other final repository-project exist in Lower-Saxonya, the Gorleben-project, a final repository for heat-generating radioactive waste. Today the Gorleben salt-dome is under examination.

General Situation

The electrical power generation from nuclear power plants has steadily increased in Germany since 1970, see table - 1 -. Due to the fact that the majority of the generated radioactive waste had to be stored in interim storage facilities a total amount of nearly 55 000 m^3 of waste with negligible heat generation exist, see table -2-.These values were given from the Radiation Protection Office in Salzgitter /2/.

Tab. 1. ELECTRICITY GENERATION IN GERMANY

Total electricity generation in 1991 : 458,7 TWh	
Nuclear	147,4 TWh
Coal Power	149,4 TWh
Lignite	84,0 TWh
others	77,9 TWh
Total electricity generation in 1992 : 461,7 TWh	
Nuclear	158,8 TWh
Coal	142,2 TWh
Lignite	86,6 TWh
others	74,1 TWh

Tab. 2. Radioactive Waste with negligible heat generation in Germany during 1991 and 1990

Producer of Rad. Waste	1990 m^3	1991 m^3
Reprocessing facility	873	799
Nuclear Power Plants	2 682 *)	1 846 **)
RID facilities	2 531	2 056
other	792	397
≤	6 878	5 098

*) final storage of 708 m^3 in Morsleben

**) final storage of 49 m^3 in Morsleben

Structures of the nuclear energy industry in Germany

Since 1988 modification of the structures of important areas of the nuclear energy industry in Germany /3/ has resulted in the following:

- the conditioning of radioactive waste is now centralized and carried out by one firm belonging to the Gesellschaft für Nuclear Service (GNS)

- the transport of radioactive material from nuclear power plants is centralized and coordinated by the German Railway Company (Deutsche Bahn AG, DB).

The main goal of restructuring is substantially to improve national safety measures by making one company in the private sector (GNS) totally and solely responsible for the conditioning of radioactive waste and one company (DB) totally and soleley responsible for their transport.

The Federal Government is of the opinion that the combination of the two approaches - i. e. improvements to national regulations and structural change in the nuclear industry - is a particularly appropriate way of achieving their safety objectives avoiding certain disadvantages of free competition.

Shipments of radioactive waste in Germany 1991 and 1992

Due to the fact that the inspection authorities in Germany could use a documentation system, a computer program for waste flow tracking and quality assurance and compliance assurance, data from radioactive waste shipments are available.

Data for 1991 show a total amount of 179 shipments including shipments in January and February 1991 to the final repository Morsleben. In 1992 the total amount of shipments of radioactive waste were 177. This means only shipments of conditioning radioactive waste from nuclear power plants to waste-handling facilities or interim storage facilities. A short survey is given in figures - 1 - and - 2 -. Standardized containers were used for those waste-shipments.

RadWaste Transport, 1992
Destination of Transport

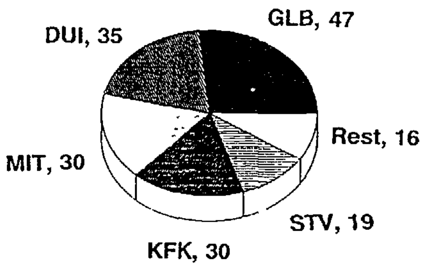

Figure 1

RadWaste Transport, 1992
Mode of Transport

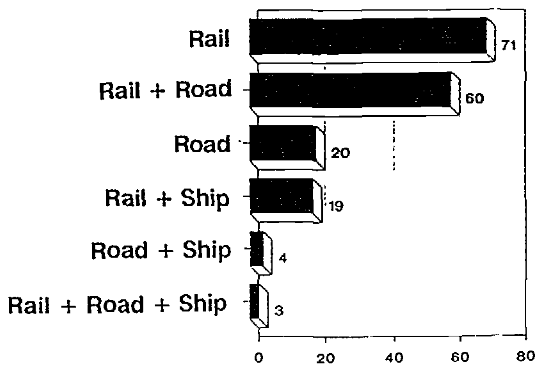

Figure 2

Konrad Transport Study

The "Gesellschaft für Anlagen- und Reaktorsicherheit, GRS" has finalized a safety analysis /4/ for the transports of non-heat-generating (low- to medium-level) radioactive waste to the planned final repository Konrad in 1991. The safety analysis has two main objectives:

- Assessement of potential radiation exposures from normal (incident-free) transportation, especially in the region of the final repository where all waste transports converge.
- Assessment of risks from transport accidents in the region of the final repository, i. e., the quantification of the frequency of accidents and of possibly resulting radiation exposures and contamination levels.

For the purpose of the study the anticipated waste transport volume and the waste properties were analysed in detail. This included information on the transport containers, waste product properties, activity inventories and local dose rates of the waste packages being transported.

The relevant IAEA transport regulations /5/ contain the requirements to be met by the transport of shipping units carrying radioactive waste.

Radiation Exposure from Normal Transport

The total annual radiation dose received by an individual as a result of waste transports passing by or stoping in this vicinity is derived from the dose rate at each location and the cumulative period of time spent by individuals at these locations during a period of one year. In this context, the study concentrated on exposure situations in which groups of persons are particularly exposed to the radiation field of the waste packages as a result of their living habits or their occupation. This corresponds to the normal procedure adopted in radiological protection in order to determine the potential doses to "critical groups of individuals". For persons who do not belong to the "critical group", the radiation exposure caused by the waste transport can generally be expected to be lower, and in most cases much lower, than for those in the critical group.

The doses determined for individuals of the critical group in the region of the final repository range approximately from 0,02 to 0,1 mSv/a.

To a large extent, doses are in the lower range of 0,02 mSv. This is either because on an average only short periods of time are spent in the immediate vicinity of the waste transports, since the vehicles generally pass by or stop only for a short time, or because of the larger distances between the individual and the waste transports.

The highest exposure conditions were determined for residents of buildings closest to the track at the Braunschweig marshalling yard. The effective dose for this population group is between 0,1 and 0,2 mSv/a.

The potential radiation exposure of critical groups of persons in the population as a result of waste transports is well below the value of 1 mSv/a recommended by the IAEA transport regulations. The additional radiation exposure of the critical groups of persons as a result of the waste transports is equivalent at the most to a small fraction of natural radiation exposure. The radiation exposure of these groups of persons, and consequently even more significantly of those inhabitants of the region around the final repository who do not belong to the relatively few individuals of the critical groups, remains practically unchanged by the waste transports.

Considering the persons who are occupationally involved with waste transportation, that is employees of the Federal German Railways and Verkehrsbetriebe Peine-Salzgitter, the dispatch and marshalling personnel at the Braunschweig marshalling yard and Beddingen interchange station, who are primarily involved in shunting and dispatching the waste wagons must be regarded as the critical occupationally exposed group of persons.

Depending on their functions, maximum doses of approximately 0,3 - 0,7 mSv/a are obtained for marshalling personnel and reception inspectors as a result of waste transport by rail. For the other transport personnel doses are significantly lower.

The results of the transport accident risk analysis will be summarized in the paper "Safety Analysis of the Transportation of Radioactive Waste to the Konrad Waste Disposal Site, IAEA-SR-189/27" (Mr. F. Lange).

Morsleben Transport Study

A transport safety analysis study has been conducted for shipments of low to medium level radioative waste materials suitable for underground disposal at the Morsleben final repository /6/. The objective of the study - refered to as Morsleben Transport Study - is the anaylsis of transport operations and the assessment of the radioalogical risks from normal transportation and potential accidents. The annual volume of waste shipments assumed for the study is 865 shipping units corresponding to a waste volume of approximately 5 000 m^3. These values are consistent with courrent estimates of the disposal capacity of the Morsleben final repository for one-shift operation.

A shipping unit generally represents a standard 20'-freight container used as overpack for transporting the varius reusable and non-reusable waste packages types. The pakkages accepted for disposal are assumed to be primarily 200 l-drums, cylindrical concrete containers, and cubical sheet steel containers.

The requirements concerning the package activity content, the characteristics of the waste packages and other relevant paramenters result from:

- the current waste acceptance criteria based on safety considerations for the Morsleben repository
- regulations for transporting hazardous materials.

The information required to describe the type, quantity and propertiers of the various waste materials suitable for disposal at the Morsleben final repository are based on a survey at major waste producers. Consistent with the preliminary waste acceptance criteria only low specific α-activity waste materials (< 40 MBq/m^3) are included in the study.

Rail transport ist the preferred shipping mode nationally. But the Morsleben final repository has no rail acces. This is the reason why in the repository region, however, i. e. the 40 - 50 km region around the disposal site, waste transportation have to be primarily by road from the marshalling yard Magdeburg-Sudenburg to the location of the final repository.

The assessment of the potential radiation exposure from normal transportation to the transport personnel and the public is based on an analysis of the transport and handling operations, the dose rate of the waste packages, and the potential exposure conditions along the transport route.

Only critical group of individuals, i. e. person exposed to radiation from the waste packages, are considered in the study. The dose estimates for members of the public are generally less than 0,08 mSv/a and the transport personnel are generally less than 1.1 mSv/a, respectively, except for truck transport personnel where doses can be as high as 5 mSv/a or even above depending on the driving schedule of the individual truck drivers.

The accident analysis relies to a large extent on probabilistic safety assessment techniques taking into account the broad range of values of model parameters which determine the radiological consequences of transport accidents and the estimated frequency of occurrence.

The expected frequency of road transport accidents resulting in minor radioactive releases in the region surrounding the repository has been estimated to be on the order of 1 in 60 per year. The doses and environmental consequences of such accidental events are very low and are limited to the site of the accident. Transport accidents and the associated relesases resulting in doses at the accident site exceeding 2 mSv/a is approximately 1 in 250 for an assumed operating period of 20 years.

An effective lifetime-dose of 50 mSv will not be exceeded under any circumstances at the site of the road transport accident that can be reasonably assumed for dose assessment.

From the results of the study it can be concluded that the overall transport risk from shipments of radioactive waste materials to the Morsleben final repository is very small.

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TRANSPORT OF RADIOACTIVE WASTE IN DUKOVANY NUCLEAR POWER PLANT

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Abstract

Article describes the system of radwaste transport in Dukovany NPP: gaseous, liquid and solid waste of low level and intermediate level activity. There are different kinds of the transport in the reactor unit and on the way from the waste treatment facilities to the disposal. The article describes also waste composition, transport containers and new Czech radwaste disposal.

In conformity with the valid Czech legislation, radioactive wastes belong to dangerous wastes. Their transport is rather different in dependence on the waste kind /state, chemical composition, and activity/ and on the route of transport.

Radwaste transport in the Czech Republic is regulated by the regulation of the Ministry of Health No. 59/1972 Coll. on "Protection Against Ionizing Radiation" and another regulation No. 67/1987 Coll. on "Ensurance of Nuclear Safety during Handling with Radioactive Wastes", issued by the former Czechoslovak Atomic Energy Committee. Parts of Czechoslovak State Standards ČSN 341730 "Regulations for Working Places with Radioactive Substances", issued in

1989, and ČSN 401400 "Transport Packing Sets for Radioactive Substances", issued in 1992, concern the above-mentioned matter. Details on transport conditions are given in the regulations of single ministries and, in reality, they give the provisions of the signed international treaties on transport of dangerous substances by train, by roads, by water, and by air.

Transport of gaseous radwastes is done automatically by a system of airconditioning ventilators and piping. During transport, the air from less contaminated spaces goes through possibly contaminated spaces into the gas cleaning plant. After cleaning it is let out by a 150 m high ventilation stack.

In accordance with regulation No. 67/1987 Coll. there is an effort to minimalize the transport of liquid radwastes in casks in the Dukovany Plant technology. All in all, this possibility is used in the design only for transport of contingent contaminated drainage water from the regional radwaste storage facility within NPP Dukovany premises, located about 400 m from the Dukovany Unit No. 4.

In conformity with the CAEC regulation No. 67/1987, transport of liquid radwastes is done through piping primarily. This is done within the reactor unit as well as to the building of radwaste solidification. A piping bridge transports liquid radioactive concentrates and ion-exchange resins through outdoor line in the length of 300 m in the summertime. It is made of stainless steel piping with certificate, connected by X-ray checked welds. All the three piping lines /for concentrates, for ion-exchange resins, and stand-by line/ are covered. For the case of contingent leakage, there is a collection trough which would lead off the liquid back to the control zone.

For transport of contaminated water from outside buildings into the power plant there is a special transport cask for liquid radwastes. It is made of stainless material with certificate and has a safety tank for collection of leakages. It can be transported by a crane and on a truck. It is planned mostly for transport of drainage water from radwaste storage facility which may be contaminated.

Solid radwastes are transported within the plant premises as follows: Common small waste is collected at designated places within the plant control zone from where it is transported in plastic bags by hand or electric trucks into the auxiliary building. After a selection of non-active objects waste is deposited here for several years. Waste sorting is done in a half-automatic dosimetric apparatus - sorting carousel and sorting box. It makes possible to sort out 50% of non-active objects from solid waste. These sorted out non-active wastes may be incinerated in a non-active incineration plant. In addition to the planned transports we had to cope with transport of large-size objects, viz. racks from spent fuel ponds. Large-size metal objects are partially decontaminated and then transported to outside radwaste storage. With regard to a low exposure rate the transport is done without a cask. Metal objects have to be completely dry after decontamination and then they are put, in a plastic cover, on wooden beams on the truck body. After checking of tyres cleanness the truck leaves the control zone and within one minute it reaches the radwaste storage place where it goes into control zone again. Metal objects are taken by a forklift onto the prepared beams. The transport is not permitted to be done during rain or snowfall.

The transport of solid combustible radwaste into the incineration plant is solved at present and its implementation begins. Combustible waste is collected in the auxiliary building. Auxiliary building No 2 stands next to the planned location of the incineration plant. Radwaste is collected in box pallets. In each pallet there are 5-8 polyethylene bags with waste. These pallets are placed into a shielded cask which, under a special regime, goes into the lift shaft, from where it goes by lift onto a load truck which will transport it to auxiliary building No. 2 by a similar lift. Box pallets with waste from all the plant Units are transported through an inner gate directly into the incineration plant where they will be destroyed.

One of the most important procedures is the transport of solidified radwaste in barrels into the regional spent fuel storage facility. It is a surface storage facility for low- and middle-active waste. It contains four rows of pits in double lines. The rows contain seven dilatation parts with four pits each. Each of 112 pits can hold approximately 1200 barrels for 200 l.

The total size of storage space is 55 450 cu.m. which corresponds to 130 000 barrels. The waste handling is done by means of a gantry crane which moves along the lengthwise walls of the pits. The crane has two crabs. The larger one with capacity of 12.5 t serves for handling of panels and casks. The lesser one with capacity up to 1 t serves for precise manipulation with barrels. This crab is provided with microtraverse and is controlled from a shielded box by means of a set of camers. The crane may be also controlled remotely from the operation building /to the length of as much as 200 m/. It is provided with several lifting attachments. Movable shield covers the opened tank and serves also for transport of barrels taken out of the cask. In addition to barrel cask there are three basic types of transport casks. The casks are put on a transport frame of a semi-trailer. An eigth-barrel cask weighing 2310 kg is used for low-active waste. For middle-active waste, four-barrel casks of a cylinder shape, weighing 3700 kg and 6200 kg, are used. For relatively highest activities, one-barrel casks, weighing 4000 kg and 6080 kg, are used. Treated radwaste in the form of bitumen composition is closed in 200 l metal barrels. In spite of a short distance between the building of radwaste treatment from the storage facility /about 500 m/, the transport is done by special technical means. The barrels are put into the storage pits in six layers in vertical position and gradually covered by a concrete mixture. Putting into carriers or casks is done by a crane in the building of radwaste treatment. The casks are put into the transport frame located on a semi-trailer in the despatch hall of this building. Under a special regime the semi-trailer goes from the restricted area towards the storage facility by the shortest way. The despatch hall is designed in such a way that the semi-trailer with transport frame need not go back but it drives through the hall. With regard to large quantity of transported radwaste, there are special safety and warning provisions for the case of an accident outside the restricted area. There are emergency procedures for accident with fall and rupture of a barrel and for the case of vehicle fire. After the rupture of a barrel the bitumen composition usually does not desintegrate because it is rather soft. It is very slightly inflammable and it is self-quenching. After a ten-minute burning the fire would be quenched by the resulting salt crust. A fall into a water source has not been analysed as the route is chosen in such a manner that it avoids all water tanks or water streams. Even with the maximum speed of 30 kms per hour the distance between the two objects is covered within 2 minutes.

Within the precincts of the radwaste storage facility wastes are taken over and the semi-trailer is transported under the gantry crane. Here the barrels are taken out of the carriers or casks and placed onto the transport platform. Barrels with platform are transported by the gantry crane to the now filled pit and put in the alloted space. Then the barrels will be covered by concrete mixture. This storage facility serves for permanent storing of all low- and middle-active wastes from NPP Dukovany and NPP Temelín. It is fully completed and prepared to store radwastes. Their bitumenation shall begin at NPP Dukovany within a few months.

At present the transport of radwastes from NPP Temelín to the storage facility at NPP Dukovany is subjected to an intensive research. Two road routes were chosen. The usual route is 176 km long and is planned for transport in good weather conditions. The reserve route is planned for transport at worse road conditions and higher weight of the semi-trailer. It is 189 km long. Until the beginning of NPP Temelín operation other roads may be constructed, with good carrying capacity of bridges, sufficient width and low slope. Nevertheless their safety checking and approval is a long-time matter, especially from the viewpoint of individual solution of all possible accidents with water streams, which cannot be fully avoided. Mostly one-barrel casks with bitumination product and also pressed waste for the incineration plant will be transported. The possibility of transport of vitrified product is evaluated which would be exceptionally resistant against water. Transport of small quantities of other radwaste and dress parts to be washed will be done in casks too, but it is not completely solved as yet.

The transport of radwaste from Temelín to Dukovany by road is a certain risk for smooth operation of the Czech Power Enterprises Co. The risk is connected with bad road conditions in winter time or during heavy rain and also with political influences and public acceptance. In spite that it is possible to stop the transport even for several months it has been decided to evolve a method of transport of radwaste by train. The preferable way is the transport of a semi-trailer without towing vehicle on a waggon. Two semi-trailers can be put on a special waggon at the same time. The variant of the transport of the casks directly on the waggon is a more complicated one as for handling, because then it is necessary to transload the casks and to provide the storage facility with a railway siding. But from the legislation point of view, any transport by train is more simple because it is not necessary, when seeking approval, to negotiate with single district authorities as in the case of transport by road.

A PLAN OF RADIOACTIVE WASTE MANAGEMENT IN CHINA

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Abstract

China has been developing nuclear industry for more than three decades. In these activities, considerable quantity of various radioactive waste has been accumulated. By 1992, the accumulated solid and liquid Low- and Intermediate-Level Waste (LILW) had totaled to 41,000m^3 in China. Along with the utilization and development of nuclear energy, the increasing quantity of radioactive waste is required to be appropriately treated and disposed of. It is estimated that the accumulated LILW in China will be 173,000m^3 by 2010. In order to better utilize and develop nuclear energy, to protect environment, China has formulated a plan on appropriate treatment and disposal of LILW, and implementation of this plan is undergoing. My introduction will be with emphasis on the plan of LILW management in China.

1 INTRODUCTION

The basic objective of radioactive waste management is, through treatment and disposal of radioactive waste with a safe and effective way, to prevent radioactive nuclide releasing to environment with an unacceptable quantity, and to make exposure to staff and public now or in future be within the authorized limit and be As Low As Reasonable Achievable (ALARA), further more, so as to protect mankind and its environment.

We have formulated the policy on solid LILW management: minimizing waste generation, collecting with waste segregation, volume reduction and stabilization, firm packaging, interim storage in situ, safe transportation, and regional disposal. For liquid LILW, concentrates processed by evaporation, ion exchange and filter are first put into interim storage in situ, then be solidified to stable form. The main solidification processes used in our country include: cementation and bituminization. After appropriate packaging, dry solid waste and solidified waste will be transported to disposal site for final disposal.

2 CONDITIONING

Some kind of waste must be solidified and packaged to convert waste to a form that is suitable for transportation, storage and disposal. The solidification technology used in our country include cementation, bituminization and solidification with other agents specially developed.

The plant to solidify low level liquid concentrate with bitumen was put into hot trial operation in 1992. The plant was equipped with two independent process lines. The main process equipment on each process line is a rotary scraper with a throughput of solidifying Low Level Liquid Waste (LLLW) of 200~250 liters per hour. The result of trial operation is very satisfactory. 500m^3 of LLLW has been solidified with bitumen in 1993.

The first nuclear power plant in China——Qinshan Nuclear Power Plant produces about 40m^3/a of LLLW concentrates with specific activity ranged from 2.3×10^6Bq/l to 2.3×10^7Bq/l. The concentrates are solidified with a in-drum cementation process. Solid waste produced in Qinshan Nuclear Power Plant is totally 220m^3(1100 drum) each year.

For intermediate level liquid waste (ILLW), two kind of cement solidification technology have been chosen: underground hydrofracture process with cement, and in situ bulk grouting with cement. Both processes combine treatment and disposal, and are not only safe but also economic. The treatment capacities of the two processes are also quite high.

Based on design, hydrofracture process can treat and disposal of 300~350 m^3 of ILLW per 8 hours. It is planned to operate 4~6 cycle (8 hours per cycle) each year. The disposal capacity of each hydrofracture well is more than 10000m^3. Waste will be disposed of in the various layers undergroud from 400~300m. The key factor of this process is whether a geologically and hydrologically suitable site can be found near a reprocessing plant. Through about 10 years relevant research and development, the suitable cementation formulation has been developed and an underground hydrofracture test with radioactive tracer has been completed. It is demonstrated that the site is suitable for disposal of ILLW with undergroud hydrofracture process. The engineering design of this project has been completed, and the construction started. It is expected that disposal of ILLW with this process will begin in 1995.

The process of in situ bulk grouting with cement can only be used in the specific site which is suitable for shallow land disposal of ILLW. Radioactive waste, cement and additives are mixed in a mixer and then continuously poured into underground pools with dimension of 8m×8m×6m. Each pool is equipped with a double vane mixer. The plant for in situ bulk grouting is located at Lanzhou Nuclear Fuel Complex (LNFC). The site is located in Gobi Desert where is sparsely populated and arid climate, and the groundwater level is between 38. 9 and 40. 2m. Engineering cold test and conceptual design of the project have been completed. Engineering design and preparations for construction are actively conducted. Construction work of this project will be actively conducted in this year, and hot operation of in situ grouting of ILLW will begin in 1995.

3 TRANSPORTATION

Solid waste and solidified waste produced in nuclear power plant is interimly stored in site for about 5 years, it shall then be transported to volume reduction center or final disposal site. Based on the locations of nuclear power plant and disposal site, transportation can be carried out either by rail, road, water or by combination of them. All transportation shall be carried out in accordance with the regulations for transportation of radioactive materials, and all packaging of radioactive materials shall strictly fulfill relevant regulations and standards.

At present, practice experience on radioactive waste transportation in China is very limited, it is necessary to learn relevant experience and lessons from other country. In the other hand, it is essential to establish a sound system of radioactive transportation, which includes transportation carrier, shielding and shipping container, transport modes and routes, authority in charge of transportation as well as a perfect surveillance system.

Because the development of nuclear power in China is still in a initial stage, the quantity of waste produced is very limited comparing with countries in which nuclear power are developed. Although the policy of LILW disposal is regional disposal, it is not necessary at present for our country to built more disposal facilities, because operation of disposal facility can be cost-benefit only when the facility has a reasonable scale. Even though the nuclear power is developed to a considerable scale in China, it is impracticable for each nuclear power plant to construct and operate a disposal site. It means that transportation of radioactive waste is also a unavoidable issue that must be solved.

Most countries have promulgated regulations on radioactive materials transportation which will also regulate radioactive waste transportation. IAEA has promulgated a Regulation for Safe Transportation of Radioactive Materials. These regulations provide requirements on the radioactive quantity limits of each packaging, shielding, and surface contaminations of packaging as well as packaging quality control requirements etc. Based on IAEA Regulations for Transportation of Radioactive Materials, China formulated "GB-11806-89 Regulation on safe transportation of radioactive materials" in 1989.

In order to smoothly conduct works relevant to LILW disposal in our country, we will make efforts on research of transportation of radioactive waste. On the basis of extensive research on experience and lessons of radioactive waste transportation of other countries, the feasibility research of transportation of LILW will be carried out. The safety of transportation of radioactive waste is achieved through strictly implementing the relevant regulations and rules. We will also formulate radioactive waste transportation rules which meets conditions of our country, based on extensive research on experience and lessons of radioactive waste transportation of other countries and experiences on radioactive materials transportation of our country. So long as packaging, handling, and transporting of radioactive waste are strictly supervised according to regulations and rules for transportation of radioactive materials, the safety of radioactive waste transportation can be ensured.

4 VOLUME REDUCTION

In some countries, the charge for waste disposal is ever increasing for a variety of reasons. However, the increasing of charge greatly promote the developing and utilizing of volume reduction technologies for LILW. Volume reduction not only can improve the stability of waste, but also decrease cost for waste disposal and interim storage. Based on our research and experiences of foreign countries, we have decided to construct a radioactive waste volume reduction center in LNFC. Technologies used in the volume reduction center include compact and incineration. Melting decontamination technology of contaminated metal may be also included in the future.

For compact, we plan to import a supercompact system with compact pressure up to 2000 tonnes, and volume reduction factor of $3 \sim 10$.

For incineration, on the basis of our studies and experiences learned from other countries, it is planned that the first prototype of incineration facility will be developed ourself. The incinerator can combust $50 \sim 100$kg combustible waste per hour, and the ash will be solidified with cement.

After being treated by volume reduction, stabilization, package, decontamination, waste is transferred to disposal site for final disposal.

It is planned that the volume reduction center will be put into operation within 3 years.

5 Disposal

In radioactive management, disposal is the final and the most important step. Countries worldwide have paid a great attention to disposal of LILW. In view of that China has a vast territory and site selection for disposal site is less difficult in our country, considering the economy supporting ability of our country and the developing trends of LILW disposal worldwide, we deem that shallow land disposal is suitable for conditions of our country. For this reason, a great efforts have been made to select LILW disposal site in west, east and south of China, and a lot of research and development works relevant to shallow land disposal have been carried out.

For Northwest Disposal Site, feasibility research has been completed, and safety analysis report to determine environmental impacts has also completed, the conceptual design work will begin soon. The disposal capacity of the first phase of the disposal site is $60,000m^3$with disposal capacity of second phase being $200,000m^3$, and disposal capacity of final phase up to $1,000,000m^3$. The Northwest Disposal Site is located in the northwest of China and near the LNFC. The climate of the site is arid and very dry, annual precipitation is very low. This region is sparsely populated. The site, which is similar Richland Disposal Site of USA in nature, is a quite ideal site for LILW disposal.

The performance objective provided by National Standard "GB9132-88 Regulations for Shallow Ground Disposal of Solid Low- and Intermediate-level Radioactive Wastes" of China is as follows:

a) The task of shallow land disposal of waste is to retain radioactive nuclide in waste within the disposal site during the period in which the waste may bring out unacceptable risk to mankind (in general, it shall be considered for $300 \sim 500$a), so as to prevent radioactive nuclide releasing to environment with an unacceptable concentration or quantity which will threaten the safety of mankind.

b) During normal operation or in the circumstance of accident, radiation protection for staffs and public shall fulfill the requirements prescribed in National Standard "GB8703-88 Regulations for Radiation Protection", and shall observe the "As Low As Reasonable Achievable" (ALARA) principle. During waste disposal, effective equivalent dose of exposure to public, which is resulted from radioactive materials releasing from disposal site through all kind of pathways, shall not exceed 0.25mSv per year.

Researches on Northwest Disposal Site show that the performance objectives can be completely achieved in the site.

It is planned that Northwest Disposal Site will receive the first batch of waste in 1995.

6 CONCLUSION

China has formulated a plan on LILW management which is smoothly being implemented. We believe that radioactive waste management in China will be improved to a new level in few years. The improvement of radioactive waste management will surely benefit the development of nuclear power and environmental protection in China.

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HANDLING AND TRANSPORTATION OF LOW LEVEL WASTE IN INDONESIA

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Abstract

At present, radioactive wastes in Indonesia are generated mainly from nuclear research and in a small amount from industrial application activities. The wastes, mostly containing short-life radionuclides, consist of large quantity of low- and small quantity of medium- level wastes. Some research centres have their own waste treatment facility, but do not have an adequate long-term storage for their packaged wastes. Other research centres, like centres at Serpong Nuclear Research Centre (SNRC), centralize their waste-treatment and -storage at the Radioactive Waste Management Centre (RWMC). The transport of wastes from waste producers to the RWMC are carried out using truck trailers.

The paper describes practices of waste handling and transportation. Principles covering waste handling before- during- and after- waste treatment are implemented by considering the health and safety of the transportation workers and the environmental safety. Waste collection has been arranged is a such a way to facilitate further waste processing. The description covers the low level wastes in the form of inorganic, organic liquid waste, spent resin, compactable and burnable solid wastes and embedded wastes.

1. INTRODUCTION

1.1. Waste Management

There are four nuclear research facilities. These centres are located in Yogyakarta, Bandung, Jakarta, and Serpong. Serpong Nuclear Research Centre (SNRC) is the newest and largest among the four. It consists of a 30 MW research reactor, a radioisotope production facility, a fuel fabrication and fuel research centre and a radioactive waste management centre (RWMC).

In a broader picture, the waste generation can be differed by location at where waste is originated, i.e.: outside SNRC and inside SNRC (Fig.1).

The quantity of radioactive wastes generated from nuclear research centres outside the SNRC and nuclear application was small. The wastes consist of low level liquid- and solid-wastes. The treatment of the wastes is simple, i.e., through collection, storage, and decaying the radionuclides with short half-lives. The longer half lives, however, were conditioned into cement matrices. The embedded wastes, finally, are sent to the RWMC, because those facilities do not have waste storage that can satisfy the safety requirement. Meanwhile, wastes generated from nuclear research facilities in the SNRC is in relatively large amount. Treatment of the waste is centralized in the RWMC.

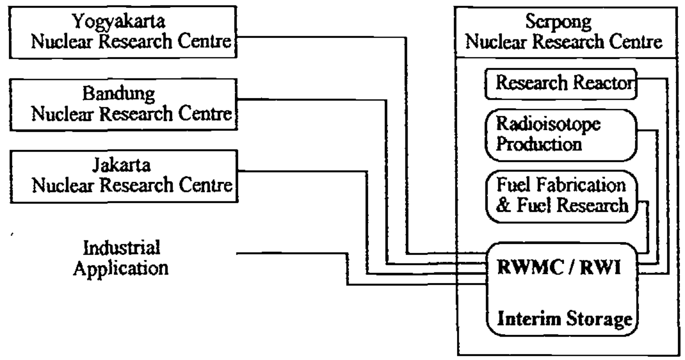

Fig. 1. A flow diagram of low level waste transported to the RWMC

1.2. Waste Treatment Facility

Processing Unit

The RWMC has a centralized installation to manage radioactive wastes (RWI). The RWI is equipped with an evaporator unit of 0.75 m^3/h capacity; a hydraulic press of 600 kN for compacting solid wastes; a cementation unit to immobilize concentrates, spent resins and solid wastes; an incinerator unit to burn organic liquid- and burnable solid wastes; and an active laundry unit to decontaminate the personnel protective devices.

Storage

To accommodate the 'raw' wastes, the RWI is provided with two storage tanks of 5 m^3 capacity each for spent resins, an underground storage tank of 50 m^3 capacity for organic-liquid wastes and five storage tanks of 50 m^3 capacity each for inorganic-liquid wastes. To to store the embedded wastes, an Interim Storage building that has a 1500 m^2 space area is available. Figure 2 shows a flow diagram of the wastes from the receiving until storage.

2. HANDLING AND TRANSPORTATION OF LOW LEVEL WASTES

In practice, prior to waste-transfer is carried out, waste producer officially request the RWMC to transfer their wastes to the RWMC. In the request, the waste producer mentions the waste description which includes activity of alpha- beta- and gamma-radiation; radionuclides content, amout of waste, physical and chemical form of waste, and other important information. The description will determine the waste handling and transportation method.

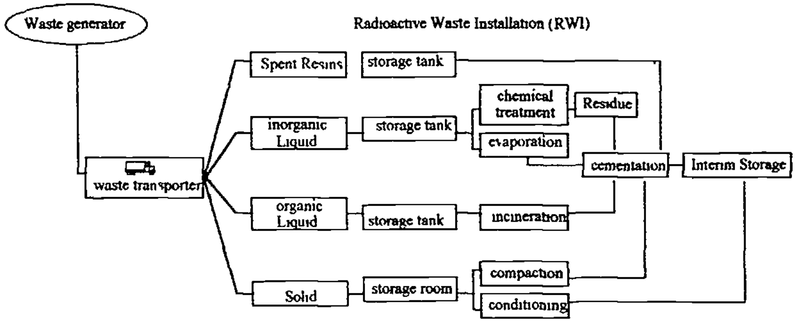

Fig 2 A flow diagram since the waste receiving, processing until storage

2.1. WASTE HANDLING

Wastes originated from centres in the SNRC are usually handled and transported in 'raw' condition, whereas wastes from outside the SNRC handled and transported in pre-treated condition.

2.1.1 Raw wastes

In practice, waste handling and transportation are carried out according to the waste treatment methods. The raw wastes which is originated from centres in the SNRC consist of

1. liquid:. inorganic
   organic
   spent resins
2. solid. compactable
   burnable
   non-compactable and non-burnable

Both inorganic liquid wastes, and spent resins can be directly transported by waste transporter as long as the specific activity does not exceed 0.1 Ci/m^3 But, other type of waste does need a well designed and tested packaging to be safely handled and transported.

<u>Inorganic Liquid Wastes</u>

The inorganic liquid wastes generated by the nuclear research centres in SNRC are collected in stainless steel storage tanks located in each centre The transporter will carry wastes from those waste generating centres to the RWMC located within a radius of 2 km. The transporter used by the RWMC is shown in Fig. 3 The transporter is capable of safely performing all operations necessary to transfer liquid waste into and out of the unit Transfer is to be effected by placing the transfer tank under partial vacuum for

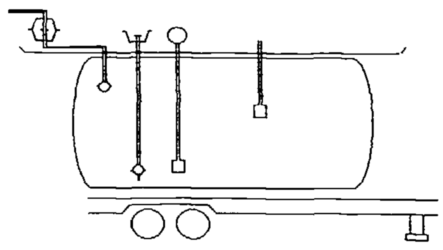

Fig 3 Liquid waste transporter used by RWMC

filling/loading, and pressurizing the tank to empty Transfer piping size is 50 mm in diameter and a flexible hose of approximately 15 m length with a self blanking quick connect/disconnect coupling is provided.

Liquid wastes generated by a radioisotope production centre has an activity level of 1.5 10^{-3} Ci which is considered to be the highest activity level among the low level liquid wastes. However, the waste activity level is still lower than the A_2 limits set-up by IAEA. Up till now, the maximum measured dose in liquid waste transportation is 0.4 m^3/h at contact and 0.3 mrem/h at 1 m from the trailer In the RWMC, the wastes are evaporated to convert the raw waste into concentrates. Finally, the concentrates are solidified in 950-l concrete shells

The inorganic liquid wastes generated by nuclear research facilities outside SNRC Serpong is processed by each facilities. Then, the waste will be transferred to the RWMC in the embedded form. Handling and transportation of this type waste is described in details in Sect 2 1.2

<u>Organic Liquid Wastes</u>

This type of waste is generated by laboratories in the SNRC The organic liquid wastes are collected in Teflon jerrycans. A stainless-steel container is provided as a packaging in which three 30 l jerrycans can be contained. Space between the container and jerrycans is filled with absorbents as a shock absorber. A complete packaging for organic liquid waste is shown in Fig. 4.

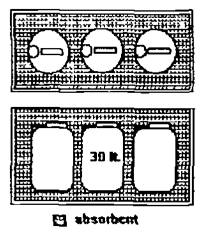

Fig 4 Packaging used to handle organic liquid wastes

Transportation of the organic liquid waste is carried out using a specifically designed truck to transport solid wastes Once the truck reaches its destination in the RWMC, using a portable pump the liquid waste is unloaded to an underground storage tank to be processed further Each time the wastes are transferred to the storage tank, samples of wastes are picked up to carry out the waste identification In RWMC, the wastes will be incinerated, then the generated ash will be solidified in a 100-l drum

Spent Resins

In addition to liquid wastes, a research reactor in SNRC also generates spent resins To meet transport criteria, dilution of the resins is needed until the specific activity of resins down to 0 1 Ci/m^3 Transfer of resins from the storage tank to the transporter is carried out indoor The resins recirculated by a pump and then trapped in transporter tank The process continues until a half of transporter tank filled with the resins The size of the tank is 1 5 m^3 It is made of SS 316 L with a wall thickness of 9 mm (Fig 5) In the RWMC, the wastes are unloaded to a storage tank in similar way but reversibly the loading process Finally, the spent resins are immobilized with cement slurries in a 350-l concrete shell

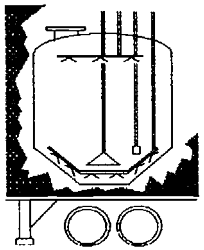

Fig 5 Spent resins transporter

Compactable Solid Waste

The wastes consist of boxes, paper, woods, etc The wastes are contaminated with β, γ emitters such as Co^{60},Cs^{137}etc Up till now, the maximum measured activity of the solid waste is 0 7 mCi that is still considered lower than A_2 value set-up by the IAEA for those emitters To avoid the spread of contaminants during handling, the wastes are put into plastic bags Then, the bag is tied using a tape Finallly, the bag is put into a100-l steel drum The packages containing the low level solid waste are then loaded to a specifically designed truck to transport solid wastes and the total number of packages aboard a single conveyance is limited so that the total sum of the transport index does not exceed 10 The maximum of measured radiation exposure is 0 4 mrem/h at contact and 0 2 mrem/h at 1 m from the freight container In RWMC, the compactable wastes in 100-l drum are placed in a 200 l drum and compacted by a 600 kN hydraulic press The 200- l drum is then moved the cementation unit for solidification in cement slurry

Burnable Solid Waste

Burnable solid waste particularly animal carcasses are generated by laboratories Other wastes are paper, linen, etc The waste is placed in plastic bags Then, the bag is tied using tapes Finally, the bag is put into boxes The packages containing the low level solid waste are then loaded to solid waste truck The same transport criteria are applied for the packages In the RWMC, the waste will be kept in refrigerated conditions until incineration process scheduled to treat this waste Then, the generated ash will be solidified in a 100-l drum

Non-Compactable and non-Burnable Solid Wastes

The wastes consist of metals, filter, glass, spent sources, etc Metals and filters are transported to the RWMC only after undergoing segmentation Those wastes and sealed spent sources is in put in plastic bags or boxes if possible then placed in a 200-l drum or a 950-l concrete shell depending on the waste activity level After removing the waste from the plastic bags, the waste will be directly solidified in 950-l concrete shells

2 1 2 Embedded Wastes

Embedded Wastes From Facilities Outside SNRC-Serpong

Liquid wastes generated by facilities outside SNRC-Serpong usually are immobilized locally by each facilities The waste was put into a 200 l steel drum and immobilized with cement slurry Conditioning in this way prevents unauthorized removal of the radioactive waste because of the bulk, weight and vigorous nature of the package and it also provide barriers against loss of containment of radioactive waste A 200-l package would have a weight of about 450 kg For reasons previously mentioned, the embedded waste need to be stored in the RWMC'Interim Storage

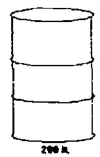

Fig 6 Embedded waste package from facilities outside SNRC-Serpong

Embedded Wastes from the RWMC

Packaged wastes generated by the RWMC itself has two different size containment, i e , 200 l drum and 950-l concrete shell It is noted, that the external dimension of 350-l concrete shell is the same as that of the 950-l concrete shell (Fig 7), but the first has a thicker wall

Wastes packaged in 200-l drum may be either cemented solid wastes or organic liquid wastes Wastes packaged in 350-l drum are cemented spent resins, whereas wastes packaged in 950-l drum may be solidified concentrates or solid wastes Each package is identified as explained in Sect 2 1 3

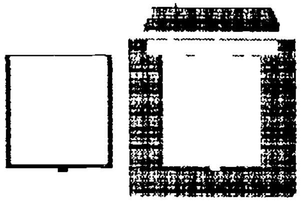

Fig 7 A 950 l concrete shell along with a Steel containment

The embedded wastes in the RWI need to be transferred to Interim Storage The distance between the RWI and the Interim Storage is only about 100 m A forklift is used because its versatility in picking up and arranging the packaged waste in the Interim Storage Within a restricted area, the transportation of embedded wastes from the waste processing unit to Interim Storage building does not need to apply the local (Indonesia) transport regulations However, the waste transportation is carried out very carefully and the safety regulations are fully followed A radiation protection personnel always accompanies the transportation workers The Interim Storage which has 40 cm wall thickness, has a space area of 1500 m^2 and capable to store about 1500 200-l drums and 500 950-l concrete shells

2 1 3 Package Identification

Waste generator is responsible to put package identification The identification is carried out immediately after the wastes are ready in the containment or package Package identification is carried out by attaching placards on different sides of the package surface or on four sides of a freight container on the package surfaces The identification may include package category, contents, and activity For the waste transporter a placard is also needed to show that the vehicle is commonly used to transport radioactive wastes In a broader scope, a transportation information system is applied on every radioactive waste transferred from the facilities Transportation workers need to fill the Radioactive Waste and Contaminated Material form in five copies, each copy will be sent to the division of waste process, the division of occupational and environmental safety, the regulatory body, the consignor, and the sub-division of transportation

2 2 WASTE TRANSPORTATION

Transport Criteria

To conduct a radioactive waste transportation, National Regulatory Body has set up a safety criteria These criteria are primarily based on the IAEA Regulations, and Indonesia transport regulations Requirements include criteria concerning loading of tanks and accumulation of packages, i e , 1) for single conveyance, the radiation level under conditions likely to be encountered in routine transport shall not exceed 200 mrem/h at any point on, and 10 mrem/h at 2 m from, the external surface of the conveyance, 2) except in the case of shipment for single conveyance with no packages containing missile material Category II-Yellow or Category III-Yellow, the total number of packages aboard a single conveyance shall be so limited that the total sum of the transport index does not exceed 50

For liquid waste transportation, transport criteria requires that there are no volatile organic that are explosive and corrosive to the tank and the specific activity of waste must not exceed 0 1 Ci/m^3

Vehicles

For liquid waste and spent resin transportation, the RWMC operates two trailers which are capable of being pulled by a common tractor Each trailer has a fixed tank made of SS 316 L Each has a tank capacity of 6 m^3 for liquid waste and of 1 5 m^3 for spent resin

For solid waste transportation, a specifically designed truck is available The truck versatility make it also possible to transport organic liquid wastes contained in jerrycans, or sealed spent sources The truck has a fixed freight container made of SS 304 The container dimension is 4 x 2 5 x 2 8 m

All of those trucks are provided with a fixed biological shielding to protect the driver/operator while driving in a safety level, therefore the specific activity of any waste transported must not exceed 0 1 Ci/m^3

Transportation Workers

Both liquid- and resins-waste transporters are operated within SNRC area only They are suitable for local operation by a single driver/operator With a Radiation Protection personnel monitoring the safety, the operator is responsible for driving the vehicle, connecting and disconnecting the transfer hose(s) and controlling the transfer of waste while communicating with the facility's operating personnel

The solid waste transporter is operated in and out of the SNRC Besides a radiation protection personnel, the driver is accompanied by a helper

Radiation Dose

The following whole body criteria is applicable to the design of the transporters

$<$ 200 mrem/h contact dose

$<$ 10 mrem/h at 2 m from the trailer

$<$ 10 mrem/h in the drivers cab and at the waste transfer station

The Role of Sub-Division of Transportation

The Sub-Division of Transportation's main task is to evaluate and continuously to solve problems and difficulties in waste transport management The sub-division is also responsible to conduct R & D activities to guarantee the safety and security of the transport of radioactive wastes. It is recognized a need to promote professionalism in waste transportation. The effort among others is done by promoting better understanding of waste transport regulations for all transportation workers.

3. SUMMARY

In terms of the magnitude of problems and the efforts already taken, waste handling and transportation practices follow Transport Regulations issued by the Indonesia National Regulatory Body.

In general, the radiation exposure of the conveyance does not exceed 200 mrem/h at any point on, and 10 mrem/h at 2 m from, the external surface of the conveyance; on every single conveyance the total number of packages aboard was so limited that the total sum of the transport index did not exceed 10.

The present experience in waste transport is assumed adequate. However, specialists in waste transportation is still needed to adopt any development in the field of waste transportation, especially in the waste transport regulations.

References:

1. Government Regulation of The Republic of Indonesia No. 13/1975 on the Transport of Radioactive Materials.
2. International Atomic Energy Agency, Regulations for the Safe Transport of Radioactive Materials, Safety Series No. 6, IAEA, Vienna (1985).
3. Radwaste Management and Technology Centre, Subdivision of Transportation, Transportation Notes and Log-Book Serpong, (1989).
4. International Atomic Energy Agency, Handling, Conditions, and Disposal of Spent Sealed Sources, TECDOC-548, IAEA, Vienna, (1990).

PROBLEMS RELATED TO THE TRANSPORT OF LOW AND INTERMEDIATE LEVEL RADIOACTIVE WASTE IN THE REPUBLIC OF CROATIA

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Abstract

In the Republic of Croatia, there is a project on the facility for low (LL) and intermediate level (IL) radioactive waste disposal. Within the tasks being performed by Hazardous Waste Management Agency, there are many preliminary proceedings related to the project of the construction of the repository for low and intermediate level radioactive waste. Among them are, for example, elaboration of criteria for the most suitable site selection, choice of appropriate type of the repository, repository project design, elaboration and supervision of the project documentation, and the like.

Part of the enterpreneurial actions refers to the problems related to the transport of low and intermediate level radioactive waste from the place of its generation or storage up to the location of final disposal site.

In this phase of the preliminary works – prior to the site selection and further working out of project documentation for the facility, it has been possible and necessary to commence with the certain study papers related to the problems of transport of low and intermediate level radioactive waste from nuclear power plant Krško and other collecting centres in Croatia up to the location of disposal site.

Therefore, during 1992/1993 the first version of the generic study related to the transport was worked out with the aim of preparing the high-quality literature, technical and investment support for further working out of the project.

The study is conceived as the literature abstract of the status on the territory related to the transport of low and intermediate level radioactive waste with the final considerations containing the recommendation of the system related to the transport for the requirements of the Republic of Croatia.

1. INTRODUCTION

Republic of Croatia is one of the newly independent countries in the middle-south of Europe. Although there is not any nuclear power plant on its territory Croatia is faced with the problem of radioactive waste disposal. That is so not only for the radioactive waste from existing industry, hospitals, research and similar institutes in Croatia, but as a consequence from the very specific problem. Namely, as a part of former Yugoslavia, Croatia financed together with the Republic of Slovenia, the erection of the nuclear power plant Krško. NP Krško is PWR 600 MW power plant, Westinghouse technology, and is located in the Republic of Slovenia, but along the frontier with Croatia, only 25 km form Croatia capital- Zagreb.

Based on that joint ownership, Croatia is obliged to participate in seeking the solution for the safe, long-term and ecologically suitable final disposal of radioactive waste generated by the operation, as well as by the decommissioning, of the nuclear power plant Krško.

Construction of the final repository for the low and intermediate level radioactive waste in the Republic of Croatia is one of possible solutions. So Hazardous Waste Management Agency was established in the Republic of Croatia with the prime task to prepare all preliminary proceedings related to the project of the construction of the repository for LL/IL radioactive waste.

Preliminary proceedings which have to be done, related to this project, include numerous activities, such as: preparing and verification of site selection criteria, selection of appropriate type and technology for final disposal, repository project design, elaboration and supervision of the project documentation, and the like.

Part of the enterpreneurial actions refers to the problems related to the transport of low and intermediate level radioactive waste from the place of its generation or storage up to the location of final disposal site. According to many restrictions and unknown facts regarding this project it has not been possible, at this moment, to prepare exact transport study. But it has been possible and necessary to commence with the certain study papers related to that problem.

So, as a first step towards a more serious consideration of that problem, the first version of the generic study related to the transport was worked out with the main aim of preparing the high-quality literature, technical and investment basis for further working out of the project.

2. STRUCTURE OF GENERIC TRANSPORT STUDY

The study [1] is conceived as the literature abstract of the status on the territory related to the transport of LL/IL radioactive waste with the final considerations containing the recommendation of the system related to the transport for the needs of the Republic of Croatia.

Since the aim of this preparation is to achieve the grounds as serious as possible for the sake of further work in that field, especially when it is known that it is the first time to cover this issue systematically in the Republic of Croatia, the study has been prepared in two basic parts. The first part comprises, let us say, the literature overview of the world status, while the second part is more directly dedicated to the problem of radwaste transport up to the repository in Croatia.

2.1. Overview of the status in the world related to the radwaste transport

The literature overview includes all important factors when considering the problem of transport of radwaste, and is consisted of the following items:

a) An overview of the international criteria, recommendations and guidelines related to the transport of radwaste - types of packing, quality of packing, control, manufacture and testing of packing, means of transportation and standard transport equipment, procedures which have to be constrained in radwaste transport [2],[3],[4],[5],[6], as well as criteria and practice of developed countries having experience in transporting of radioactive materials [7],[8],[9],[10][11];

b) Review of the literature data base referring to that field, with print-out of all abstracts related to the radwaste transport published during last ten years;

c) Preliminary - generic risk assessment for LL/IL radwaste transport, in which fundamental principles of control and physical protection in radwaste transport, world experience with accidents during transport and generic analysis of possible accidental situations and risk assessment according to that, have been presented.

2.2. Problem definition and recommendations for the Republic of Croatia

In this part, the analysis of the situation in Croatia has been presented and according to that, recommendations for the transport system and the transport dynamics have been proposed. This part includes:

a) Review of the Croatian legislation for that field,

b) Analysis of possible transport means (road, train, river), routes and transport equipment, based on the most important factors: safety, frequency of transport and price,

c) Recommendation of radwaste transport model for Croatia.

3. STRESSES FROM THE LITERATURE PART OF THE STUDY

3.1. Some basic facts of radwaste transport

Besides to present the different criteria, possible equipment and procedures for the radwaste transport, the basic aim of this part was to stress two very important points, especially having in mind that this problem is for the first time being considered in our country:

a) to present the basic approach philosophy, and

b) to show basic facts concerning safety of such transport.

a) Talking about basic approach philosophy, these are the milestones :

Transport of radioactive materials is strictly regulated with the numerous international conventions and regulations;

<u>Elementary facts of transport philosophy</u>

* Transport of radioactive materials must be **safe**;
* When this safety is achieved, transport must be **quick and simple**, without unnecessary restrictions, which could negatively effect on safety and efficiency of transport;

<u>Primary purpose of the regulations</u>

Protection of all persons which are participants in the transport, as well as public, which could, in this way or another, come in touch with the cargo, and public goods which could be, directly or indirectly, exposed to radiation during transport.

<u>Primary aim of the regulations</u>

To achieve direct protection of:

a) scattering of radioactive materials,

b) potential exposure to radiation caused by careless handling during transport.

<u>Directions of protection</u>

* Restrictions on the **quantity** of radioactive materials which is allowed to be transported;
* Restrictions on the material **activity**;
* Requirements on the **transport equipment quality**;
* **Procedures** which must be obligated by all participants in the transport

b) Talking about safety these are the clue facts:

* Transport of radioactive materials is routine experience in the world, being practised almost fifty years;

* Few millions of radioactive shipments are transported annually;

* Although there were some accidents and incidents in the radioactive material transport, till now is unknown neither one serious accident which would result in serious injuries or death caused by radiation;

3.2. Review of materials published during last ten years, with analysis of titles and themes

Review of the literature data-base with print-out of all abstracts related to the radwaste transport published during last ten years has been done. About 150 titles have been recognized, from books, annual reports to Governments, laws, guidance, regulations, cost-benefit analysis, scientific papers, review articles, problems and solutions from practice to advertisement materials from transport firms, equipment and monitoring producers etc.

Reviewing these titles it was possible to identify the most interesting sources being incident to problems of radwaste transport, from publications to international meetings.

Also, the critical analysis according to themes which were the most frequently published according to different countries, has been done. The frequency of presence of some subject in the published materials directly points out the most important problems for specific country during observed period. For example, the United States, although they have very developed legislation, obviously have great problems with defining authorities and responsibilities between detached competent bodies, as well as with relations between federation and single states, so most of the published materials is related to these problems – relations between states, procedures for licensing the permits, coordination of different states legislations with federal one, cost- benefit analysis for interstate transport and so on.

3.3. Generic risk assessment and safety in the radwaste transport

As the question of safety of radwaste transport is one of the problems which considers the public most, we have to provide some basic answers. The preliminary – generic risk assessment for LL/IL radwaste transport has encompassed the following:

a) World experience with accidents in radwaste transport, including report on few incidents and accidents;

b) Fundamental principles of control and physical protection in radwaste transport,

c) Generic analysis of possible accidental situations and risk assessment according to that.

Without going into details of generic risk assessment, results of the overwiev showed that there was extremely small number of accidents, according to the number of shipments, and that the consequences were not so serious. The analysis of possible accidental scenarios and risks connected to that, shows that risks from radwaste transport, when all safety and physical protection measures are obliged, are very small. Such assessments are in accordance with real situation, when being met with an accident in radwaste transport [3],[5],[12],[13], [14], [15].

4. RADWASTE TRANSPORT IN CROATIA

4.1 Present situation

4.1.1 Existing legislation concerning radwaste transport in the Republic of Croatia

In the Republic of Croatia there is no single law or some other legal act of the radwaste transport regulation . That field is "covered" by detached parts in different laws and regulations. The most important are "Law on the transport of dangerous goods" [16] from 1993 and "Law on the radiation protection and special safety measures in the consumption of nuclear energy" [17] from 1984, which is now under the process of making amendments. Besides these two laws, there are several Regulations [18] which regulating some part of radwaste transport cycles and which have to be incorporated into complete procedure of radwaste transport.

4.1.2 Types, quantities, activities and form of radwaste in Croatia

a) Low and intermediate radioactive waste from NPP Krško

Types of low and intermediate level radioactive waste which are generating during normal operating of NPP Krško could be divided into following categories: spent ion exchange resins (SR), evaporator bottoms (EB), compactable waste (CW), waste which could be supercompacted (SC), filters from HVAC systems (F), others- undefined types of wastes (O). The waste is filling up into 55 gal drums (provided with different additional protection according to activity) in the Krško radwaste process unit and stored in the Krško temporary storage. About 800 drums LLW/ILW is produced annually and Krško storage is almost full. The quantities and activity of accumulated waste are presented in the table I [19].

TABLE I. Summary data of LL/IL radwaste in the NPP Krško storage

TYPES OF WASTE	DRUMS – number of–	VOLUME (m^3)	ACTIVITY (Bq)	SPECIFIC ACTIVITY (Bq/m^3)
SR	831	170	$2,5 \cdot 10^{13}$	$1,5 \cdot 10^{11}$
EB	6.011	1.232	$7,7 \cdot 10^{12}$	$6,2 \cdot 10^{9}$
SC	617	127	$5,3 \cdot 10^{11}$	$4,2 \cdot 10^{9}$
CW	585	120	$4,5 \cdot 10^{11}$	$3,7 \cdot 10^{9}$
O	111	23	$2,0 \cdot 10^{10}$	$8,9 \cdot 10^{8}$
F	86	18	$1,8 \cdot 10^{12}$	$1,0 \cdot 10^{11}$
SUMMARY	8.241		$3,5 \cdot 10^{13}$	$2,6 \cdot 10^{11}$

Although, according to the activity measurements made at the moment of filling up the drums in the NPP radwaste process unit, almost 85 % drums belongs to the category of intermediate level waste, that fact have not great significance for planning the transport equipment. The more relevant data is that in the NPP storage only about 9%[1] drums have surface contact dose greater than 2 mSv/h, which require special protection in handling and transport[2].

[1] *According to data from NPP, on July 1st 1992, from 8.351 drums 7.629 had contact dose smaller or equal to 2mSv/h.*

[2] *2mSv/h is law threshold value above which it is necessary to apply additional safety measures in handling and transport of radioactive waste.*

a) Low and intermediate radioactive waste from hospitals, institutes and industry

Radioactive waste, generated in medicine, institutes, research and industry is now stored in two temporary storages in Zagreb. Data are presented in the table II [20].

TABLE II. Summary data about radioactive materials and radioactive waste storaged in the Republic of Croatia

STORAGE	VOLUME (m^3)	ACTIVITY (Bq)	SPECIFIC ACTIVITY (Bq/m^3)
1. IMI*	8	$1{,}1 \cdot 10^{12}$	$3{,}5 \cdot 10^{11}$
2. IRB**	52	$0{,}3 \cdot 10^{12}$	$3{,}7 \cdot 10^{8}$
3. Other institutions	2		
SUMMARY	62	$1{,}4 \cdot 10^{12}$	

* Institute for Medical Research, Zagreb
**Institute Rudjer Bošković, Zagreb

4.2. Key project data

4.2.1. Assumptions, restrictions, requirements and criteria for the project work out

As there were a lot of unknowns in this project, and according to necessity to make some project limits, number of restrictions, requirements and criteria [21],[22] were implemented during project work out.

The major project restrictions
* Unknown location of the Repository;

* Undetermined status of the Croatian nuclear programme (question of quantities of radwaste);

* Unknown definite agreement between Croatia and Slovenia– concerning NPP Krško radwaste sharing;

* Unknown routes for radwaste transport (unknown repository location!).

Requirements and criteria for project work out
1. The location of the future repository will be within 200 up to 250 km from NPP Krško;
2. The repository will be located in the relatively thinly populated area, including implicitly the relatively trafficless area as well, out of main transport routes and out of the railway net;
3. Transport equipment must be in accordance with existing form of radwaste in the NPP Krško storage;
4. Analysis of transport dynamics is made on basis of overall quantities of radwaste in Croatia plus 50% radwaste from NPP Krško;
5. Repository would be in function in the year 2000;
6. It would be possible to accept max. 2 transport vehicles with special protection and undefined number of unprotected vehicles on the repository site;
7. All radwaste collected in the NPP storage would be transported to the repository during one year.

4.2.2 Quantities and form of radwaste for future transport

TABLE III. Estimation of radwaste quantities [19],[20]

SOURCE	Quantities in the year 2000 m^3	number of drums	Quantities in the year 2050. m^3	numbers of drums
Radwaste from NPP Krško (half quantities) – operating period – decommiss. period	1.650	8.250	3.300 5.500	16.500 27.500
Radwaste from other sources	92	460	307	1.535
TOTAL	**1.742**	**8.710**	**9.107**	**45.535**

4.3. Analysis of possible transport models

The analysis have been made upon possible transport models, needed transport equipment and transport frequency.

4.3.1 Transport model

Possible transport models: by rail / road / river, or some type of combination, have been analyzed, according to the basic factors – safety, frequency and price.

Railway transport, common in some Western European countries, is suitable for greater amounts of radwaste, transported more or less continuously, with existing railway network. In our case such model shows numerous disadvantages:

1. According to literature data, even in cases of great amounts of radwaste, this type of transport is more expensive than transport by road;
2. Analysing potential sites in Croatia, 200 km to 250 km from NPP Krško, it would probably be hilly area, relatively thinly populated, very probably without existing railway net, so model of railway transport should include the erection of the missing rails, and that, for transport of such a little quantities of radwaste, is not economically reasonable;
3. Mixed transport model, which include construction of reloading station for loading the cargo to road vehicles, includes not only construction costs for such a station, but also maintenance costs, monitoring costs, costs for man-power, security costs and so. Combined transport model brings up additional requirements on the unification of transport equipment, which in that case should be adequate for railway, as well as road transport;
4. As one more step in process of handling with radwaste, reloading station is additional risk generating place.

At the other hand almost all areas in Croatia are connected with road network. Additionally, road transport is more flexible, what is in our case, due to many unknowns, of great significance.

So, the solution we consider the best at this moment, is the transport by road. Although, that model would demand some additional works and costs its obvious that expenses will be the further to less.

4.3.2 Transport vehicle and equipment

There are a lot of different types of transport vehicles and equipment for LL/IL radwaste on the market. As form of storage waste is defined (55 gal drums, with or without shields) the proposed equipment consists of container type trailer for LL waste and standard protection vessel for IL waste transport. Transport vessel is placed on the special three-axle trailer-platform which is usually used for the transport of heavy cargo.

The same standard transport vehicle, power about 280 KS, is suitable for pulling both types of trailers.

4.3.3 Transport dynamics

According to recommended way of transport and transport equipment, calculated beginning of shipments in the year 2000 and the requirement that all accumulated waste have to be transferred from the NPP storage to the repository during first operational year, observed period of time is divided into three phases:

Phase	Period of time	Threshold year	Event
I	2000 - 2001	2001	First working year of the repository
II	2001 - 2020	2020	Closing the NPP Krško
III	2020 - 2050	2050	Closing the repository

In considering the transport dynamics it is obvious that the first year (in which all accumulated waste must be transported) is critical for planning transport frequency. Calculation of the transport dynamics is presented in the table IV:

TABLE IV Calculation of transport dynamics

Radwaste for transport	Numbers of drums	Number of drums per one shipment	Number of shipment
Without additional protection	11.000	96	**115** (114,6)
With additional protection	1.100**	14	**80** (78,6)
TOTAL	12.000*		**195**

* Total estimated amount of radwaste drums (9.100 drums) from Table III, has been enlarged for 30 % for the case of unpredicted rise of waste for any reason;
** According to findings that about 9% drums need special protection

As calculation is made on the basis of 22 working days per month, eleven months per year, (242 working days/year) **it would be possible to transport all accumulated radwaste during one year with only one transport vehicle.**

Transport dynamics for the second phase - from year 2001 till 2020 - is not on the critical pathway due to small quantities of waste. Radwaste could be collected in the temporary storage during one, or more years, and then transported in one transport campaign. The third phase is long time off and is not to be considered now.

4.4. Recommended transport model

The road transport model, i.e. the standard vehicle with two types of trailers, have been suggested.

TABLE V Characteristics of the vehicle and trailers

PROPERTY	VEHICLE	TRAILER TYPE I	TRAILER TYPE II
Power (ks)	280	/	/
Length (m)	cca 7	12	12
Width (m)	2,5	2,5	2,5
Height (m)	3,1	3,9	3,1
Weight (kg)			6.300
Carrying capacity (kg)	/		26.000
No.of drums	/	-96-	-14-

The type I trailer, for the transport of LL radwaste is container type, without any additional protection; It can accept totally 96 drums (55 gal), posited in two levels.

The type II trailer, for the transport of IL radwaste, is three axle trailer- platform- on which the standard protection transport vessel for radioactive waste, with total capacity of 14 drums (55 gal), can be put.

The characteristics of the recommended vehicle and trailers are shown in the Table V, and the appearance and dimension of vehicle and vessel in the figures 1 and 2.

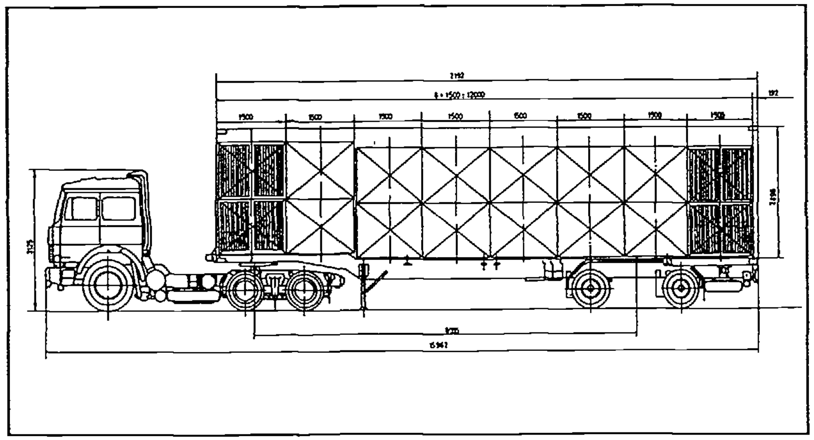

FIGURE 1. Transport vehicle for LL radwaste, container type trailer

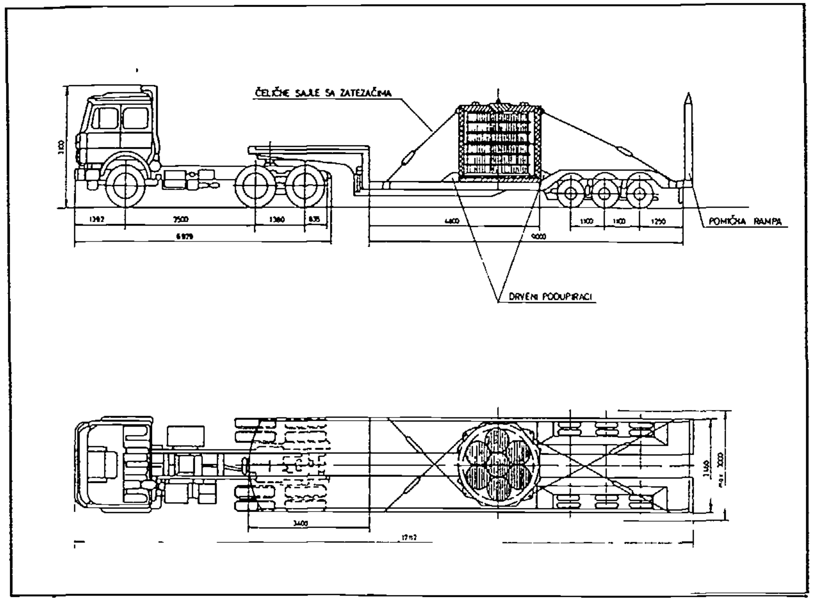

Figure 2. Transport vehicle for IL radwaste, with transport protection vessel

5. CONCLUSION

Low and intermediate level radioactive waste transport problems, from the place of the storage up to the site of the potential location for final repository, have not been more seriously analyzed so far in the Republic of Croatia. As the first step towards the systematic approach to that problem, the extensive literature data and other relevant backgrounds have been collected at the one place and have been processed. Also, the analysis of possible systems for the transport in the Republic of Croatia have been made.

The system of road transport by means of standard vehicle with two types of trailers–three–axle platforms for the transport of intermediate level radioactive waste and container type for low level radioactive waste, have been suggested, and needed transport dynamics calculated.

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THE TRANSPORT OF LOW AND INTERMEDIATE LEVEL WASTE FROM SPANISH NUCLEAR FACILITIES

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Abstract

ENRESA is a state company responsible for the management of low and intermediate level wastes including transport. An average of 9000 (220 litres) drums/year will be transported to a disposal facility. Management includes the assurance of compliance with the acceptance criteria of the facility, planning as well as compliance with transport legislation. Emergency response is also taken into account.

1. INTRODUCTION

Royal decree of July 4, 1984 stablished the National Company for Radioactive Waste Management (ENRESA) as a state fully owned corporation. Among other duties indicated in this Royal Deree, ENRESA is responsible for the organisation of transport systems as well as the operation of the waste disposal facilities.

All those activities are supervised by the Nuclear Safety Council and the Ministry of Energy and Industry.

2. THE TRANSPORT SYSTEM

ENRESA has commissioned in October 1992, a surface disposal facility of El Cabril for low and intermediate level wastes in Southern Spain, while all the Nuclear Centres are situated in Center and Northern Spain. The distance from the eleven Nuclear Facilities to El Cabril ranges between, 350 Km (Almaraz NPP) to 950 Km (Ascó NPP). (Fig. 1)

For over three years ENRESA has been studing, together with the Railway company, the use of railway in the transport of low and intermediate level waste.

The viability study had following constrains:

- All but two nuclear facilities are situated far away from any railway.
- A branch close to El Cabril, nowadays only in use for coal transport, should be repaired for some 50 Kms.
- A bimodal road + railway + road system should be implemented with special trains and the condition that all transport should be done in less than one and a half day journey.

SPANISH NUCLEAR FACILITIES

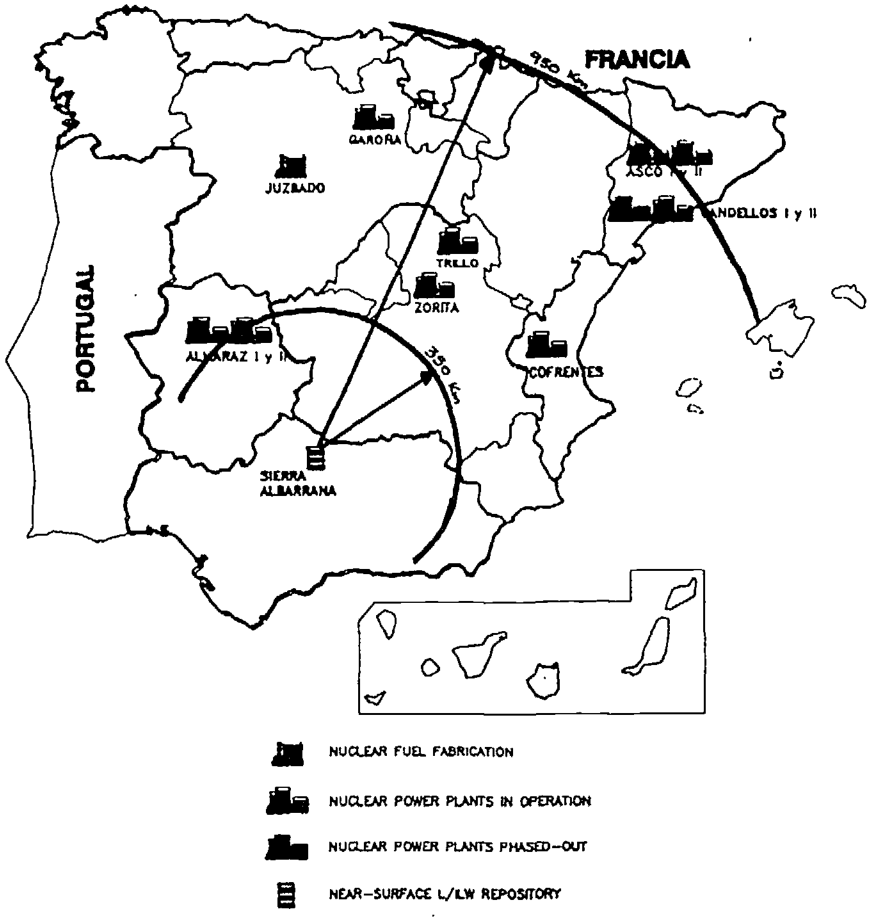

FIG.- 1

The conclusions of the study have demonstrated that, from the economical point of view, the use of railway was not viable because of the necessary investments (the infrastructure increased the cost of road transport in some 100%), and the complexity of the logistics. In any case, more than 15% of the over all distance had to be done by road using the secondary network.

In that a case the unique transport mode available is road transport.

3. CHARACTERISTICS OF THE STORAGE FACILITY

The facility of El Cabril is a surface storage facility with a capacity of 35.000 m^3. Once the waste packages have arrived to the conditionning building, they are introduced in a cubicle shaped concrete container, with a capacity of eighteen 220 l. drums or forty five supercompaction pellets. Containers with wastes are filled with cement mortar. The final 25 Tms container has passed successfully the IP-II tests.

After curing time the container is transported within the facility to the storage platforms.

Three different halls are available for discharge, depending on dose rate of the package and the conditionning process.

Hall for compactibles wastes

In this nave, vehicles that transport 220 l. drums with compactible waste and dose rate of less than 2 mSv/h in contact are discharged. A crane picks up the drums and puts them onto a roller conveyor that feeds the 1200 T supercompactor.

Hall for low dose type of wastes

In this nave, vehicles that transport conditioned wastes packages with a done rate of less than 2 mSv/h are discharged. The drums are introduced in the concrete container directly from the truck.

Hall for higher dose type of wastes

All waste packages with a dose rate in contact ranging from 2 to 50 mSv/h, are only discharged in this nave, in a similar way as low dose type.

In all three cases the trucks are discharged just upon arrival.

4. ACCEPTANCE OF THE PACKAGES

The authorization of El Cabril gives indications of how the packages have to be accepted by ENRESA for long term storage. Depending on the specific activity (table 1), packages are classified in two levels of characterisation.

In both cases, the acceptance follows similar paths:

- Description Document: It is the document prepared by the producer which indicates the fabrication process of the package and the main characteristics of the wastes and the conditioning matrix. It has to be approved by ENRESA before production starts.

TABLE 1

CHARACTERISATION LEVEL

Level 1

Solid wastes, or wastes which have been solidified by being incorporated of immobilised in a characterised solid matrix, satisfying sufficient stability requirements and having mass activities below the following values (NOTE 1):

- Total alpha activity .... 1.85 10^2 Bq/g (0,005 Ci/t) (NOTE 2)
- Individual beta-gamma emitter activity with a period exceeding 5 years (except Tritium) .... 1.85 10^4 Bq/g (0.5 Ci/t)
- Total radionuclide beta-gamma actyvity (nuclides with a period exceeding 5 years) .... 7.4 x 10^4 Bq/g (2 Ci/t)
- Tritium activity .... 7.4 10^3 Bq/g (0.2 Ci/t)

Level 2

Solid wastes, or wastes which have been solidified by being incorporated of immobilised in a solid characterised matrix satisfying strict stability requirements and having mass activity levels equal to or higher than the values given above, and below the following limits (NOTE 1):

- Total alpha activity .... 3.7 x 10^3 Bq/g (0,1 Ci/t)
- Co-60 activity .... 3.7 x 10^5 Bq/g (10 Ci/t)
- Sr-90 activity .... 3.7 x 10^5 Bq/g (10 Ci/t)
- Cs-137 activity .... 3.7 x 10^5 Bq/g (10 Ci/t)

NOTE 1: This activty measured or calculated on the date of package production.

NOTE 2: This weight considered is that of the wastes, the drum or metallic packaging and the immobilization or solidefication material. The weight of shielding materials will not be considered.

- Characterisation Protocol: This document is prepared by ENRESA; it includes the tests to be done over specimens of wastes or on real packages and their results. It also includes the results of the normal transport conditions tests.
- Interpretation of the results and Document of Acceptance.

5. TRANSPORT MANAGEMENT

There are three main activities in transport management:

Planning

Today over 60.000 drums are storaged in the Nuclear Power Plants. Some of those plants have very small remaining storage capacity. More over, the overall production is of some 6.000 drums/year.

The maximum capacity for conditioning is over 10.000 drum/year, equivalent to less than 300 conveyances/year.

Taking into account those data as well as the status of acceptance of the packages, it has been stablished together with the Power Plants a long term planning of 5 years, adjusted every twelve months.

Every six months ENRESA adjusts a twelve months in advance monthly planning. Deliveries have to be definitively set up with forty five days, because the authorities require the final programme with this delay. This programme includes the exact date of delivery and arrival to El Cabril, and the type and quantity of wastes transported.

Reception

The contract between ENRESA and its customers stablishes that packages are handed over to ENRESA when the truck passes the fence of the Power Plant. Packages cannot be given back to the producer if, later, it is discovered that they do not comply with the acceptance criteria. Because of that, ENRESA has to be assured that the packages to be transported are fabricated in accordance with the corresponding documents of acceptance.

Several times a year, while wastes are being conditioned at the power plant, ENRESA makes in service inspections where it checks the quality control programme of the customer, deviations from the acceptance documents and the fabrication of some of the packages.

Furthermore, two other inspections are settled. The first one, right after receiving the packing list and having checked that all the drums are included in ENRESA data base and that they comply with the specific activities of the acceptance documents. In that inspection, the physical state of the packages is verified, as well as their identification, dose rate, weight and labelling.

In the second inspection, at the very moment of the loading of the truck, ENRESA verifies the loading that has to follow a loading chart, the stowage and the sealing of the transport box, as well as the compliance of the conveyance with the legislation.

Transport

Even though ENRESA is responsible for transport, it does not own a fleet. It has stablished contracts with specialised companies which are reevaluated from time to

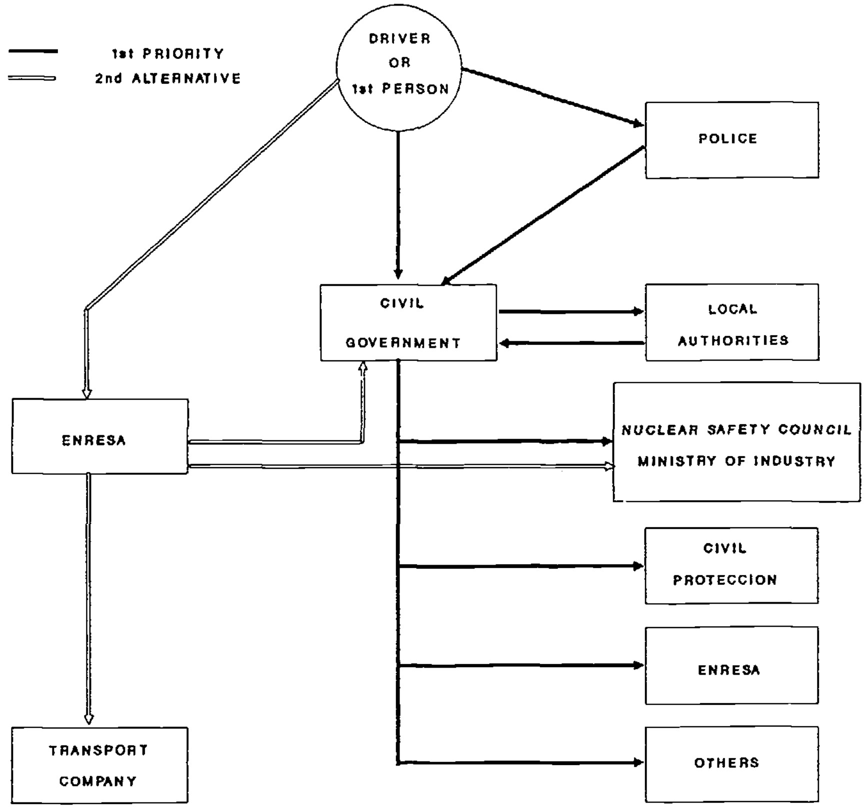

FIG.- 2
EMERGENCY RESPONSE ORGANISATION

time. The trucking companies have quality assurance programmes applicated to the personnel training, maintenance and organization.

ENRESA has also stablished specific training programmes on low and intermediate level wastes transport that are granted to the personnel of the subcontractors.

Transport equipments are of ENRESA own design in order to facilitate the operations at the storage facility. They are put into operation together with the transport company.

Depending on the dose rate of the packages to be transported, there are semitrailers of 3 cm thick iron biological protection, with automatic opening, the possibility of vertical or horizontal loading, and liquid retention valve. For dose rate packages of up to 5 mSv/h a 14 Ton transport box semitrailer of the same characteristics, is under construction and will be in operation during this year. Those semitrailers have a capacity of 45 and 27, respectively, packages of an average of 400 Kg each.

For next year, new transport boxes one that will transport packages up to 10 Sv/h and another one from 10 to 50 mSv/h.

This last one will have to stand Type IP-II tests.

6. EMERGENCY PLANNING

Back in 1988 the Nuclear Safety Council, the Civil Protection Authority and ENRESA stablished a Contingency Plan in which it is clearly indicated the responsibilities and lines of action in case of an emergency (Fig. 2). Today, with the experience gained with over 300 conveyances this plan is under revision.

Independently that any emergency or incident has to be notified to the Nuclear Safety Council who is the leader of the Spanish Radiological Group for Nuclear Events, and emergency response team is ready 24 hours a day. This team who includes Radiological Protection experts as well as waste transport responsibles, can make the first evaluations and coordinate the necessary actions with fire brigade or police.

WASTE GENERATION VOLUMES, CHARACTERISTICS AND DISPOSAL REQUIREMENTS

(SESSION V)

Chairman

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Germany

TRANSPORT AND DISPOSAL REQUIREMENTS FOR SOME SELECTED WASTE SHIPMENTS TO THE KONRAD REPOSITORY

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Abstract

The Konrad repository is planned to accept radioactive waste with negligible heat generation originating from nuclear power plant operation, decommissioning, nuclear fuel cycle and application of radioisotopes in medicine, industry and research. Standardized packagings of different sizes and requirements are used for shipment and final disposal.

Preliminary waste acceptance criteria were derived from safety analyses for the Konrad repository. These criteria for some typical wastes are compared with transportation requirements based on IAEA-Regulations in particular for waste form, packaging and radionuclide inventory. The range of applicability of LSA/SCO-regulations for these waste shipments is considered and some aspects concerning the further development of these regulations within the IAEA Revision Process of Safety Series No. 6 are discussed.

1. INTRODUCTION

The Konrad repository in Germany is planned to dispose of radioactive waste with negligible heat generation, that means radioactive waste of low and intermediate level activity. The waste arises from nuclear power plant operation, decommissioning, nuclear fuel cycle industry and application of radioisotopes in medicine, industry and research.

In the whole waste management field different processes have to be considered like conditioning, interim storage and final disposal together with transportation as the linking process. It is necessary to pay attention to all the relevant requirements of these processes in particular for a safe waste management.

In the following the transportation and disposal requirements for waste shipments to the planned Konrad repository will be described and compared. The application of LSA-transport regulations to those waste packages will be discussed and some conclusions will be drawn concerning some aspects of the further development of the corresponding transport regulations.

2. DISPOSAL REQUIREMENTS FOR WASTE PACKAGES

The safety principles and basic criteria for underground disposal in Germany are described in /1/. Based on these criteria a site specific safety assessment of the Konrad repository was performed taking into account the operational phase of the repository as well as the post-operational period. It includes the waste packages intended to be disposed of, the technical concept of the repository and the overall geological situation /2/. Based on the results of this assessment preliminary waste acceptance criteria were derived concerning waste form, packaging and radionuclide inventory /3/.

The permissible radionuclide inventory per waste package is specified

(a) for various waste forms and packaging tightness levels resulting from safety assessment of normal operation, and
(b) for various waste form groups and packaging integrity levels (waste container class I and II) resulting from safety assessment of incidents, and
(c) for various packaging types to limit decay heat resulting from safety assessment of decay heat influence upon host rock, and
(d) for various packaging tyes to quarantee criticality safety in case of fissile contents resulting from criticality safety assessment.

These limits are independent of one another and the most restrictive one has to be applied /3/.

The waste itself must be in a solid form, has to meet further basic requirements and must be assigned to one of the six waste form groups:

01 - Bitumen and plastic product
02 - Solid matter
03 - Metallic solid matter
04 - Compacted waste
05 - Cemented / concreted waste
06 - Concentrates

as described in detail in /3/. Especially the following two waste form requirements are of relevance to transport issues. This is the requirement of a uniform activity distribution in case of cemented/concreted waste and the limitation of fissile material concentration to 50 g per 0.1 m^3 of waste volume.

There are various types of packagings as shown in Table I and Figure 1 which are standardized according to operational requirements of the Konrad repository. Two of the main basic requirements /3/ they have to meet are

- to comply with the external dimensions and gross volumes as give in Table I, and
- to be designed in such a way that, when filled, they can be stacked over a hight of at least 6 m without adverse effect their tightness and integrity.

The waste packagings can be assigned to two waste container classes having different integrity levels concerning their mechanical and thermal stability under incident conditions in addition to the basic requirements /3/:

Waste container class I:
The design of the waste packagings is such that, up to an impact velocity of 4 m/s, their integrity is

Table I: Standardized packagings for the disposal of radioactive waste in the Konrad repository /2/

No	Designation	External dimensions Length/ diameter mm	Width mm	Height mm	Gross volume m^3
01	Cylindrical concrete packaging type I	Ø 1060	–	1370 [1]	1,2
02	Cylindrical concrete packaging type II	Ø 1060	–	1510 [2]	1,3
03	Cylindrical cast iron packaging type I	Ø 900	–	1150	0,7
04	Cylindrical cast iron packaging type II	Ø 1060	–	1500 [3]	1,3
05	Cylindrical cast iron packaging type III	Ø 1000	–	1240	1,0
06	Container type I	1600	1700	1450 [4]	3,9
07	Container type II	1600	1700	1700	4,6
08	Container type III	3000	1700	1700	8,7
09	Container type IV	3000	1700	1450 [4]	7,4
10	Container type V	3200	2000	1700	10 9
11	Container type VI	1600	2000	1700	5,4

[1]) Height 1370 mm + 90 mm lifting lug = 1460 mm
[2]) Height 1510 mm + 90 mm lifting lug = 1600 mm
[3]) Height 1370 mm for KfK type
[4]) Stacking height 1400 mm for KfK type

Container materials are e g sheet steel, reinforced concrete or cast iron.

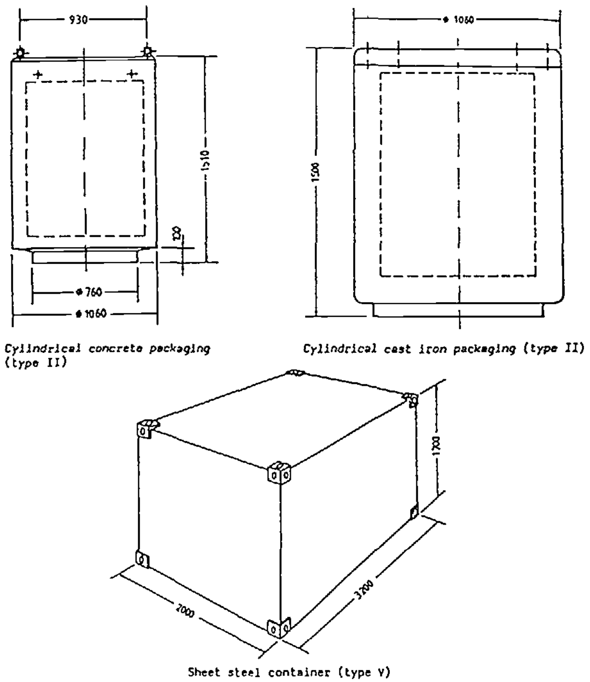

Fig. 1: Basic design of standardized packagings for radioactive waste /2/

preserved to an extent that in the case of a subsequent thermal impact the access of oxygen to the waste form is so limited that flammable waste forms with melting points above 300°C do not burn away with a naked flame, but pyrolyse.

Waste container class II: The design must
- withstand a dropping from a height of 5 m onto an unyielding target in such a way that the leak rate after the drop does not exceed 10^{-4} Pa·m^3/s,
- ensure in the case of fire with a temperature of 800°C during one hour that the leak rate prior to the fire is less than 10^{-5} Pa·m^3/s and that the integral leakage rate from the gas plenum of the packaging during the fire and a cooling phase of 24 hours does not exceed a value of 1 mole.

Finally each waste package must not exceed the following dose rate and contamination limits, according to /3/:

Dose rate limits:

at surface - maximum value	: 10 mSv/h
at surface - mean value	: 2 mSv/h
at 1 m distance from cylindrical packages	: 0.1 mSv/h
at 2 m distance from containers	: 0.1 mSv/h

The non fixed surface contamination is limited to
- 0.5 Bq/cm^2 for alpha emitters having an exemption limit[1)] of 5 x 10^3 Bq
- 50 Bq/cm^2 for beta emitters and electron capture nuclides having an exemption limit[1)] of 5 x 10^6 Bq
- 5 Bq/cm^2 for other radionuclides

Based on all these waste acceptance requirements a safe handling and disposal of all waste packages in the Konrad repository is ensured.

3. TRANSPORT REQUIREMENTS FOR WASTE PACKAGES

The radioactive waste in Germany has to be shipped according to the GGVS /4/ on road and the GGVE /5/ on rail, which are based on the IAEA Transport Regulations, Safety Series No. 6 (SS6) /6/.

The regulations for low specific activity material (LSA) of SS6 are mainly applied in case of waste shipments which leads to the following requirements for the relevant waste packages.

a) the waste form has a limited specific activity and can be assigned to LSA-II or LSA-III category according to para 131 (b) or (c) of SS6.

[1)] Exemption limit according to the german radiation protection ordinance

In case of LSA-II the activity is distributed throughout the waste form and the estimated average specific activity does not exceed $10^{-4} A_2/g$.

In case of LSA-III it is required that
(i) The radioactive material is distributed throughout a solid or a collection of solid objects, or is essentially uniformly distributed in a solid compact binding agent (such as concrete, bitumen, ceramic, etc.);
(ii) The radioactive material is relatively insoluble, or it is intrinsically contained in a relatively insoluble matrix, so that, even under loss of packaging, the loss of radioactive material per package by leaching when placed in water for seven days would not exceed 0.1 A_2; and
(iii) The estimated average specific activity of the solid, excluding any shielding material, does not exceed $2 \times 10^{-3} A_2/g$.

b) LSA-II and LSA-III material has to be shipped in industrial packagings Type 2 and 3 (IP-2, IP-3) according to para 426 of SS6. They must meet the general design requirements (para 505-514) and, in addition, in case of IP-2 the free drops test (para 622) and the stacking test (para 623), and in case of IP-3 all Type A tests (paras 621-624) and Type A design requirements (paras 523-537) for solids. By these tests it is demonstrated that the package can withstand normal conditions of transport. Alternatively freight containers may also be used as IP-2 or IP-3 provided they meet the general design requirements and the ISO 1496/1-1978 requirements ("Series 1 Freight containers - Specifications and Testing - Part 1: General Cargo Containers") according to para 523.

c) The total quantity of LSA material in a single IP-2 or IP-3 package shall be so restricted that the external radiation level at 3 m from the unshielded material does not exceed 10 mSv/h (para 422). In addition in case of combustible solid LSA-waste in IP-2 or IP-3 the total activity per conveyance is limited to 100 A_2 (para 427).

d) IP-2 or IP-3 waste packages must not exceed the following external radiation and contamination limits:
Dose rate limits:
(i) under exclusive use conditions:
- at the package surface : 10 mSv/h
- at the external surface of the conveyance : 2 mSv/h
- at 2 m from the ext. surface of the conveyance : 0.1 mSv/h

(ii) under non exclusive use conditions:
- at the package surface : 2 mSv/h
- at 1 m from the package surface : 0.1 mSv/h
- at the external surface of the conveyance : 2 mSv/h
- at 2 m from the ext. surface of the conveyance : 0.1 mSv/h

The non fixed surface contamination is limited to
- 4 Bq/cm^2 for beta and gamma emitters and low toxicity alpha emitters, and
- 0.4 Bq/cm^2 for all other alpha emitters

If radioactive waste materials can be classified as surface contaminated object (SCO) according to para 144 of IAEA Regulations than also IP-2 packagings have to be used with the requirements as decribed above for LSA-material.

4. COMPARISON AND APPLICATION OF DISPOSAL AND TRANSPORT REQUIREMENTS TO WASTE PACKAGE SHIPMENTS

A waste package intended to be disposed of in the Konrad repository has to meet both transport and disposal requirements. In comparing both the following conclusions can be drawn concerning the main criteria of a waste package.

a) External radiation and surface contamination level:

These criteria are limited in a similar way, so that in most cases compliance with transport requirements results also in compliance with disposal needs.

b) Waste form characteristics:

Basically each waste form group may be categorized as LSA-Material provided that compliance with LSA-Material defintion (see chapter 3.a)) can be demonstrated. In case of waste form group 05 e.g. uniform activity distribution is already required by disposal needs, which is also an important criteria for LSA-Material. To classify such waste as LSA Material it remains to check specific activity limits and in case of LSA-III leachability criteria (chapter 3.a)(ii)).
Due to the disposal limitation of fissile material concentration to 50 g per 0.1 m^3 which is in compliance with the transport requirement of fissile exempted material (para 560 (d) /6/) such a waste package is excepted from transport requirements for packages containing fissile material, provided the activity is uniformly distributed.

c) Activity inventory:

The activity limits are specified differently. For disposal, as described in chapter 2, there are various limits derived from considerations on normal operation,

incidents, decay heat and criticality, depending in particular on waste form and package characteristics. For transportation, as described in chapter 3, the activity inventory is limited by the specific activity of waste according to LSA-II or LSA-III specifications in an IP-2 or IP-3 packaging and by the limit of 10 mSv/h at 3 m from the unshielded waste contents. In addition, in case of combustible waste the total activity limit per conveyance of 100 A_2 has to be considered. All these limits have to be obeyed and the most restrictive one determines the permissible activity inventory of the waste package. This has to be done based on the real waste package specifications in particular the radionuclide mixture contained in the waste and the material and the geometry of the waste and the packaging. As a simplified example those activity limits are given in Tables 2 and 3 for Co-60 and Cs-137 as relevant nuclides of the activity inventory of waste from nuclear power plant operation. In this case the total activity for most of the waste forms is mainly restricted by the external unshielded radiation limit of 10 mSv/h from transport requirements and by the decay heat limits from disposal requirements.
In principle, compliance with this 10 mSv/h radiation limit is to be expected for waste shipments to the Konrad repository, because the standardized packagings have a limited shielding capability. The cylindrical concrete packagings are designed with a wall thickness up to 200 mm and the cylindrical cast iron packagings up to 160 mm wall thickness. Experience with those waste package from nuclear power plant operation showes that the unshielded contents meets the 10 mSv/h criteria /7/.

d) packaging

For disposal, as described in chapter 2, the package has to meet basic requirements which take into account normal operation conditions of the repository and in addition the mechanical and thermal test requirements of waste container class I or II representing assumend incident conditions within the Konrad repository. For transport (see chapter 3.b)) general design requirements and specific test requirements of IP-2 or IP-3 package have to be met, to demonstrate that the waste package can withstand normal conditions of transport including minor mishaps.

In comparing these requirements the following conclusions may be drawn. The IP-2 or IP-3 qualification of a package gives high credit to meet the basic package requirement from disposal and in particular meets the mechanical integrity requirements under impact velocity condition of 4 m/s for waste container class I up to a package mass of 10 t (height of free drop is more than 0.8 m). In addition, for waste container class I qualification, the

Table II: Activity limits per waste package in Bq from disposal requirements for Co-60 and Cs-137

		Co-60	Cs-137
Normal operation limits depending on packaging tightness		$3{,}7 \times 10^{13} ... 3{,}7 \times 10^{17}$	$3{,}7 \times 10^{13} ... 3{,}7 \times 10^{17}$
Incident limits waste container class I, for various waste form groups	01	$3{,}5 \times 10^{10}$	$3{,}6 \times 10^{10}$
	02	$1{,}2 \times 10^{12}$	$1{,}3 \times 10^{12}$
	03	$4{,}3 \times 10^{12}$	$4{,}5 \times 10^{12}$
	04	$1{,}1 \times 10^{13}$	$1{,}2 \times 10^{13}$
	05	$3{,}5 \times 10^{13}$	$3{,}6 \times 10^{13}$
	06	$3{,}5 \times 10^{13}$	$3{,}6 \times 10^{13}$
Waste container class II, for all waste form groups		$8{,}6 \times 10^{14}$	$9{,}1 \times 10^{14}$
Decay heat limits for cylindrical concrete packaging		$2{,}6 \times 10^{12} ... 2{,}9 \times 10^{12}$	$4{,}5 \times 10^{12} ... 4{,}9 \times 10^{12}$
cylindrical cast iron packaging		$1{,}7 \times 10^{12} ... 2{,}9 \times 10^{12}$	$2{,}8 \times 10^{12} ... 4{,}9 \times 10^{12}$
container		$7{,}8 \times 10^{12} ... 2{,}2 \times 10^{13}$	$1{,}3 \times 10^{13} ... 3{,}7 \times 10^{13}$

Table III: Activity limits per waste package from transport requirements (LSA-Material for Co-60 and Cs-137

	Co-60	Cs-137
Specific activity limit of waste		
LSA-II ($10^{-4} A_2$/g)	4×10^7 Bq/g	5×10^7 Bq/g
LSA-III ($2 \times 10^{-3} A_2$/g)	8×10^8 Bq/g	1×10^9 Bq/g
Total activity limit according to the external radiation limit of 10 mSv/h at 3 m from the unshielded waste	depending on material (binding agent) and geometry of waste (e.g. in case of Co-60 for concreted waste about 1×10^{12} Bq ... 4×10^{12} Bq per waste package)	

behaviour of the waste package under fire conditions (800°C, 1h) has to be taken into account.
For waste container class II the test requirements are much more severe. But in that case it has taken into account that activity inventories are permitted which exceed the LSA Material specification, resulting in the necessity to use a Type B package for transport which is required to withstand severe accident conditions.

On the basis of model data provided by waste generators, categorized and summarized according to /8/, an estimation concerning the applicability of LSA-criteria to planned waste package shipments to the Konrad repository was performed. It comprises waste with negligible heat generation from reprocessing, nuclear power plant operation, decommissioning, research centres, nuclear industry and collecting depots. Using these model data on activity inventory and waste volume it can be expected that more than 95 % of the waste complies with LSA-requirements concerning the limits of specific activity and dose rate at 3 m from the unshielded waste volume.

5. CONCLUSIONS

For radioactive waste intended for disposal in the planned Konrad repository both transport and disposal requirements have been taken into account in the qualification of the waste package. Concerning the shipment of those waste packages with negligible heat generation it can be expected that more than 95 % of the waste complies with LSA-transport regulations of Safety Series No. 6 /6/ regarding the limits of specific activity and dose rate at 3 m from the unshielded waste volume. Taking into account both transportation and disposal requirements the following conclusions can be drawn with respect to some aspects of the further development of LSA/SCO-requirements discussed within the current revision process of Safety Series No. 6 /9/.

(a) It seems to be more appropriate to keep the present dose rate limit of 10 mSv/h at 3 m from the unshielded LSA/SCO-Material (/6/, para 422) instead to introduce a new limit on the basis of a multiple of A_1 as proposed in /9/.

(b) Waste packages of the container type (see Table 1) are designed according to ISO-Standards but they don't need to meet necessarely ISO-dimension requirements. Dimensions are specified according to technical and operational needs in storage, transport and disposal. To take account of those dimensions the alternative IP-2/IP-3 qualification of freight containers according to para 523 of SS6 should also be applicable to containers with other than ISO-dimensions, provided their external dimensions don't exceed the maximum dimension specification of ISO-Standard.

(c) In case of waste package shipments with negligible heat generation to the Konrad repository there seems to be no need to introduce a so called "transport system approach" /9/ in the Transport Regulations.

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INVENTORY AND CHARACTERISTICS OF CURRENT AND PROJECTED LOW-LEVEL RADIOACTIVE MATERIALS AND WASTE IN THE UNITED STATES OF AMERICA

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Abstract

This paper addresses the quantities and forms of low-level radioactive wastes (LLW) and mixed low-level wastes (mixed LLW) which have been generated and could yet be generated in the U. S. and which could require packaging and transport at some time in the future to storage and/or disposal sites. The current inventory of buried and stored LLW and mixed-LLW in the U. S. that needs to be, or already has been disposed of, in surface facilities was approximately 4.5 million m^3 in 1991. The inventory of LLW is projected to grow to approximately 6.4 million m^3 by the year 2000, and to approximately 25 million m^3 by the year 2010. This will result in an average growth in LLW inventories requiring packaging and transport of about of o.2 million m_3 per year through 2000, and more than 1.8 million m_3 per year during the first decade of the next century. The majority of this growth will result from environmental restoration activities of U. S. Department of Energy facilities. The paper also addresses Greater-than Class C LLW and Special Case LLW which will require disposal in other than surface facilities. Efforts are still underway in the U. S. to better understand the classification and quantities of these materials, but it currently appears that there could be as much as 1 million m^3 of these materials which will require packaging and transport in the future.

The Integrated Data Base (IDB), under U.S. Department of Energy (DOE) funding and guidance, provides an annual update of compiled data on current and projected inventories and characteristics of DOE and commercially owned radioactive wastes. The data base addresses also the inventories of DOE** and commercial spent fuel. These data are derived from reliable information from government sources, open literature, technical reports, and direct contacts.

The radioactive materials considered are spent nuclear fuel, high-level waste (HLW), transuranic (TRU) waste, low-level waste (LLW), commercial uranium mill tailings, environmental restoration wastes, and mixed-LLW. This paper primarily focuses on LLW inventory and characterization. The definitions for various waste classifications [1] are:

** For Revision 9 of the IDB, quantities of DOE production reactor spent fuel that will not be reprocessed will be included; these data will be included in the February 1994 presentation to the International Atomic Energy Agency (AEA).

Spent Fuel: Irradiated fuel discharged from a nuclear reactor. The fuels are assumed to be permanently discharged and eligible for repository disposal. Spent fuel to be processed from government production reactors for national defense is not part of this inventory.

HLW: Highly radioactive material resulting from the reprocessing of spent nuclear fuel. These wastes are mainly liquid wastes resulting from the recovery of uranium and plutonium in a fuel reprocessing plant. They contain fission products that require heavy shielding and provisions for decay-heat dissipation.

TRU: Radioactive wastes that contain more than 100 nCi/g of alpha-emitting isotopes with atomic numbers greater than 92 and half-life greater than 20 years.

LLW: Radioactive waste not classified as spent nuclear fuel, HLW, TRU waste, or by-product uranium mill tailings. In the United States, there are four classes of LLW as defined by the U.S. Nuclear Regulatory Commission (NRC). These are Class A, Class B, Class C, and Greater than Class C (GTCC), and each requires varying degrees of confinement and monitoring.

Commercial Uranium Mill Tailings: Earthen residues that remain after the extraction of uranium from the ores. The isotopes of major concern are ^{226}Ra and ^{222}Rn.

Mixed LLW: Wastes containing both low-level radioactive materials and hazardous chemicals. The hazardous chemicals might be polychlorinated biphenyl (PCB), asbestos, any other U.S. Environmental Protection Agency (EPA) listed or characteristic waste, or any wastes deemed hazardous by state regulations.

Table I shows the summary of all the current and projected low-level and mixed wastes [except GTCC and Special Case wastes (SCW), which are discussed later] shown in the latest revision of the annual IDB spent fuel and radioactive waste inventory report [1]. The inventory does not include any LLW-contaminated soils or any contaminated liquid/gas in storage. The data show a large quantity of radioactive wastes that will have to be disposed. The DOE and NRC are currently addressing the packaging and transportation issues relative to such large quantities of waste.

Table I. Current and projected cumulative quantities of U.S. low-level and mixed low-level radioactive waste [1].
(Quantities are expressed as volume (1 ×10^{3}m^3).)

Waste categories	Actual waste volume 1991	Projected waste 2000	Projected waste 2010
U.S. Government owned:			
LLW (buried and stored)	3,000	3,787[a]	4,769[a]
Mixed LLW	101	Not available	Not available
Environmental Restoration: LLW	Not available	920	18,000
Commercially owned:			
LLW	1,423	1,722	2,055
Mixed LLW	2.1[b]	Not available	Not available

[a] Projections exclude contributions from stored LLW and wastes presently managed as TRU wastes which may be eventually reclassified as LLW.

[b] Stored volume for 1990.

The remainder of this paper will address specifically the inventory and characterization of GTCC and SCWs. As noted above, LLW is classified as Class A, B, C, or GTCC, as defined by the NRC. Wastes in the GTCC category are defined as commercially generated wastes that exceed the definition of Class C wastes and are therefore not suitable for near-surface land disposal. GTCC-like wastes that are generated by the U.S. government (DOE) are called SCWs. The enactment of Public Law 99-240 (Low-Level Waste Policy Amendments Act of 1985) by the U.S. Congress made DOE responsible for disposal of both commercial GTCC LLW and U.S. DOE-owned or generated LLW. Individual states in the United States are responsible, either individually or in cooperation with other states, for the disposition of Class A, B, and C LLW generated from commercial sources. DOE is responsible for the disposal of Special Case wastes.

The U.S. regulations that determine waste classification for near-surface disposal involve two considerations: (1) the concentrations of long-lived radionuclides (and their shorter-lived precursors) whose potential hazard will persist long after precautions such as institutional controls, improved waste form, and deeper disposal have ceased to be effective, and (2) the concentration of shorter-lived radionuclides for which requirements on institutional controls, waste form, and disposal methods are well defined. Waste classes A, B, and C have well-defined near-surface land-disposal requirements. The most stringently regulated class, Class C, must be disposed so that the top of the waste is at a minimum of 5 m below the top surface of the cover or must be disposed using intruder barriers that are designed to protect against an inadvertent intrusion for at least 500 years. LLW exceeding the limits for Class C is not generally acceptable for near-surface disposal. In the absence of specific regulatory requirements, wastes exceeding Class C must be disposed in a geologic repository unless other disposal sites are approved by the NRC.

In 1993, the U.S. DOE Transportation Management Division established the GTCC Packaging and Transportation Working Group to address packaging and transportation issues related to both GTTC LLW and Special Case wastes. The initial key element of this effort is to characterize and quantify both waste categories. The next phase will develop an action plan with recommendations on packaging and transporting of such wastes. Over the past few years, a number of DOE-sponsored reports have been prepared to describe, locate, and quantify low-level radioactive wastes held in both the commercial and government (DOE) sectors. References 2 and 3 document the known commercially generated GTCC. These reports, which were based on several sources, are the most complete available documentation currently for the characterization and quantification of GTCC LLW.

Special Case wastes include a wide variety of forms, sources, and isotopic mixtures. Characterization of these wastes is incomplete, and published documentation has not yet been issued. In this paper, SCWs are divided into the following categories [2]:

Non-Certifiable Defense TRU. Defense-program-generated TRU wastes that are not certifiable for disposal at the Waste Isolation Pilot Plant (WIPP) or for transport in the TRUPACT-II shipping container. Without WIPP acceptance, these wastes currently have no disposal alternative.

Non-defense TRU. DOE-titled TRU wastes generated by DOE Energy Research (ER) Programs, Nuclear Energy (NE) Programs, or an NRC licensee. Currently, WIPP will only dispose of DOE Defense Programs (DP) generated wastes. Therefore, DOE non-defense TRU wastes currently have no disposal alternative.

Specific Performance Assessment Required (SPAR). DOE-titled wastes that contain radionuclides in concentrations greater than those specified for NRC-defined Class C wastes. These are Special Case wastes because they are not generally acceptable for near surface disposal.

Performance Assessment Limiting (PAL). Absorbed tritiated liquids waste, hot-cell wastes from destructive examination of fuels, sludges containing mixed fission products, ion-exchange resins containing transuranics, gauges and dials containing ^{226}Ra, and uranium solids with associated decay products.

Fuel and Fuel Debris. Primarily include DOE-titled fuel and fuel debris used for research and development. These wastes are similar in some respects to waste destined for the spent-fuel and HLW repository. However, most of these wastes are in packaging configurations that are unlike normal commercial fuel elements and may not meet repository waste acceptance criteria (WAC). These wastes are SCWs because of undefined disposal and facility acceptance criteria.

Uncharacterized Wastes. Any containers of waste with unknown contents. These wastes are believed to contain nuclear materials at or near the limits of SPAR or TRU wastes. Further characterization of these wastes will determine their material forms, approximate mass, and levels of radioactivity.

Excess Nuclear Materials. Nuclear materials above the economic discard level (EDL) that either are no longer useful to the current custodians or require processing that is not available because of a lack of capabilities or capacities to recover the useable nuclear materials. Excess nuclear materials include unirradiated nuclear materials, irradiated nuclear materials, and nuclear materials containing decay products. Some of the materials contain Resource Conservation Recovery Act (RCRA)-regulated constituents, which preclude processing because recovery facilities are not permitted by the Act.

Sealed sources. Encapsulated radioactive material whose main purpose is to generate known amounts of radiation. These sources are of special interest because the concentrations of their radioactive material usually make them SPAR waste at the time of disposal.

DOE-Titled, Held by Licensees. Includes wastes or materials that are DOE-titled but held by NRC licensees. These wastes or materials include sealed sources and spent fuel. DOE has provided nuclear materials to licensees through a variety of mechanisms, including contracts, loans, leases, and grants for use in nuclear-research-related fields.

Because of the responsibilities given to it under the Low-Level Waste Policy Amendments Act of 1985, DOE established the National LLW Management Program (NLLWMP) to develop best estimates of GTCC LLW volumes and radioactivities to use in planning for the disposal of this waste. In its report of August 1991, the NLLWMP grouped GTCC LLW into the following generator categories [3]:

Nuclear Utilities: Operators of light water reactors (LWRs) (pressurized-water reactors and boiling-water reactors) are GTCC LLW generators. Operating procedures of LWRs vary from reactor to reactor. Different operating practices and manufacturing designs of individual reactors dictate the potential GTCC LLW that may be produced. These wastes include metals from standard operation procedures, ion-exchange resins, and cartridge filters from decontamination efforts and other decommissioning wastes.

Sealed Sources: Sealed sources are small capsules, generally stainless steel, that contain high concentrations of a single nuclide. Sealed sources are used in a wide range of industrial and medical applications and become waste when they are no longer usable. These sealed sources include well-logging devices, moisture gauges, and medical therapy and calibration devices.

DOE-Held GTCC Waste: Several commercial facilities have generated GTCC LLW, and through contractual agreements DOE has taken possession of those wastes and currently stores them at various DOE sites. It has not been completely determined whether an NRC-licensed facility will be required for disposition of these wastes.

Other Generator Waste: These are generated wastes that do not belong in the previous three categories. These waste generators include Carbon-14 users, fuel fabricators, nuclear research/test reactors, and sealed source distributors.

Table II presents a summary of the current inventory of GTCC LLW and Special Case waste based on refs. 2 and 3 and on unpublished data obtained from the Idaho National Engineering Laboratory (INEL).

Special Case Waste Classification Under Safety Series No. 6

Effort is underway to classify SCW under the Safety Series No. 6 guidelines for Low Specific Activity (LSA). Safety Series No. 6 1985 edition (amended 1990) classifies LSA materials into 3 groups:

(a) LSA-I
- (i) Ores containing naturally occurring radionuclides (e.g., uranium, thorium), and uranium or thorium concentrates of such ores;
- (ii) Solid unirradiated **natural uranium** or **depleted uranium** or natural thorium or their solid or liquid compounds or mixtures; or
- (iii) **Radioactive material**, other than **fissile material**, for which the A_2 value is unlimited.

(b) LSA-II
- (i) Water with tritium concentration up to 0.8 TBq/L (20 Ci/L) or
- (ii) Other material in which the activity is distributed throughout and the estimated average **specific activity** does not exceed 10^{-4} A_2/g for solids and gases, and 10^{-5} A_2/g for liquids

(c) LSA-III

Solids (e.g., consolidated wastes, activated materials) in which
- (i) The **radioactive material** is distributed throughout a solid or a collection of solid objects, or is essentially uniformly distributed in a solid compact binding agent (such as concrete, bitumen, ceramic, etc.);
- (ii) The **radioactive material** is relatively insoluble, or it is intrinsically contained in a relatively insoluble matrix, so that, even under loss of **packaging**, the loss of **radioactive material** per **package** by leaching when placed in water for seven days would not exceed 0.1 A_2; and
- (iii) The estimated average **specific activity** of the solid, excluding any shielding material, does not exceed 2×10^{-3} A_2/g.

Table II shows that for all SCW categories, gross volume (m³) and overall radioactivity (Ci) content are the only two parameters that are known. To classify the wastes as LSA, the following additional information will be required:

(a) Radionuclide(s) present in the various waste streams;
(b) Activity levels (Ci) for each waste stream; and
(c) Mass of the waste stream.

Table II. Summary of estimates of commercial GTCC LLW and DOE-generated/owned SCWs through 1990 [1,3].

Waste category	Volume (m^3)	Number of items	Radio-activity (10^6 Ci)
SCW summary:			
Non-Certifiable Defense TRU	35,500		0.580
Non-Defense TRU	850		0.117
SPAR	34,200		122.0
PAL	5,600		0.08
Fuel And Fuel Debris	8,300		21.6
Uncharacterized	871,000		201.0
Excess Nuclear Materials	125,000		0.796
Sealed Sources	NA[a]	2,570	1.02
DOE Titled, by Others	NA[a]	4,160	0.013
Subtotal	**1,079,450[b]**	**6,730**	**347.0**
Commercially Generated GTCC LLW			
Nuclear Utility Wastes	1,853		65.0
Sealed Sources	6	27,000	0.303
DOE Held GTCC Waste	1,076		0.538
Other Generator Waste	307		0.003
Sub-Total	**3,242**	**27,000**	**66.0**
Total	**1,082,692**	**33,730**	**413.0**

[a] NA = not available

[b] INEL has estimated that the quantity of SCWs could increase by approximately 12,000 m³ during the 5-year period 1991–1995.

This information will determine the LSA classification (I, II, or III) and the required packaging (IP-1, IP-2, or IP-3).

SUMMARY

Table II indicates that approximately 1.08×10^6 m³ of DOE SCW and 3,240 m³ of GTCC LLW waste could potentially exist in the U.S. inventory. INEL has estimated that the quantity of SCWs could increase by approximately 12,000 m³ during the 5-year period 1991–1995. Most of the SCW is uncharacterized waste from underground storage tanks at the Hanford, Washington, site. This inventory is expected to rise as more waste is characterized at various government sites as a result of environmental restoration efforts. The inventory of SCW far exceeds the quantity of GTCC LLW; therefore the GTCC Packaging and Transportation Working Group will concentrate on updating the inventory and characterization of SCW. The characterization will help determine the waste classification (i.e., LSA type) under IAEA regulations. This classification will be instrumental in defining packaging and transportation requirements to transport such large quantities of waste to a future disposal site. This characterization effort can also provide valuable input to IAEA in its evolving definitions of LSA/surface-contaminated objects (SCO) based upon the impact to GTCC/SCW packaging requirements.

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RADIOACTIVE WASTE ASSAY TO VERIFY THE FULFILLMENT OF WASTE ACCEPTANCE REQUIREMENTS

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Abstract

For intermediate storage, transport and final disposal of radioactive waste, the waste package has to fulfill the waste acceptance criteria, as defined by the nuclear regulatory institutions of the various countries. Important especially for transport, storage or final disposal is the radiochemical content of the waste package.

The fulfilment of the waste assay criteria concerning radiochemical content can be verified by the execution of a quality assurance programme from the waste production up to the waste package, ready for final disposal, complemented by computations and/or measurements at the beginning, during and at the end of the waste treatment. A programme to develop measurement systems, instruments and analysis methods is executed in order to make an experimental assay of waste packages possible with sufficient accuracy.

Different assay methods for the determination of fissile and non fissile radionuclide containment in waste are discussed.

1. INTRODUCTION

Radioactive waste management including the transport of radioactive waste is of essential importance in the nuclear industry. During each production through the waste treatment and conditioning, up to the final storage, and consequent transportation of waste, the radioactive inventory has to be known.

To assure the correctness of the inventory it is necessary to execute a good quality assurance programme complemented by computations and/or measurements at the beginning during and at the end of the waste treatment. A good quality assurance system may reduce partially but never totally the need for experimental verification, and is even needed were the measurement methods are unable to reach the required precision.

Further, the waste package to be transported or to be stores has to be verified by an independent institution. This institution needs to dispose of verified measurement method to control the waste package.

2. QUALITY ASSURANCE

The scope of a quality assurance programme is to reduce the costly and time consuming experimental assay of radiocontaminants in waste. The experimental methods have not always reached the required accuracy, which make a quality assurance programme mandatory.

The IAEA (and several Countries), have started a programme in Quality Assurance, preparing a set of documents as a guide for the execution of a Quality Assurance programme for waste treatment, storage and transportation [1-5].

However a quality assurance programme alone is not sufficient and has to be supported by calculations and measurements at the beginning, during and at the end of the waste stream. Each nuclear installation (e.g. a reprocessing plant) has to declare the waste produced.

In the next chapters the instruments and methods and their capabilities which are developed at JRC ISPRA for experimental waste assay are described.

3. WASTE ASSAY

Waste is subdivided in waste containing fissile and non fissile radionuclides. The fissile isotopes can be either plutonium or uranium (or eventually other fissile isotopes of minor importance) or both.

Plutonium in radioactive waste is assayed by passsive neutron techniques, eventually complemented by gamma ray measurements in order to determine the Pu isotopic composition. At JRC Ispa the Time Correlation Analysis (TCA) is applied for the analysis of signal pulse trains of detected neutrons. The basis of the TCA technique is the frequency distributions of signals in fixed observation intervals. Two analysis methods are used. One method utilises a comparison of the measured frequency distributions and Monte Carlo calculated distributions [18, 25]. The second method uses the factorial moments of the measured frequency distributions, and derives from analytical equations the fissile mass [23].

The problem for uranium containing waste is much more difficult. The measurement is based on active neutron interrogation with an external neutron source. Two factors define the correct determination of ^{235}U: the matrix with its moderating properties and the distribution of uranium in the matrix. An R & D project is in execution to attack this problem with the final goal to produce an instrument to assay uranium in waste.

Non fissile waste is measured by gamma scanning. Again two factors determine the precision with which the content of gamma contaminants in waste can be determinated: the absorption and moderation of the matrix and the source distributionl. A gamma scanner and adapted software is currently used.

4. PASSIVE NEUTRON INTERROGATION

An analytical analysis method has been developed for the passive neutron assay technique [13, 23, 24, 26, 27]. It determines the spontaneous fission rate of Pu isotopes with even mass numbers (^{238}Pu, ^{240}Pu, ^{242}Pu). Determination of the total Pu mass requires knowledge of the isotopic composition. The analysis involves treatment of neutrons generated by (α,n) reactions on light nuclei, neutron mulitplication by induced fission, reduced neutron detection probability due to leakage and absorption in the waste matrix, and dead times in the signal pulse train.

Neutrons emitted by a test item are slowed down in a moderator, diffuse there as thermal neutrons and are partially absorbed by neutron detectors incorporated in the moderator assembly. The neutrons absorbed in the detectors are transformed in real time into electric signals, are amplified, shaped and converted into a signal pulse train. Several methods [6-12] are in use for the analysis of this pulse train : the shift register, the reduced variance method, and the variable dead time counter. These instruments deliver two experimental quantities, the total count and the correlated count. This permits the determination of the spontaneous fission rate F_s and either the $(\alpha$-n)-reaction S_α or the probability p that a primary source neutron generates an induced fission. In case of α-contaminated waste, neutron multiplication is less important, however neutron absorption in the waste material modifies the neutron detection probability ε of the moderator-detector assembly. In such measurement conditions a 3 parameter analysis is suitable for a determination of F_s. S_α and ε.

With the triple neutron correlation technique three quantities are available as experimental data : the effective number of singlets, doublets and triplets. This permits to determine three unknowns of Pu contaminated waste i.e. either the spontaneous fission rate F_s the (α,n)reaction rate S_α and the detection probability ε of the unknown Pu distribution in the waste barrel or knowing the isotopic composition F_s, ε and the neutron *multiplication* factor M. Pair and triple correlation neasurements can be performed contemporarily with the Euraton Time Correlation Analyzer (TCA) according to two different methods and analyzed with the developed algortihms [13-17, 24]. A passive neutron assay monitor consists of 3 principal units : the neutron detection head, the analog chains and the neutron signal analyzer.

4.1. The neutron detector head

The fast neutrons generated inside a waste barrel are partly slowed down and absorbed in the matrix of the waste. A fraction of the fast neutrons and of the partially slowed down neutrons enter into a surrounding detector head. Such a detector head has ideally a 4n geometry and consists of a polythene moderator surrounding the barrel. The moderator has thermal neutron detectors incorporated lined, with Cd at all free surfaces. The fraction of fast and epi cadmium neutrons leaking from the waste barrel into the neutron detector head are slowed down in the moderator , diffuse as thermal neutrons and are partially absorbed by the neutron detectors. The Cd liner and the moderator dimensions permit.

1. to fix the exponential decay constant λ to a desired values
2. to ensure a single exponential of the time response function over more than two decades.

The decay time $1/\lambda$ is of the order of 70 µs at an optimized detection probability ε_{max} for spontaneous fission neutrons. An undermoderated detector head with 70% of ε max would have a decay time $1/\lambda$ in the order of 40 µs.

4.2. The Euratom Time Correlation Analyser

The Euratom Time Correlation Analyser [18] is a signal frequency analyser, which measures the signal frequency according to two different principles. In method 1 each signal of a signal pulse train existing in the time interval (t,t+dt) opens with a setable delay T at time t+T, an observation interval and closes at time t + T + τ.

In method 2 the intervals τ are opened periodically without any rest time from interval to interval.

The use of 16 different time intervals and of the two independent methods has several advantages [19, 29, 30].

Correlation analysis with shift register electronics is performed as well at the Los Alamos National Laboratory [20, 21].

4.3. Theory

The interpretation models used during the period 1970 till 1980 were based at the JRC on Monte Carlo simulations [22] of the signal Pulse train for the fixed dead time counter, the shift register and the more general pulse train analysis. Boehnel [8] Dowdy et al [11] Ensslin et al [9] developed analytical interpretation models which took into consideration approximately the neutron multiplication effects for the assay of fissile material. At the JRC Ispra rigorous analytical models were developed [23, 24] which permitted to reduce the measured data to following expressions :

$$R_1 = \varepsilon F_s M \nu_{s(1)}(1+\alpha) \tag{1}$$

$$R_2 = \varepsilon^2 F_s M^2 \nu_{s(2)}\left[1+(M-1)(1+\alpha)\frac{\nu_{s(1)}\nu_{I(2)}}{\nu_{s(2)}\left(\nu_{I(1)}-1\right)}\right] \tag{2}$$

$$R_3 = \varepsilon^3 F_s M^3 \nu_{s(3)}\left[1+2(M-1)\frac{\nu_{s(2)}\nu_{I(2)}}{\nu_{s(3)}\left(\nu_{I(1)}-1\right)} + (M-1)(1+\alpha)\frac{\nu_{s(1)}\nu_{I(3)}}{\nu_{s(3)}\left(\nu_{I(1)}-1\right)}\left[1+2(M-1)\frac{\nu_{I(2)}^2}{\nu_{I(3)}\left(\nu_{I(1)}-1\right)}\right]\right] \tag{3}$$

$$M = \frac{1-p}{1-p\nu_{I(1)}} \tag{4}$$

$$\alpha = \frac{S_\alpha}{\nu_{s(1)}F_s} \tag{5}$$

$$\nu_{j(\mu)} = \sum_{\mu=\nu}\binom{\nu}{\mu}P_{j\nu} \tag{6}$$

R_μ is the rate of correlated signal multiplets of order μ. The symbols used are :

S_α	=	(α,n) neutron emission rate of test item
F_s	=	spontaneous fission rate of test item
p	=	probability that a neutron generates an induced fission event
ε	=	probability for detection of a neutron
$P_{j\nu}$	=	probabiity for the emission of ν fast neutrons per prompt fission caused by a primary neutron generated by reaction j(j = I when induced fission, j = s for spontaneous fission)
$\nu_{j(\mu)}$	=	μth factorial moment of the $P_{j\mu}$ distribution.

R_1 and R_2 are obtained from the Shift Register, from the Reduced Variance meter and from the Variable Dead Time counter. The Euratom time correlation analyzer gives 16 values of R_1 and R_2 for both techniques, the shift register and the reduced variance meter. In addition it can deliver 16 values of R_3, R.... for a generalization of each of the two techniques. However for detection probabilities of $\varepsilon < 0.2$ the R_4-values are heavily influenced by cosmic radiation neutron bursts. The JRC has developed [24, 25] explicit interpretation models which are valid for waste and safeguards applications permitting an analysis of either 2 or 3 unknown parameters of waste or fuel items.

One of the main problems of Pu waste assay is to develop techniques, which are able to measure Pu quantities in the mg range in any type of waste matrix with any unknown location of the Pu debris. This is mainly a problem to eliminate the influence of the variation of the cosmic neutron emission. This can be performed by special filtering techniques of the signal pulse train [16] and a theoretical treatment of the measured data [27] to obtain the correct correlated multiplets $R\mu$ irrespective of the used filter, having an updating signal dead time.

Measurements in field were successfully executed with the Time Correlation Analyser [28-30].

5. ACTIVE NEUTRON INTERROGATION

Non-destructive neutron interrogation of uranium is only effective if active neutron interrogation is applied. The existing active neutron interrogation devices are: Active Well Coincidence Counter (AWCC), Cf-shuffler, the Differential Die-Away and the PHONID [31-35]. The latter one has been developed at the JRC Ispra and has shown its capabilities to measure homogeneous bulk material samples for safeguards purposes. If well characterized representative samples are used for calibration, a precision of the order of 1% can be obtained. For waste material the accuracy of the existing devices however does not reach the performance imposed by the authorities for disposal of the material.

This is due to the fact that the mentioned devices measure in principle only one signal, the total counts, from the induced fission neutrons. This one signal is not enough to yield information about the moderating power of the sample, attenuation of the interrogating neutron beam, efficiency etc. Even for bulk material all the existing devices show their shortcomings if some moderating matrix materials are present in the sample. To overcome this problem a research program has been set up by the NFC Unit of JRC Ispra,

PHONID 3b is the most recent model of a family of non-destructive interrogation devices built at the JRC Ispra already since 1970. A ^{124}Sb-^{9}Be photoneutron source performes an active interrogation of the sample with epithermal neutrons. Without moderating material in the measuring cavity the neutron source spectrum has an average neutron energy of 12 keV in the centre of the assay chamber. The source neutrons induce only fissions in the fissile material, in particular ^{235}U contained in the sample. The fission neutrons are counted by ^{4}He detectors. By using ^{4}He fast neutron detectors the fission neutrons can be separated from the source neutrons and the gamma background by energy discrimination. After correction for the decay of the external photoneutron source (T 1/2 = 60.2 d) the neutron count rate can be related to the ^{235}U mass in the sample.

The response of the PHONID 3b can be expressed by :

$$T = \varepsilon S L. \tag{7}$$

with

T	=	the total count rate
ε	=	the probability for detection of a neutron
S	=	the number of induced fission neutrons per unit time generated by the incoming neutron flux of the photoneutron source
L	=	probability that the generated induced fission neutrons leak out from the fuel element

The term S is proportional to the mass of the fissile material, but depends in addition on the geometrical arrangement of the U debris, of the neutron moderating and absorbing material. In this operation mode calibration with standards is necessary.

Extensive infield measurements on uranium bulk material and uranium waste have been performed to determine the precision of the device for material in a container with a diameter of 10 cm [35]. All the measured items have been analysed entirely by DA.

Homogeneous bulk material can be measured with an accuracy of 1%, if items with a specific U-factor are analysed individually and if the calibration curve is based on at least two well characterised representative standards.

Low-enriched powder waste material can be measured with an accuracy of 2.5%, while the liquid items can be measured within 10%. For the mass analysis at least one well characterised representative standard is necessary.

Residues from five different production processes were measured. Without separation of the production processes the measurement results have no significance. By an individual analysis of the production processes reasonable accuracies were obtained. For three production processes the accuracy was in the order of 5 %. Two production processes resulted in accuracy values not better then 30%. This is due to the different degree of humidity of the samples for these processes.

The results of the DA analysis on items containing rocks of material also show that the solution for the mass spectrometry measurements and concentration analysis has to be prepared very carefully. For some items more then 50 % of the uranium was missing in the solution representing the item.

The results show the necessity for the development of a new device to upgrade the performance of existing devices. Such a R & D project is in course. Using neutron spectroscopy and neutron position sensitive detectors the moderation and absorption power of the matrix and lumps of fissile material in the waste can be determined experimentally.

6. CHARACTERISATION OF GAMMA ACTIVE WASTE [36]

During the decommissioning and operation of 'hot' installations [e.g. 'hot' cells] waste contaminated with fission products, mainly Cs and Co are produced. Before storing and conditioning the 220 l compressed waste drums with a matrix of hydrogeneous materials (gloves, paper etc) have to be measured to determine the most important gamma emitters present.

The boundary conditions are

- fast measurement execution
- determination of whether or not a prefixed level of contamination is exceeded
- determination of individual contaminant nuclides.

A simple system has been set up based on a Ge detector without collimator, detecting the activity of the whole drum. The intrinsic efficiency of the Ge detector was determined by standard sources, for a distance detector-drum centre of 42 cm.

The waste matrix had a diameter of 38 cm and a height of 76 cm. The intrinsic efficiency for sources placed on different radii in the detector-drum centre plane did not change by more then 3% if the drum was rotated around its axis. For sources placed up to 70 cm above the drum centre on the axis of the drum the deviation from the centre intrinsic efficiency was not more than 10%. The greatest deviation of about 15% occurred when the sources were placed at the most outside position of the waste matrix, at the top or bottom and at a radius of 19 cm. For a distance detector/drum centre larger than 42 cm the intrinsic efficiency becomes more homogeneous.

As the intrinsic efficiency does not vary much in function of the position of a source in the waste drum, the detector was calibrated with known gamma emitting sources placed in the centre of an empty drum. The standard deviation between the fitted and measured efficiencies for the 7 source energies was 1.9% for sources with a declared uncertainty of 2.5% and a statistical counting uncertainty of 0.6%.

In the matrix material the gamma rays are absorbed or lose energy by Compton scattering. for a typical non-contaminated matrix the total absorpiton coefficient and the matrix homogeneity have been determined. Gamma sources were place in the centre of the rotating drum as well as in a position outside the rotating drum. The average absorption value currently expressed in $cm^2\ g^{-1}$, is a relative value to the value of the iron absorption [37-38]. Indeed it results that within experimental uncertainties the shape of the matrix absorption follows that of iron. The uncertainty of the absorption relatives to iron was determined, taking account into the absorption calculated for the different gamma peaks for one single path or measurement, it amounts to about 1% standard deviation.

By utilizing 4 or 5 paths, as can practically be done, the true matrix homogeneity was normally underestimated. Only for measurements in which the path crossed the drum in different directions a reasonable estimate of the homogeneity was possible. However the average absorption of the matrix was determined with an acceptable precision for all the measurements.

The largest source of uncertainty is due to the unknown location of nuclide contaminants in the drum. This uncertainty was about 30% for the two extreme cases, i.e. contaminants in the centre of the drum or on the border of the drum relative to a supposed homogeneous distribution. In order to eliminate this source of uncertainty the use of a sophisticated system with collimator and tomographic techniques is necessary leading to long measurement times.

Due to the fact that in a real waste drum the source position is unknown an analysis method was chosen in which the maximum content (i.,e. the 'UPPER BOUND') of the gamma contaminants is determined. It was decided to divide the drum in five horizontal sectors of equal heigth measuring the gamma spectrum with the detector in front of each sector of the rotating drum. Assuming five sources in the centre of each sector the upper bound of the contaminant activity in the drum was determined. The analysis method is simple and is reduced to a system of five equations with five unknowns.

On a laboratory scale 10 real waste drums have been characterised by measuring the absorption method is simple and is reduced to a system of five equations with five unknons of the matrix by neutron transmission with an Europium source. The inhomogeneity (as measured with 5 paths) is of the order of 20%. The average absorption value relative to iron determined for the 10 drums was

$\Sigma = 0.96 = 0.2$

in agreement with the absorption for a typical matrix. In agreement with ENEA-DISP using a single identical absorption value for all drums (e.g. $\Sigma = 1.0$) each single drum is measured with a certain error. However if a total activity value is measured by cumulating a certain number of drum measurements, the average activity of a single drum can be correctly determined since the positive and negative single drum errors compensates each other. The instrument has been installed at the Radioactive Waste Conditioning Building of the Ispra site and is now used on a routine basis.

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CHARACTERISTICS, CONDITIONING AND ACCEPTANCE REQUIREMENTS ON RADIOACTIVE WASTE TO BE DISPOSED OF

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Abstract

In the Federal Republic of Germany it is intended to dispose of all radioactive waste in deep geological formations. To provide the necessary data base for the disposal related planning work conducted by the Bundesamt für Strahlenschutz (BfS - Federal Office for Radiation Protection), an extensive characterization of this waste has been performed. Detailed information on the various compositions of primary waste, waste treatment and conditioning processes, waste forms, packagings and radionuclide inventories per waste package as well as on the respective waste arisings have been compiled. The acceptability for radioactive waste to be emplaced in a repository has been checked within the scope of site-specific safety assessments covering a repository's operational and post-closure phase. The results of these investigations served in particular to define the requirements to be met by the waste intended for disposal. Waste acceptance requirements have been established for the planned Konrad repository and the Morsleben repository. For the conditioning of radioactive waste various strategies and techniques have been applied. According to the available interim storage capacities, the Konrad waste acceptance requirements and the waste type catalogue, the waste generators and the conditioners have started to select appropriate conditioning processes and/or to adjust existing methods to the requirements. Special emphasis was given to volume-reducing waste conditioning techniques. According to the Morsleben repository having quite recently resumed operation, and its waste acceptance requirements, conditioning techniques must not necessarily aim at volume reduction measures. Thus, possible modifications and/or changes in conditioning strategies may be expected.

1. INTRODUCTION

From its very beginning, the German approach to disposal is based on the decision that all types of radioactive waste are to be disposed of in deep geological formations. Shallow land burial is not practised because of high population density, climatic conditions and existing appropriate deep geological formations. The Bundesamt für Strahlenschutz (BfS - Federal Office for Radiation Protection) is legally responsible for the establishment and operation of federal installations for engineered storage and disposal of radioactive waste.

2. RADIOACTIVE WASTE TO BE DISPOSED OF

An essential prerequisite for the development of disposal strategies or the planning and construction of repositories is the provision of a realistic data base. Data on the origin, type and expected amount of radioactive waste are therefore necessary.

2.1. Origin and type of radioactive waste

In the Federal Republic of Germany, current and potential radioactive waste origins are

(a) the operation of light-water reactors (pressurized water and boiling water reactors),

(b) reprocessing of spent fuel elements from German nuclear power plants in other European countries (British Nuclear Fuels plc (BNFL); Compagnie Générale des Matières Nucléaires (COGEMA)),

(c) basic and applied investigations in the Karlsruhe and Jülich nuclear research centres,

(d) other research centres, universities, industrial companies or medical applications of radioisotopes (note: This waste is generally handed over to the collecting depots of the federal states),

(e) uranium enrichment and the production of fuel elements, as well as research and development work in the nuclear fuel cycle industry,

(f) decommissioning and dismantling of nuclear facilities, and

(g) other waste producers, e.g., the German Federal Armed Forces and the pharmaceutical industry.

In addition, the direct disposal of spent fuel elements including the waste originating from their conditioning are in future to be taken into account.

The term "radioactive waste" covers a wide range of various materials which are quite different according to their nature and quantity of the radioactivity associated with them. The major bulk of waste arisings comprise, e.g., the following waste types (primary waste):

(a) liquids, concentrates and sludges,

(b) ion exchange resins,

(c) compressible and/or combustible materials,

(d) dimensionally stable solids,

(e) filters and multiple tube filters,

(f) ashes, powders and granules,

(g) scrap, insulating materials, debris, rubble and contaminated soil, and

(h) other waste types.

As the different waste types must be adequately separated to allow appropriate pretreatment and conditioning for interim storage and disposal, the development of a waste type catalogue seems to be of great advantage [1]. Such a catalogue surveys the radioactive wastes from their arising in a nuclear facility to their disposal in a repository. It specifies the primary wastes, the pretreatment methods, the resulting intermediate products, the conditioning methods and the resulting waste packages, thus describing the "flow" of radioactive waste.

The subdivision of primary wastes is hierarchically organized into waste groups, waste subgroups and waste types. As an excerpt from the waste type catalogue, the waste group "inorganic solid wastes" is described in Table I. The assignment of the primary waste to a waste group, waste subgroup or waste type must be done in accordance with the requirements of the waste pretreatment and the conditioning process. If necessary, primary wastes must be assorted. They must be specified in more detail, for example, if they are categorized as chemicals, filters or metals. They can be pretreated to form an intermediate product and must be conditioned (treatment and packing) to a waste package (waste form and packaging). The waste type catalogue, therefore, forms the basis for the description and tracking of radioactive waste from the primary waste to a waste package suitable for disposal. It helps to provide the necessary transparency in waste management and, in particular, gives guidance to the waste conditioners.

Further information being of importance for a data base, comprise relevant waste properties. For the characterization of radioactive waste, the radionuclide inventory per waste package is of particular importance. It represents essential basic information for the radiological evaluation with regard to the disposability of a waste package and for the formulation of waste acceptance requirements to be met by the waste package.

Data on relevant properties, e.g.,

Table I: Excerpt from the waste type catalogue subdivision of inorganic solid waste

Waste group	Waste subgroup	Waste type
Inorganic solid wastes	Metals	Ferritic metals Austenitic metals Nonferrous metals Heavy metals Light metals
	Nonmetals	Rubble Gravel and sand Soil Glass Ceramics Insulation
	Filters	Laboratory filters Air filters Box filters Multiple tube filters
	Filter aids	Ion exchangers Diatomite Silica gel

(a) density, porosity and compressive strength of waste forms,

(b) specific heat capacity and thermal conductivity of waste forms and packaging materials,

(c) softening, melting and flash points of waste forms,

(d) leach and corrosion rates of waste forms and packaging materials,

(e) gas formation rates, e. g., due to corrosion, microbial activities or radiolysis, and

(f) release rates of volatile and aerosol-bound radionuclides from waste packages

should be available for safety assessments. Not all the properties stated above must necessarily be known for each waste package.

Waste form characterization work including the determination of source terms has been performed within the scope of research and development programs. The results of these investigations, i.e., waste form or waste package properties relevant to disposal, have provided most of the data necessary for the performance of disposal-related safety assessments.

2.2. Conditioning techniques

Conditioning of radioactive waste includes processing and/or packing of the waste, eventually after a pretreatment or a sorting. Various strategies and techniques are applied. The selection of a conditioning process is dependent upon factors like the requirements for interim storage and disposal, acceptance of the process, and volume of the resulting waste packages. Therefore, it is not surprising that different conditioning techniques for the same type of waste may be applied. Furthermore, the necessity to minimize the volume of the conditioned waste because of lacking repositories stimulates the development of new and advanced conditioning techniques.

Primary waste must be collected and pretreated in such a way that it is suitable for the selected conditioning process. Principal pretreatment methods are decontamination, crushing, compression, evaporation/distillation/rectification, decantation/dewatering/filtration and incineration/pyrolysis.

Especially the incineration is attractive for all types of combustible waste. Solid or liquid waste and also alpha-bearing waste may be incinerated. The large volume reduction and, in particular, the inorganic and inert character of the intermediate product (e.g., ashes or slags) are reasons to recommend the incineration process from a repository-related point of view. Nevertheless, the off-gas treatment and the secondary waste must be taken into account.

The cementation of radioactive waste is the most well-known immobilization process being widely applied. It is used for the solidification of liquids, the embedding of solids as well as the grouting of voids in scrap, rubble or filters. Various cementation techniques are used and the equipment might be mobile or stationary. If necessary special cement formulas and/or suitable additives are to be used. Reactions with the cement, e.g., gas generation by amphoteric metals in the ashes must be taken into account. Possible chemical reactions between the radioactive waste, the immobilization material and the packaging must be limited to permissible levels.

The high-pressure compaction with 1500 Mg to 2000 Mg compactors is a new development to minimize waste amounts. Solid materials are compacted to a stable pellet. This technique is applied to, e.g., metallic materials, paper, plastic, rubble and even ashes from the incineration of organic radioactive waste. Due to possible gas generation occurring in compacted waste, a segregation before compaction is reasonable, i.e., to separate metallic and wet organic materials. Alternatively the compacted pellets may be dried.

The drying of liquid radioactive waste is another new development for waste minimization. The liquids are fed into a packaging which is heated. The evaporation is supported by application of a slight vacuum. The resulting dry residue consists of the constituents dissolved or dispersed in the liquid. The resulting product contains a higher activity concentration than, e.g., the cemented waste form and needs therefore a superior shielding which is often made of cast iron.

The melting of activated and/or contaminated metallic material is of special interest for the decommissioning and dismantling as well as the repair and maintenance of nuclear facilities. Depending on the radioactivity level in the melt it is of special interest to re-use the metals by casting packagings for radioactive waste disposal. If the radionuclide content is too high the melt might be poured into a packaging as radioactive waste. The resulting slag has also to be conditioned and disposed of.

Radioactive waste has to be packed for handling, transportation and storage. The necessary quality of a packaging is dependent on the type of waste and its radionuclide inventory. Sheet steel, reinforced concrete and cast iron are common as packaging material. Cylindrical and box-shaped packagings of different sizes and weights are being used. A standardization of the packagings has successfully been realized in order to harmonize the equipment as well as the repository-related handling and emplacement techniques.

2.3. Waste amounts

On behalf of the Bundesminister für Umwelt, Naturschutz und Reaktorsicherheit (BMU - Federal Minister for Environment, Nature Protection and Nuclear Safety) the BfS carries out an annual inquiry into the amount of unconditioned and conditioned radioactive wastes generated in Germany. In the following, only low and intermediate level wastes (i.e., radioactive waste with negligible heat generation) will be considered.

According to the latest inquiry [2], the amount of unconditioned radioactive remnants and primary wastes was about 27,800 m^3 on December 31, 1992. This was mainly produced by the nuclear power plants (11,485 m^3) and the nuclear research centres (10,543 m^3). In addition, an amount of about 200 m^3 of short-lived waste was stored for decay.

The volume of conditioned radioactive waste amounted to about 58,400 m^3 on December 31, 1992 (Figure 1). Of this, waste originating from the research centres contributed 24,104 m^3; waste from the operation of nuclear power plants 18,949 m^3 and that from reprocessing spent fuel elements 10,494 m^3. In comparison to the amount of radioactive waste occurring in other countries, the total waste package volume in Germany is rather small. This indicates the use of modern conditioning techniques contributing to the avoidance or reduction of

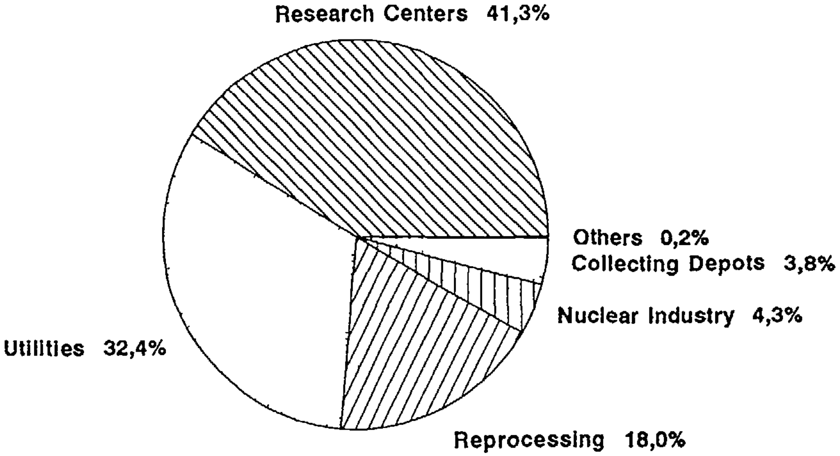

Fig. 1: Origin and amount of conditioned radioactive waste with negligible heat generation on December 31, 1992 (total amount: about 58,400 m^3)

wastes and resulting in continuously declining specific amounts of conditioned radioactive waste to be disposed of.

Based on the results of the annual waste inquiries from 1984 to 1992 and taking data on the return of radioactive waste due to reprocessing of spent fuel elements from German nuclear power plants in foreign countries into account, a forecast of future waste arisings was carried out [2]. This prognosis predicts an amount of conditioned waste at the end of the year 2010 to about 273,000 m^3 (Figure 2). Major contributions are made by spent fuel reprocessing (102,600 m^3), nuclear research centres (55,600 m^3), operation of nuclear power plants (50,200 m^3) as well as decommissioning and dismantling of nuclear facilities (50,000 m^3).

2.4. Waste disposal

According to the German disposal concept, all radioactive waste has to be emplaced in a repository constructed and operated in deep geological formations. As liquid and gaseous wastes are excluded from disposal in such a mine, only solid or solidified radioactive waste is accepted. In the Federal Republic of Germany, two sites are presently considered for disposal of low and intermediate level waste:

(a) In the abandoned Konrad iron ore mine in Lower Saxony, it is planned to dispose of radioactive waste with negligible heat generation.

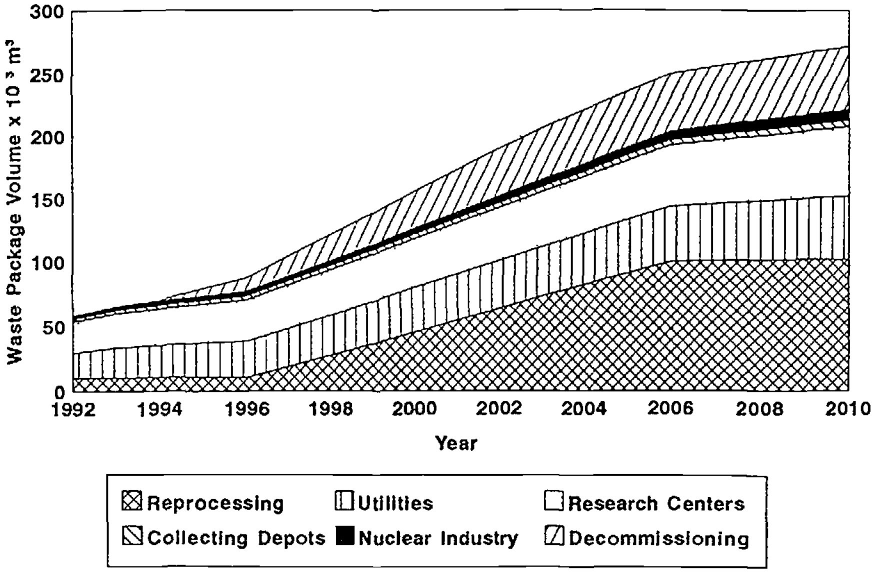

Fig. 2: Expected amount of conditioned radioactive waste with negligible heat generation accumulated up to the year 2010 (prognosticated amount: about 273,000 m^3)

(b) The emplacement of waste in the former Morsleben salt mine in Saxony-Anhalt which was operated as a repository for short-lived low and intermediate level waste with low alpha emitter concentrations has been resumed.

The planned Konrad repository is assigned to accept radioactive waste with negligible heat generation, i. e. waste packages which do not increase the host rock temperature by more than 3 K on an average. Iron ore, i. e. coral oolite, at a depth of 800 m to 1,300 m is the host rock for this repository. Waste packages will be disposed of in drifts with an excavated volume of about 1,100,000 m^3 allowing an emplacement of about 650,000 m^3 waste package volume. Operation of the repository is scheduled at least 40 years. A total activity in the order of 10^{18} Bq and an alpha emitter activity of about 10^{17} Bq are anticipated in this facility.

In the former German Democratic Republic an abandoned salt mine located near the village of Morsleben was re-used for waste emplacement. From 1981 until 1991, radioactive waste with a total emplacement volume of approximately 14,500 m^3 and about 6,700 spent sealed radiation sources were disposed of. Of this, the activity of alpha emitters amounts to $1.6 \cdot 10^{11}$

Bq and that of beta/gamma emitters amounts to $4.8 \cdot 10^{14}$ Bq. Subsequent to German unity the Morsleben facility has the status of a federal repository, the continuation of its former license being now limited by law until June 30, 2000.

Due to these planning data and marginal conditions the major bulk of low and intermediate level waste, i.e. radioactive waste with negligible heat generation, is intended to be disposed of in the planned Konrad repository. This is in particular relevant to alpha-bearing waste. The operation of the Morsleben repository will in the first instance contribute to the discharge of interim storage facilities.

3. WASTE ACCEPTANCE REQUIREMENTS

Persuant to the Safety Criteria for the Disposal of Radioactive Wastes in a Mine [3], the safety of a repository in the operational and post-closure phase must be proved within the scope of a site-specific safety assessment. Such an assessment comprises the undisturbed performance of the planned facility, assumed incidents, the thermal influence upon the host rock, the nuclear criticality safety and the radiological long-term effects in the post-closure phase.

The results of the respective Konrad safety assessment have been converted into both the design of the surface and underground facilities of this planned repository, and a system of preliminary waste acceptance requirements [4]. They describe the general basic aspects and the general requirements to be fulfilled and then develop into more specific requirements on waste forms, packagings, radionuclide-specific activity limitations, documentation and delivery of waste packages to the repository. A survey on the structure of these requirements is given in Table II. The Konrad waste acceptance requirements can only be compiled in a final form after the license for this facility has been granted.

The structure of the Morsleben waste acceptance requirements [5] is similar to that of the Konrad requirements. Nevertheless, there are two decisive differences:

(a) The operation of the Morsleben repository is regulated by the respective license granted on April 22, 1986, and by further documents pertinent to the license. This represents the legally binding framework which must be adhered to. The waste acceptance requirements includes both marginal conditions prescribed in the license and results of additional safety assessments which keep the above-mentioned framework. According to this, at first sight, the Morsleben requirements appear to be rather complicated.

(b) The Morsleben waste acceptance requirements clearly distinguish between requirements on solid radioactive waste and on sealed radiation sources. Such a difference is not explicitely made within the Konrad requirements; they are formulated in a more general sense.

Nevertheless, it should be pointed out that both the Konrad and the Morsleben waste acceptance requirements include the fulfillment of the requirements resulting not only from the safety assessments but also from the transport regulations.

Table II: Survey on the structure of the Konrad preliminary waste acceptance requirements

1. General Basic Requirements on Radioactive Wastes to Be Disposed of

2. General Requirements on Waste Packages
   - (a) Local Dose Rate
   - (b) Surface Contamination
   - (c) Depressurized Delivery

3. Requirements on Waste Forms
   - (a) Basic Requirements
     - (aa) without immobilization material
     - (ab) with immobilization material
   - (b) Waste Form Groups
   - (c) Exhausting of Activity Limiting Values
   - (d) Filling of Waste Packagings

4. Requirements on Waste Packagings
   - (a) Basic Requirements
   - (b) Waste Container Classes
   - (c) Inner Packagings

5. Limitations of Activity
   - (a) Permissible Activities
     - (aa) undisturbed performance
     - (ab) assumed incidents
     - (ac) thermal influence upon the host rock
     - (ad) nuclear criticality safety
   - (b) Declaration of Radionuclides

6. Delivery of Waste Packages
   - (a) Compliance with Transport Regulations
   - (b) Permits
   - (c) Marking of Waste Packages
   - (d) Requirements on Shipping Units

The waste acceptance requirements were elaborated in such a way that a flexible system of requirements could be established. Such a system includes several alternatives and different options for the waste packages which ensure the required level of safety for the respective repository. The waste generators thus have the possibility of applying and fulfilling those requirements which are specifically applicable to the waste packages produced by them and to be disposed of.

Bearing these possibilities in mind, it is not excludable that the waste generators and conditioners will re-evaluate and optimize/rationalize present conditioning strategies and procedures. Up to now, those strategies and procedures have been determined by the available interim storage capacities, lacking repositories and the Konrad waste acceptance requirements. As a consequence, conditioning techniques are in particular aiming at volume reduction and observing the permissible activities per waste package due to the Konrad requirements. From now on, the operation of the Morsleben repository offers potential new developments or modifications of existing conditioning techniques. From a conditioner's point of view it is meaningful to analyze the Morsleben waste acceptance requirements [5] and to adopt conditioning strategies and techniques to these requirements. In addition to technical aspects, due to the Morsleben costs for disposal fixed at DM 12,500 per m^3, i.e. DM 2,500 per 200 litre drum, it is henceforth possible to select appropriate conditioning procedures taking economic aspects into consideration, too. For example, as to combustible waste, it could be meaningful to use high-pressure compaction instead of incineration.

4. STATUS OF THE KONRAD AND MORSLEBEN REPOSITORIES

4.1. Konrad repository project

On August 31, 1982, the licensing procedure for the planned Konrad repository was started. Meanwhile this procedure is in a well-advanced stage. The public debate of the licensing documents from September 25, 1992, to March 6, 1993. The hearing took 75 days and was the longest public debate within a nuclear licensing procedure ever held in the Federal Republic of Germany.

About 10,000 objections to the licensing documents were raised. The number of different objections amounted to about 3,600 and that of objectors to about 290,000. Due to an analysis of the BfS the objections contained about 1,000 different arguments against the Konrad repository project. The 290,000 objectors were represented during the public debate by an average of about 10 to 15 people increasing to several hundreds on special days.

The debate dealt with formal issues, types of waste to be disposed of, the German waste management concept, long-term safety, the operation of the repository, incidents/accidents, waste transport outside of the facility, protection against catastrophic events, individual involvements and those of the local region and communities, other than nuclear energy specific legal matters, environmental impact, and other topics (e.g., electricity from nuclear power plants.) Concerning the post-closure period, the discussion centered around time horizons for the safety assessment, quality and completeness of geological data, relevance of shafts and boreholes for radionuclide migration and, finally, gas production and possible chemical reactions of the waste.

In the applicant's opinion, the public debate resulted in a confirmation of his work. No deficiencies could be identified causing an interruption or a premature termination of this debate. The BfS can therefore expect a positive decision.

At present, the experts of the licensing authority, the Lower Saxonian Ministry for Environment, are preparing their final expertises. These documents should be finished in 1994. A final decision on the licensing procedure may possibly be expected in 1995. Thus, operation of the Konrad repository may be assumed to start towards the end of the nineties. However, the political will within the state government of Lower Saxony (a coalition of The Social Democrats and The Greens) tries to prevent the Konrad repository project, so that the outcome of the process appears to remain open.

4.2. Morsleben repository

To answer pending questions with regard to safety but also with regard to licensing, the emplacement of radioactive waste in the Morsleben repository was stopped in February 1991. The safety-related features of this facility have accordingly been under scrutiny of, e.g., the Reaktorsicherheitskommission (RSK - Reactor Safety Commission). A number of recommendations and improvements have been given, also referring to improvements concerning the backfilling and sealing of the mine after emplacement will have been ceased. Since the safety-related provisions against the further emplacement of waste could be dispelled in the meantime and the legal objections are unfounded according to the final judgement of the Federal Administrative Court of June 25, 1992, the Morsleben repository can continue operation as a federal installation.

A decisive step towards the resumption of the repository's operation was the judgement of the superior administrative court in the federal state of Saxony-Anhalt. According to its judgement of December 13, 1993, radioactive waste originating from the old federal states can be disposed of in this facility, too.

Thus, first emplacement of radioactive waste from the shut-down Greifswald nuclear power plant took place on January 13, 1994. Since the operation has smoothly been resumed, until mid-February 1994 five shipments were carried out and 202 waste packages emplaced (operational waste originating from nuclear power plants in the new and the old federal states packed into 200 litre drums, e.g. scrap, protective clothing, insulation material, worn-out equipment, paper, polyvinylchloride, rubble and contaminated soil). Concerning the continuation of waste disposal in this facility, it is anticipated to increase the number of shipments and waste packages, respectively, including radioactive waste from other waste generators, too. Until June 30, 2000, according to present plannings a radioactive waste volume of 40,000 m^3 is envisaged to be disposed of. The estimated maximum activity of alpha emitters amounts to about 10^{13} Bq, that of beta/gamma emitters to about 10^{16} Bq.

5. CONCLUDING REMARKS

The Konrad preliminary waste acceptance requirements (as of October 1993) are subject to the pending licensing decision. For planning and project implementation, the compulsory nature and reliability of these requirements are of great importance to the waste generators and conditioners. They have already started to adopt and to convert the guidance resulting from these requirements. According to a successful continuation and final realization of their efforts, the license for the Konrad repository is an important factor being a highly political and not a technical subject.

The operation of the Morsleben repository is an important step in the realization of a proper radioactive waste management system. Therefore, parallel to the Konrad work, present activities of the BfS are in particular intended to increase the number of waste shipments to the repository according to the planned performance as well as to initiate further developments and improvements within the emplacement of radioactive waste in this facility.

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THE ACCEPTANCE, BEFORE TRANSPORT, OF THE WASTE RADIOACTIVE PACKAGES ON THE DISPOSAL FACILITIES OF ANDRA

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Abstract

ANDRA has set procedures with the waste generators and with the carriers in order to be sure that all the packages are in accordance with the technical prescriptions and with the specifications elaborated by the Agency. In addition, these procedures guarantee that the safety rules for the storage, and the regulation applicable to the transport of dangerous goods will be respected. The first procedure concerns the package acceptance and is decomposed in four steps : Acceptance of the individual package/Validation of the shipment/Confirmation of the shipment and Control at the arrival. The second procedure is a qualification procedure for the carriers, and concerns the applicable regulations and ANDRA'own requirements concerning : Driving personnal/Vehicles/Tranport conditions and Instructions in case of a road accident. Moreover, quality assurance programs have been established by the carriers for transport operation. To assist them, ANDRA has drafted a quality assurance Standard.

ANDRA - National Agency for the Management of Radioactive Wastes - is responsible for, among other missions, the long term management of disposal facilities, in France.

In 1994, two disposal facilities are operationnal :

- Le Centre de Stockage de la Manche, located near the reprocessing plant of La Hague, which has been in operation since 1969, will be full in 1994, june.
- Le Centre de Stockage de l'Aube, which opened in 1992, will be in operation for about 40 years.

The largest waste generators are E.D.F. (the Power National Company), C.E.A. (Atomic Energy Commission), and COGEMA.

The Agency has set procedures with the waste generators and with the carriers in order to be sure that all the packages convoyed to the disposal centers are in accordance with the technical prescriptions and with the specifications elaborated by ANDRA. In addition, these procedures guarantee that the safety rules for the storage, and the regulation applicable to the transport of dangerous goods will be respected.

These procedures are :

- on the one hand, a package acceptance procedure done automatically by computer.
- on the other hand, a qualification procedure for carriers, which include essentially the training of the drivers, the equipment of the vehicles, and the instructions which have to be applied during transportation.

I. PROCEDURE FOR PACKAGE ACCEPTANCE

This procedure is applied up and down the transport operation, by computer (most of the waste generators are connected directly to ANDRA through software designed specifically for this purpose).

It is broken down into four steps.

I.1 Acceptance of individual package

In order, for a package, to be accepted by ANDRA, the waste generator gives it a specific number when it is made. This number corresponds to a file on the computer network which is created for the package, and contains the following information :

- its designation :
  * origin (ex. : process waste produced by the operation of industrial facilities)
  * identification number
  * volume
  * weight
- its composition :
  * waste nature (ex. : plastics, cotton, filters, ...)
  * embedding (ex. : concrete, bitumen, ...)
- its radiological characteristics :
  * spectrum, date
  * total activity
  * activity of each isotope
  * dose rate

ANDRA's computer checks that all these data are in accordance with specifications, and if this is the case, ANDRA accepts the package.

I.2 Validation of the shipment

When the waste generator has enough of the same type of package, to organize a shipment, he sends the list of packages included in the shipment to ANDRA.

ANDRA's computer then verifies that every package designated on this list has been previously accepted, and if so, validates the shipment.

I.3 Confirmation of the shipment

When the packages have been loaded on to the vehicle, the waste generator confirms to ANDRA that he is ready to proceed to the shipment.

The information related to the carriage is transmitted at the same time by the central computer, located at the headquarters of ANDRA, to the disposal center (both of the disposal centers are connected with this computer). That is to say :

- the reference of the shipment,
- the waste generator,
- the packages : quantity and numbers,
- the characteristics of each package : weight, nature of waste, origin, spectrum, activity.

Furthermore, the transport documents, where all this information appears, are given to the driver.

Each package bears a double identification :

- a clear identification (numerical),
- a bar code label.

I.4 Controle of the packages upon arrival

Upon arrival at the disposal facility, each package is subjected to three checks :

- a physical check, in order to verify that the package has not been damaged during the transport,
- a radiological check (dose rate, contamination), in accordance with the regulations for the safe transport of radioactive material,
- an identification check (by reading the bar code with a laser system), to make sure that no mistake occured during the loading.

The storage of a package will not be authorized if any of these three checks reveals an anomaly.

This individualized and computerized tracking of the waste packages guarantees maximum safety, because :

- it allows ANDRA to verifiy, before storage, that each package respects the radioactive limits authorized for long life radioisotopes,
- it permits ANDRA to make different statistics about the level of radioactive stored (whole radioactivity for each waste generator, for each disposal structure, etc ...).

II. QUALIFICATION PROCEDURE FOR CARRIERS

In order to get the highest level of safety in road transports, ANDRA established a qualification procedure for carriers. Only the carriers qualified by ANDRA are allowed to carry the wastes to the disposal centers.

When a carrier proposes its collaboration, ANDRA supplies him with a file reminding it of the applicable regulations and as well ANDRA's own requirements, in particular :

- driving personnel (training - in France a special licence is required for driving vehicles loaded with radioactive material -, medical and radiological surveillance),
- vehicles (general condition, maintenance),
- specific equipment (securing, stowing, protection, intervention),
- transport conditions,(instructions for loading and unloading, accompanying documents, itineraries),
- special instructions in case of a road accident.

An audit is then carried out by an approved organisation in the presence of ANDRA. If the carrier satisfies the obligations specified in the qualification file, he receives a certificate valid for one year.

ANDRA reviews each carrier prior to renewal of the certificate. Unscheduled verifications take place during the year.

Moreover, according to new regulations published by the International Atomic Energy Agency concerning the transport of radioactive material, quality assurance programs have been established by the carriers for transport operation. To assist them, ANDRA has drafted a quality assurance standard based on the ISO norm 9002, and on the French regulations for the transport of dangerous goods.

What are, in detail, these requirements that must be complied with ?

II.1 Drivers'training

This one week training is provided by the specialists of the French Atomic Energy Commission. It has been obligatory since 1979. The main subjects covered include :

- Basic knowledge about radioactivity. The different radiations.
- The regulations about the transport of radioactive material. The different types of packagings.
- The obligations of the loader concerning the preparation of the packages (dose rates, contamination levels, labelling, transport declaration).
- The obligations of the carriers (equipment of the vehicles, stowing rules, instructions in case of an accident).

At the end of this training, an examination is carried out. If successful, a four years licence is delivered to the driver. When this licence expires, he must take follow another training course.

II.2 Vehicle's characteristics

According to the qualification procedure, the vehicles which have to transport radioactive waste have to be equipped, firstly, in accordance with the general conditions required for all transport of dangerous goods, and, secondly, in accordance with specific conditions necessary for the transport of radioactive material. The former are :

- an uncontaminable platform,

- a protective barrier between the cabin and the packages (lead screen),

stowing schemes, specific to each kind of package transported,

two special fire-extinguishers (powder) - one for the engine and another for the carriage,

- an emergency signals which include, mainly, lamps, chains and stakes for demarking a safety area in case of the loss of packages, and panels ("Emergency - Radioactive" and "Accident - Don't approach")

II 3 Instructions for transport

The driver must have the following documents with him during the transport

- Instructions for loading and unloading (addresses, phone numbers of the loader and the consignee)
- Instructions for completing the accompanying documents
- Instructions concerning itineraries and stops
- Instructions in case of an accident (concerning the signalization of the vehicle and also the notification procedure)

II.4 Quality Assurance

According to the regulations for transport of radioactive material published in 1985 by I A E A "Quality assurance programmes shall be established for transport operations to ensure compliance with the relevant provisions of these regulations"

So, ANDRA set up a specific document for the carriers which defines the requirements in this field

The different items of this "standard" are

1 Generalities

2 - Definitions

3 - Scope of application

4 - Reference documents

5 Legislative and regulatory responsibilities

6 - Quality system requirements

Management responsabilities
Quality system
Contract rewiew
Mastering of the documents
Purchasing
Control of the transport's preparation
Verifications, maintenance, checks
Treatment of anomalies
Incidents -Accidents
Transport documents
Handling - Storage of transport vehicles
Internal quality - Control audits

In 1994, all the carriers qualified by ANDRA have a quality system in operation

Beyond that, ANDRA plays an important part of safety by maintaining a permanent staff, on duty night and day This staff knows the loading details of every vehicle and is able to give any necessary information about possibility and conditions of an intervention in the case of an incident or accident (every year, about 80 000 packages of low level wastes are carried in France)

INITIAL ESTIMATES OF SAMPLES AND RESIDUES REQUIRING TRANSPORT ARISING FROM THE UNITED STATES DEPARTMENT OF ENERGY'S ANALYTICAL SERVICES PROGRAM

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Abstract

This paper addresses initial estimates of materials to be packaged and shipped in the U S. as a result of the taking of hazardous, or potentially hazardous, samples for the purpose of analyzing the levels and extent of hazard posed by specific water, soils, materials, etc. at each site the U S. Department of Energy is responsible for The taking of such samples is required to facilitate remediation of these sites. The sampling process requires that samples be shipped to analytical laboratories and that unused portions of samples, residues from sample analyses, and any secondary waste streams also be transported away from the laboratories. Although the planning for and tracking of such shipments has not been completed, it appears from the data assembled to date that tens of thousands of shipments per year of samples to laboratories and associated shipments from laboratories following analyses can be expected. Many of this shipments will be mixed wastes. A number of packaging and operational issues have been identified and are discussed in the paper

INTRODUCTION

The United States (U.S.) Department of Energy (DOE) has created the Office of Environmental Restoration and Waste Management (EM) to address the diverse complex problems associated with the remediation of DOE sites The remediation requires the taking and analyzing of samples of hazardous materials, and then storing or disposing of the unused portions of samples, residues from sample analyses, and any secondary waste streams

Included in the EM office are two companion divisions that are playing vital roles in preparing for and monitoring and implementing the remediation process. These divisions are the Transportation Management Division (TMD) and the Laboratory Management Division (LMD) Because significant use of commercial sector laboratories is envisioned for performing the laboratory analyses [1], a large number of shipments to and from the laboratories will be required The materials to be shipped will be generally hazardous, and many of them will be radioactive

The LMD has proceeded in its efforts and has issued detailed plans for proceeding further. This includes the development of a five year plan [1], and progress toward the scoping of sample needs on a site-by-site basis (e.g , see Reference 2). As a result, the packaging and transportation planning by TMD, in support of LMD, has now begun in earnest. The first step has been to start to define packaging and transportation requirements to guide the development of packagings for the future.

The initial efforts and findings, in terms of initial estimates of materials to be shipped are provided in this paper In addition, some of the major issues relating to package design which have been identified are briefly discussed

GENERAL CHARACTERISTICS OF MATERIALS TO BE SHIPPED

The scoping of the characteristics and quantities of samples needed for analysis and the resulting residues from the analyses is under way. The first estimates for the Idaho National Engineering Laboratory (INEL), have been issued [2]. Data for the other DOE sites is not yet available, although some early estimates for a second site, the Hanford site in Richland, Washington are available, and limited estimates of numbers of transuranic (TRU) samples to be taken are also available.

A wide variation is being identified in the types of samples to be obtained, packaged, and transported to analytical laboratories The INEL study categorized the sample sources as "routine," "waste," and "other" as follows [2)].

Routine Monitoring: samples made to satisfy air and water monitoring requirements. The INEL report indicated that these samples will be typically nonhazardous and, specifically, nonradioactive.

Hazardous Waste· nonradioactive samples that contain either toxic, corrosive, flammable or reactive chemicals or polychlorinated biphenyls (PCBs) above limits defined by the U.S Environmental Protection Agency (EPA).

Radioactive Waste: radioactive samples that contain source, special nuclear, or by-product materials These are further categorized into low-level wastes (LLW), contact handled transuranic (CHTRU) wastes, remote handled transuranic (RHTRU), and high-level wastes (HLW).

Mixed Waste: radioactive samples that also contain chemically and/or physically hazardous materials. These are further categorized into mixed LLW, mixed CHTRU wastes, mixed RHTRU wastes, and mixed HLW

Other: samples required by remediation that are not covered by the previous categories

The hazardous, radioactive and mixed waste samples are, of course, uncharacterized and may or may not hazardous; their exact characteristics can only be ascertained after shipment to and analysis by a laboratory Because of their origin and until they are properly analyzed, conservative approaches generally will be necessary in handling, packaging, and transporting them by the sampling sites. If there is even the slightest potential of hazardous materials being in the samples, personnel at the sampling site will need to assume the materials are hazardous and will therefore handle, package and transport them using a conservative approach. This will result in many of the samples being presumed to be of the mixed hazardous category.

In addition to the shipment of samples from the sampling site to an analytical laboratory, there will also be significant transport of materials from the laboratory to storage or disposal sites. Generally, the (1) unused portions of samples, (2) residues from sample analyses, and (3) any secondary waste streams will all need to be returned to the originating site or to some other site for storage or disposal. Because the materials resulting from the sample analyses will be known as a result of the analysis, the proper packaging requirements will be readily established for them. It is expected that many of these will be of the mixed waste category. Also, the contents of the unused portion of the samples will have be known as a result of the characterization, and the proper packaging and transportation requirements will be readily established for them.

ESTIMATED QUANTITIES AND TYPES OF MIXED WASTES AND OTHER MATERIALS TO BE SHIPPED

As noted above, the estimating of the quantities and characteristics of sample analyses to be requested by various U.S. DOE sites is under way. To date, first estimates for only one site, the INEL, have been documented. This evaluation for the INEL provides a first perspective of the magnitude of the challenge to be faced.

The report for the INEL notes that "the annual number of analytical requests from the INEL programs may be biased low." [2] That is, the number of analytical requests will probably be higher than those shown in the report. The following three subsections, which summarize the perspective for the INEL site, are derived from Ref. 2; preliminary, unpublished data for the Hanford site; and preliminary, unpublished data for all shipments of TRU wastes, DOE-wide.

Quantities for the INEL

Reference 2 documents the number of analytical sample requests, not the number of samples requiring shipment. It is expected that multiple analyses will be made from a single sample. Hence, the number of samples that will be shipped will generally be significantly smaller (possibly, on average, by about a factor of 10) than the number of analytical requests.

The data from the INEL evaluation can be used at this time only as an indicator of the magnitude of the task. The INEL study, which provides projected analytical sample requests per year, over a 10-year period for this one site, are summarized in Table 1.

These data demonstrate that, during the next 10 years, about 1.3 million analytical sample requests can be expected for the INEL site alone. Of these, 89 % (1,126,428 requests) will be categorized hazardous waste, either radioactive or mixed. These are expected to be:

Radioactive,	1%;
Hazardous,	5%;
Mixed LLW,	20%;
Mixed TRU waste,	71%; and
Mixed HLW,	3%.

Table 1. Ten-year summary of projected number of analytical sample requests to be made by the INEL [2]

Year	Sample Category						Totals
	Routine Monitoring	Hazardous Waste	Mixed LLW	Mixed TRU Waste	Mixed HLW	Radioactive Waste	
1993	15,482	8,487	20,770	5,487	3,000	1,307	54,533
1994	16,010	6,740	47,243	6,900	3,000	3,556	83,449
1995	14,609	5,586	47,176	10,302	3,135	1,806	82,614
1996	15,219	4,801	27,436	3,985	3,135	806	55,382
1997	15,436	4,585	16,367	51,565	3,135	806	91,894
1998	15,287	6,410	13,795	51,565	3,135	1,306	91,498
1999	15,530	6,184	12,515	51,565	3,135	806	89,735
2000	13,409	7,499	15,144	205,149	3,135	806	245,142
2001	13,652	3,594	12,395	206,749	3,135	806	240,331
2002	10,922	4,999	12,395	205,149	3,135	806	237,406
Grand Totals	**145,556**	**58,885**	**225,236**	**798,416**	**31,080**	**12,811**	**1,271,984**

Thus, from these data it can be readily seen that, at least for this one DOE site, the majority of the analytical requests are projected to be categorized as mixed waste and that the majority of these mixed waste samples will be mixed TRU wastes. The data illustrate that the number of potential samples that must be packaged and transported to analytical laboratories is certainly significant. The samples may be sent to DOE laboratories on the sampling site, to DOE laboratories away from the sampling site, or to commercial laboratories.

As noted previously, many analyses may be obtained from a single sample. Also, multiple samples may be shipped in a single package. Considering the need for timely analyses and the need to ship unused portions of samples and secondary waste streams from the analytical laboratories, the hazardous/mixed/radioactive waste-material shipments involved in the sampling program for the INEL site alone could approach 100,000 over the next 10-years (i.e., about 10,000 shipments per year). This compares with 1992 DOE nationwide shipping activity of 23,000 shipments of all hazardous materials. Thus, the burden on the DOE packaging and transportation system imposed by sample analysis will result in a significant increase over current activities and capabilities.

Quantities for the Hanford Site

Data for the Hanford site have been obtained informally which show that the number of analytical sample requests for the Hanford site will grow from about 1 million in 1993 to 2 million in 1998. The radioactive nature of the samples for these determinations in 1998 is projected to be approximately:

Number of Sample Requests	Radioactive Characteristics of the Samples
1,000,000	<10 mrem/h and <100 nCi/g;
200,000	10 to 200 mrem/h and <100 nCi/g;
350,000	>200 mrem/h;
500,000	>100 nCi/g of TRU.

Again, because multiple analyses may be obtained from a single sample and multiple samples may be shipped in a single package, the number of shipments involved in the sampling program for the Hanford site will be significantly less than the number of sample requests. Thus, as with the INEL, it can be projected that the number of shipments per year for the Hanford site associated with samples and analytical residues arising from analyses could number in the tens of thousands per year.

In addition to the samples required for the remediation of the Hanford site in general, an additional challenge concerns the shipment of core samples from the HLW tanks to analytical laboratories for characterization. These samples will generally require shipment in shielded casks unless the samples continue to be handled under special arrangements, on-site, at Hanford. The core samples are currently transported in 0.48 m (19-in.) segments. A minimum of two full cores are to be taken from each tank. Thus, if future shipping utilizes the 0.48 m core segments, and there are an average of five 0.48 m core segments per core sample, a minimum of about 1,770 core segments will require packaging and shipping for the 177 tanks at Hanford. These shipments are projected to occur by 1998. In addition to the core segment shipments, the unused portions of the samples and the analytical residues also will require shipment from the analytical laboratories. In all cases, these shipments will be categorized as mixed wastes.

Quantities of TRU Waste Samples to be Shipped

Based upon preliminary, undocumented data obtained by the working group, the number of required TRU field samples required will grow from about 2300 per year in 1993 (based upon the 5,487 analytical sample requests shown in Table 1, where it is assumed that approximately two analyses are obtained per sample, all from the INEL) to more than 20,000 per year beginning in about 2003 (projected to come from multiple sites throughout the U.S. including the Hanford, INEL, Los Alamos, Oak Ridge, Rocky Flats, and Savannah River sites.

Thus, it appears that one of DOE's major challenges will be the shipment of TRU waste samples.

ISSUES AND CHALLENGES IN SHIPPING MIXED WASTES RESULTING FROM SAMPLING ACTIVITIES

A number of significant issues relating to the packaging and transportation of the samples, their residues, and secondary waste streams have been identified. A few of the key issues are discussed here.

Need for Efficient Chilled Sample Packages

The U.S. Environmental Protection Agency (EPA) imposes a requirement (40 CFR Part 136) [3] upon many samples that, from the time they are obtained until they are tested, they must be maintained at 4°C (±2°C). This requirement is imposed to allow for the evaluation of the presence of volatiles in the sample.

The samples, many of which could be mixed waste, must therefore be transported in some type of cooled (or chilled) container, and they must be transported rapidly from origin to destination because the low temperature will most likely be maintained using a passive system (e.g., ice). Air transport is therefore required. Currently, no package exists that can be used effectively and efficiently to ship these samples by air; existing packages, which will accommodate ice for a period of days, do not simultaneously satisfy both the U.S. Department of Transportation (DOT) [4] and International Air Transport Association (IATA) [5] regulations. Thus, a new package design, or perhaps multiple package designs, for chilled samples will be required.

Need for a New, Diverse Package Design for Shipment of Liquid Samples

The ability of sampling sites to ship multiple samples of mixed wastes in a single package is constrained currently because adequate package designs to accommodate such multiple samples do not exist. It has been determined that the design of a new package, which could carry large quantities of liquid samples and have the capability of containing either single or multiple bottles and different types of bottles would be useful.

Need to Prepare for Future Enhanced Mixed Waste Shipping Demand

A significant enhancement in the need for packages and shipments of mixed wastes will be needed as the characterization and remediation of DOE sites progress. These needs are driven by many factors. For example, as noted previously, rapid growth in the number of alpha-contaminated, mixed waste samples needing shipment is projected. In addition, as characterization proceeds, more hazardous samples are expected to be obtained. This will require the use of new, enhanced package designs with more shielding or increased capabilities.

In addition, the system must be prepared to accommodate the unexpected. When samples are taken, it may be determined in some cases that the samples may be more hazardous than anticipated. Consequently, packagings must be available to receive these samples. All these factors indicate that there is a need to prepare a "family" of package designs that can accommodate a wide range of mixed wastes.

Need for HLW Sample and Analytical Product Packages

As the number of samples from the HLW tanks at the Hanford site grows, the need to ship samples off-site to other laboratories can be expected. When this occurs (and it could occur within the next two to three years), there will be a need to

- have available U.S. Nuclear Regulatory Commission-certified package designs (current certified package designs for shipment of these samples do not exist),
- have NRC-certified package designs for multiple 0.48 m samples (currently, all shipments are made on-site in non-certified package designs, one sample at a time),

- have designed and fabricated more efficient packages for the on-site shipments (this could reduce costs and personnel exposure),

- address package-facility interface issues (current analytical laboratory could not conveniently accommodate certified casks),

- consider upgrading and recertifying existing, commercially available Type B package designs (some commercially available package designs might be adapted to satisfy the HLW-sample shipping requirements), and

- prepare for off-site shipments of unused samples and secondary wastes (if additional analytical laboratories are used, then these materials will require shipment to some location for storage or disposal; certified package designs will be required for these shipments; currently, such shipments are made on-site in uncertified packagings).

Concerning the latter issue, little experience is available in packaging or shipping secondary wastes (generally mixed wastes), waste products from analytical laboratories, and unused samples. It has been determined that a significant effort may be required to prepare for these activities. As this effort progresses, a number of issues will need to be addressed, including defining where, when and how the mixed wastes arising from analytical laboratory efforts are to be shipped.

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RECYCLED SCRAP METAL AND SOILS/DEBRIS WITH LOW RADIOACTIVE CONTENTS

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Abstract

Two types of large volume bulk shipments of materials with low radioactivity have characteristics that complicate compliance with normal transportation regulations. Recycle scrap metal sometimes contains radioactive material that was not known or identified by the shipper prior to it being offered for transportation to a scrap recycle processor. If the radioactive material is not detected before the scrap is processed, radiological and economic problems may occur. If detected before processing, the scrap metal often will be returned to the shipper. Uranium mill-tailings and contaminated soils and debris have created public health problems that required moving large volumes of bulk material to isolated safe locations. Similarly, old radium processing sites have created contamination problems needing remediation. The U.S. Department of Transportation has issued exemptions to shippers and carriers for returning rejected scrap metal to original shippers. Other exemptions simplify transportation of mill-tailings and debris from sites being remediated. These exemptions provide relief from detailed radioassay of the radioactive content in each conveyance as well as relief from the normal requirements for packaging, shipping documents, marking, labeling, and placarding which would be required for some of the shipments if the exemptions were not issued.

1. Introduction

Sometimes bulk recycled scrap metal containing undetected radioactive material has been processed at steel mills or other facilities causing serious radiological problems or contamination of the facility which may require shut down of the facility and clean-up costing millions of dollars. If the radiation from the shipment of scrap metal is detected prior to processing, the materials are usually returned to the shipper. Since the original shipper did not know that radioactive material was present, the identity and activity of the radionuclides in the returned shipment is not readily available. These return

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shipments containing unidentified radioactive material are difficult to classify and offer for transportation within existing transportation regulations.

Processing uranium for nuclear programs has resulted in massive volumes of uranium mill-tailings. Processing of radium many years ago resulted in residues that were disposed of with little concern for low level radiation exposure. Unacceptable radiation hazards to the general public exist where uranium, thorium, and radium waste have intentionally and unintentionally been deposited. To mitigate this public health problem, large volumes of tailings, soil, and debris need to be transported to radiologically safe locations. Soil and debris shipped from sites being decommissioned and decontaminated often contain naturally occurring radionuclides with low but uncertain specific activity. Sometimes these materials are below the 70 Bq/g (2 nCi/g) definition of radioactive material, and in other cases the activity greatly exceeds the definition. Without knowledge of the activity in the material in each vehicle or bulk container, it is difficult to classify the materials and offer the shipments in accordance with the regulations.

The U.S. Department of Transportation (DOT) Hazardous Materials Regulations have statutory provisions that allow the issuing of exemptions that grant relief from regulatory requirements. Information submitted in applications for exemptions must demonstrate that the level of safety for shipments under the exemption will be equivalent to the level of safety if shipments met all normal transportation requirements. Exemptions issued to Federal and State organizations allowed shippers and carriers to make shipments in a controlled and operationally effective manner. These exemptions allow the shipments to be made without analysis of the material in each vehicle or bulk container. Further, they provide relief from the usual requirements for packaging, shipping documentation, marking, labeling, and placarding. The exemptions are similar to Special Arrangements under Safety Series No. 6 or the transportation systems approvals being considered for the regulations in 1996.

2. Recycled Scrap Metal Problem

Radioactive material in recycled scrap metal is occurring too often in steel production. However, it has also been found during the recycling of aluminum, copper, lead, and gold. The problem is international and is evidenced by radioactivity being detected in scrap and other metals entering the United States from about ten other countries. The radioactivity is usually detected in recycled scrap prior to the production of new products, but in other cases it has been found in products and waste/byproducts associated with the production.[1]

The radiation hazards associated with the unknown radioactivity have ranged from potentially acute effects to radiation levels representing a small fraction of background radiation. The radioactive material creating these conditions has ranged from sealed sources with activities in excess of a TBq to materials such as refractory bricks that are not normally considered radioactive, but because of thorium content emit very low level gamma radiation. One well known example of a large source being processed during recycling was the ^{60}Co from a teletherapy unit in Juarez, Mexico in 1983. The undetected discrete sources that have caused most of the shut downs these past few years have been ^{137}Cs sources, probably from radiation transmission gauging devices, with activities in the range of 20 GBq (0.5 Ci). The products that most commonly caused the radiation detection systems to alarm contain radium--examples are sections of oil-well pipe with scale and other metal from systems that process materials with trace amounts of naturally occurring radioactive materials.

To avoid radiological problems and costly shut downs many facilities have installed high sensitivity radiation monitoring systems, often computer controlled, to look for the presence of radiation from vehicles bringing feed stock into the facility and for monitoring products leaving the facility. The sensitivity of these systems hopefully will detect the presence of a shielded source that is further shielded by large amounts of other material in the load. The computer controlled monitoring systems can detect radiation levels that are a small fraction above normal background radiation. When the radiation level from a conveyance exceeds the alarm threshold such as 0.02 uSv/h (2 urem/h) above a normal background of typically 0.15 uSv/h (15 urem/h), other radiation monitoring instruments are used to verify and/or identify the presence or absence of radiation. Obviously, setting the alarm thresholds extremely low can cause problems with excessive numbers of false alarms. When the radiation monitoring systems alarm, the system cannot distinguish whether the radiation comes from a large well shielded source in the center of a heavy load of scrap or from a slightly contaminated piece of metal at the external wall of the conveyance. When the presence of radiation has been verified by the additional measurements, the shipments are typically rejected as a condition of purchase between the scrap recycler and the scrap supplier.

The suppliers of scrap metal are seldom aware in advance of the occurrences when radioactive material is present when the bulk shipments of scrap are offered for transportation. The shipments with the detected radioactive material are often returned to the original shipper because the processing facility does not have the resources or chooses not to accept the responsibility for handling the radioactive scrap. In a few cases, the processors have established areas where the problem loads are carefully unloaded and sorted to find the radioactive materials under the scrutiny of a technical specialist with radiation monitoring equipment. Such specialists are usually approved by the radiological health authority of the State where the load is being processed. In these cases, the special handling and radioactive material disposal must be handled between the processor and original shipper. Under an exemption issued by the DOT [2], evaluation of the radiological conditions and the coordination of activities between the processor and original shipper is delegated to the State radiological health official in the State where the radioactive material is detected.

Under normal transportation regulations a regulatory dilemma is present when the recycling facility returns a load to the original shipper. First, radioactive material is clearly being transported because the radiation monitoring equipment detected radioactive material, but the identity and amount of radioactive material is not known. Second, regulations prohibit transporting radioactive material unless specific information is known and described on documentation accompanying the shipment, and when necessary markings, labeling, placarding, etc. are provided.

The DOT exemption for these shipments relies heavily on the radiological health officials in each of the fifty States. These officials have the basic responsibility for the health and safety of the public and the environment in their State for all types of radioactive material and other sources of radiation. They are also usually the persons responsible for resolving radiological conditions during emergencies. The DOT exemption authorizes shippers to offer and carriers to transport these scrap metals without compliance with a number of regulatory provisions. However, in the exemption DOT limits the external radiation levels outside the vehicle to no more than 0.5 mSv/h (50 mrem/h) and the radioactive material must not be readily dispersible. The exemption further requires that the shipper and carrier comply with the conditions prescribed by the State radiation official on a shipment approval document that must accompany the exemption with the transport documents with the shipment. The transport documents must include the following description of the consignment:

> "Scrap metal for recycle containing unidentified radioactive material causing low levels of radiation outside the transport vehicle. Shipment is under Exemption DOT-E 10656 without a determination of materials meeting or not meeting the regulatory definition of radioactive material. The shipment is a minor radiological concern based on considerations of the U.S. Department of Transportation and the State Official signing the attached Shipment Approval document".

The Shipment Approval form signed by the State official has the names and telephone numbers of the facilities and the responsible individuals that originally offered and received the shipment. It describes the radiation levels that were detected outside the vehicle that is carrying the scrap. It identifies the name of the responsible individual at the destination where the material is to be moved under the exemption. It also identifies the name and telephone number of the responsible State official where the shipment is being sent. Any special conditions which the authorizing State official deems appropriate to assure safety of the shipment also appears on the approval form along with the names, titles, and telephone numbers of the investigating and approving State officials.

During calendar year 1993 more than 40 shipments of scrap metal were returned to the sender under the exemption. During the same period several steel mills inadvertently processed radioactive scrap which forced shut downs. The numbers of shipments with detected radioactive material that were "reworked" during 1993 at the processing site without use of the exemption are not known. It may be noted that more radiation monitoring systems are being installed at recycle scrap metal collection facilities as well as at the processing facilities.

3. Uranium Mill-Tailings Problem

The processing of uranium and thorium ores for more than half a century has resulted in millions of cubic meters of processed waste commonly referred to as tailings. In some locations these tailings are considered to present unacceptable hazards to the public and the environment. Most of these tailings are in well defined sites near the mills where the ore was processed years ago. However, some of the tailings had physical and chemical properties, not related to the radiological properties, that were attractive for uses in residential and public locations. Some of these uses included land-fill, soil conditioning, and incorporation into masonry and construction products.

Some of the massive piles of tailings near populated areas were considered to present an unreasonable radiological risk due to radon in the air and radionuclides leaching into surface and ground water. The tailings that were used as land-fill and other materials around homes and residential areas created radon and other low-level radiation hazards that public health officials declared to be unacceptable. An Act of the United States Congress required remediation of these problems by moving the tailings and contaminated soils and debris to isolated hydrologically acceptable locations.

The contractors who processed the ores and generated the tailings for the most part operated under government contracts. The U.S. Department of Energy (DOE) was assigned the responsibility to remediate these uranium mill tailings problems. The first 16 sites that DOE chose to remediate involved both piles of tailings and locations where the tailings had created unacceptable radiation hazards in residential areas. The DOE, and its present contractors, evaluated the conditions and prioritized the sites to be cleaned-up. The DOE recognized that some of these materials had specific activities that required they be transported as regulated radioactive materials while others had specific activities less than the definition of radioactive material for purposes of transportation. DOE recognized it was difficult to know precisely what the specific activity and total activity would be from one load to another in a given area, and it would be excessively costly to assay each and every load being moved to the acceptable locations.

The DOE requested an exemption from DOT that would allow these shipments involving millions of cubic meters of materials to be transported in a safe and effective manner. Relief was requested from the requirements for packaging, shipping papers, marking, labeling, and placarding. In characterizing the hazards of the 16 sites to be remediated, the DOE performed corings and other assays which indicated that most of the materials to be remediated had gross specific activities ranging from 2 Bq/g (40 pCi/g) to 300 Bq/g (8,000 pCi/g), and occasionally materials would contain 2 kBq/g (50 nCi/g). In remediating the sites, the clean-up criteria was based on ^{226}Ra and was specified to be 0.2 Bq/g (5 pCi/g) for ground surface soils and a value slightly higher for soils deeper than 15 cm.

The exemption authorized DOE contractors to transport the materials by motor vehicle and rail without an analysis of the radioactive content of each load.[3] The site characterization surveys were considered to provide adequate information about the material being transported. The cognizant DOE remediation offices had sufficient information to analyze radiological risks if any mishaps occurred during transportation between a clean-up site and destination. Transportation documents accompanying the shipment did not need to contain the information normally required for the transport of radioactive materials. Rather, the shipping documents had a generic description of the materials and actions to be taken in the event of an emergency. Further, the names and telephone numbers of Federal and State officials to be contacted for additional information or assistance in the event of an emergency were included with the documents. There were no specific criteria for the closed vehicles or rail cars that were to be used during transport; other than the materials should be protected so that radiologically significant amounts would not be lost during transport. The contractor carriers were obligated to report all spills or releases exceeding approximately 7 kg (15 lbs) to DOE. DOE was not obligated to report spills to DOT unless the spills were much greater. According to the DOE plans, the same procedures were to be used for handling spills regardless of the activity being above or below the transportation definition for radioactive material. The conveyances were not required to be marked or placarded as would be normally required for bulk packagings. The usual radioactive material placard was considered to be an excessive warning for a material for which the hazards were considerably lower than most materials represented by the placard. Instead, a poster was required on opposite sides which indicated the content was low level radioactive material under the regulatory authority of DOE and DOT. It also stated that additional information could be obtained from an indicated telephone number or on documents in the cab of the conveyance. The exemption also required that the operating personnel have documented training concerning the hazards of the materials and the procedures to be followed in the event of emergencies. Before these bulk transport vehicles used in this program could be used for transporting other materials, they had to be thoroughly cleaned.

4. Radium In Soils and Debris Problem

During the early part of the twentieth century there were a number of locations in the United States where radium was extracted from ores. There are other locations where the radium was processed and used for a variety of purposes, some of which are considered very questionable in light of today's knowledge. As decades have passed, the nature and location of many of these activities were lost, but the long half-life radionuclides that were in the buildings and the refuse from the processing remain. Also, the character of the communities changed, and the old process sites have showed up in everything from fine residential areas to heavy industrial locations.

Following surveys by public health officials, many of these sites have been declared to be unsafe, and, by an Act of Congress, programs are underway to remediate these sites and transport the contaminated soils and debris to safe locations. The U.S. Environmental Protection Agency (EPA) was assigned the responsibility for the remediation of these radium contaminated sites. The standard for radium clean-up of sites was set at 0.2 Bq/g (5 pCi/g) for ^{226}Ra at the surface of the ground and not more than 0.6 Bq/g (15 pCi/g) at depths greater than 15 cm. Since most of the corporations or individuals who were responsible for the contamination problems are no longer in business, the Federal Government has also assumed the financial expenses of the clean-up.

One of the most notable public health hazards existing in the radium contaminated areas was high radon concentrations in air at levels greatly exceeding the standards recommended by EPA. These conditions existed mostly in homes, but were also found in business and public buildings. In the course of investigating the radium and radon problems, EPA and other public health officials learned of the presence/location of contaminated areas from measurements taken with high sensitivity radiation detection instruments and from documented and word of mouth historical information.

In some cases, contaminated buildings and facilities were remediated and the transportation of the wastes to the disposal sites was done in compliance with DOT regulations. In other cases where there were a large number of sites and/or where large volumes of soils and debris needed to be transported, it was very ineffective to perform detailed assays and to transport the materials within the transportation regulations.

Typically to assay a small area of land to be cleaned-up, a grid pattern was established for the area. Radiation dose rates at ground surface or at a fixed distance above points on the grid were recorded, and corings were taken at selected points to determine the specific activity of the materials as a function of depth. From such measurements at an area being remediated, a "source term" could be computed to indicate the total amount of activity to be moved to the acceptable location. Such measurements could also identify the highest as well as an average specific activity of the materials to be transported. From such measurements, the EPA was aware that some of the materials to be transported exceeded the 70 Bq/g (2 nCi/g) level requiring regulated transportation, but most of the materials were below the radioactive materials definition for transportation. However, as would be expected, the occasional "hot spots" high specific activity in the radium clean-up areas were much higher than the "hot spots" specific activity found during uranium mill-tailings clean-up. The specific activity of the radioactive material within a site being characterized varies widely. Differences in chemical processing methods and differences in dispersal into the environment (including weathering and leaching) caused the distribution of the radionuclides present to be different than the classical decay schemes for uranium, thorium, and radium.

For two separate areas where there were a large number of sites to be cleaned-up and where the volumes of the materials to be moved exceeded 10^6 cubic meters, the EPA requested exemptions from DOT that would allow the shipments to be made in a safe and more cost effective manner. In the first application, EPA presented data that indicated that the "source term" determined by site characterization measurements provided a more conservative (larger) estimate of materials moved from the site than a "source term" determined by patterned sampling of materials during loading into the bulk transport vehicles. Both applications requested relief from detailed assays of materials in each bulk container, as well as relief from the usual requirements for detailed shipping documents, packaging, marking, labeling, and placarding.

The exemptions authorized EPA contractors to transport the materials by motor vehicle and rail without an analysis of the radioactive content of each load.[4][5] The site characterization surveys were considered to provide adequate information about the material being transported. The cognizant EPA field offices had sufficient information to analyze radiological risks if any mishaps occurred during transportation between a clean-up site and destination. Transportation documents accompanying the shipment did not need to contain the information normally required for the transport of radioactive materials. Rather, the shipping documents required a specific statement that described the low hazards of the materials being transported and actions to be taken in the event of an emergency. It also included a telephone number to be called for information or in case of an emergency. There were no specific criteria for the closed vehicles or rail cars that were to be used during transport; other than the materials should be protected so that radiologically significant amounts would not be lost during transport. The exemptions required contract carriers to diligently report all spills or releases of material to EPA, but the EPA reporting of occurrences to DOT was to be done only when the consequences were significant. According to the EPA plans, the same procedures were to be used for handling spills regardless of the activity being above or below the transportation definition for radioactive material. The conveyances were not required to be marked or placarded as would be normally required for bulk packagings. The usual radioactive material placard was considered to be an excessive warning for a material with hazards considerably lower than most materials represented by the placard. Rather, a poster was required on opposite sides which indicated the content was low level radioactive material under the regulatory authority of EPA and DOT. It also stated that the material presents minimal risks to workers, the public, and the environment and additional information could be obtained from an indicated telephone number. The exemptions also required that the motor vehicle and rail workers be informed of the hazards of the materials associated with shipment. Before these bulk transport vehicles used in this program could be used for transporting other materials, they had to be thoroughly cleaned. EPA is obligated to provide rail carriers with EPA points of contact for dealing with emergencies or abnormal occurrences. For emergency response personnel and State and local officials along the route to the disposal site, the cognizant EPA office must be prepared to provide hazards information and emergency response guidance for dealing with mishaps during transport.

5. Summary and Other Information

The exemptions for these transportation programs provide shippers relief from detailed analysis of radioactive content, relief from shipping documents, marking, labeling, and placarding. For containment of the radioactive material, judgement is exercised on a case by case basis for the scrap metal shipments, but for the soil/debris the vehicles or bulk containers must assure no radiologically significant dispersal during transport. To aid emergency response personnel in the event of an accident, the vehicles and bulk containers transporting soils and debris are posted with information about the low radiation hazards and the telephone numbers to be contacted for information and/or emergency assistance.

The scrap metal exemption issued by DOT has been received favorably by the recycle scrap industry, the States, and the U.S. Nuclear Regulatory Commission, the Federal agency that regulates the possession and use of most radioactive materials other than naturally occurring radionuclides. It has been a convenient regulating partnership between Federal and State agencies. It helps State officials and industry to track down and eliminate the conditions that allow radioactive materials to enter the recycle scrap metal stream. Further, it relieves the DOT from issuing case-by-case exemptions to shippers and carriers. Unless there is a major mishap during the exempted shipment, the DOT is not notified of the specific actions and approvals issued by the States.

A document was recently prepared by the Institute of Scrap Recycling Industries which describes the problems associated with radioactive material in scrap metal.[6] It is intended for persons at processing facilities as well as collectors. It describes the kinds of radioactive material found in scrap, hazards to personnel and processing facilities, sophisticated and common radiation detection instruments, recommended procedures and practices for dealing with the problems, and a listing of State officials and technical consultants who may be able to provide help. A video training tape is being prepared which covers much of the information in the document. A Spanish translation of the document is expected soon.

The uranium mill-tailings and the radium contaminated sites that are covered by the exemptions that have been issued represent a small fraction of the sites in the United States that are known to be contaminated and need to be remediated. Some of these other sites have materials with specific activities that are higher and some that are lower than the materials covered under the existing exemptions. In some cases the remediation problems are further complicated by non-nuclear contaminants that present hazards equal to, greater than, or less than the radiological hazards. Writing regulations that will adequately address the problems with the hazards and the practical aspects of transportation will be very difficult. For the near future it is expected that transportation programs for these materials will involve issuance of exemptions, rather than revision of the regulations. The exemption process has the advantage of allowing the evaluation of many conditions and public risk, and the use of value judgements on a case-by-case basis.

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RISK ASSESSMENT

(Session 6)

Chairman

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PROBABILISTIC SAFETY ASSESSMENT FOR THE TRANSPORT OF RADIOACTIVE WASTE TO A UK REPOSITORY AT SELLAFIELD

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Abstract

A study has been undertaken to provide a detailed understanding of the radiological and non-radiological risks associated with the transport of radioactive waste from the sites at which waste is produced in the UK to a proposed deep repository at Sellafield, and to ensure that these risks meet the design targets specified by Nirex. The routine transport collective dose to members of the public was assessed to be 0.2 man Sv per year, which is only about 0.004% of the natural background dose. Accident frequencies were calculated using event tree methodology. The radiological consequences of accidents were assessed using the probabilistic computer code CONDOR. The risk expectation value was calculated to be 1.5×10^{5} *-* 8.6×10^{6} *latent cancer fatalities per year (depending on the transport mode scenario). These values are significantly lower than the corresponding predictions for non-radiological accident fatality rates, 0.05 - 0.035 fatalities per year. The radiological accident risk for the most exposed individual member of the public was assessed to be* 5×10^{11} *-* 1.7×10^{11} *per year, very much less than the Nirex target of* 5×10^{7} *per year. Plots of societal risk were shown to lie in the region of 'negligible risk', as defined by the UK Health and Safety Commission for non-radioactive dangerous goods transport.*

1. INTRODUCTION

UK Nirex Ltd (Nirex) has been established to develop and operate a deep repository for the disposal of intermediate and low level waste (ILW and LLW) arising in the United Kingdom. Nirex is also responsible for producing standards for the design of waste packages, and for developing integrated transport arrangements for the movement of packaged ILW and LLW from UK waste producing sites to the repository. Nirex is concentrating its investigations on Sellafield in West Cumbria as a potential location for the deep repository.

This paper describes a study undertaken to provide a detailed understanding of the radiological and non-radiological risks associated with the transport of ILW and LLW to the proposed repository at Sellafield, and to ensure that these risks meet the design targets specified by Nirex as part of company policy.

Two other papers presented at this seminar describe the integrated transport arrangements [1] and the package designs [2] being planned by Nirex.

All waste must be transported in conformity with the IAEA Transport Regulations [3] applicable at the time. This study and its subsequent phases will provide a useful basis when considering the safety aspects of any future changes in the Regulations.

The following sections of this paper describe:

- The input data (including the selection of representative waste streams)
- The calculation of routine transport doses
- The assumed response of the packages to impact and fire accident conditions
- The assessment of accident frequencies and radiological consequences
- Risks presented in several forms
- Planned future development work
- The main conclusions of the study.

2. INPUT DATA

Three classes of waste package were included in the assessment:

- Type B packages containing ILW (reusable shielded transport containers holding waste in four 500 litre drums)
- Industrial packages containing ILW
- Industrial packages containing LLW.

This represents a simplification of the set of packages currently envisaged [2], but the principal packages are covered. Also, accident response information is not yet available for all package designs, some of which are at an early conceptual stage.

Nirex policy is that rail transport shall be used wherever practical for the transport of waste to the repository, although it is not possible to utilise rail transport for all waste transport. Two transport mode scenarios were considered for assessment purposes:

- Rail-only: all packages would be transported by rail.

- Road/rail: all packages below the European Union (EU) weight limit of 40 tonnes for unrestricted road transport would be transported by road, and all heavier packages would be transported by rail.

Even in the rail-only scenario, some transport by special road vehicles will be required where there is no on-site railhead for the direct loading of heavy packages on to rail wagons. The risks of these short road journeys are not included in the assessment. However, doses to workers involved in the road-to-rail trans-shipment operation have been included in the results of the rail-only scenario under 'Handling'.

A maximum waste disposal volume scenario was assumed, corresponding to 1240 ILW and 480 LLW packages to be transported per year [4]. These packages originate at nearly 30 sites in the UK and the corresponding annual transport distances are:

	ILW package km per year		LLW package km per year	
	Road	Rail	Road	Rail
Road/rail scenario	315000	331000	206000	
Rail-only scenario		625000		194000

In excess of 350 separate waste streams were identified [4] so it was necessary to group these and identify representative streams for detailed analysis. Two methods were adopted.

First, an approximate risk ranking parameter was computed for each waste stream. This parameter was defined to include the important factors affecting transport risk, but was sufficiently simple to be evaluated for all the waste streams. The factors included were the package km, the quantity of activity (in Bq) per package, the fractional release in defined accident conditions, and a hazard index for the waste stream (based on the A_2 value [3]). The absolute value of the parameter has no meaning, but the relative values can be used to rank the streams in terms of their importance as contributors to the total transport risk. It was found that the computed values of the parameter spanned eleven orders of magnitude for the 350 waste streams. About 30 streams gave rise to 99% of the total hazard index, so these formed the focus for further study.

Second, the streams were formed into groups on the basis of their origin and general characteristics. The following groups were identified:

- Fuel element debris
- Plutonium contaminated material
- Ion exchange resins
- Sludges and flocs
- Special wastes and miscellaneous contaminated items
- Uncategorised (a small number of special, high activity wastes)
- Other ILW
- LLW.

A single waste stream was selected to be representative of each of these groups, using the ranking data. Initially three streams were selected to represent all LLW, with one being further divided into combustible and non-combustible material. However, the differences in radionuclide composition and in the radiological consequences of releases for these streams were so small that finally only one stream was selected to represent all LLW, the stream giving the worst predicted consequences.

3. ROUTINE TRANSPORT DOSES

Routine transport doses were calculated using a methodology similar to that employed in the IAEA INTERTRAN computer code [5], but with changes to make the algorithms more appropriate for UK road and rail transport conditions.

Table 1 Routine Transport Collective Doses

Scenario, waste and mode	Annual Collective Doses (man-Sv)						
	Workers			Members of the Public			
	Crew	Handlers	Total Workers	Alongside Route	Sharing Route	During Stops	Total Public
<u>Road/rail</u>							
LLW by road	0.08	0	0.08	0.015	0.017	0.02	0.052
ILW by road	0.11	0	0.11	0.02	0.022	0.027	0.069
ILW by rail	0.062	0.019	0.081	0.0049	0.053	0.021	0.079
Totals	0.25	0.019	0.27	0.04	0.092	0.068	0.2
<u>Rail Only</u>							
LLW	0.041	0.016	0.057	0.0033	0.035	0.017	0.055
ILW	0.12	0.036	0.16	0.0092	0.099	0.038	0.15
Totals	0.16	0.052	0.22	0.013	0.13	0.055	0.21

An average ILW package external dose rate of 31 μSv h^{-1} at 2m from the surface was obtained from Nirex shielding calculations. An average external dose rate of 80 μSv h^{-1} on the surface of LLW packages was obtained from available Nirex inventory data.

The resulting collective doses are shown in Table 1. The collective dose for members of the public was assessed to be about 0.2 man Sv per year for both road/rail and rail-only scenarios. Using a risk factor of 0.05 Sv^{-1} this corresponds to an expectation value of 0.01 fatalities per year. For comparison, the collective dose to members of the public due to naturally-occurring sources of radiation along the transport routes was calculated to be 5500 man Sv per year. The additional collective dose due to the waste package transport therefore represents an increase of only 0.004%.

In addition, estimates of the maximum individual dose were made. Three hypothetical individuals were considered: a rail commuter regularly positioned on a station platform while waste packages passed by; a person living near traffic lights on the road approach to the repository; and a person living near a road-to-rail trans-shipment point. The maximum individual doses in all cases were estimated to be less than the Nirex target dose for members of the public of 0.05 mSv y^{-1}. However, it was recognised that further work is desirable to identify more closely the exposure times and distances for the critical groups.

4. PACKAGE ACCIDENT RESPONSE

The accident response of the waste packages was based on the IAEA Transport Regulations [3]. It was pessimistically assumed that the package containment would fail completely in accident conditions marginally more severe than those of the IAEA package tests (ie an impact more severe than 2.4 m s^{-1} against an unyielding target or a non-trivial fire for an industrial package, and an impact more severe than 13 m s^{-1} against an unyielding target and a 30 minute fully engulfing fire for a Type B package). This is a conservative assumption, for the design process is likely to result in packages with margins beyond these limits.

5. ACCIDENT FREQUENCIES

Accidents were identified which have the potential to exceed the IAEA Transport Regulation test conditions and therefore result in radiological consequences. These included:

- Fall from a high bridge
- Impact with a lineside or roadside object (tunnel abutment etc)
- Collision with a second train or road vehicle
- Railhead transfer accident
- Major fire (involving another vehicle carrying flammable material)
- Minor fire.

Eight accident condition categories were defined using impact and fire severity parameters. These categories covered all identified accident scenarios, including very low probability extreme conditions. The two most severe conditions considered were a fire in a tunnel involving a second train of flammable goods tankers, and impact against an unyielding target at 40 m s^{-1}.

The historical record for the world-wide transport of radioactive materials is very good, so there are few instances of transport accidents with radiological consequences. However, that presents difficulties for an assessment such as this. For example, no meaningful estimate of the probability of a high-speed impact of a waste package on a tunnel abutment can be derived from the fact that such an event has never occurred. Fortunately, event tree methodology can be used to estimate such probabilities, as described in the following paragraphs.

Event tree methodology involves dividing the accident development into a number of steps, beginning with an initiating event (such as derailment) and assigning probabilities for the severities of conditions (such as the speed) which are relevant to further steps.

Historical data for UK transport were used to derive initiating event probabilities as follows:

- A rail wagon derailment probability of 1.54×10^{-7} per wagon km for bogie freight wagons
- A rail (same-line) collision probability of 2.4×10^{-7} per train km
- A fatal or serious road accident probability of 7×10^{-8} per vehicle km for motorways and 2.1×10^{-7} per vehicle km for major roads.

Additional information (speed distributions, fire probabilities, etc) was obtained from published and unpublished British Rail and UK Department of Transport sources. Data concerning specific hazards along the routes (eg location and heights of bridges, and nature of underlying surface) were obtained from detailed route and map surveys.

Probabilities for all the identified accident scenarios were developed in this way, and an example is shown below. Where data were uncertain, pessimistic values were adopted.

Probability of impact at 13-27 m s^{-1} with an unyielding tunnel abutment

=	derailment probability (1.54×10^{-7} per wagon km)
x	probability of sufficient wagon displacement from line of travel to strike the abutment (0.16)
x	probability of presence of tunnel abutment at derailment location (0.0031)
x	probability of abutment being effectively unyielding (0.72)
x	probability of wagon speed at derailment being in the range 13-27 m s^{-1} (0.11)
=	6×10^{-12} per wagon km.

Accident frequencies were simply obtained by multiplying these probabilities by the total distances travelled per year.

6. RADIOLOGICAL CONSEQUENCES OF ACCIDENTS

6.1 Release Fractions

The industrial packages and reusable Type B transport containers will form the outer containment boundary. In accidents slightly more severe than the IAEA Transport Regulation test requirements it was assumed that this boundary will fail. However, for the very strong Type B packages in particular, it is virtually inconceivable that the packages will burst open completely under accident conditions. Damage is likely to be confined to lid-body interface gaps opening up, except in the most extreme cases. Since the retention of the radioactive contents in an accident cannot be quantified at the present stage of package development and testing, no retention was assumed for the purpose of the assessment. Therefore the results are probably quite pessimistic in this respect.

Within the outer container containment boundary is the wasteform itself. Nirex has conducted programmes of work to investigate the impact response of immobilized ILW, including experiments, theoretical studies and literature reviews. Several different designs of 500 litre drum, all meeting the Nirex waste package specification but containing various inactive simulated wasteforms, have been dropped in different orientations from a range of heights. The drums were then subjected to detailed examination. Particle size analysis of the loose debris was undertaken using sieving, laser particle sizing and aerosol analyser techniques. A distribution of results was obtained and the worst-case respirable release fraction (1.8×10^{-5} for a 9 m drop) was pessimistically applied to all the example ILW streams.

Nirex has also conducted a major programme of work to investigate the effects of heat on immobilized ILW. Small-scale and full-scale inactive simulant samples, and small-scale active samples, have been heated to 300°C and 1000°C. A computer heat transfer model to predict the temperature distribution in a 500 litre drum of immobilized waste has been developed and verified using the experimental data. In general it was found that the release fractions were dependent both on the radionuclide and on the type of wasteform. Example release fractions derived from this work for a two-hour 1000°C fire and employed in the safety assessment study are:

- 1.5×10^{-11} for Co-60 in fuel element debris
- $3.? \times 10^{-6}$ for Pu-239 in plutonium contaminated material
- 1.66×10^{-3} for Cs-137 in sludges.

For LLW, a respirable release fraction of 10^{-3} was adopted under impact conditions, based on a literature search and flowing-air entrainment data. These data did not include supercompacted waste; it is expected that most or all LLW will be supercompacted before transport to the repository, and that the resulting wasteform will have a much lower impact release fraction than assumed here.

For fires involving LLW, pessimistic respirable release factions were adopted for caesium, for other nuclides in non-combustible material and for other nuclides in combustible material, based on data in the literature. Since this study was completed Nirex has investigated the behaviour of supercompacted LLW in fires. The preliminary results indicate that the assumed release fractions are likely to overestimate the releases from supercompacted LLW by at least one or two orders of magnitude.

6.2 *Dispersion and Health Effects Calculations*

The radiological consequences of release were evaluated using the CONDOR computer code [6]. This is a probabilistic consequence assessment code which was developed jointly in the UK by AEA Technology (SRD), Nuclear Electric and the National Radiological Protection Board. It models the downwind dispersion of released activity, taking account of dry and wet deposition processes, radioactive decay and any variation in the meteorological conditions. From the resulting distribution of the released material in the environment, the code evaluates the radiation doses to man via a number of exposure pathways: cloudshine, groundshine, inhalation, resuspension and ingestion of contaminated food.

CONDOR evaluates the radiological consequences for a large number of different meteorological sequences, in which the weather conditions change hourly, and calculates a consequence probability distribution. Hourly meteorological data from a representative site over an eight-year period were sampled to derive the input data for the assessment.

CONDOR calculations were run using the population distribution around three representative sites. The population distribution data were taken from the UK census, The three sites were chosen to be representative of 'urban', 'intermediate', and 'rural' locations along the waste transport routes. The population densities (in people km^{-2}) for the respective sites were:

- 4165, 1433 and 118 averaged out to 1 km
- 5181, 1026, and 62 averaged out to 10 km.

It was pessimistically assumed that no countermeasures would be taken, such as evacuation and food bans.

The calculations included both individual doses (doses which would be received by an individual who was located at a specified distance from the release) and societal doses (probabilistic distributions of frequency against dose for the exposed population).

For all the releases assessed, the results indicated no early fatalities, only risks of latent cancer deaths.

If a release were to occur, the wastestreams selected to be representative of plutonium contaminated material and uncategorised material (see Section 2) would give rise to the most significant radiological consequences. The inhalation exposure pathway led to nearly 99% of the total predicted dose for the plutonium contaminated material (plutonium radionuclides being by far the most important in this stream). For the uncategorised material stream the groundshine pathway resulted in about 63% of the total dose, with 30% resulting from the ingestion pathway and almost all the remainder from inhalation. Caesium and cerium radionuclides were the most important in this stream.

Averaged over all weather conditions, the predicted individual at a distance of 0.1 km from a release ranged from 9 mSv to 1 nSv for the different representative waste streams.

The expectation values of the societal consequence distributions, assuming a release to have occurred, were all less than one latent cancer fatality. They ranged from about 0.3 to less than 10^{-6} fatalities for the different representative waste streams, the largest values being associated with releases in urban areas of high population density. For each stream and release location, the probability of ten or more latent cancer fatalities was less than 10^{-5}, if a release were to occur.

7. RISKS

7.1 Introduction

The frequencies of potential accidents (see Section 5) and radiological consequences which would result if the accident occurred (see Section 6) were combined to evaluate risks. Three separate risk measures were evaluated and compared with available criteria and levels of acceptance:

- The risk expectation value (the risk derived from the average value of the societal consequence distribution)
- The individual risk (the risk to the most exposed hypothetical individual)
- The societal risk (the probability distribution of frequency against number of fatal cancers in the exposed population).

7.2 Risk Expectation Value

The risk expectation value R_B was calculated as follows:

$$R_B = \Sigma_i \Sigma_j \Sigma_k F_{ij} P_k N_{ijk}$$

where F_{ij} = frequency of each accident category i for each waste group j

P_k = probability of population distribution in region k

N_{ijk} = number of latent cancer fatalities conditional upon accident category i for example wastestream j in region k.

Hence R_B = 1.5×10^{-5} latent cancer fatalities per year for the road/rail transport scenario, and 8.6×10^{-6} for the rail-only scenario.

The plutonium contaminated material group provided the largest single contribution to the total predicted value of R_B.

There is no UK or Nirex risk criterion for comparison with R_B. However, comparisons with routine transport dose risks (see Section 3) and non-radiological transport accident risks help to place the radiological accident risk R_B in perspective. Accidents will inevitably occur during the transport of any commodity, and UK statistics for fatal accidents in general freight were analysed to derive fatality frequencies for waste transport accidents in which radiation exposure is not a factor.

	Road/rail scenario	**Rail-only scenario**
	Expected number of fatalities per year	
Non-radiological transport accidents	0.05	0.035
Routine transport radiation exposure	0.01	0.01
Radiation exposure in accidents	0.000015	0.0000086

R_B provides a useful measure of the average risk. However, it does not differentiate between the contributions to the total risk from the higher consequence, lower frequency events and those from lower consequence, higher frequency events. Therefore other risk measures were also evaluated, as described in the following sections.

7.3 Individual Risk

The transport system was approximated by a straight line running the length of the UK, with accidents assumed to be equally likely at any point along the line. The most exposed hypothetical individual was assumed to be located at the centre of the line, near the repository at Sellafield. The risk per year experienced by this hypothetical individual is given by:

$$R_I = 2 L^{-1} \Sigma_i \Sigma_j F_{ij} Q_{ij}$$

where $Q_{ij} = 0.5 \Sigma_m (W_{mij} + W_{m+1,ij})(X_{m+1} - X_m)$

W_{mij} = fatal cancer risk to an individual at the centre of the straight line due to accident scenario i occurring to wastestream j at point m along the line

X_m = distance from centre of line to point m on straight line

L = length of straight line

Hence R_I = 5×10^{-11} per year for the road/rail scenario and 1.7×10^{-11} per year for the rail-only scenario.

These values are very much less than the Nirex company target of 5×10^{-7} per year for the maximum risk to an *individual member of the public. In addition, they are very* much less than the figure of 10^{-6} per year which the UK Health and Safety Executive has advised is broadly acceptable for a member of the public, provided there is benefit to be gained and proper precautions are taken [7].

The total individual risk is spread along the route because accidents could occur at different locations. Partly for this reason, *the UK Health and Safety Commission (HSC)* chose to assess societal rather than individual risk in a recent study of (non-radioactive) dangerous goods transport in the UK [8]. Societal risk for the Nirex waste transport operation is discussed in the following section.

7.4 Societal Risk

Societal risk plots of the frequency of N or more latent cancer fatalities against N are shown in Figure 1.

The predicted frequencies of multiple-fatality accidents due to the transport of ILW and LLW are extremely small. Although there are no criteria for the acceptability of societal risks for application to radioactive materials transport in the UK, the HSC report on dangerous goods transport in the UK included societal risk acceptability guidelines [8]. However, this study did not include radioactive materials transport. The perceived nature of the hazard from radioactive material, and the perceived benefits of transporting the material, are rather different from those associated with the transport of, for example, petroleum products. The societal risk guidelines developed in the HSC document are therefore not directly transferable to radioactive waste transport, but they still provide one of the few authoritative reference points available.

The HSC guidelines define three regions on the societal risk plot [8]. An upper line defines risks which are considered to be just tolerable - transport operations with risks above

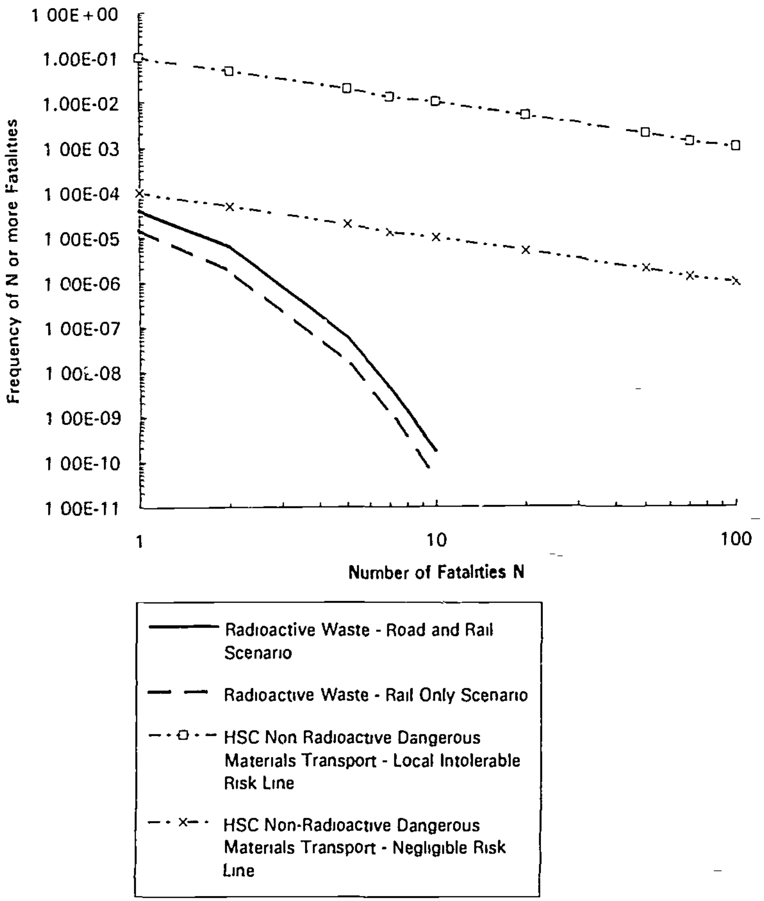

Figure 1 Societal Risk for Total Waste Transport

this line should be banned or modified to reduce the risks irrespective of cost. The frequencies for one and ten fatalities on this just-tolerable line are 10^{-1} and 10^{-2} respectively. Below this line is a region where risks should be reduced so far as is reasonably practicable. A lower line defines negligible risks - transport operational risks below this line "should be ignored" [8, para 80]. The frequencies for one and ten fatalities on this negligible-risk line are 10^{-4} and 10^{-5} respectively. These three regions of risk are over-plotted on the results of Figure 1.

It is clear from Figure 1 that both of the radioactive waste transport scenarios have societal risk curves which are well below the negligible-risk line. The HSC lines are intended for application to one locality (such as a port area) and the report indicates that higher societal risks are acceptable for nationally distributed transport operations, such as those proposed by Nirex.

7.5 Comparison of Transport Mode Options

The study evaluated the risks for transport options based on assumed scenarios involving a mixture of road and rail transport, and rail transport only. The rail-only risks were assessed to be lower by up to a factor of about three. However, the absolute magnitudes of all the assessed risks are very small indeed. All the risks are well below Nirex company targets and the UK regulatory limits and guidelines, and are so small that it would not normally be considered worthwhile to undertake formal assessments to ensure that they are as low as reasonably practicable. The differences between the road/rail and rail-only risks are therefore not judged to be significant, and other factors are likely to be more important in making the transport mode decisions.

8. FURTHER WORK

Nirex plans for future work include the revision and updating of the safety assessment study. Data which will be incorporated include:

- The most recent accident data
- Data from the package development programmes
- The most recent UK Radioactive Waste Inventory data.

Work is also under way to quantify the transport risks associated with incorrect packaging of waste. Although rigorous quality assurance arrangements will be in place at the waste arising sites to meet Nirex and regulatory requirements, there will remain the possibility of using the wrong type of package, incorrectly specifying the waste, or incorrectly assembling the package. The consequences of these unlikely events being followed in turn by an accident will therefore be assessed.

9. CONCLUSIONS

A comprehensive probabilistic safety assessment for the transport of radioactive waste to a UK repository at Sellafield has been completed in its initial phase.

The predicted radiological risks, measured in terms of the expectation value, individual or societal risks, are extremely low. All the risks are well below Nirex company targets and UK regulatory limits and guidelines, and are so small that it would not normally be considered worthwhile to undertake formal assessments to ensure that they are as low as reasonably practicable. It will be necessary to implement operational procedures at the time of transport (such as monitoring and restriction of access to stationary packages) to ensure

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that doses are as low as reasonably practicable, but no weaknesses in the package designs were identified which should be addressed to reduce transport risks.

Non-radiological risks predominate. To a first approximation these are proportional to the total package-kilometres, so reductions in this parameter can result in significant improvements in overall transport safety. The proposed siting of the repository at Sellafield, very close to the main UK waste producing site, is beneficial in this respect.

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SAFETY ANALYSIS OF THE TRANSPORTATION OF RADIOACTIVE WASTE TO THE KONRAD WASTE DISPOSAL SITE

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Abstract

A safety analysis has been conducted for the transports of low - to medium-level radioactive waste to the planned final repository KONRAD at Salzgitter in Germany. The expected annual transport volume is about 3400 shipping units The main objective of the study was the assessment of radiological risks from transport accidents in the region of the final repository. Two shipping scenarios - 100% transportation by rail and 80% transportation by rail, 20% by road were analyzed. The availability of an extensive and detailed waste data survey of the German Federal Office for Radiation Protection (BFS) was of fundamental importance to the safety analysis. To a large extent probabilistic safety assessment techniques have been applied to take into account the variational range of quantities and parameters which determine potential radiological consequences of transport accidents and the expected frequencies of their occurrence Influencing quantities and parameters are. 217 waste categories representing different types of waste, packaging and radionuclide inventory; 9 severity categories to classify the spectrum of accidental impacts; 8 waste package groups representing the release behaviour of containers and waste products; different loading configurations of vehicles and varying numbers of waste wagons within a train, expected frequencies of accident severity categories and, with respect to rail transports, of the number of waste wagons affected in an accident; variability of atmospheric dispersion conditions and consequently of potential radiation exposures from accidental releases. The results of the probabilistic risk analysis of transport accidents in the region of the final repository KONRAD are expressed as cumulative complementary frequency distributions. These distribution curves show expected frequencies of radiological consequences such as potential effective doses of individuals in the region of the repository.

1 INTRODUCTION

The former iron ore mine KONRAD situated within the city limits of Salzgitter in Germany is planned to be used as deep underground final repository for radioactive wastes with negligible heat generation About 95% of the radioactive waste volume resulting from nuclear industry including reprocessing of German spent fuel abroad, from research, medicine and other applications could be disposed of in the KONRAD waste repository Detailed safety analyses were performed concerning both the operating and the post-operating phase of the planned repository. This work has resulted in the establishment of preliminary waste acceptance criteria. Presently the licensing procedure is still underway.

Although questions concerning possible risks associated with waste transportation are not a formal part of the licensing procedure, they nevertheless play an important role in public debate, especially in the local region of the repository. On behalf of the German Federal Ministry for Environment, Nature Protection and Nuclear Safety the GRS has conducted an extensive safety analysis of the transportation of radioactive wastes to the KONRAD waste disposal site [1] The main objective of the study was the assessment of radiological risks from transport accidents in the region of the final repository. Two shipping scenarios, which can be considered to bound the real conditions were analyzed:

- 100% transportation by rail
- 80% transportation by rail, 20% by road

The expected annual transport volume to be shipped to the final waste repository is about 3400 shipping units, a shipping unit being either a cubical container or one or two cylindrical containers on a pool palette.

2 SAFETY REQUIREMENTS FOR WASTE PACKAGES

Requirements concerning activity contents, waste products and qualification of waste containers result from:

- the waste acceptance criteria derived from a detailed safety analysis (operating and post-operating phase) for the KONRAD repository
- the transport regulations for dangerous goods

Both, the transport regulations for the shipment of radioactive materials which are based on the Safety Series No. 6 of the IAEA [2] and the "KONRAD preliminary waste acceptance criteria" [3] represent the framework of the safety requirements for the waste packages

2.1 KONRAD Preliminary Waste Acceptance Criteria

The waste acceptance criteria are the result of the safety analysis for the final waste repository. They represent a set of requirements which originate from

- incident analysis (operating phase)
- thermal influence to the host rock
- criticality safety
- limitation of releases of volatile radionuclides from the repository (operating phase)
- limitation of dose rates of packages

The systematics of the waste acceptance criteria distinguish between two categories of waste containers.

- waste container class I, basically equivalent to strong industrial packages
- waste container class II, packages with increased qualification to withstand severe mechanical or thermal impacts

There are three main types of standardized transport containers accepted for disposal: Cylindrical concrete containers, cylindrical cast iron containers and cubical containers (sheet steel, concrete or cast iron). In addition because of differences in release behaviour following mechanical and/or thermal (fire) impact six different waste form groups (e g bituminized, cemented, high pressure compacted waste) are distinguished for waste container class I Also different levels of leak tightness of containers are provided for in case of high activity contents of volatile radionuclides With respect to the acceptable radionuclide inventories of waste containers requirements resulting from different safety domains such as incident analysis or limitation of heat generation have to be observed simultaneously. This, of course, means that the most limiting of the parallel requirements will restrict acceptable radionuclide contents of waste containers

2.2 Transport Regulations

The national requirements concerning the transportation of waste containers to the final repository are essentially identical to the international IAEA Regulations for the Safe Transport of Radioactive Material. The systematics of these requirements is quite analogous to the waste acceptance criteria:

- limitations of dose rates
- different package categories (e g strong industrial, Type A, Type B packages)
- distinction of the physical/chemical form of activity contents (e.g. special form, LSA-II, LSA-III, SCO-I, SCO-II)
- limitation of activity contents or of activity concentrations in relation to properties of package and physical/chemical form of radionuclides

3 WASTE DATA BASE

Both sets of requirements - the transport regulations and the waste acceptance criteria - do not provide any information on the type, quantity and properties of the radioactive waste actually produced and requiring disposal. Consequently, the availability of an extensive and detailed waste data survey of the German Federal Office for Radiation Protection (BFS) was of fundamental importance to the safety analysis. Completed in summer 1990, the survey was conducted with the aim of obtaining comprehensive data on the radioactive waste produced and to be anticipated in the foreseeable future in the Federal Republic of Germany. The spectrum of radioactive wastes suitable for disposal in the KONRAD waste repository comprises 217 reference waste types. For each reference waste the following information is available:

- origin/originator
- type of waste
- conditioning/immobilization type
- type of packaging
- radionuclide inventory
- local dose rate of the package
- mean annual number of packages

4 METHODS AND DATA FOR PROBABILISTIC ACCIDENT RISK ASSESSMENT

The risk of transport accidents is determined by the frequency of accidents leading to a release of radioactive substances and the potential radiological consequences, such as radiation exposure of persons and contamination of the biosphere. To assess the risk associated with transport accidents, the region in the proximity of the final repository KONRAD is considered and this is defined as the area within a radius of 25 km around the installation. This region, for which the accident risk is calculated, is chosen since it covers all waste transports converging in the vicinity of the final repository and the rail and road transport routes representative for this region. This includes the Braunschweig marshalling yard, through which a large proportion of the waste transport is expected to be routed.

To a large extent probabilistic safety assessment techniques have been applied to take into account the variational range of quantities and parameters which determine potential radiological consequences of transport accidents and the expected frequencies of their occurrence. Influencing quantities and parameters are:

- 217 waste categories representing different types of waste, packagings and radionuclide inventories,
- 9 severity categories to classify the spectrum of accident impacts,
- 8 waste container groups representing the release behaviour of containers and waste products,
- different loading configurations of vehicles and varying numbers of waste wagons within a train,
- expected frequencies of accident severity categories and, with respect to rail transports, of the number of waste wagons affected in an accident,
- variability of atmospheric dispersion conditions and consequently of potential radiation exposures from accident releases.

4.1 Severity categories and accident frequencies

The mechanical and/or thermal impact on the waste containers caused by the accident together with the properties of the waste containers and the waste product they contain (e.g. cement/concrete, bitumen, compacted waste etc.) determine the extent to which radioactive materials are released into the environment. To permit a quantitative evaluation of accident risks, the broad spectrum of possible accident impacts must be condensed into a finite number of severity categories, each of which in turn encompasses a wide range of possible effects on waste containers caused by accidents. For the purposes of the present study, nine severity categories (SC) were defined with the characteristics shown in Fig.1.

Detailed analyses have been performed to determine expected accident frequencies per vehicle-km for heavy trucks (articulated lorries), per goods-train-km and per rail car-km and to assess the relative frequencies of the 9 severity categories in each case. The overall accident rate for articulated lorries (damage to vehicle > 4000 DM) on federal motorways was determined to be $3.5 \cdot 10^{-7}$ km^{-1}. For freight trains an accident rate (damage to rail car > 3000 DM) of $5 \cdot 10^{-7}$ per train-km and of $2.5 \cdot 10^{-8}$ per rail-car-km was established. Details of the analysis made for this purpose of German rail accident statistics of goods-trains covering the 10 year period 1979 to 1988 are given in [4]. From this accident analysis also the relative frequencies of the 9 severity categories were determined. Taking accidents of freight trains as an example these relative frequencies are included in Fig. 1. All events where only a fire occured without prior mechanical impact were included into severity categories 2 or 3 (impact velocity < 35 km/h)

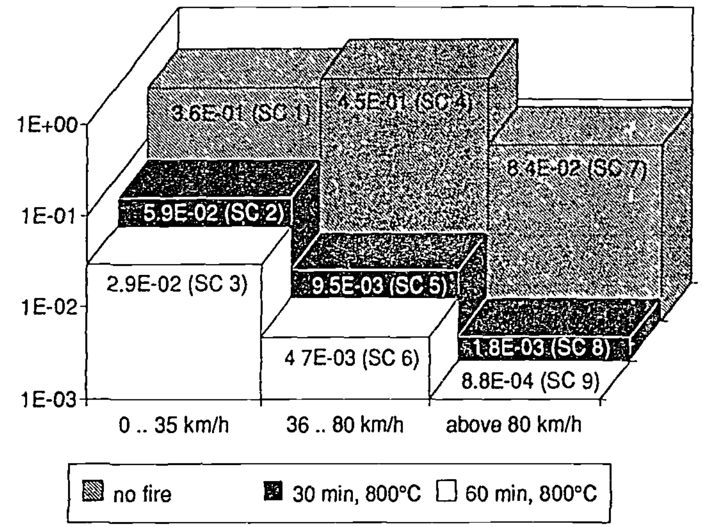


Fig.1 : Relative frequencies of severity categories (SC) for goods train accidents

4.2 Waste container groups and release fractions

Releases of activity from accident impacts depend on the properties of the transport container and the waste product which it contains. For this reason, the range of waste containers in use is divided into <u>waste container groups (WCG)</u> with the aim of categorizing waste containers with similar release characteristics in a single group. Eight waste container groups are distinguished:

WCG 1	Bituminized waste in sheet steel cubical containers
WCG 2	Non-immobilized and non-compactable metallic and non-metallic waste in sheet steel cubical containers
WCG 3	Metallic waste in sheet steel cubical containers
WCG 4	Compacted waste in sheet steel cubical containers
WCG 5	Waste immobilized in cement in sheet steel cubical containers
WCG 6	Bituminized waste in concrete containers
WCG 7	Waste immobilized in cement in concrete containers
WCG 8	Waste in cast iron containers

Airborne fractional releases from waste packages suffering a transport accident were determined for the 8 waste container groups and 9 severity categories defined above for particles in the following size range intervals of aerodynamic equivalent diameter (AED). 0 - 10 µm, 10 - 20 µm, 20 - 50 µm and 50 - 70 µm. For particles below 10 µm (excluding H3, C14, halogens) the fractional releases are shown

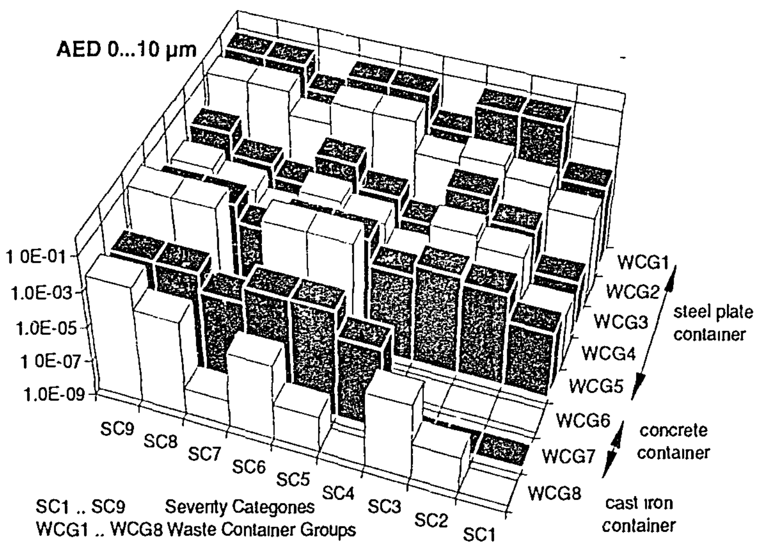


Fig. 2: Release fractions from waste containers for different severity categories

in Fig. 2. In each case it is assumed that the mechanical impact to waste containers is equivalent to an impact onto an unyielding target with a speed equal to the upper limit of the respective velocity interval. Accidents with speeds above 80 km/h are treated as accidents with a velocity of 110 km/h. Releases from fire impact are modelled assuming a fire fully engulfing waste packages with a thermal energy input equivalent to a fire either of 800°C and 30 min duration or of 800°C and 60 min duration.

4.3 Probabilistic source term determination

A computer code was developed to simulate a wide spectrum of waste transport and accident configurations using Monte Carlo sampling techniques In a first step a large number (e.g. 10000) of source terms are generated to represent possible releases of radionuclides from transport accidents. Accident events in which the integrity of waste packages is retained and consequently no releases occur are also recorded. Source terms are determined separately for road and rail transports.

A source term generated by the accident simulation program represents the released activities of individual radionuclides for the simulated accident configuration. The radionuclide-specific activities are determined by the activity content of the waste packages involved in the accident and the fraction assumed to be released into the atmosphere.

For the purpose of subsequent analysis of possible radiological consequences and their expected frequencies of occurrence the following information is assigned to each source term·

- The severity category (k = 1,2,3.. 9)
- The conditional probability of the accident configuration (given an accident occurs)
- A radiological hazard index calculated from the radionuclide-specific activity which permits an approximate relative classification of different source terms with respect to potential radiological consequences

To facilitate the analysis of environmental consequences, the large number of source terms must first be appropriately grouped into a limited number of source term groups. In a next step for each source term group a representative source term is determined designated as release category

The source terms are first arranged in ascending order according to the radiological hazard index. This is done separately for purely mechanical and combined mechanical/thermal severity categories. The reason for this is that in the calculation of radiological consequences an effective release height of 2 m is assumed for accidents with only mechanical impact and of 50 m in the case of mechanical impact followed by a fire

Source term groups are then formed by combining source terms with approximately equal hazard indices in a way that the range of radiological hazard indices of source terms having high hazard indices does not differ substantially. This procedure is intended to assure representativeness particularly for the source terms resulting in higher radiological consequences.

In a next step for each source term group a representative source term, called release category, is derived. Without going into detail here it can be demonstrated that the limited number of release categories determined in this way very well represent the spectrum of potential releases from transport accidents including their probabilities of occurrence. In summary, ten such release categories each have been generated by the simulation program for accidents during transportation by goods train, by truck, and in the Braunschweig marshalling yard. In each case 5 release categories are representative for accidents with purely mechanical impact on shipping units, and 5 release categories for accidents with mechanical impact and subsequent fire. The expected frequency of occurrence has been determined for each of these release categories.

5 RESULTS

Potential radiological consequences such as radiation exposure of persons and ground contamination have been calculated by using the accident consequence code UFOMOD. In calculating radiation exposure, the following exposure pathways are considered.

- cloudshine (radiation from the passing cloud)
- inhalation (intake of activity with respiratory air)
- groundshine (external radiation from radionuclides deposited on surfaces, 70 a)
- ingestion (intake of activity with food, integration time 70 a)
- resuspension (reentry of radionuclides deposited on surfaces into the air with subsequent inhalation, 70 a integration time)

The calculations take into account the relative frequency of different atmospheric dispersion conditions in the region of the final repository on the basis of long-term measurements of a meteorological station near Braunschweig The calculations with the accident consequence code *UFOMOD* are made for each of the 10 release categories representative for the following shipping scenarios

- 100% transportation by rail
- 80% rail / 20% road
- marshalling yard of Braunschweig

For each scenario the results for the 10 release categories are then superimposed taking into account the relative frequency of occurrence of each release category. The final presentation of risks from transport accidents is in the form of cumulative complementary frequency distribution (ccfd) relating radiological consequences and the associated expected frequencies of their occurrence The expected frequencies refer to the region (25 km zone) of the final waste repository Fig 3 shows as an example for the 80% rail / 20% road scenario the expected frequencies of effective doses which could result anywhere within the 25 km radius in downwind direction from the location of an accident

By displaying frequency distributions for different downwind distances of 250 m, 1250 m and 6250 m the additional information is given how radiological consequences decrease on average with distance from the location of an accident From Fig 3 the following information can be derived.

- The frequencies shown on the vertical axis refer to the entire region of the waste repository, that is to say to the zone within a radius of 25 km around the installation
- The effective dose given on the horizontal axis indicates the potential dose to a person residing permanently in close proximity to the accident site in the direction of atmospheric transport of the contaminant
- Accidents of trucks or rail cars carrying radioactive waste are expected to happen on average every 75 years
- As a result of the accident analysis every second accident involving a truck or rail cars loaded with radioactive waste containers would lead to a release of radioactivity But it has to be stressed that this rather high fraction is the result of a cumulation of conservative assumptions within the risk analysis. Nevertheless, in many accidents with airborne release of radioactive material potential radiological consequences would be quite small
- The chances that for this shipping scenario an accident in the region of the repository would lead without countermeasures to an effective dose in 250 m downwind distance from the location of the accident equivalent to or exceeding the natural radiation exposure of one year are about 1 in 75 for an operating period of 40 years
- Effective lifetime doses of 50 mSv in 250 m down-wind direction from the location of an accident would be expected with a chance of 1 in 10000 during an operating period of the repository of 40 years. As can be seen from Fig 3 potential radiation exposures decrease on average rapidly with distance from the location of the accident, starting from 250 m up to about 1200 m by a factor of 10 and a further factor of 10 at a distance of about 6200 m.
- No protective countermeasures are assumed in calculating the potential radiation doses. That is to say, that the removal of radioactive substances deposited on vegetation and other surfaces after an accident or other measures to reduce potential radiation exposure are not assumed

It would also have been possible to calculate cumulative complementary frequency distributions of collective effective doses resulting from transport accidents anywhere within the 25 km zone around

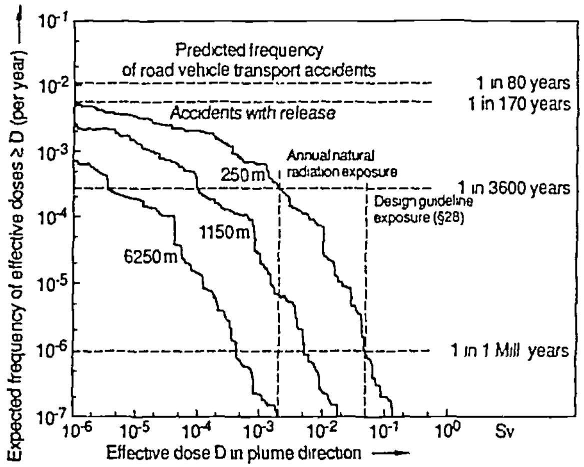


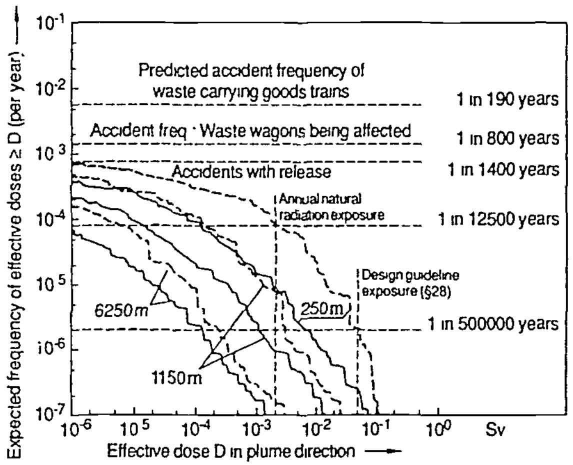
Fig. 3: Frequency distribution of the effective lifetime dose from waste transport accidents: 80% rail / 20% road transport.

the repository by assuming a uniform population density. But collective doses are much more difficult to interpret than potential individual doses. In addition the information about the decrease of radiation doses with distance from the location of an accident is lost in this case

Effective doses have also been calculated for distances less than 250 m But in this case only radiation exposure from inhalation of airborne radionuclides was calculated, whereas the long-term exposure pathways groundshine and ingestion resulting from dry or wet deposition of released radioactive material were not taken into account. For these long-term exposure pathways (70 a integration time) it would be too unrealistic to assume that a person takes all foodstuff from agricultural products from such a limited area or would be exposed there to groundshine for such a period without any countermeasures.

In order to judge the average influence of countermeasures to limit radiation doses resulting from the long-term exposure pathways ingestion and groundshine cumulative complementary frequency distributions of effective doses from all exposure pathways and from inhalation alone are compared in Fig. 4. The results refer to the 100 % rail scenario The following conclusions can be derived from the results summarized in Fig 4 as regards the accident risk from waste transportation by goods train·

- Referred to the waste transport volume for one year, the predicted frequency with which an accident involving the release of radioactive substances occurs in the region of the waste repository is $7 \cdot 10^{-4}$ per year.
- A comparison with Fig. 3 shows that expected frequencies of accidents with release of radioactive material are lower for rail transport compared to road transport.
- Since the quantity of radioactive substances released is generally small, as a result the potential radiation exposures are accordingly low. Thus, the calculated effective lifetime doses at a distance



- - - - all exposure pathways without countermeasures ____ Inhalation only

Fig 4. Frequency distribution of the effective lifetime dose from waste transport accidents 100% rail transport.

of 250 m from the accident site in the direction of atmospheric dispersion are in the range below 2 mSv in approximately 90% of accidents with release. This value corresponds to the effective dose that a person receives on average from natural radiation sources in one year. This can be read off the curve for a distance of 250 m, whereby effective doses of 2 mSv or higher can occur with an anticipated frequency of $8 \cdot 10^{-5}$ per year referred to the waste transport volume of one year.

- At larger distances from the accident site, the predicted annual frequency of being exposed to potential doses subsequent accident releases comparable to annual doses from natural radiation is accordingly lower. A transport accident that could lead without countermeasures to radiation doses equivalent to the natural radiation exposure of one year at a distance of approximately 1000 m could be anticipated in the local region of the waste repository with a frequency of occurrence of approximately $8 \cdot 10^{-6}$ per year. This is equivalent to one such event in approx. 125000 years assuming a hypothetical continuous operation of the waste repository.

- Potential doses of 50 mSv do not occur at greater distances from the accident site even with extremely low frequencies on the order of 10^{-7} per year. Such radiation exposure could only occur in the immediate vicinity of accident locations with very low predicted frequency of occurrence of $2 \cdot 10^{-6}$ per annum (250m, in wind direction, no countermeasures).

- In order to assess the possible influence of measures taken after an accident to reduce radiation exposure the curves for the total dose are compared with curves which state the contribution of the inhalation dose. The horizontal distance of the pairs of curves at 250 m, 1150 m and 6250 m reveals the fraction of the total radiation exposure that results on average from the long-term exposure pathways, namely groundshine, ingestion and resuspension, and can therefore be reduced by countermeasures such as decontamination or suspending agricultural production. Since the curves of the frequency distributions represent a large spectrum of accident sequences and atmospheric dispersion conditions, only general conclusions about the effectiveness of countermeasures can be derived from a comparison of the total dose without countermeasures and the inhalation dose. The relative contributions of the individual exposure pathways to the total dose vary according to which radionuclides are released in the specific case, and depending on the ratio of deposited to airborne radioactivity. Deposition of radionuclides on vegetation and ground is more effective for wet conditions as a result of rain or by enhanced dry deposition processes of particles with larger aerodynamic diameters.

- The difference between the potential total dose and the inhalation dose is larger for shorter distances from the accident site, such as 250 m, than for larger distances, such as 1150 m and 6250 m. This can be primarily attributed to the fact that particles with larger aerodynamic diameters which do not remain airborne for long are deposited more profusely in the close vicinity of the accident site.

- Thus, at a distance of around 250 m from the accident site in the direction of the dispersing plume, measures which influence the long-term exposure pathways can typically achieve a dose reduction of about 10 and more in approximately 99 out of 100 accidents causing a release. At a distance range around 1000 m the possible reduction of potential radiation exposure by means of countermeasures amounts to about a factor of 5.

Throughout the analysis parameter values and assumptions have been adopted in such a way that the results of the analysis imply an overestimation of the frequency of accidents with release of radioactive material and also to a large extent of the associated activity release. As a result, it is unlikely that an accident with release of radioactivity will occur in the region of the repository during an operating period of about 40 years. For a large fraction of accidents with airborne release of radioactive material potential radiological consequences would be quite small. The chances that a traffic accident in the region of the repository would lead without counter-measures to an effective dose in 250 m down-wind distance from the location of the accident equivalent to or exceeding the natural radiation exposure of one year are about 1 in 70 for an operating period of 40 years. In summary it can be concluded from the study results that for the region of the waste repository KONRAD additional risks resulting from accidents involving waste transports are very small.

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RISK ASSESSMENT ASSOCIATED WITH THE TRANSPORTATION OF LOW AND INTERMEDIATE LEVEL RADIOACTIVE WASTE TO THE CENTRE DE L'AUBE DISPOSAL FACILITY, FRANCE

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Abstract

Since 1991, the French Low Specific Activity waste have been stored in the near-surface waste disposal site in the Aube region (CSA) In 1995, the CSA plans to receive approximately 25,000 m^3 of waste from the three major producers which are EDF (Electricité de France), COGEMA (COmpagnie GEnérale des MAtières nucléaires), and the CEA (Commissariat à l'Energie Atomique). Four different kinds of package are broadly represented · the 200 l drums to be compacted, the 200 l drums filled with fixed wastes, concrete shells and metallic boxes As the radiological exposures resulting from waste transportation could stem from both incident-free transportation and accidental situation, two separated studies have been conducted. Using the INTERTRAN code (IAEA software) for normal transportation, the overall effective collective doses related to the whole transportation activity have been calculated and a risk of 0 48 man Sv/year has been deduced To cope with the diversity of the LSA (Low Specific Activity material) transportation, a representative package has been selected in each of the previously mentioned categories of package and its mechanical behaviour evaluated A former accident database was analysed to extract the accident rates and the accident distributions within different accidental scenarios It was therefore possible, in regard to each package to connect these scenarios with an amount of release material and an associated collective dose Based on the evaluation for each selected package, a generalisation to the whole transportation was performed. Thus, the total transportation of radioactive waste to the CSA leads to an accidental risk of 1.10^{-5} man Sv/ year and to a maximal individual dose of 0.2 mSv. As a result, the total effective collective dose, mainly induced by normal transportation, has been assessed at about 0 48 man Sv/year. In conclusion, it appears that the transportation of LSA waste to the CSA does not present any really significant risk of radiological exposure

1. INTRODUCTION

In France, the low and intermediate level radioactive waste, mainly produced by the nuclear power industry, scientific research and medical or industrial sectors, have been stored since 1991 in the near-surface waste disposal site in the Aube region (CSA). An inventory of these wastes, their packaging and their transportation from many different places in France, has been established using a 1995 prediction (full operational year of the CSA), by taking into account information from the waste producers and the ANDRA (Agence Nationale pour la gestion des Déchets RAdioactifs). As the radiological exposures resulting from waste transportation could stem from both incident-free transportation and accidental situation, two separated studies have been conducted using all the gathered information First, the assessment of radiological doses induced by normal transportation (incident-free) of LSA (Low Specific Activity) material to the CSA [1] And lastly, the risk assessment of accident associated with this waste transportation [2] This study has been performed by the Nuclear Protection Evaluation Center (CEPN) for the Division of Radioactive Transportation Safety of the Nuclear Safety and Protection Institut (IPSN/DSMR/SSTR).

2. COLLECTED INFORMATION

In 1995, the CSA plans to receive approximately 23,000 m^3 of LSA waste from the two major producers (approximately 40% each) which are EDF (Electricité de France) and COGEMA (COmpagnie GEnérale des MAtières nucléaires), as well from the CEA (Commissariat à l'Energie Atomique). Four different classes of package are broadly represented · the 200 l drums to be compacted, the 200 l drums filled with fixed waste, concrete shells and metallic boxes. The transportation of these packages covers about 500,000 km per year, either by rail to the Brienne-le-Château station (15 km from the site) to be then carried by truck to the CSA (70 %) or directly by road (30 %)

The origins of the wastes are illustrated in Figure 1 The thickness of the arrows illustrates the relative amount of waste transported The largest transportation originates in the southern of France. In this area, there are five nuclear sites, different fuel fabrication plants and two research centers. As the kinds of package are numerous, it proves to be necessary to gather them within few distinct categories in order to extract some representative packages for particular analysis. Table 1 presents the distribution of packages and transports for each producers. It must be born in mind that these values are ANDRA estimation for the reference year 1995.

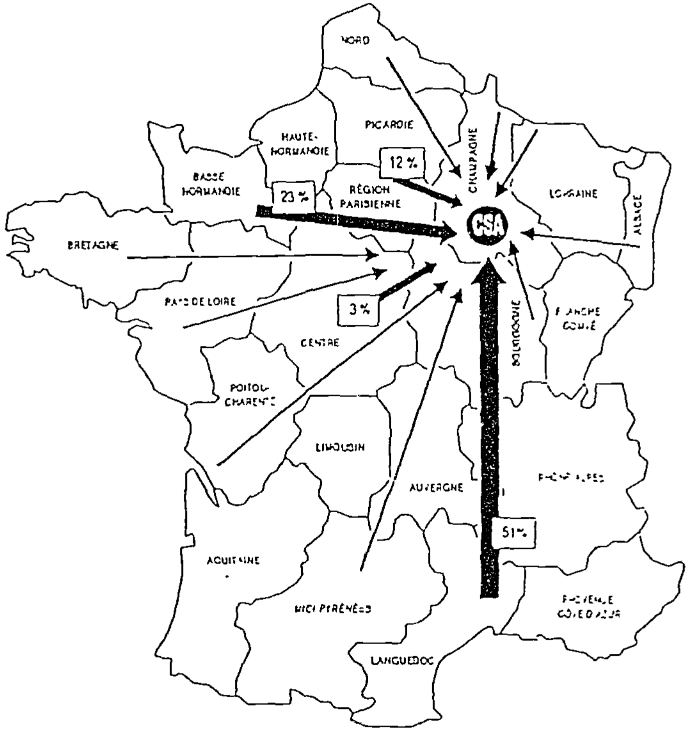

Figure 1 : Origins of the waste destinated to the CSA

3. NORMAL OPERATION

Using the INTERTRAN code [3] (IAEA software to assess dose and risk associated with the transportation of radioactive materials) for normal transportation the total effective collective doses related to road and rail transportation for reference year 1995 have been calculated. The main inputs for these appraisals are the volumes of the waste; the kilometers travelled; the corresponding transport index providing the radiation level (in 10^{-2} mSv/h) at 1 m distance from the packages; the associated numbers of workers; the speed and stop times and the population densities along the itineraries Four different itineraries have been selected as representative of the global transport Table 2 presents these routes and the associated collective doses for both public and occupational As shown on Table 3 a total risk of

Table 1 : Packages distribution intended to the CSA

	Annual number of package	% of the total	Number of package per wagon	Number of package per truck	Required number of wagon	Required number of truck	% of the total
Metallic drums to be compacted							
200 l CEA	4100	8.76	-	150	-	28	3.18
200 l EDF	*21350*	*45.62*	*216*	*144*	*55*	*67*	*13.87*
Cimented wastes							
200 l CEA	900	1 92	-	30	-	30	3 41
213 l COG	*5280*	*11.28*	*100*	*-*	*53*	*-*	*6.02*
200 l EDF	1900	4.06	100	30	11	29	4 55
Concrete shells							
CEA	600	1.29	-	24	-	25	2.84
COG	*6570*	*14.02*	*36*	*-*	*183*	*-*	*20.81*
EDF	3250	6 95	36	24	50	61	12 62
Metallic boxes							
CEA 5 m3	*280*	*0.60*	*-*	*5*	*-*	*56*	*6.37*
CEA 10 m3	*80*	*0.17*	*-*	*2 à 3*	*-*	*32*	*3.65*
EDF	170	0.36	7	5	14	16	3.41
COG 5 m3	920	1.97	6	-	154	-	17.52
Other packages							
Other prod	1400	3 00	-	-	-	15	1.71
TOTAL							
	46800	100			520	359	100

0.48 man.Sv/year has been deduced. this dose being quite equally distributed between occupational and public. Moreover in incident-free situation. it should be taken into account that even if. the overall rail transportation results in higher collective dose (0 26 man Sv/year) than road transportation (0 22 man Sv/year) the collective dose per container and per km associated with rail transportation induces significant lower doses [1] (2.6 10^{-4} man. mSv and 7.10^{-4} man mSv for road transportation)

Table 2 : Radiological impact (man.mSv/year) associated with normal transportation for the four selected itineraries

	Tricastin to CSA	CEA center to CSA	La Hague to CSA	Brienne-le-Ch to CSA
Transportation mode	road	road	rail	road
Length (km)	520	230	560	15
Frequency (/year)	12	10	280	1015
Public doses	4 5	0.7	45 8	2 3
Occupational doses	4.0	0 7	61 3	5.3
Total dose	8.5	1 4	107.1	7 6

Table 3 : Total radiological impact for road and rail transportation in France for reference year 1995

Collective doses man.mSv/year	Road	Rail	Total
Occupational collective dose	102	148	250
Public collective dose	120	114	234
Total	222	262	484

4. ACCIDENTAL CASES

As regards to radiological risks associated with accidental situations. the pathway is more complex as it concerns a significant number of package types. radionuclides. itineraries. kind of accident and modes of transportation In order to resolve this intricate spectrum of casualties. some simplifications were needed.

First in each of the previously mentioned categories of package. a representative itinerary and content has been selected. The description of the different transport systems is presented in Table 4

For both road and railway transportation former accident databases were analysed to appraise the accident rates. This assessment revealed a rate of 1.10^{-7} acc./ veh.km for motorways transportation and 3.10^{-7} acc./ veh.km for standard roads. As far as rail transportation was concerned a value of $1.9\ 10^{-8}$ was retained. Moreover these accident databases were

Table 4: Description of the different transport systems to the CSA

Designation	Type	Origin	Category	No. of transport per year	Transportation means	Maximum Activity (A2)
C1	200 l drums	EDF CPN Tricastin (PWR)	Industrial Type 2 *to be compacted*	9	Road	0 33
C2	200 l drums fixed wastes	COGEMA Pierrelatte (fuel cycle)	Industrial Type 2 *cemented waste*	24	Train	75
C3	concrete shells	COGEMA La Hague (reprocessing)	Industrial Type 2 *concrete shells*	145	Train	100
C4	metallic boxes	CEA CEN FAR (laboratory)	Industrial Type 2 *metallic boxes*	6	Road	100

Table 5 : Definition and distribution of the different scenarios [3]

Impact speed	Insignificant thermal stress	Thermal stress 30 min.. 800 °C	Thermal stress 60 min.. 800 °C
0-35 km/h	Sc1 0 20 0.36	Sc2 $4.4\ 10^{-2}$ $5.9.10^{-2}$	Sc3 $3\,5.\ 10^{-4}$ $2.9.10^{-2}$
36-80 km/h	Sc4 0 68 0.45	Sc5 $1\,5\ 10^{-2}$ $9.5.10^{-3}$	Sc6 $11.8.\ 10^{-4}$ $4.7.10^{-3}$
>80 km/h	Sc7 $9\,6\ 10^{-2}$ $8\,4.10^{-2}$	Sc8 $2.1\ 10^{-3}$ $1.8.10^{-3}$	Sc9 $1\,7.\ 10^{-4}$ $8\,8.10^{-4}$

ROAD TRANSPORTATION
RAILWAY TRANSPORTATION

[3] With an accident rate of : 1.10^{-7} acc/veh km for motorways and $1.9.\ 10^{-8}$ acc/wagon km for railways transportation

examined to extract the accident distributions within different scenarios [4. 5]. These accident scenarios were selected to be representative of a kind of stress potentially harmful to the package [6]. They were so expressed as speed of impact and fire duration. The conditional probabilities for each scenario and transportation means are presented in Table 5

It was therefore possible. with regard to each package. to connect these scenarios with an amount of release material (Table 6). The associated collective doses for inhalation were calculated using an atmospheric dispersion model with an appropriate density of population

Table 6 : Release quantities (g)

Scenario	C1	C2	C3	C4
Sc1	1 3 10^{-9}	1 5 10^{-6}	0	5 7 10^{-7}
Sc2	1 9 10^{8}	5 3 10^{-5}	0	1 3 10^{-4}
Sc3	7.6 10^{-8}	1 2 10^{-4}	0	2 6. 10^{-3}
Sc4	2 7 10^{-8}	2 6 10^{-5}	1 1 10^{-5}	1 2 10^{-5}
Sc5	1 5 10^{6}	6 9. 10^{-4}	3 0 10^{-4}	1 4 10^{-4}
Sc6	1 5 10^{6}	6 9 10^{-4}	3 0 10^{-4}	2 6 10^{-3}
Sc7	8 1 10^{8}	7 7 10^{5}	3 5 10^{-5}	3 5 10^{-5}
Sc8	1 6 10^{6}	7 4 10^{-4}	3 5 10^{-4}	1 6 10^{-4}
Sc9	1 6 10^{-6}	7 4 10^{-4}	3 5 10^{-4}	2 6 10^{-3}

Finally, a generalisation of the whole waste transportation to the CSA was conducted using the assumption that each selected package is representative of its class (as far as population density, radionuclides and released fraction are concerned).

As a result the Farmer curve (Figure 2), which gives the annual cumulative occurrence of a given collective dose allows analysis of the acceptable levels of probability and exposure. The overall release rate is about once every 67 years (1.5 10^{-2} release/years)

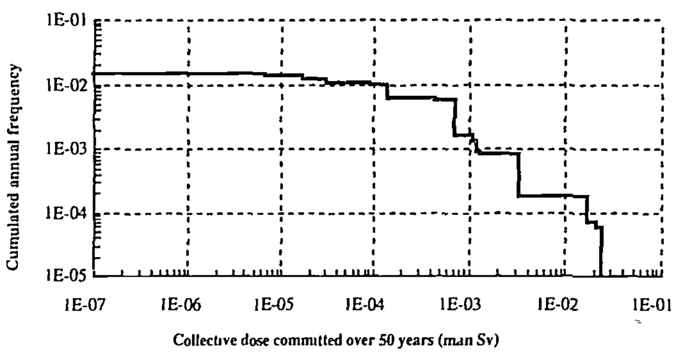

Figure 2: Farmer curve related to collective dose for the overall transportation of LSA to the CSA

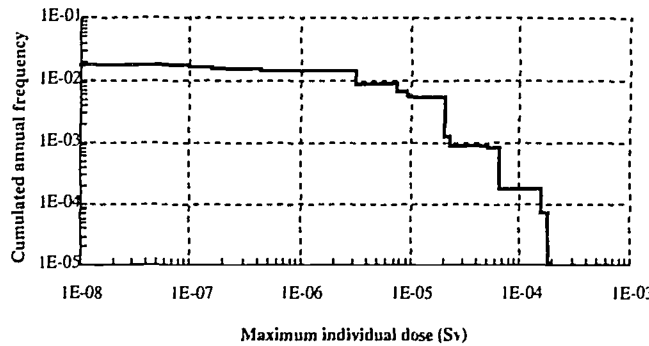

Figure 3: Farmer curve related to individual dose for the overall transportation of LSA to the CSA

leading to an effective collective dose lower than 1.10^{-3} man.Sv in 90 % of the cases, while the maximal potential collective dose is about $2.4.10^{-2}$ man.Sv for a release rate about once every 20.000 years. Moreover, the assessment of the individual doses for different distances and accident scenarios reveals a maximum dose of $1.8.10^{-4}$ Sv (Figure 3) at 100 meter from the release. Finally, the total transportation of radioactive waste to the CSA leads to an accidental risk of 1.10^{-5} man Sv/ year [2].

5. CONCLUSION

This study is confirming the accuracy of the transportation regulation as far as LSA transportation to the CSA is concerned. Indeed, the global risk appears to be low and no probability of a major accident appears Quite obviously, normal transportation seems to be nearly responsible for the total radiological impact, but it is important to point out that the impact associated with a normal transportation is more uniformly shared among the population than in an accidental situation, and consequently the maximal individual dose is lower. The total effective collective dose has been assessed at about 0 48 man Sv/year, this dose being equally reparted between public and workers. The higher individual doses are induced by accidental situations and reach a maximum value of 1 8. 10^{4} Sv at 100 meter from the release In conclusion, it appears that the transportation of LSA waste to the CSA does not present any really significant risk of radiological exposure.

The characteristics of most commonly transported radionuclides are located in a data library in the RADTRAN code and have been reviewed for correctness The RADTRAN 4 DEFINE function still allows isotopes to be added, so the analyst retains flexibility Use of a data library of this sort improves quality assurance because:

a) data-entry errors are virtually eliminated, and
b) fundamental errors are reduced, except for isotopes defined by the analyst, since the data have already been reviewed for correctness.

Among the radionuclide data are certain health-physics parameters that may require updating from time to time, as new findings become available Such updates must be a regular part of system maintenance.

The characteristics of many types of packages used in international commerce are embedded in a series of sample files that are publicly available on the United States' system. Currently, the user may begin with a sample file and edit it to suit his or her purposes Since this could mean extensive editing for users in other countries, a better approach for international applications might be to extract the package-related data and place them in a separate file from which a user would select the package that most closely matches his or her needs The data could then be electronically transferred to an input-data set that the user is constructing. Of course, the user would retain the ability to edit imported data, but a message noting that the imported data had been edited would be automatically printed The intent is to complete an electronic QA "trail." The latter consideration assumes importance in the event of disagreement with activist litigators and other challengers. Comment lines that specifically describe the edit changes, while optional, would be desirable in this regard.

Both air and sea transportation enjoy a high degree of international standardization. The handling at a seaport of a standard ISO* container is very much the same at all container-cargo terminals throughout the world, for example, movements of cargo ships proceed in much the same fashion in all international waters; air-freight procedures are relatively standardized, etc. Therefore, input parameters covering these modes that are suitable for most if not all users can be assembled and made available in a manner similar to that proposed for package data Like the latter, modal data are currently embedded in sample files, and also like them could be converted to separate files from which a user might select mode-related parameters to meet his or her needs. The quality of analyses performed with standardized package and modal data is improved by eliminating data-entry errors and reducing basic errors by using previously QAed data.

Uncertainty in all types of input-parameter values can be handled by assigning probability distributions to input parameters and sampling out of these distributions to produce a large number of input data sets. Each data set is entered into RADTRAN 4, the code is run, and the outputs displayed graphically to illustrate the uncertainty (range and distribution) of the resulting risk estimates. Earlier Monte Carlo methods required at least 1000 runs. However, the Latin Hypercube Sampling (LHS) method only requires about 100 samples to yield reliable results [Iman and Shortencarier, 1984] An LHS code that can be used in conjunction with RADTRAN has been demonstrated in a proof-of-concept exercise [Wheeler, Neuhauser and Kanipe, unpublished data], and this code will be made available for the United States' RADTRAN 4 system in the near future This technology is independent of national or regional considerations and may be made part of a similar computational system anywhere in the world.

*International Organization for Standardization

Country-Specific or Region-Specific Data

Data of this type depend on the landscape through which the route passes, on a segment-by-segment basis Example of these classes of data are.

- Population densities of route segments;
- Traffic densities of route segments (highway mode only);
- Shielding provided by surrounding structures,
- Accident rates along route segments,
- Meteorological characteristics of area in which route segment is located.

Population, accident-rate, and meteorological data are usually collected by local authorities and published by regional or national authorities. The absolute requirement for these data in transportation risk analysis can present a challenge to risk analysts in countries where such information is not yet compiled in electronic form. The alternative is laborious hand calculations from maps and tabulations of census data, traffic counts, etc. The results are seldom satisfactory. The input data are likely to contain errors of calculation and errors of transcription. It is practically axiomatic in the field of transportation analysis that one of the most useful things that an analyst can obtain are transportation network data and surrounding population densities in electronic form This goal can only be realized with active support and funding from national authorities

Route-segment lengths and population densities are calculated in the United States with codes such as HIGHWAY and INTERLINE [Johnson et al., 1992a and 1992b], which contain details of U.S. highway and railroad networks and are capable of generating output in a form that can be electronically transferred directly to a RADTRAN input-data set. Similar codes have been developed in most developed countries, they need to be examined for adaptability to producing output in a directly RADTRAN-compatible format

A related issue is ensuring consistent use of the terms rural, suburban, and urban The current definitions, which are given in Table 1, have been used for about 15 years [NRC, 1977] and are recommended in both RADTRAN and INTERTRAN documentation [Neuhauser and Kanipe, 1992; Ericcson and Elert, 1983] The user is not, however, forced to adhere to these definitions.

TABLE 1
POPULATION DENSITY ZONE DEFINITIONS

Density Zone	Range (persons/km^2)	Mean (persons/km^2)
Rural	<54	6
Suburban	54 to 1284	719
Urban	>1284	3861

XA9540093

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THE RADTRAN 4 COMPUTATIONAL SYSTEM FOR TRANSPORTATION RISK ASSESSMENT: A PROTOTYPE FOR THE INFORMATION SUPERHIGHWAY*

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Abstract

The RADTRAN 4 computer code for transportation risk assessment [Neuhauser and Kanipe, 1992] is the central code in a system that contains both other codes and data libraries [Cashwell, Neuhauser and Kern, 1988; Cashwell, 1989]. Some of these codes and data libraries supply input data for RADTRAN; others perform supplemental calculations. RADTRAN 4 will be released by the IAEA in an international version known as INTERTRAN 2 in 1995. In the United States, RADTRAN 4 and its supporting system may be accessed via the INTERNET, a precursor to the Information Superhighway. Similar networks are being contemplated elsewhere in the world, and the RADTRAN System may serve as a prototype for systems on these networks. A system is desirable for the following reasons. Some classes of data and data-handling methods are country-specific and some are not - ancillary codes and data libraries that provide the latter are not affected by national and regional borders while the former must be provided on a country-by-country basis. Making the invariant portions available to all users in an international system would simplify quality assurance (QA) and, therefore, the reliability and consistency of risk results.

Among the classes of data used in RADTRAN 4 (and INTERTRAN 2) and the supplemental calculational capabilities that are essentially invariant for all countries and regions are:

- radionuclide characteristics such as half-life, photon energy, and dose-conversion factors;
- characteristics of radioactive-material packages found in international commerce;
- features of highly standardized international transportation modes (primarily sea and air);
- uncertainty analysis.

We have found in the United States that making these features available in a form that either can be electronically transferred to or is already embedded in a RADTRAN input-data set increases the ease with which risk analyses can be performed, reduces the frequency of data-entry errors, and standardizes to the extent possible the calculation of transportation risk. These features and their related QA benefits are discussed individually in the next paragraphs.

* This work was performed at Sandia National Laboratories, a US Department of Energy facility, under contract DE-AC04-94AL85000.

Analysts who redefine the three zones for some special purpose should use the comment-line feature of RADTRAN to make a note of the deviation each time they run the code with the redefined values in order to facilitate comparisons between countries

Meteorological data can be supplied to RADTRAN in one of two ways -either as a single "look-up" table of downwind areas and dilution factors or as a set of six frequencies for the Pasquill atmospheric stability classes A through F [Pasquill, 1960] The latter refers the code to a second set of "look-up" tables, one for each Pasquill category evaluated at a single release height (ground-level, small-diameter puff) for the minimum windspeed in each class As part of the IAEA Coordinated Research Program effort that is producing INTERTRAN 2, France has developed a computer code, TRANSAT, which can supply these data in a form directly usable in RADTRAN/INTERTRAN [DeGrange et al., 1993]

Making the TRANSAT code available within the system would assist the user to calculate dispersion values. Good-quality meteorological data often are not available for a particular locale, especially in areas remote from cities and airports that are nevertheless traversed by highways, rail lines, and sea lanes. It is tempting to blindly follow arbitrary rules that yield ostensibly "accurate" data in these situations -- for example, to extrapolate from the nearest weather station, regardless of how far away or how dissimilar the terrain around the station may be -- but such practices should be avoided. The result is precision without accuracy. Mesoscale weather research will ultimately provide more satisfactory methods for handling this problem, but in the meantime, regional averages are often the best that one can do. TRANSAT can be used to produce regional-average dispersion parameters in a form electronically compatible with RADTRAN; but it also can be used to generate locale-specific dispersion parameters for those users who have information on that level of detail and resolution.

Supplemental Calculations

The Transportation Individual Center-Line Dose (TICLD) code fills the needs of risk analyst to respond to questions and regulations based on individual doses while at the same time ensuring that the individual dose estimates are obtained in a manner that is methodologically consistent with population-dose estimates [Erickson, Kanipe and Neuhauser, in preparation].

TICLD also allows dose thresholds to be correlated with distance and time-integrated concentration. In the United States, for example, there are two thresholds of particular interest. The first is the Negligible Individual Dose of 1 mrem (.01 mSv) [NCRP,1993]. The downwind distance at which the individual dose estimate drops below 1 mrem can be identified with TICLD and used in an iterative process to refine the original risk estimate by truncating the cumulation of population dose at that distance. The second threshold of interest in the United States is the radial distance within which the 100 mrem (1 mSv) recommended annual dose limit for a member of the public [ICRP, 1990] is exceeded Radial distances within which other, higher dose thresholds are exceeded may, of course, also be identified, although these often either do not occur or occur within a very short distance of the release point. These calculations can assist emergency-response personnel and policy makers by identifying radial distances within which efforts should be concentrated in the event of an accident of the type analyzed

Summary

The RADTRAN/INTERTRAN system is more powerful than the sum of its parts and provides the user with several benefits. The system allows all shipment configurations to be evaluated within a consistent methodological framework while also allowing route-related features to be treated as specifically as the available data permit. Additionally, quality assurance of most of the system components needs only to be done once, or after regular updates, at most. Thus, when a country establishes or joins a network containing the system, their analysts need be concerned only with QA of their own country-specific data Quality assurance thus becomes less time-consuming and expensive than it would be otherwise. With proven risk analysis tools and facilitated QA, RADTRAN- or INTERTRAN-based systems can provide flexible, high-quality transportation risk analysis capabilities to the international community This concept could serve as a prototype for other systems that might be made available on the future network now being referred to as the Information Superhighway.

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TRANSPORT AND WASTE PACKAGES

(Session 7)

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OPTIONAL MULTILAYER CASK FOR TRANSPORTATION OF SPENT SEALED NEUTRON SOURCES

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Abstract

The present investigation has been carried out to meet the safety requirements in handling or transporting the spent neutron sources (Cf-252 and Pu-Be). A model multilayer cylindrical shield cask with different design configuration options composed of repeated layers - of lead, carbon and lithium or cadmium or indium - has been constructed to resolve the optimum thickness required to establish the nuclear regulatory safety limits. The spent sources remained radiactivities varries from 0.1 A1 to A1 type A container or cask and from A1 to 10 A1 type B container. The calculations were made using ANISN Code after modification with some modules together with the DLC-75 and ISOTXS data libraries. The spectrum has been categorized to 7 energy group structure for neutrons and 21 energy group structure for the gamma spectrum. The deep penetration in the multilayer shield container are presented. Also the shield design parameters are studied, discussed and analysed to reach safety limitations for transporting such spent sources. The design parameters include neutron and gamma attenuation in container materials, the type of container shield materials and their thicknesses, the quantity of radioactivity remained in spent sources and the transport index.

1. INTRODUCTION

Neutron sealed sources may have different shapes and size according to its usage or purpose such neutron radiography, subcritical or critical application and nuclear reactor operation or research. These sources - after usage - should be transported from the operating facilities to the waste disposal places.

The range of radioactivity quantities remained in the spent sealed sources are usually varrying from 0.1 A1 to 10 A1 which must be transported using type A or type B containers. A1 values have been determined for most common radionuclides and are tabulated in S.S.No.6[1]. A total quantity of up to A1 may be transported in type A containers whereas type B containers may transport any quantity of radioactivity more than A1[2].

In principle,these containers - which are considered as shield for radioactive sources - besides satisfying all regulations of structural integrity, they must confirm safety requirements from radiological point of view. Thus, the neutrons must be thermalized and absorbed and the gamma rays must be attenuatted through the container or cask material such that the outer surface total fluxes are kept to minimum and the transport index (TI) kept within the regulatory safety limits of transportation.

In this work, four options for a cylindrical container with variable layers are suggested to transport spent sealed neutron sources (Cf-252 and Pu-Be). The neutron radioactivities varies from 0.1A1 to A1 type A container and from A1 to 10 A1 type B container. It is designed such that the multilayer cylindrical shield is composed of repeated layers of lead to attenuate gamma rays, carbon to thermalize neutrons, boron and lithium or cadmium or indium to absorb such neutrons. The four design configurations were constructed to solve for the optimum cask thickness required to establish the nuclear regulatory safety limits of transportation[1].

The neutron and gamma spectra are categorized to 7 and 21 energy group structure respectively and the calculation are performed using the ANISN code[3] after modification with a new version of the FE-CM1[4] modules together with the DLC-75[5] and ISOTXS[6] data libraries.

2. THEORETICAL FOUNDATION AND METHOD OF SOLUTION

The dose rate attenuation in the multilayer shield container for the spent sealed neutron sources has been determined using the discrete ordinate one dimensional program ANISN in P3 S16 approximation. The discrete ordinates Sn method is a means of effecting a numerical solution of energy-dependence linear Boltzman transport equation. This equation is solved numerically using the discrete ordinate method adopted by ANISN code. Such code discretization has been performed on three steps. Energy dependence is discretized by the usual multigroup approximation method, angular dependence is discretized and by the discrete ordinates approximation method and spatial dependence is discretized by the finite difference method. The steady state multigroup finite difference discrete ordinates equations are then solved iteratively.

Some logic functional modules (named FE-CM package) have been constructed and coupled to the ANISN code. The FM-CM package is modular by function and logic by selection. It has been developed and modified with the intent of realizing the convinience with ANISN code, ease and simplicity and optionality. Moreover it is also modified to raise up its efficiency and widen its applicability to include the source term calculations and the (TI) determination of such kind of problems. The FE-CM package include some modules for data management by assorting, segregation, categorization and preparation, modules for calculations, modules for conversion to patch filling and a control logic module.

The STM module permit the calculation of an external source term S(r,E) for neutrons or gamma photons. The external volume source is introduced as a function of energy group (E) and sptial mesh interval (i) and is denoted by (S^i_g) where (g) corresponds to the total number of energy groups. It is given by :

$$S^i_g = (\Delta V_i)^{-1} \int_{E_{g-1}}^{E_g} dE \int_{\Delta V_i} dv \; S(r,E)$$

where S(r,E) is the source in n or γ/cm^2.sec.Mev, and V_i is the volume relative to the interval (i). In cylindrical geometry, $dv = 2\pi r \, dr.1cm$.

The neutron and gamma spectra of both spent neutron sources Cf-252 and Pu-Be have been managed through the STM and categorized to 7 energy group structure for neutron and 21 energy groups structure for gamma photons from 0.1 A1 to 1 A1 for type A container and from 1A1 to 10 A1 for type B container. Tables I and II give the neutron and gamma spectrum of both studied sources[7,8]. In the present work, it is assumed that the spent source has been used for 7 half lives, i.e. decayed to (1/128) of its initial spectrum values. Similar categorization has also been performed to the data of cross section through a c.s. management module (CSMM). The design logic module (DLM) controls all the FE-CM package and it is an artificial intellegence package[9].

Table (I) Neutron Spectrum of Cf-252 and Pu- Be sources.

Group No.	upper Energy Limit[4]. EV	Cf-252[7] n/cm^2.sec.gm	Pu-Be[8] n/cm^2.sec
1	1.73329E07	1.2100E10	8.5000E06
2	7.94000E06	5.3000E10	3.0000E06
3	6.06530E06	1.0000E12	2.8500E07
4	3.07880E06	4.6000E11	1.0400E07
5	1.92050E06	7.6000E11	0.0000E00
6	1.00260E06	3.7000E11	0.0000E00
7	4.00310E05	2.8000E11	0.0000E00

Table (II) Gamma-ray Spectrum of Cf-252 and Pu- Be sources.

Group No.	upper Energy Limit[4]. MEV	Cf-252[7] Photons/(cm^2.sec.gm)	Pu-Be[8] Photons/(cm^2.sec.)
1	8.65385	0.0000E00	0.0000E00
2	7.50000	0.0000E00	0.0000E00
3	6.50000	1.0000E09	0.0000E00
4	5.50000	3.3500E09	3.3750E07
5	4.50000	1.1000E10	2.1250E08
6	3.50000	4.3000E10	6.5875E07
7	2.75000	1.1000E11	4.4750E07
8	2.25000	2.2000E11	1.7275E08
9	1.75000	4.2000E11	2.5000E07
10	1.25000	7.7000E11	2.5000E07
11	0.75000	1.7000E12	1.5750E07
12	0.50660	3.3000E12	0.0000E00
13	0.34240	0.0000E00	0.0000E00
14	0.23120	0.0000E00	0.0000E00
15	0.15620	0.0000E00	0.0000E00
16	0.10550	0.0000E00	0.0000E00
17	0.07127	0.0000E00	0.0000E00
18	0.04815	0.0000E00	0.0000E00
19	0.03252	0.0000E00	0.0000E00
20	0.02197	0.0000E00	0.0000E00
21	0.01484	0.0000E00	0.0000E00

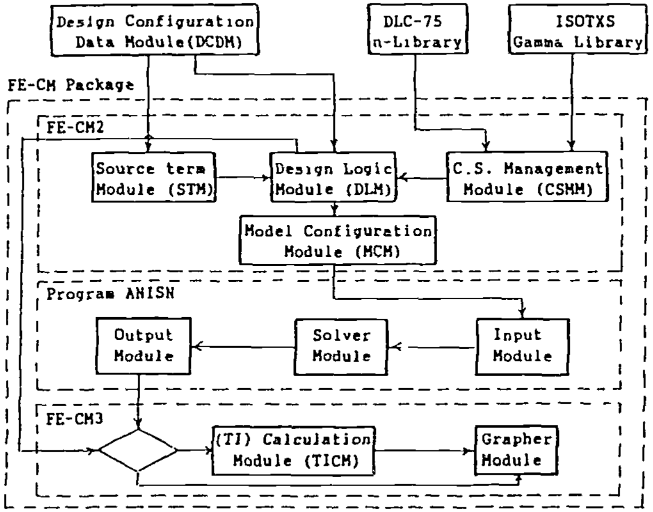

Fig (1) Flow Chart of Coupling ANISN Code With FE-CM Package

The transport index for both spent sources is calculated from[10]:

$$TI = (M \cdot \phi_{ts} \cdot D(r)) \,/\, d^2$$

where M is the spent source mass in (mgm), ϕ_{ts} is the total specific stradiant flux on the container outer surface {n(photon)/cm^2.sec.mgm.stradian}, D(r) is the dose rate factor in {([mrem/hr]/[n(photon)/sec]).cm^2}and d is the distance of *100 cm*.

Figure (1) shows the flow chart for the coupling between the ANISN code and the FE-CM package.

3. THE DESIGN CONFIGURATION MODELS

Four design configuration models are suggested as a shield container (type A or B) to transport the studied spent neutron sources. The design configuration model consists of a spent neutron source located in the center-line of a cylindrical multilayer shield container of different dense materials and with overall radial dimensions greater than 10cm to meet the IAEA safety requirements[1]. The analyses have been made for the following four design options:

(i) Design configuration(I); of a cylindrical laminated shell with 10cm total thickness. It compossed of 4 zones, namely 1cm thick. lead (Pb), 4cm thick. carbon (C), 4cm thick. boron (B-10), followed by lead (Pb) 1cm thickness.

(ii) Design configuration (II); composed of two coaxial laminated shells (typically repeated design configuration(I)) with total shield thickness of 20 cm and 8 zones.

(iii) Design configuration (III); of a double thickness (20cm) cylindrical multilayer shell with 4 zones. It is the same as design configuration (I) but each zone with double thickness.

(iv) Design configuration (IV); of a double thickness cylindrical multilayer shell with 5 zones; lead (2cm), carbon (8cm), boron (4cm), a neutron absorbing material such as lithium or cadmium or indium (4cm) followed by lead (2m).

The detailed design configuration models are given in Table III. The proposed design configuration (I) is suggested to type A containers whereas desgin configurations (II),(III) and (IV) are suggested to type B containers.As observed from the table, each mesh interval size is equivalent to 0.5cm. of shield thickness.

4. RESULTS AND DISCUSSIONS

The multigroup neutron and gamma radial attenuations through different zones of shield materials and the transport index calculations for the different transport container configurations - designed to transport Cf-252 and Pu-Be spent sources having different radioactive quantities varrying from 0.1A1 to 1A1 type A container and from 1A1 to 10A1 type B container - are computed using the ANISN code and FE-CM package with DLC-75 and ISOTXS data Libraries. A resume of results are shown in Figs.(2) to (7).

Figure(2) shows the multigroup neutron (7 groups, Fig.(2-a)) and gamma (21 groups, Figs.(2-b) and (2-c)) attenuations in shield design configuration(I) for type A container to transport 0.1A1 quantity Cf-252 spent source. Other similar sets of curves are performed for the other radioactivity quantities 0.2A1, 0.3A1, ... and up to 1A1. From all these curves, the total neutron and gamma fluxes ϕ_{tns} and $\phi_{t\gamma s}$ on the outer surface of the container shield are then computed together with the corresponding transport index for both neutron $(TI)_n$ and gamma $(TI)_\gamma$ radiations.

Table (III) Detailed Design Configuration Models

Config No.	Type of Cont.	zone No.	Shield Thick	No. Mesh Inter.	Material Type
I	A	1	1 cm	2	Pb
		2	4 cm	8	C-12
		3	4 cm	8	B-10
		4	1 cm	2	Pb
II	B	1	1 cm	2	Pb
		2	4 cm	8	C-12
		3	4 cm	8	B-10
		4	1 cm	2	Pb
		5	1 cm	2	Pb
		6	4 cm	8	C-12
		7	4 cm	8	B-10
		8	1 cm	2	Pb
III	B	1	2 cm	4	Pb
		2	8 cm	16	C-12
		3	8 cm	16	B-10
		4	2 cm	4	Pb
IV	B	1	2 cm	4	Pb
		2	8 cm	16	C-12
		3	4 cm	8	B-10
		4	4 cm	8	Li/In/Cd
		5	2 cm	4	Pb

The results of these calculated neutron and gamma transport indices are drawn against the corresponding variation of the radioactivity quantities (0.1Al to 1Al) and shown in Fig.(3-a) for design configuration(I) type A container. As observed from the figure, The $(TI)_{\gamma}$ curve has higer values than that of $(TI)_n$ curve. Since, according to the regulations, the transport index of any designed system is always taken for the higher value of any of neutron transport index or gamma one, so in Fig.(3-b), the neutron total flux at the outer surface of the container (type A, design configuration(I)) is drawn against the gamma transport index. As clearly seen from the figures, at total surface neutron flux TSNF ϕ_{tns} of 3.484E04 $n/cm^2.sec$, the gamma (or system) transport index is 1.6695E-02 which is far below the regulatory

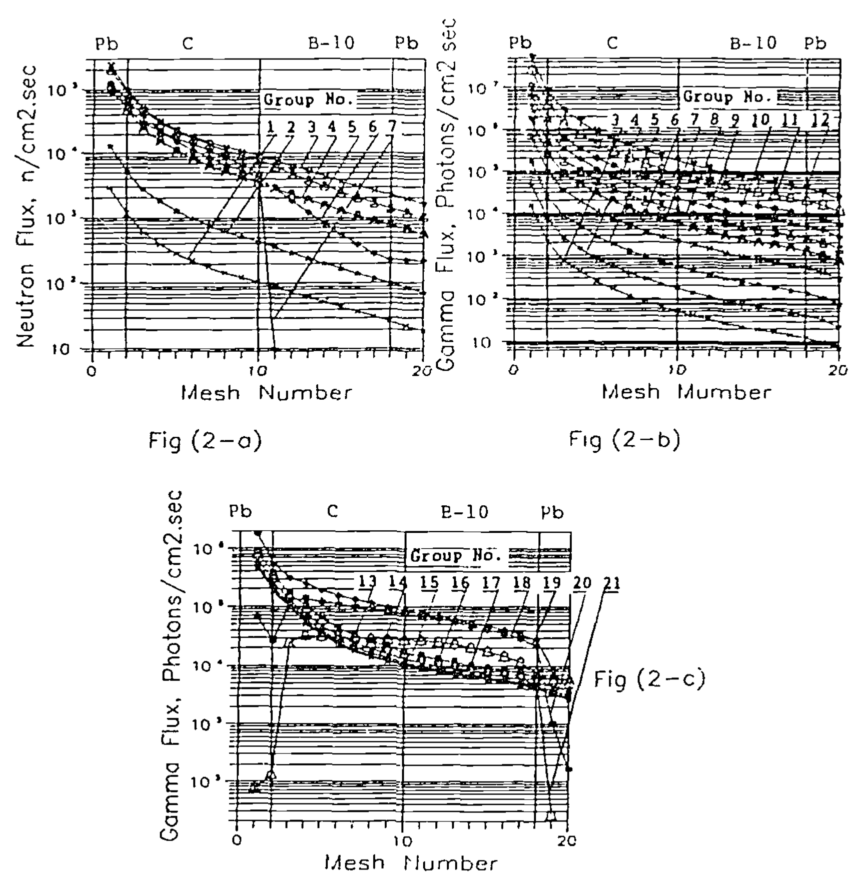

Fig (2) Neutron and Gamma Fluxes Attenuation in Shield Design Configuration (I) for Type A Container to Transport 0.1Al Quantity of Cf-252 Source

safety limits (TI<10) of transporting radioactive materials. This means, the design configuration(I) for type A container may carry a Cf-252 spent source of radioactive quantity up to 1Al (2Ci[1]) to yield a system transport index up to 1.6695E-02.

The deep penetration of the multigroup neutron fluxes in the three shield design configurations(II,III&IV) proposed for type B container to transport spent Cf-252 source of radioactivity

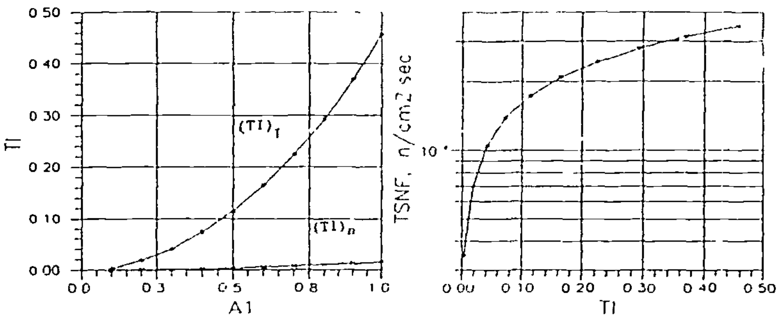

Fig (3-a) Fig (3-b)

Fig (3) Type A Container Shield Design Parameters for Configuration (I) and Safety Limitations of Cf-252 Source

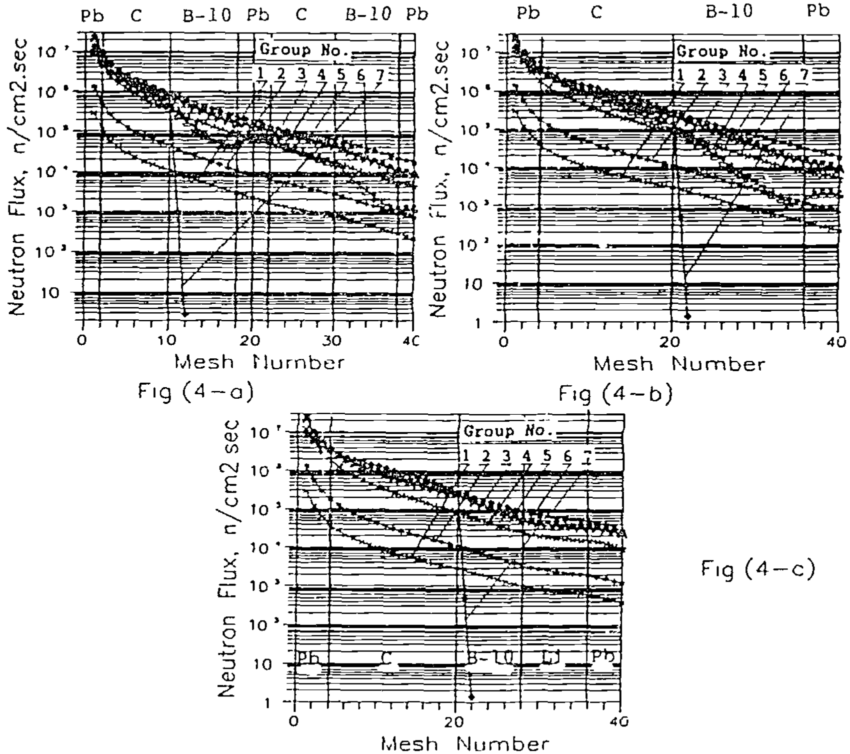

Fig (4-a) Fig (4-b) Fig (4-c)

Fig (4) Neutron Fluxes Attenuation in Different Shield Design Configurations for Type B Container to Transport 10A1 Quantity of Cf-252 Source

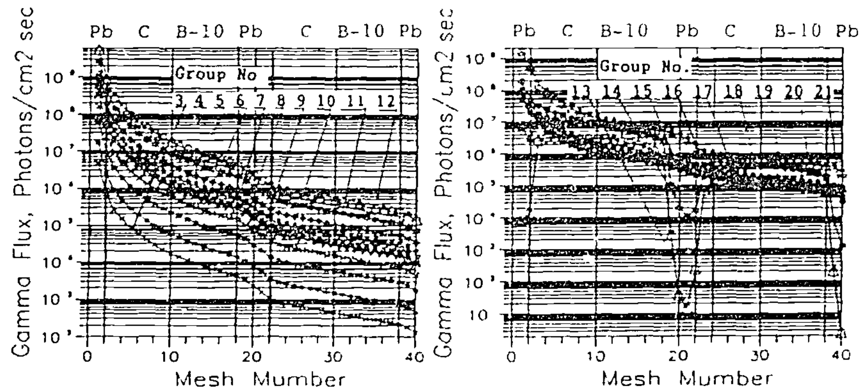

Fig (5-a) Fig (5-b)

Fig (5) *Gamma-rays Attenuation in Shield Design* Configuration (I I) for Type B Container to Transport 10A1 Quantity of *Cf-252 Source*

quantity 10A1 are shown in Fig.(4). In order to optimize between these three options to choose the suitable design configuration capable to safely transport such 10A1 quantity, the total surface neutron flux TSNF (ϕ_{tns}) and the corresponding transport index $(TI)_n$ are computed for each design. It is found that TSNF (ϕ_{tns}) equals 3.1277E04, 3.4557E04 and 6.99908E4 n/cm^2.sec with a corresponding $(TI)_n$ of 0.12924315, 0.1655432 and 0.33489 for configurations(II),(III) and (IV) respectively. Since configuration(II) has the lowest values of TSNF (ϕ_{tns}) and $(TI)_n$, so it is chosen and recomended for type B container, since it maintained the *largest safety reuirement margin*. For this chosen design configuration(II), the multigroup (21 groups) gamma attenuation curves are shown in Fig.(5) which shows a sharper cut off for gamma fluxes. This reveals assuring that the design configuration(II) type B container is the best and efficient shield container for multigroup neutron and gamma attenuations as well as it is the safest design through transportation.

The design paramaters important for safe transportation of spent Cf-252 source to be carried in type B container of shield design configuration(II) are shown in Fig.(6) which reflects its capability to transport spent Cf-252 sources of radioactive quantities up to 10A1 values. It must be noted that curves fo Fig.(6) are constructed in a similar way to that of Fig.(3). As clearly observed from Fig.(6) the maximum achievable transport

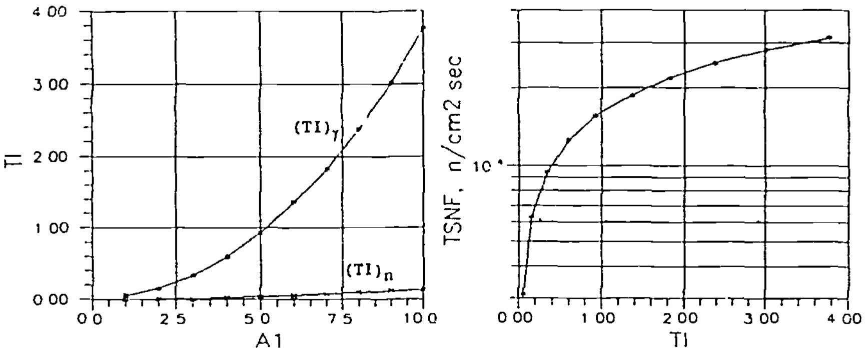

Fig (6–a) Fig (6–b)

Fig (6) Type B Container Shield Design Parameters for Configuration (I I) and Safety Limitation of Cf-252 Source

index for 10A1 (20Ci) spent Cf-252 source is TI= 3.7651652 for the chosen design configuration(II). At this TI value, which is far below the permitted value (TI<10) recommended by the IAEA for safe transportation[1-2], the total neutron flux on the outer surface TSNF of type B container is 3.1277E04 n/cm^2.sec.

Similar results are also obtained for Pu-Be spent source. An example for these results is presented in Fig.(7). In this figure the multigroup neutron and gamma fluxes attenuation in shield design configuration(I) - for type A container and proposed to transport a Pu-Be spent neutron source of 0.2Ci remained activity- are shown. The results shows that this design configuration(I) type A container is suitable to transport 0.2Ci spent Pu-Be source whereas the design configuration(II) type B container showed also to be the recommended design to transport a spent Pu-Be source of 2Ci remained radioactivity.

5. CONCLUSIONS

From the analysis of the results of the four design configuration options for multilayer container (or cask) to transport spent sealed neutron sources, the following conclusions are obtained:

1. According to the regulatory safety requirements, the design criteria for spent source transporting container design are attained by keeping to a minimum both the system transport index and the total neutron flux on container outer surface.

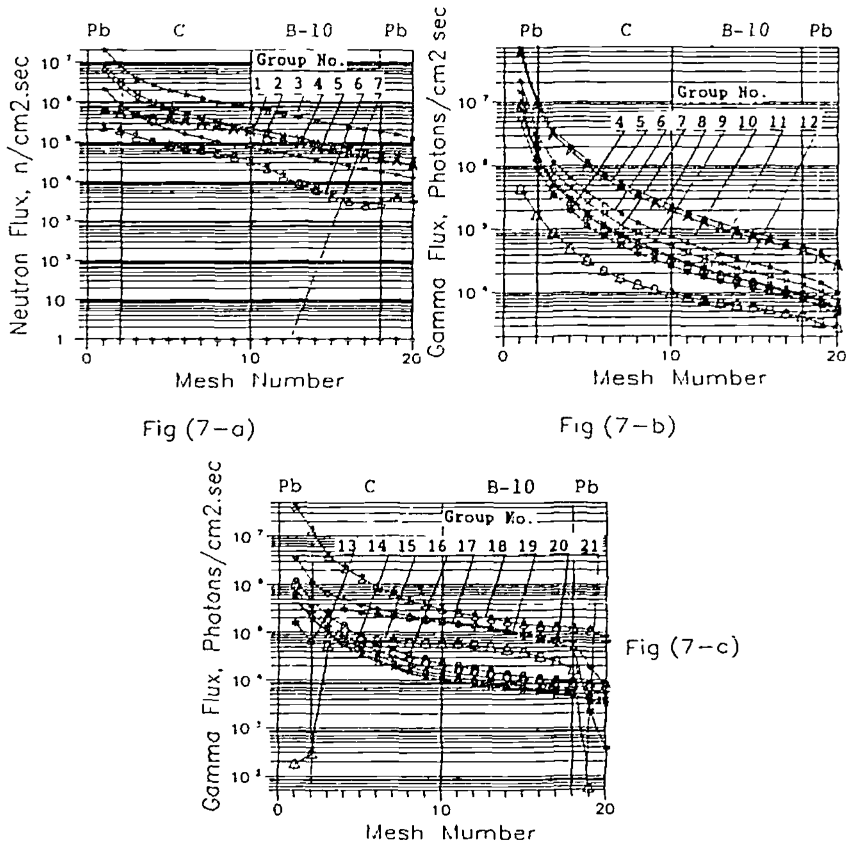

Fig (7–a) Fig (7–b) Fig (7–c)

Fig (7) Neutron and Gamma Fluxes Attenuation in Shield Design Configuration (I) for Type A Container to Transport 0 2Ci Activity of Pu-Be Source

2. These criteria are achieved for spent Cf-252 in the design configuration(I) proposed to type A container (for 0.1A1 - 1A1 radoactivity quantity range) and the design configuration(II) proposed to type B container (for 1A1 - 10A1 quantity range).

3. For 0.2Ci spent Cf-252 source (1A1 quantity), the total neutron flux on container surface (type A, design configuration (I)) is found to be 3.4854E04 n/cm^2.sec and the system transport index is 4.57456E-1. On the other hand, for 20 Ci spent Cf-252 source (10A1 quantity), the total neutron flux on container

surface (type B, design configuration(II)) is 3.1277E04 n/cm^2.sec and the system transport index is 3.7651652.

4. The two coaxial laminated shells (design configuration(II)) results in the best attenuation characteristics for both neutrons and gamma radiations for the higher range of radioactivity quantities (1A1 - 10A1).

5. The type B container shield design configuration(II) shows capability to satisfy the design criteria and the regulatory safety limits are adequately achieved for both Cf-252 (2Ci-20Ci) and Pu-Be (2Ci) spent sealed sources. Also, the type A container design configuration(I) is suitable to transport (0.2Ci) spent sealed Pu-Be source.

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STATUS OF THE BENEFICIAL USES SHIPPING SYSTEM CASK (BUSS)*

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Abstract

The Beneficial Uses Shipping System cask is a Type B packaging developed by Sandia National Laboratories for the U. S. Department of Energy. The cask is designed to transport special form radioactive source capsules (cesium chloride and strontium fluoride) produced by the Department of Energy's Hanford Waste Encapsulation and Storage Facility. This paper describes the cask system and the analyses performed to predict the response of the cask in impact, puncture, and fire accident conditions as specified in the regulations The cask prototype has been fabricated and Certificates of Compliance have been obtained.

INTRODUCTION

This paper presents a status report on the development of the Beneficial Uses Shipping System (BUSS) cask. The purpose of U. S. Department of Energy's (DOE) Beneficial Uses of Nuclear Byproducts Program is to develop and encourage beneficial uses of nuclear byproduct isotopes such as cesium-137 and strontium-90. Applications include the use of gamma irradiation to improve the quality of certain food products, to disinfect municipal sewage sludge, and to sterilize medical products.

The transportation of cesium chloride or strontium fluoride capsules produced by the DOE Waste Encapsulation and Storage Facility (WESF) at Hanford, WA, to and from commercial licensed facilities is performed in a Type B packaging (cask) certified by the U. S. Nuclear Regulatory Commission (NRC). Sandia National Laboratories was funded by the DOE to develop the BUSS cask to support transportation of these sources

The BUSS cask was designed to maximize payload within prescribed weight and size limits established by WESF and to serve as a safe, reliable, and efficient alternative to existing transportation systems. The cask design must be consistent with DOE policies for containment and as low as reasonably achievable radiation exposure and must comply with applicable regulations. A major goal of the BUSS cask development program was to obtain regulatory approval of the design through verification by means of state-of-the-art analysis techniques.

CONTENTS DESCRIPTION

The approved contents to be transported in the BUSS cask are special form capsules of either melt-cast cesium chloride or press-filled strontium fluoride [1,2]. Each source is doubly encapsulated with a 316-L stainless-steel outer layer and an inner steel capsule made from Hastelloy C-276 for the strontium fluoride or from 316-L for the cesium chloride. The capsule assemblies are about 7 cm in diameter, 53 cm long, and weigh about 8 kg. A cutaway sketch of a typical capsule is shown in Figure 1 Containment for the BUSS cask is provided by the special form nature of the source capsules.

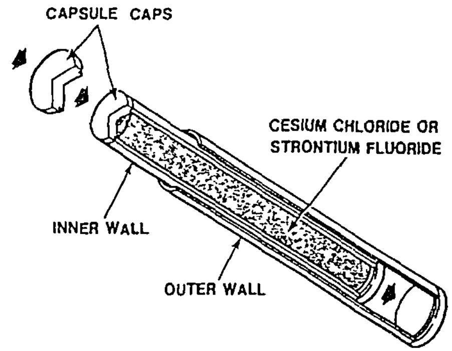

Figure 1. Schematic of a typical Waste Encapsulation and Storage Facility capsule.

CASK DESCRIPTION

The major components of the BUSS cask system include the cask body and lid, basket, impact limiters, personnel barrier, and shipping skid. Figure 2 shows an exploded view of the BUSS cask.

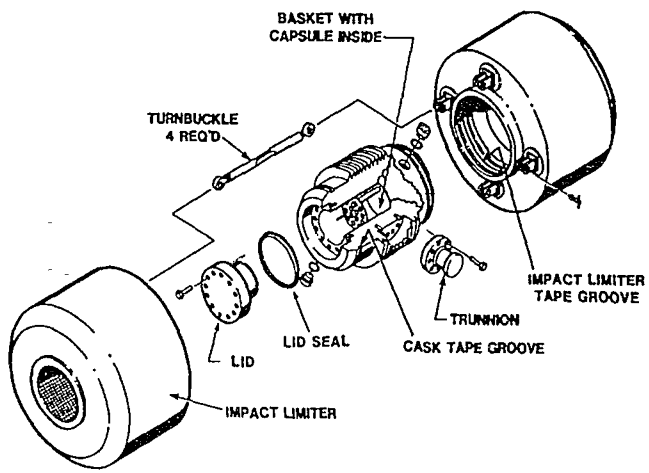

Figure 2. Exploded view of the BUSS cask.

The cask body is constructed from a one-piece 304 stainless-steel cylindrical forging. The wall and end of the cask body are a minimum of 33 cm thick. Eleven integral circumferential fins are

*This work was performed at Sandia National Laboratories, a US Department of Energy facility, under contract DE-AC04-94AL85000.

machined on the outer surface of the cask body for heat dissipation The cask closure is a one-piece, 33-cm-thick 304 stainless-steel forged lid, weighing about 680 kg. The lid is bolted through the 10-cm-thick lid flange to the cask body with 12 ASME SA-286 steel 1-1/2 in. (3.81-cm) dia bolts All openings into the cask interior are fitted with bolted-on lids, each having a combination metallic-elastomeric double seal. The inner containment seals used in the cask closure and port lid covers are high-temperature copper Helicoflex seals rated for a 450° C operating temperature [3] The outer elastomeric seal serves to provide the test volume for leak testing.

The cask's contents (capsules) are carried in one of four removable solid stainless-steel basket configurations An example of a 16-hole basket is shown in Figure 3.

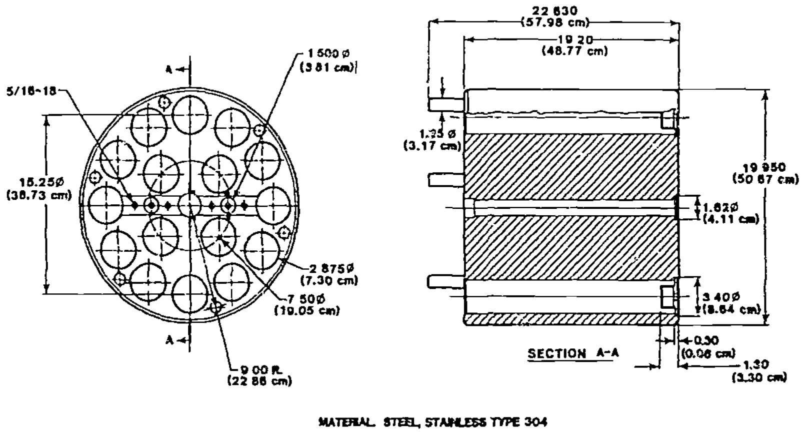

Figure 3. Typical basket configuration (16-capsule CsCl).

Depending on the thermal power level of cesium or strontium capsules to be transported, one of four different basket configurations may be used (see Table 1 for cask content limits).

Table 1 Beneficial Uses Shipping System Cask Radioactive Material Limits

Capsule Type	Basket Capacity/Maximum Capsule Thermal Power [no of capsules (W)]		Thermal Power (kW)	Activity (millions of Ci)*
Cesium chloride	16	(250)	4 0	0 85
(Cs-137)	12	(333)	4 0	0 85
Strontium fluoride	6	(650)	3 9	0 65
(Sr-90)	4	(850)	3 4	0 56

* 1 Ci = 37 GBq

Steel-encased polyurethane-foam impact limiters are attached to each end of the cask body to provide impact protection to the cask system These impact limiters are retained by four turnbuckles and two tape joints [4]. The turnbuckles are used to secure the impact limiters to the cask body during normal handling operations. The tape joints become effective during accident environments to hold the limiters onto the cask body during impact. These tape joints are loose fitting and do not take effect until large forces are imposed on the impact limiters The tape joints are unique in cask design and are normally found in applications required to withstand large shear loads. Because of the near one-to-one aspect ratio of the cask body, the ends of the cask do not extend into the impact limiters sufficiently to produce large resisting forces to counteract the moments developed on the impact limiters during side drops. It was determined analytically that these moments may be practically resisted through devices which generate large shear forces such as tape joints.

The assembled cask with impact limiters is transported on its shipping skid as shown in Figure 4. A personnel barrier is used to prevent unauthorized access to the hot surfaces of the loaded cask. The weight of the loaded cask including its shipping skid and personnel barrier is approximately 15,310 kg. The cask can be dry- or wet-loaded and unloaded, depending on the facility.

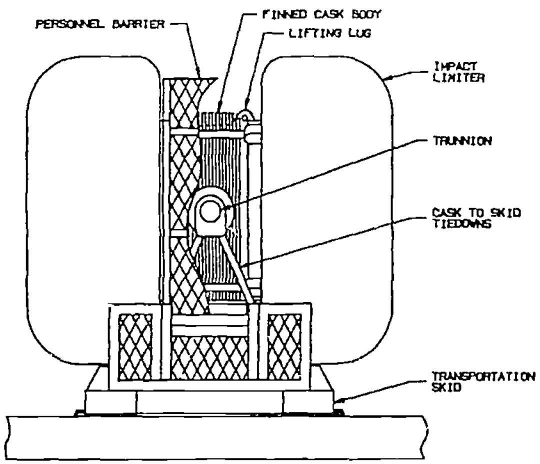

Figure 4. Cask assembly showing personnel barrier and shipping skid.

CASK DESIGN AND DEVELOPMENT

The design process included shielding, structural, and thermal analyses of the BUSS response to regulatory normal and hypothetical accident conditions. Results of these analyses for the final design were incorporated in the Safety Analysis Report for Packaging (SARP) [5].

The principal design criteria for structural integrity and shielding used during development of the BUSS cask were specified in 10 CFR Part 71 [6]. Of those conditions, the hypothetical accident conditions are the most stringent. They include 9-m drop, 1-m puncture, 30-minute thermal, and water immersion tests.

Shielding Analysis

We performed detailed shielding analyses of the BUSS cask loaded with 16 cesium chloride capsules to determine the radiation environment external to the package. The 16-capsule cask was found to be a more extreme shielding problem than the system loaded with six strontium fluoride capsules. The shielding assessments included multienergy group discrete ordinates and Monte Carlo computer analyses [7,8,9]. The radiation transport analyses of the BUSS cask were performed (1) to evaluate the shielding capabilities of the package for both normal and accident conditions, and (2) to determine the energy deposition profiles in the container for use in the thermal evaluation of the system. Separate one-, two-, and three-dimensional (1-D, 2-D, 3-D) finite-difference models were developed for both cases. Table 2 gives the calculated results for normal operation and post-accident radiation levels for the BUSS cask loaded with 16 cesium chloride capsules as well as maximum levels specified by the regulations

Table 2. Summary of Radiation Levels (mrem/hr) for the Beneficial Uses Shipping System

Source	Normal Conditions				Accident Condition	
	Package Surface		2 m from Surface		1 m from Surface	
	End	Side	End	Side	End	Side
Gamma**	64	29	2 9	1 2	11	4 6
10CFR71 Regulation	200		10		1000	

As shown above, the cask meets the applicable shielding performance requirements specified in 10 CFR Part 71.

Structural Analysis

The principal structural members of the BUSS cask include the body, lid, and the impact limiters. The integrity of the system is assured for both normal operation and accidents by the performance of the structural members and the presence of high-quality metallic seals at every opening into the cask interior. In combination with the impact limiters, we show that this boundary is virtually unaffected by the normal and hypothetical accident conditions specified in 10 CFR Part 71.

9 Meter Free-Drop

We evaluated the performance and structural integrity of a BUSS cask subjected to the hypothetical accident free-drop test with 2-D and 3-D finite-element analysis techniques. Three orientations at impact were evaluated. (1) end, (2) side, and (3) center of gravity over corner. The finite-element models were generated by using QMESH [10] and PATRAN [11], they were analyzed with Hondo II [12] and DYNA3D [13]. The deformed shapes and stress distributions were plotted with MOVIE BYU [14]. Accelerations were also obtained to evaluate lid integrity.

Table 3 shows the predicted values for impact limiter crush, cask body acceleration, and von Mises equivalent stress in the cask for three impact orientations at the most severe operating temperature condition.

Table 3 Predicted Foam Crush and Peak von Mises Stress for the Beneficial Uses Shipping System Cask in the Hypothetical Accident 9 Meter Free-Drop at -40° C

Orientation	Crush (cm)	Acceleration (g)	Stress (MPa)
End	15.5	105	59 3
Side	19 1	97	15.2
Corner	28.7	75	20.0

As shown above, the cask wall is stressed to values significantly less than yield during the 9-m drop event. Evaluation of the bolting arrangement using deceleration values on the cask lid indicated that the seal will maintain its integrity in all impact orientations. Thus, the cask body is essentially undamaged when subjected to the second event of the hypothetical accident sequence, the 1-m drop onto a mild-steel pin.

1 Meter Puncture

We determined the structural response of the BUSS cask to the hypothetical accident puncture test with analyses similar to that used for the drop. To ensure analysis of the most severe accident, we evaluated puncture in three orientations: (1) cask impacting the punch on its side, (2) corner of the cask directly below the center of gravity impacting the punch, and (3) closure end impacting the punch. Each analysis was performed without the impact limiter in place to produce maximum damage to the cask. For the side punch, the cooling fins were not modeled.

In every case, we found that either the cask body or lid during closure end impact would be plastically deformed near the impact point. The damage would be limited to a shallow circular indentation corresponding to the cross-sectional dimensions of the end of the puncture bar. Elsewhere in the cask, the bulk of the material remains elastic (The average stress is 15 MPa or less).

Since the cask body is only moderately stressed and retains its containment and structural integrity, the cask body configuration (geometry) when subjected to the hypothetical accident thermal test is virtually unchanged.

Thermal Analysis

We evaluated the thermal responses of the BUSS cask for normal conditions of transport and hypothetical accidents with finite-difference modeling techniques and pre- and post-processing software. From the geometric description of the cask, we used PATRAN to generate a 2-D finite-element mesh into finite-difference data and Q/TRAN [15] to analyze the finite-difference model.

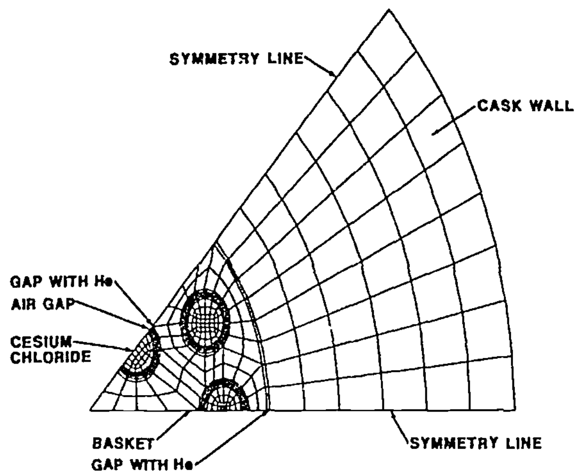

Figure 5. Cask thermal analysis model (16 capsule basket).

**Corresponds to the total dose rate since contents do not include neutron-emitting materials

Once the model was analyzed, an inverse translator was used to convert the data back into a finite-element representation for post-processing. We used PATRAN to post-process the analysis data

The finite-difference models used for the evaluation of normal and accident conditions were similar The model used in the hypothetical thermal accident evaluation was essentially the same as the normal condition model except that the exterior boundary condition was changed to simulate thermal input from the fire as defined by the regulations. We modeled the body, basket, capsules, cooling fins, and the effects of the circumferential gaps between the body components (Figure 5) We conservatively assumed that heat was transported across the gaps only by conduction and radiation Omitting convective transport across the gaps ensures greater predicted temperatures for the cask interior than expected under actual conditions. Heat loss from the exterior surface of the cask was modeled by assuming free convection and thermal radiation.

The thermal load of the cesium-137 in the BUSS cask was distributed throughout the interior portions of the cask based on the energy-deposition profile About 50% of the decay energy is transported to and absorbed by the cladding, the basket, or the cask wall. The remainder of the heat released during the radioactive decay of cesium-137 was deposited within the cesium chloride

Heat released by the decay of the strontium was assumed as absorbed entirely by the strontium fluoride. This assumption is justified since strontium-90 and its daughters emit betas which deposit their energy quickly. Such an assumption overestimates the capsule temperatures; some energy is actually deposited elsewhere in the the cask by gamma and bremsstrahlung radiation. The results of the thermal analyses (both normal and accident) indicate the temperatures of the cask body, basket, seals, and the cesium chloride and strontium fluoride contents. The temperatures in the capsules are less than the design allowables of 450°C at the cesium/metal capsule interface and 800°C at the strontium material centerline Helium is used in the cask cavity to reduce capsule temperatures and thus increases the margin of safety during transport

Containment

The containment of the special form contents is provided by the rugged double encapsulation of the radioactive materials. This encapsulation is supplemented by the cask structure and seals tested to a leak rate of 1×10^{-4} atm-cm³/s. Essentially there are triple barriers to the release of the radioactive contents.

The primary function of the cask structure and seals is to retain helium for cavity heat-transfer purposes under normal transport conditions. Under these conditions, a design basis leakage limit of 1×10^{-4} atm-cm³/s is low enough to ensure adequate helium retention during the transport period.

SUMMARY

The design of the cask system and the analyses performed to predict the response of the cask in impact, puncture, and fire accident conditions as specified in the regulations have been described. We have demonstrated in a SARP that the integrity of the BUSS cask system is maintained during normal and hypothetical accident conditions of 10 CFR Part 71. A cask prototype has been fabricated, and Certificates of Compliance have been obtained from both the DOE and the NRC.

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PACKAGING OF RADIOACTIVE WASTE IN THE UNITED KINGDOM FOR DISPOSAL AT THE DEEP REPOSITORY

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Abstract

UK Nirex Ltd is responsible for developing a deep repository for the safe disposal of intermediate and low level radioactive wastes (ILW and LLW), and is concentrating its investigations on Sellafield as a potential location. Nirex in co-operation with the producers of radioactive waste, has developed a range of standard containers suitable for the wide variety of wastes arising and predicted to arise in the future. The standard range includes unshielded containers, such as the 500 litre drum and 3m³ box as well as shielded containers such as the 4m box for ILW and the 4m box for LLW. The unshielded containers require to be packaged into reusable shielded containers for transport whereas the shielded containers form transport packages in their own right. Waste Package Specifications have been produced for each of the standard packages and these provide the essential link between waste package design and the design of the repository. Nirex will eventually be producing Conditions for Acceptance but these cannot be defined until the authorisation and other regulatory requirements are known. Until that time, the Waste Package Specifications provide a sound basis for the packaging of waste. Detailed information on stocks and predicted future arisings of waste are published in the United Kingdom Radioactive Waste Inventory. The information provided in the Inventory is used by Nirex to determine the volume of waste which will require disposal at the repository. Disposal volumes ranging from 400,000m³ to 2,000,000m³ are currently being considered. Detailed data available in the Inventory are used to classify the waste for transport and to estimate the numbers and types of packages requiring transport and disposal.

1. INTRODUCTION

United Kingdom Nirex Limited (Nirex) has been established by the major organisations in the nuclear industry, with the agreement of Government, with the task of developing a deep repository for disposal of solid intermediate and low level radioactive wastes (ILW and LLW). Nirex is concentrating its investigations on a site adjacent to the Sellafield reprocessing plant operated by British Nuclear Fuels plc.

Nirex working in co-operation with the producers of radioactive waste has developed a range of standard containers suitable for the wide variety of wastes arising in the UK, as well as those predicted to arise in the future. Each standard package meets an identified need and is defined by a Waste Package Specification.

The Waste Package Specifications provide the essential link between waste package design and repository design, and define dimensional, functional and performance criteria. They establish a minimum level of performance for all package designs and provide a firm basis for repository and transport system design. Conditions of Acceptance will be issued but cannot be provided until the conditions of the authorisation and other regulatory requirements are known.

This paper describes the range of standard waste packages adopted by Nirex and explains the role of the Waste Package Specifications. Waste volumes currently being considered by Nirex are given and estimates presented on the numbers and types of package requiring transport and disposal.

2. NIREX STANDARD WASTE CONTAINERS

Nirex has long recognised the desirability of standardising on a limited range of waste containers and, apart from some minor changes, the standard range has remained essentially unchanged since 1988. The need for standardisation has been confirmed by recent design studies which have had to consider methods for the emplacement of around 100 disposal packages per day. Standardisation is essential if the repository is to achieve the necessary throughput and is also desirable from safety, logistic and economic reasons. The range of standard containers which has been developed by Nirex and the waste producers is given below. Each standard container has been justified on the basis of an identified need for packaging a particular range of wastes.

Nirex Range of Standard Containers		
ILW	500 litre drum	0.8m diam by 1.2m high
	3 cubic metre box	1.72 x 1.72m plan by 1.225m high
	3 cubic metre drum	1.72m diam by 1.225m high
	4 metre box	4.013 x 2.438m plan by 2.2m high
LLW	4 metre box	4.013 x 2.438m plan by 2.2m high
	2 metre box	1.969 x 2.438m plan by 2.2m high

500 litre Drum for ILW

The principal container for ILW is the 500 litre drum. It will be used mainly for operational wastes, arising from day-to-day operations of nuclear facilities. At the

Sellafield Magnox Encapsulation Plant, 500 litre drums are currently being filled with waste and stored. Other plants are also nearing the stage where the packaging of waste can commence

It has not been possible to standardise on one single design of 500 litre drum because the processes for immobilising different wastes require variations to the drum, mainly in the lid area and in the internal drum furniture. A limited number of drum shapes have been adopted, but all have common lifting and handling arrangements.

The 500 litre drum is not designed to provide any radiation shielding, but is to be transported within one of a family of reusable shielded transport containers being developed for the purpose by Nirex [1]. The transport container, which will carry four drums located in a handling stillage, will be designed to Type B requirements as required by IAEA Safety Series 6 [3]. The transport container family is currently assumed to consist of four thicknesses comprising 70mm, 145mm, 210mm and 285mm steel shielding.

Figure 1 shows the 500 litre drums in a transport container, and also serves to illustrate the terms *waste container, wasteform, waste package, transport container and transport package* which are used in this paper.

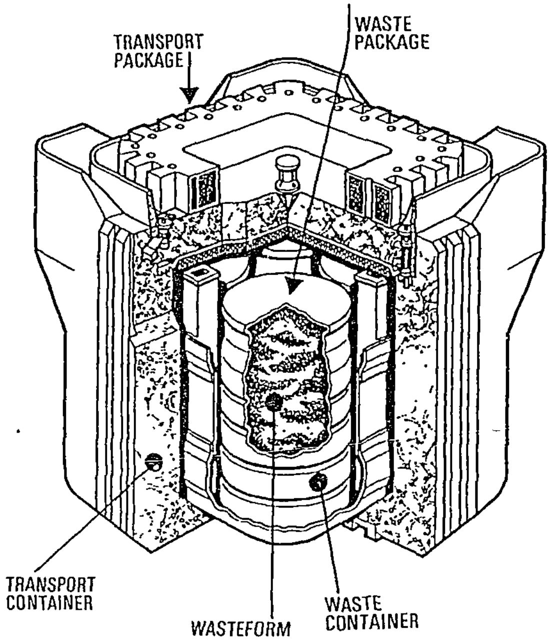

Figure 1 - 500 litre Drums in a Re-usable Transport Container

3 cubic metre Box

While the 500 litre drum is the principal container for ILW, some items will be too large for it, for example redundant items of equipment or decommissioning waste. For these wastes a standard box of nominal capacity 3m^3 will be used. The box is cuboidal with rounded corners. The shape corresponds to the outline of a stillage containing four 500 litre drums and hence the 3m^3 box can be transported within the same reusable transport container.

The 3m^3 box, in contrast to the 500 litre drum is not yet in use, but Nirex has produced a preliminary design in order to demonstrate the feasibility of the concept and has manufactured and successfully tested two prototypes [2]. The design is currently being revisited to take account of the latest information from repository design and site investigation.

The 3m^3 box will be transported within the same family of Type B reusable shielded transport containers as will be used for the 500 litre drums.

3 cubic metre Drum

The 3m^3 drum is a cylindrical version of the 3m^3 box which is suited to in-drum mixing of waste. Although currently being developed for a particular application, it has potential uses throughout the industry and has therefore been adopted as a Nirex standard waste container. As with the 3m^3 box, one 3m^3 drum can be transported within a reusable Nirex transport container and when compared against a payload of four 500 litre drums, it is some 30% more efficient.

4 metre Box for ILW

The three ILW containers described so far - the 500 litre drum, 3m^3 box and 3m^3 drum - all need to be transported within a reusable transport container which is being designed to Type B requirements. However, when filled with waste, the 4m ILW box complies with IAEA Safety Series 6 [3] as a transport package in its own right, and is designed to be disposed of without being opened.

The 4m box is designed as a non-fissile Industrial Package Type 2 (IP-2), according to the 1985 Edition of the IAEA Transport Regulations. It is intended to be used for short-lived ILW complying with the requirements for low specific activity material (LSA II or LSA III) or for surface contaminated objects (SCO) complying with SCO I or SCO II requirements.

The box incorporates its own shielding, with reinforced concrete walls up to 250mm thick. The box dimensions have been specified to follow the principles established for Series 1 ISO freight containers [4]. The standard ISO width of 8 feet (2.438m) has been adopted and the length has been specified as a two-thirds module of the standard 20 foot (6m) ISO container. The box incorporates ISO-style corner fittings to permit lifting by twistlock frame and tie-down to a transport vehicle. The box is likely to have a maximum weight of 65 tonnes when filled with grouted waste, although it is currently limited to 50 tonnes pending clarification of handling limitations at the repository [5]

Containment of radioactivity is provided by the monolithic nature of the wasteform and by the reinforced concrete walls. Qualitative methods are proposed for the demonstration of compliance with regulatory containment requirements when subject to normal conditions of transport tests [3]. The reinforced concrete has been designed in accordance with the requirements of paragraph 523 of Safety Series 6 [3], so the package can withstand the loads specified in ISO 1496/1 [6] with crack widths limited to 0.2mm. In practice the most onerous load case will be that of lifting from above.

This box is likely to be used mainly for reactor decommissioning and similar wastes. Acceptable contents are currently judged to be activated materials such as steel, graphite, concrete and similar materials for which it can be claimed that the activity is uniformly distributed. It may be some time before there is need for this package but Nirex is currently developing the design in order to provide a basis for repository design.

4 metre Box for LLW

The 4m LLW box is designed as a non-fissile Industrial Package Type 2 (IP-2) for use with LSA-II, LSA III, SCO-I or SCO-II waste material. The design is based on ISO container principles and is in accordance with the requirements for an IP-2 freight container [para 523, ref 3]. Containment of the radioactive contents is provided by a sheet steel structure which is fabricated by continuous seal welds at sheet interfaces. The complete fabrication is intended to be bubble tested to check for leaks. The lid closure is affected by an elastomeric seal which is compressed by lid clamps; the seal can be pressure tested at manufacture.

Low level waste does not require shielding and, unlike ILW, grouting will not normally be required for transport or disposal in the deep repository. Since it does not incorporate shielding or carry a grouted wasteform, the gross weight of the package is likely to be in the region of 20 - 26 tonnes. This means that the box will be suitable for transport by an ordinary Heavy Goods Vehicle of 38 tonnes gross vehicle weight. A version of the box weighing up to 65 tonnes is also being considered for the disposal of heavy wastes.

An important element of the design is commonality of dimensions and handling features with the 4m ILW box. This has major benefits since it permits the use of standard equipment for transport and handling operations Furthermore, since the boxes are based on the ISO container standards, advantage can be taken of proven technologies from the freight handling industry.

2 metre Box for LLW

This container is a half-length version of the 4m box for LLW, and similarly is a transport package in its own right. It is envisaged that the 2m box will be used for packaging of denser wastes which would exceed the weight limit if packaged into the 4m LLW box. It has been sized such that it can be filled with high density steel waste and still remain suitable for road transport; like the 4m LLW box its maximum weight will thus be 26 tonnes. It is also likely to be of particular benefit in decommissioning situations where a smaller box is required at the 'working face'.

3. NON-STANDARD WASTE PACKAGES

All the waste containers described so far are Nirex standard designs that will be suitable for the vast majority of waste producers needs. Non-standard packages will however have to be used for some wastes. These may arise for instance where the design predates the establishment of Nirex standards, or where the waste is of a size and shape which would require expensive size reduction before it could be packaged into a standard container. An example of such a container is the WAGR box, a concrete IP-2 container designed for packaging waste from the decommissioning of the Windscale Advanced Gas-cooled Reactor [7].

Nirex will consider requests to use non-standard packages, but additional costs are likely to be incurred at the repository owing to the need to make special arrangements to handle and dispose of the packages. In every case, the level of safety provided by the non-standard package will have to be equivalent to that of a standard package.

4. SPECIFICATIONS

Following on from the need for standardisation of containers, there is a need for specifications to define these standard containers and set minimum standards of performance. The Nirex Waste Package Specifications provide the essential link between waste package design and repository design. For waste producers, the specifications are required in order that waste packaging plants can be built and waste containers designed. For Nirex, the specifications provide a key element of the basis of design for the repository and transport system and also for the various safety cases to be made to the regulatory authorities.

Two facts in particular have been important in the development of the Nirex Waste Package Specifications:

(i) Long before the repository is ready to receive any shipments of waste, many producers wish to package waste and store it on site.

(ii) In the process of designing any waste processing or handling facility - including the repository itself - an important early input is the waste container dimensions and characteristics. Once design work gets under way the scope for accommodating changes becomes limited, and such changes are costly.

For these reasons the designers of waste containers, waste packaging facilities, transport containers and repository facilities have needed, as early as possible, the detailed dimensional, functional and performance criteria for waste packages together with any related requirements and supporting information The Nirex Waste Package Specifications fulfil this role, providing the link between waste package design and repository design.

Specifications have been produced for each of the standard packages. They incorporate the various requirements for disposal and transport and are compatible with the requirements for waste packaging, storage and handling at the sites of origin. They are based on design and safety case needs; supported by research and other technical studies on the performance of waste packages under repository (handling and disposal) and transport conditions.

Each specification defines dimensional, functional and performance criteria for the waste container and wasteform. These criteria include activity content, dose rate, surface contamination, heat output, dimensions, shape, handling features, venting and filtration, impact performance, integrity and stackability.

In formulating the specifications, the aim has been to minimise the risk that the requirements of the eventual Conditions for Acceptance will be more restrictive. This has been achieved by ensuring that the specifications allow for the inevitable uncertainties that exist at the present stage of repository development. This is necessary if future design options are not to be curtailed.

5. DEVELOPMENT OF ACCEPTANCE CRITERIA

Acceptance criteria will be identified by Nirex once all the constraints governing the disposal of waste at the repository have been identified. It is anticipated that the acceptance criteria will be made available to customers in the form of Conditions for Acceptance.

The Conditions for Acceptance will have to take account of any constraints imposed by the authorisation for disposal. The authorisation which will be issued under the Radioactive Substances Act 1993 [8] may result in limits being placed on:

- total volumes of waste for disposal
- total radioactivity
- activity of certain radionuclides
- mass of particular materials

In addition to the authorisation, the Conditions for Acceptance will have to encompass all the other constraints including conditions imposed by the Nuclear Site Licence, Transport and other Regulations as they exist at the time, operational and post-closure safety, and repository design and operational requirements. It is clear therefore, that the Conditions for Acceptance will not be available until much closer to the start of repository operations.

The Conditions for Acceptance will incorporate the standards and Waste Package Specifications. Inevitably there will be some changes in the latter as the development of the repository progresses and the regulatory requirements are updated. However, since the specifications have purposely been designed to be robust to uncertainties, it is expected that any changes will lead to a relaxation in requirements rather than a tightening.

6. NUMBER OF PACKAGES

As noted previously, intermediate level waste is already being packaged for disposal in the deep repository. As of January 1991 some 80,000m^3 of ILW was in stock, and by January 1994 some 2,000m^3 had been immobilised and packaged for disposal in standard 500 litre drums. By the time the repository opens for business in the early years of the next century (around the year 2010), accumulated wastes will have increased to about 156.000m^3 and further wastes will contine to arise throughout the lifetime of the repository.

Estimates of the volumes of waste which will arise over the next 15 years and during the nominal 50 year lifetime of the repository are central to all Nirex's work on transport and repository design. Information on existing stocks and predicted future arisings of radioactive waste are published in the United Kingdom Radioactive Waste Inventory [9]. This inventory, together with companion documents providing data on radionuclide content and the physical and chemical characteristics of waste [10, 11], are produced by Nirex in conjunction with the UK Department of the Environment. The Inventory is regularly updated and is published every 2 or 3 years.

The Inventory contains detailed information on accumulated stocks and predicted arisings on a waste stream basis. The 1991 Radioactive Waste Inventory details nearly 1000 separate ILW and LLW waste streams. The projections of future arisings of radioactive wastes are made by the producers of the waste on the basis of their programmes and policies, and their best estimates of the nature and magnitude of future operations and activities. The Inventory includes estimates of the quantities of future arisings up to the year 2030. This is the furthest date into the future to which waste producers could reasonably be expected to make predictions. However, estimates of decommissioning wastes predicted to arise from reactors and other facilities operating before 2030 are included.

The volumes of waste recorded in the 1991 Radioactive Waste Inventory are shown in Table 1.

The Radioactive Waste Inventory is not in itself sufficient for Nirex to determine the required capacity for the deep repository. This will ultimately be determined by customer demand and so Nirex has to plan its future strategy in a flexible manner so that it can accommodate changes in waste arisings. In order to provide a basis for planning, design and safety case preparation, Nirex is considering a range of waste

Table 1 : Waste Volumes Recorded in the UK Radioactive Waste Inventory

Waste Type	Operational Wastes			Decommissioning Wastes			Total all wastes to 2030 (cu m)	Total all wastes up to and beyond 2030 (cu m)
	Stocks at 1 1 91 (cu m)	Stocks plus arisings to 2030 (cu m)	Arisings post 2030 (cu m)	Stocks at 1 1 91 (cu m)	Stocks plus arisings to 2030 (cu m)	Arisings post 2030 (cu m)		
ILW	78,500	202,000	NE	1,530	55,100	163,000	257,000	420,000
LLW	6,250	218,000	NE	259	510,000	1,120,000	728,000	1,848,000

note volumes are expressed in conditioned form
NE = not estimated

Table 2 : Scenarios for Repository Capacity

Waste Type	Scenario 1 (cu m)	Scenario 2 (cu m)
ILW	300,000	600,000
LLW	100,000	1,400,000
Total	400,000	2,000,000

note volumes are expressed in conditioned form

arisings as shown in Table 2. The range being considered by Nirex varies from 400,000m³ to 2,000,000m³. The volumes shown in Tables 1 and 2 are given for the waste in a conditioned form, that is to say the volumes take account of any solidification medium used in ILW conditioning but exclude the volume of outer packaging. In the case of LLW, supercompaction is assumed to be used wherever possible.

Waste volumes naturally provide an important input to transport system and repository design but information on the physical and chemical characteristics of the waste and on its radionuclide content is just as important. This information is available within the Inventory and companion volumes and because information is provided on a waste stream basis it can be used in assessments to determine the categorisation and packaging requirements of waste for transport. Such an exercise is currently in progress and could be repeated if necessary to examine the implications of recent proposals [12] for revisions to IAEA Safety Series 6.

A summary table showing the results of one such waste categorisation exercise is given in Table 3. This assessment is based on the assumption of 300,000m³ ILW and 100,000m³ LLW: the minimum scenario from Table 2

The following points should be noted.

- The majority of ILW will be packaged in unshielded containers. These packages will be transported in reusable shielded containers from the site of arising to the repository.

- On-going work is examining the unshielded ILW in more detail to determine the proportion which will meet the current definitions of Low Specific Activity (LSA) material and which does not require shielding. Consideration is being given to the development of a reusable IP-2 transport container for these wastes.

- Shielded packages such as the 4m ILW box make up a small proportion of the packages to be transported to the repository but are nevertheless a key element of the UK waste management strategy. Based on ISO freight container principles they have been demonstrated to provide an efficient means of transporting and disposing of bulky items, minimising the amount of size reduction required for packaging.

- Inevitably there will be a requirement for the Nirex repository to be able to accept non-standard packages. The WAGR box is included on Table 3 as an example. This packaging concept has been developed to suit special circumstances.

Table 3 : Estimated Number of Transport Packages to the Repository

Transport Package	Container Classification	Contents Description	Contents Category	Number of Transport Packages
intermediate level waste				
reusable transport container	Type B	*unshielded packages*		
		4 x 500 litre drums	Type B and LSA	90,000
		3 cu m box / beta-gamma box	Type B and LSA	30,000
		3 cu m drum	Type B and LSA	5,000
4m ILW box	IP-2	decommissioning ILW	LSA II and LSA III	1,500
WAGR box	IP-2	decommissioning ILW	LSA II and LSA III	200
low level waste				
4m LLW box	IP-2	supercompacted or loose waste	LSA II and LSA III	4,000
2m LLW box	IP-2	supercompacted or loose waste	LSA II and LSA III	4,000

notes based on volume of 400,000 cu m of waste
includes wastes from BNFL Sellafield

The data presented in Table 3 can only be regarded as an estimate due to the inevitable uncertainties in the volume and nature of wastes requiring disposal. Irrespective of the exact numbers it is clear that there will be a significant transport operation associated with the repository and Nirex is devoting much effort to the development of a safe and efficient transport system which meets the needs of the UK in a cost effective manner.

7. CONCLUSIONS

Nirex has established, in conjunction with its customers a range of standard containers suitable for packaging the various wastes arising in the UK. Each standard package meets an identified need and is defined by a Waste Package Specification.

The Waste Package Specifications provide the essential link between waste package design and repository design, and define dimensional, functional and performance criteria. They establish a minimum level of performance for all package designs and provide a firm basis for repository and transport system design. Conditions for Acceptance will be issued but cannot be provided until the conditions of the authorisation and other regulatory requirements are known. Until that time, the Waste Package Specifications provide a sound basis for the packaging of waste.

The UK Radioactive Waste Inventory provides detailed information on accumulated stocks of radioactive waste and estimates of waste predicted to arise in the future. However the disposal capacity of the deep repository will ultimately be determined by customer demand and therefore flexibility is an essential element of repository and transport system design. Nirex is currently considering disposal volumes ranging from $400{,}000m^3$ to $2{,}000{,}000m^3$.

The UK Radioactive Waste Inventory provides detailed information on a waste stream basis and has been used to determine the categorisation and packaging requirements of waste for transport.

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PACKAGINGS FOR IRRADIATING WASTE: INDUSTRIAL EXPERIENCE AND EVOLUTION OF THE REGULATIONS

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Abstract

Paragraph 422 of 1985 Edition of IAEA Regulations for the Safe Transport of Radioactive Material specifies that "the quantity of LSA material or SCO in a single industrial package... shall be so restricted that the external radiation level at 3 m from the unshielded material ... does not exceed 10 mSv/h (1 rem/h)". This requirement, introduced in the last edition of IAEA regulations, precludes the transport of the most irradiating waste in an IP container. This is a concern for many waste producers like EDF in France, or waste management organizations like SKB in Sweden. The paper presents solutions which have been developed by TRANSNUCLEAIRE with the aims to keep in use the primary container (whose compliance with storage requirements has been strictly demonstrated), to continue to have easy operations (to maintain a low level of dose uptake for the workers), and to show that the requirements of the transport regulations are fully satisfied. It also points out our views concerning the lack of need of a new type of package, intermediate between Industrial Package and type B ; this new type is not necessary if the properties of non dispersibility of the content are taken into account in the safety analysis.

1. INTRODUCTION

Paragraph 422 of IAEA Regulations for the Safe Transport of Radioactive Material (Safety Series n°6, 1985 Edition - As amended 1990) specifies that "the quantity of LSA material or SCO in a single industrial package ... shall be so restricted that the external radiation level at 3 m from the unshielded material ... does not exceed 10 mSv/h (1 rem/h)".

This requirement, introduced in the last edition of IAEA regulations, results in the impossibility to transport the most irradiating waste in an IP container.

This is a concern for many waste producers like EDF in France, or waste management organizations like SKB in Sweden.

These two companies asked TRANSNUCLEAIRE to suggest a solution. Both organizations strongly expressed their wish to keep in use the primary container (whose compliance with storage requirements has been strictly demonstrated), to continue to have easy operations (to maintain a low level of dose uptake for the workers), and to show that the requirements of the transport regulations are fully satisfied.

The purpose of this communication is to present the solutions which are to be implemented, and to show how compliance with the regulations is obtained. The relationship between the philosophy of these solutions and some proposals for modifying the regulations will also be discussed.

2. MEDIUM LEVEL WASTES IN FRANCE

2.1 General

For a long time, medium level wastes generated by EDF reactors have been immobilized within a solid matrix and packed in concrete cylinders. These concrete cylinders are then sent to ANDRA low/medium level waste storage site.

According to the previous transport regulation (IAEA 1973), these packagings were classified as Industrial Packagings, and they were transported as such by railways and road.

Since the implementation of IAEA 1985, some of these packagings - with the most irradiating contents - are no longer classified as IP but as type B packagings, because the dose rate at 3 m of the unshielded content exceeds 10 mSv/h (1 rem/h).

The concrete cylinders are considered as a standard equipment by EDF, and furthermore they have been licensed by ANDRA for long term disposal in compliance with the storage regulations. Therefore, any change in the design of the disposal conditioning would entail operational and financial burdens.

The solution proposed by TRANSNUCLEAIRE and its justification as regards IAEA regulations will be described and explained hereafter.

2.2 Description of the cylinders

The most irradiating medium level wastes produced by EDF during operations of their power plants are ion-exchanger resin and primary loop filters.

These wastes, respectively mixed with a polyester resin or surrounded by cement, are placed into two main families of concrete cylinders named C4 (outside diameter 1100 mm, height 1300 mm) and C1 (outside diameter 1400 mm, height 1300 mm).

The main basic features of these cylinders are common to all types :

- cylindrical container with a wall and a bottom made of concrete slightly reinforced by steel rods,
- additional shielding, consisting of an internal steel shell and top and bottom plates, either in steel or lead,
- the cylinders are plugged with cement poured in place after loading the wastes in the cavity,
- at the top side of the cylinder is a groove used for lifting and handling.

Figure 1 shows C1 and C4 cylinders.

The total mass of the present and future cylinders ranges from 2500 kg to more than 6000 kg.

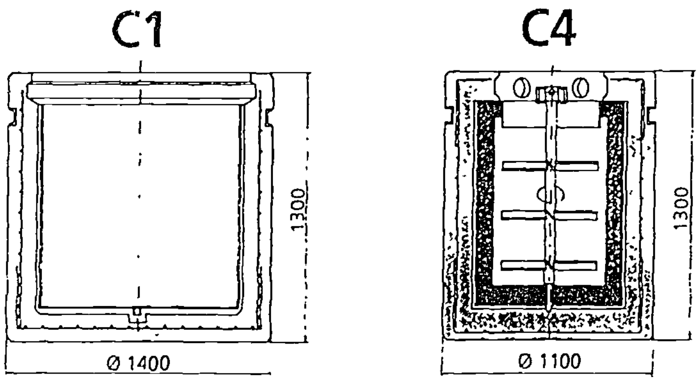

FIGURE 1

C1 AND C4 CONCRETE CYLINDERS

2.3 Principles of solution for a type B container

Very quickly, it appeared obvious that the classical solution for a type B package (i.e tight containment vessel with gaskets, shock absorbers...) will not satisfy both handling conditions and price objectives.

It therefore seemed necessary to use the particularity of the content itself to demonstrate that the cylinder is in compliance with type B requirements.

As already mentioned, the content of the cylinder is a radioactive material mixed with or surrounded by a binder such as polyester resin or cement.

This type of concrete cylinder with its content had been tested in accordance with the very severe criteria of the agency responsible for their storage (ANDRA). It had been shown that the contents of these cylinders withstand lixiviation tests specified by ANDRA and therefore has a strong ability to retain the activity by itself.

Therefore, the idea was to demonstrate that in accident conditions in transport, the content of the cylinder was still able to present a very low activity release. For that, it was necessary to show that, after drop, puncture and fire tests, the content of the cylinder was still undamaged.

2.4 Development of overpack

It is clear that if an unprotected cylinder is submitted to a 9 m drop, its concrete shell will break and the content could be exposed to the flame during the fire test. Direct exposure to the flame is not acceptable because in this case, resin or cement would reach a high temperature and release more activity than allowed.

TRANSNUCLEAIRE developed its overpack with the following philosophy.

- Prevention of activity release is made by the content itself (radioactive wastes immobilized in resin or cement matrix).

  On the basis of the previous tests, it is shown that if the matrix remains below approximately 150°C, no significant activity release occurs from the matrix.

- The concrete cylinder must remain in place around the content after the drop and puncture tests in order to act as thermal insulator during the fire test. However, some limited cracks can occur in the concrete without further consequences during fire test.

- In order to maintain the integrity of the concrete during drop and puncture tests, it can be surrounded by a steel overpack.

A sketch of this overpack is shown in figure 2.

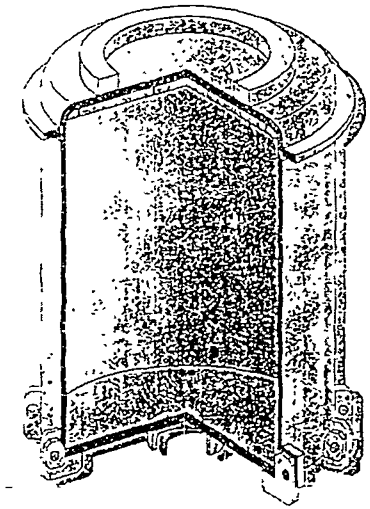

FIGURE 2

STAINLESS STEEL OVERPACK

Two full scale prototypes (one for C1 and an other one for C4) are under manufacture to perform drop, puncture and fire tests, with temperature instrumentation of the representative inactive content in order to show that it remains below allowable limits.

Preliminary 9 m drop test campaigns have been performed on 1/2.5 scale models showing that, with this steel overpack, the resin or the cement matrix of the content remains undamaged and well protected in order to withstand the subsequent fire test. Tests on full scale prototypes are expected to confirm these results.

3. MEDIUM LEVEL WASTE IN SWEDEN

3.1 General

Wastes generated by Swedish utilities are immobilized within a solid matrix and packed in concrete or steel cubic boxes, or steel drums. These boxes or drums are transported in containers, such as ATB-12K, and then sent to SFR low/medium level waste underground storage site.

It appeared that the implementation of IAEA 1985 created in Sweden problems similar to those described for France.

- The use of Industrial Packages, as ATB-12K are classified at this time, is no longer possible, due to excessive dose rates around the unshielded content.

- The primary content (box or drum) has been licensed according to storage regulations.

- The containers have been studied to be fully compatible with the organization implemented in Sweden for the transport from the nuclear plant to SFR facility (transfer from the plant to the SIGYN ship, stowage in the ship, and transfer from the ship to the storage facility).

- These containers amount to a fairly high investment.

Therefore the challenge, raised by SKB, was to adapt the whole system to the new requirements of the regulation, without modifying it !

.2 Description of the conditioning of the wastes and of the container

Wastes generated by Swedish producers and concerned by the limitation of dose rate around the unshielded content are :

- medium level ion exchange resins or metal scraps mixed with cement and cast in cubical concrete boxes 100 or 250 mm thick and 1.2 m wide,

- low level ion exchange resins mixed with cement or bitumen, or metal scraps mixed with cement, cast in cubical steel boxes 5 mm thick and 1.2 m wide,

- low level ion exchange resins mixed with cement or bitumen and cast in 200 liter drums.

Twelve cubical boxes or forty eight drums can be placed in one ATB-12K container.

This container (figure 3) is made of 130 mm thick steel plates, welded together, and a cover (which is another 130 mm thick steel plate) connected to the box through lockers.

Main internal useful dimensions of the "box" of the container are :

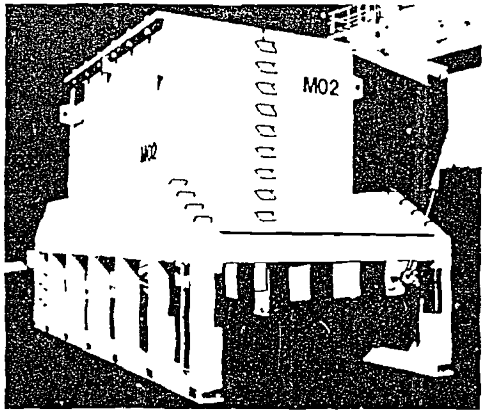

FIGURE 3

ATB CONTAINER

- length : 3 960 mm
- width : 2 570 mm
- height : 2 450 mm

A frame is welded to the container and allows its easy transfer and stowage, as it is a standard equipment designed to comply with SKB transport system.

Maximum total weight of the ATB-12K container with its content is 120 tonnes, with the following breakdown :

- container, lid and internal arrangement : 61 tonnes
- transport frame : 8 tonnes
- contents (12 concrete boxes) : 51 tonnes

3.3 Principles of solution for a type B container

The principles chosen were to take advantage of the characteristics and properties of the radioactive waste, when it is mixed or bound with cement or bitumen. This allowed to find a solution without large modifications of the present design.

These materials provide a good containment of the radioactive material as long as they remain in place, on one hand, and their temperature does not reach unacceptable values.

Consequently, in this case also, the aim was to demonstrate that after drop, puncture and fire tests the content of the container was still undamaged.

It also appeared that the container itself, due to its large weight, and therefore high thermal capacity, can act as an efficient thermal insulator for the content, not only in the case of concrete boxes, but also for steel boxes or drums, even in the case of bitumen.

Therefore, the target was to assure that the container remains in place, after the mechanical tests, so as to avoid the dispersal of the radioactive contents and to provide the requested thermal protection.

3.4 Present and further development

TRANSNUCLEAIRE developed for SKB the modifications of ATB-12K with a philosophy parallel to that explained in the case of French wastes :

- Prevention of activity release is made by the content itself (radioactive wastes immobilized in cement or bitumen matrix).

  On the basis of tests performed in Sweden, it is shown that if the temperature of the matrix remains low enough, no activity release occurs from the matrix.

- The steel container must remain in place around the content after the drop and puncture tests in order to provide thermal insulation during the fire test.

- In order to maintain the global integrity of the container during drop and puncture tests, it must be reinforced.

Preliminary tests, with one sixth scale models have shown that for existing containers, the reinforcement of the connection between walls with screws, and of the tightening system of the cover, with more efficient lockers, could enable to reach this goal.

Furthermore, calculations have shown that the temperature of the content, when the package is submitted to the regulatory fire test, does not exceed 110 °C. This temperature guarantees that no unacceptable activity release will be observed.

These possibilities of reinforcements, which have been tested, may be considered either in the case of the modification of the existing containers, or in the development of a new design (ATB-8K) for the transport of more active wastes.

In the future, it is foreseen to modify and optimize the reinforcements in two directions :

- number of screws, in the case of an existing container, or increasing of the resistance of the welds in the case of a new design,

- modification of the reinforced tightening system, in order to make its use as close as possible to the present one.

4. MODIFICATION OF THE REGULATIONS OR OF THE MINDS ?

The International Atomic Energy Agency and the member states are working on the revision of the Regulations. The transport of Low Specific Activity (LSA) materials and Surface Contaminated Objects (SCO) is one of the main topics of this revision.

A Technical Committee Meeting (TCM) took place in October 1993 to discuss this matter. Among many other subjects, the question was raised to know whether the transport of LSA materials or SCO, in a quantity such that the radiation exceeds the limits prescribed in paragraph 422, should require a new specific type of packagings. Taking in consideration the solutions proposed by designers, such as those described in this paper, and the requirements of Competent Authorities, it was concluded that an additional modification of the regulations was neither desirable, nor necessary.

It is our opinion that an intermidiate level, between Industrial Packaging and Type B packaging, could lead to useless confusion as regards the safety of transport of radioactive waste.

We are in a position to design and use type B packages dedicated to the transport of LSA materials, which comply with the regulation without being too sophisticated. For this purpose, it is necessary to take advantage of the properties of both the content (LSA material) and the packaging.

It is therefore advisable to definitely avoid the confusion between the compliance with the regulations concerning the release of activity on one hand, and a high degree of leak tightness implying the use of seals on the other hand.

Our conclusion can be summarized in a short catch-phrase : keep the regulations, change the minds !

THE SWISS CONCEPT: CONTAINER CONCEPT AND THE TRANSPORT OF L/ILW TO THE PLANNED FINAL REPOSITORY

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Abstract

The design of the Swiss final repository for short-lived L/ILW is based on a Nagra container and package concept. The package handling operations have been restricted to a minimum through the design of special handling tools e.g. a gripper for 9 drums. The routine transport weight by rail is 56 t, and for non-routine transport 80 t (maximum). The transport of drums and reprocessing waste will be in reusable steel containers and that of decommissioning waste in dual purpose transport and disposal containers. Most of the containers have standardized dimensions and corner fittings which are based on the ISO dimensions. The modes of transport for the containers and packages within the repository include overhead cranes, an air cushion platform for precise manoeuvering in limited spaces and internal rail transport. The handling and transport will mostly be remotely controlled and monitored by video cameras from the control room. Hence, the exposure times of the operating personnel in the radiation environment is minimized.

2. INTRODUCTION

In June 1993, the proposed site for the Swiss final repository for L/ILW was selected and a formal application for a general license will be submitted in mid-1994 to the Swiss Government. The proposed location is at Wellenberg, in central Switzerland, 25 km south of Lucerne (see Fig.1).

The repository is designed to take all operational L/ILW from the 5 Swiss nuclear power plants (3 GWe), low-level reprocessing waste, waste from medicine, industry and research and decommissioning waste from all nuclear facilities. The total projected amount of L/ILW is approximately 100,000 m^3, derived from a model waste inventory data bank which itself includes larger quantities of wastes to allow for planning reserves (see Table I). The inventory scenario allows for a 40 year lifetime of all 5 operating NPPs; it does not at present consider further use of nuclear power after the year 2024.

Two operating phases of the repository are planned: one starting approximately in the year 2002 and lasting until 2013 and the second from about 2017 until 2032.

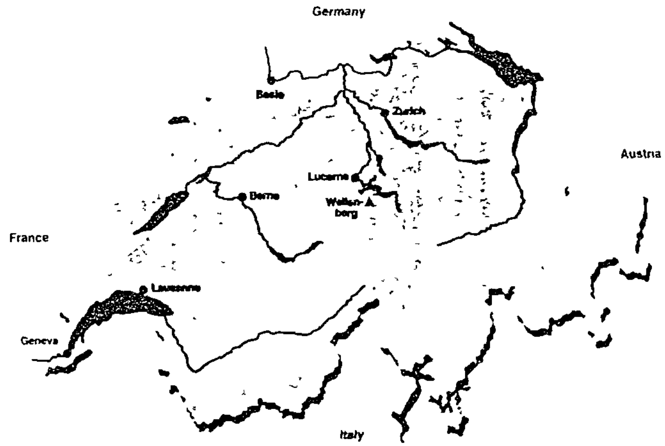

Fig. 1.
Proposed site for the Swiss final repository at Wellenberg for L/ILW

TABLE I
Model Radioactive Waste Volume (m3) for the Swiss L/ILW Final Repository

Waste Sorts	Waste Volume (m3) ODL* < 2 mSv/h	Waste Volume (m3) ODL* > 2 mSv/h	Waste Volume (m3) Sum	Remarks Number of Packages
Routine Operational Waste NPP (BA)	4450	4610	9060	ca 44000 200l-Drums
Concrete Shielded Container (BA-C)	200	0	200	ca 205 980l Packages
Waste from Medicine, Industry and Research (MIF)	5300	3500	8800	ca. 35100 200l-Drums, Remainder PSI-Container (small)
Reprocessing Waste				
BNFL	17500	0	17500	ca 14770 Packages
CAC	6500	0	6500	ca 9700 Packages
Non-Fuel Reactor Core Waste (RA) (200l-Drums, Mosaik, SA-Container)	1810	510	2320	ca 2300 200l-Drums
Decommissioning Waste (SA)				
SA NPP	43000	0	43000	total ca
SA PSI	14200	0	14200	3100 SA-Container
Total (m3)	92960	8620	101580	109175

*ODL=Surface Dose Rate at the Package, Values are at the Time of Production

The time gap between the operating phases is necessary to allow construction of the second set of disposal caverns. In phase 1, mainly operational waste from nuclear power plants and waste from medicine, industry and research in the form of 200 l-drums (approximately 80,000 drums) and reprocessing waste will be emplaced in the facility. Phase 2 is principally foreseen for emplacement of decommissioning waste (approximately 58,000 m^3) and reprocessing waste.

3. CONTAINER CONCEPT

3.1. Transport Container

The transport container concept is based on a set of reusable containers for transporting 200-l drums and reprocessing waste to the final repository. The transport containers are designed to hold a maximum number of waste packages in order to minimize the transports to the repository and to simplify the handling operations. Most of the transport containers have standardized dimensions and corner fittings which are based on the ISO dimensions (see Fig. 2). The handling tools reflect standardized ISO technology which has been used repeatedly for handling ISO freight containers.

The surface dose rate of the waste drum determines which container will be used for transport. There are transport containers which are designed to allow a surface dose rate up to 1 Sv/h at the waste drum. With one exception, the containers are designed as industrial packages type IP-3 in compliance with the 1985 IAEA Safety Series No. 6 Regulations, as amended in 1990. One transport container, a GNS Mosaik II, is designed to meet the requirements of a type B (U) package. The construction materials of all these types of containers are steel plate, cast steel or ductile cast iron.

A special design is the transport container (ZWI-TC) for holding a pallet with 72x200-l drums (low-level operational waste) which is used in Swiss NPPs for interim storage purposes. The pallet itself already has the maximum design dimensions of an ISO 20-foot container. Therefore, the transport container had to be designed with excess width and length dimensions. The handling of the ZWI-TC cover uses the same handling tool as for the pallet, thus requiring no change of handling tools.

For the internal repository transport of the packages (mainly 200-l drums) with higher dose rates, a shielded transport container (TAI) was designed. The TAI can be handled by the overhead crane via its ISO corner fittings. The lid is divided into two, one section being fixed on the container body, the other can be moved with a hydraulic drive (see Fig. 3).

The unloading of the 200-l drums from the transport containers takes place with a special handling tool which can handle 9 drums at a time.

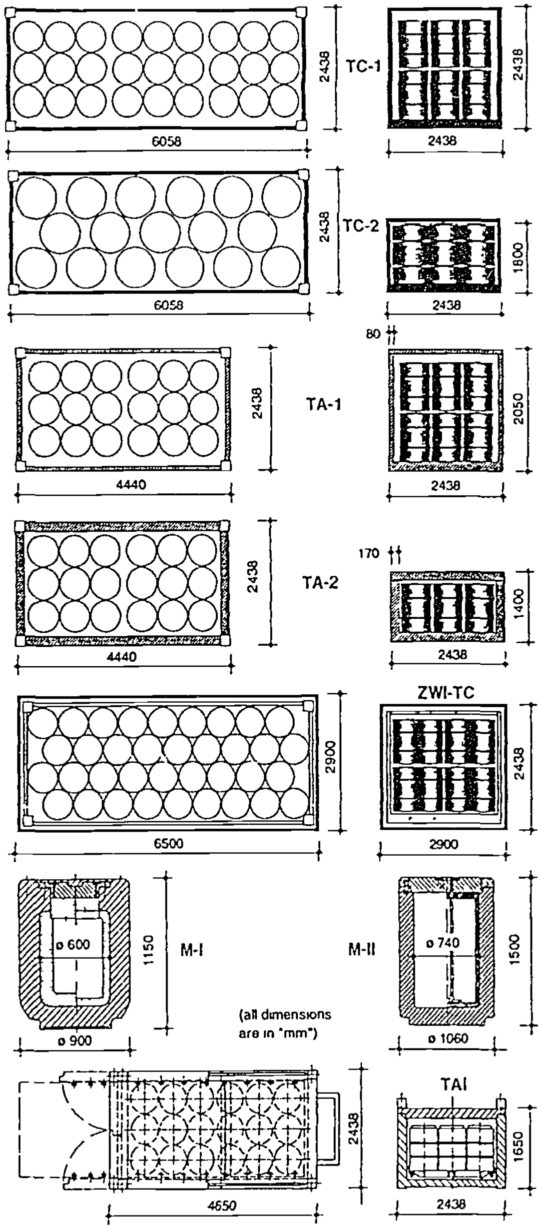

Fig 2 Transport Container Concept for the Swiss L/ILW Final Repository

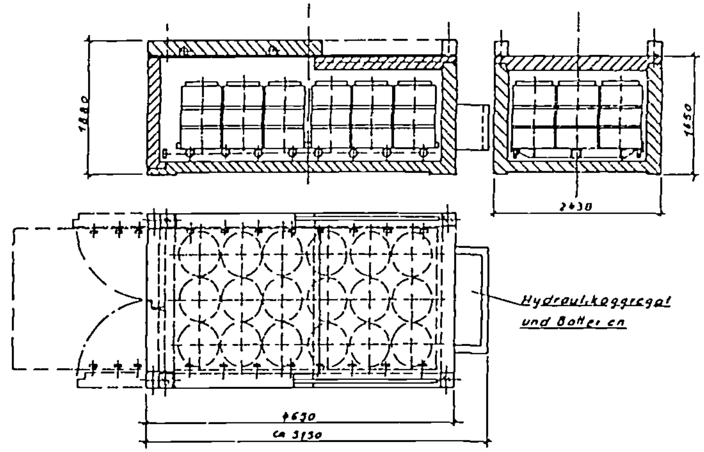

Fig. 3. Shielded transport container for the internal repository transport of higher dose rate packages

3.2. Final Disposal Container

There are two options for disposing of the 200-l drums in the lower part of the repository caverns One is the disposal of the 200-l drums in final disposal containers (EC container) and the other is the direct disposal of the drums in sets of 9 in the cavern or handling and emplacement in pallets holding 9 drums.

The EC containers are designed to hold either 200-l drums or decommissioning waste (see Fig.4). The EC containers holding 200-l drums are for internal repository use only The EC containers holding decommissioning waste are designed as dual purpose transport and disposal containers in compliance with the 1985 IAEA Safety Series No 6 Regulations, as amended in 1990. The construction material of the final disposal containers is concrete. The maximum weight for a single loaded container is 80 t

The EC containers are filled remotely with 200-l drums in the preparation facility of the operations cavern The void volume of the concrete container loaded with drums will then be filled with a porous mortar. Thereafter, the concrete container will be moved by air cushion platform into a concrete-hardening position outside the preparation facility. After hardening, the container will be lifted by an 80 t capacity overhead crane onto the internal rail system and brought into the final storage cavern

4. TRANSPORT CONCEPT

Repository Design

The repository consists of a system of tunnels and caverns which will be excavated into the mountain The repository will be accessed horizontally from the

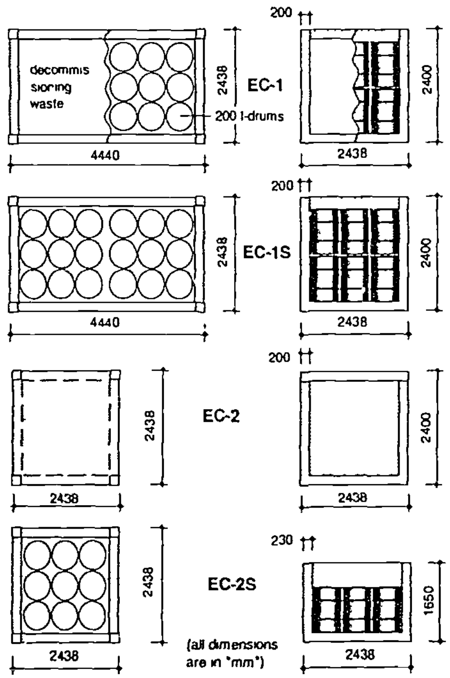

Fig. 4. Final disposal container concept for the Swiss L/ILW repository

valley floor through two access tunnels. Most of the repository facilities will be inside the Wellenberg, except for an administration building and the air intake and exhaust air facilities which are outside the mountain (see Fig. 5 and 6). The repository is comprised of the access tunnels, the reception facility, connecting tunnels and the final storage caverns. The reception facility is divided into a waste transfer cavern, an operations cavern and an auxiliary cavern. The operations cavern hosts the central control room, from where all handling and storage operations will be remotely controlled and monitored by video cameras.
Figure 7 shows a computer-generated figure of the final repository facilities inside the Wellenberg.

Transport Modes

The final repository can be reached by rail or road, with rail transport being the prefered mode of transportation.

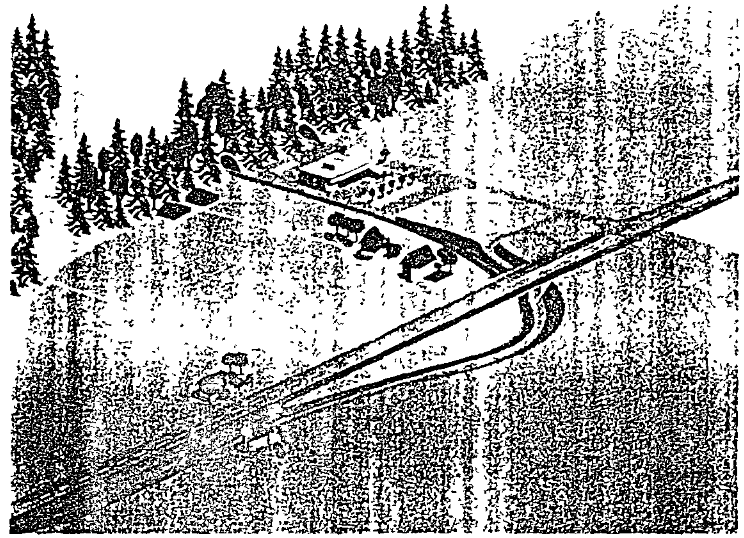

Fig 5 Computer-generated figure of the repository facilities at the mountain entrance area

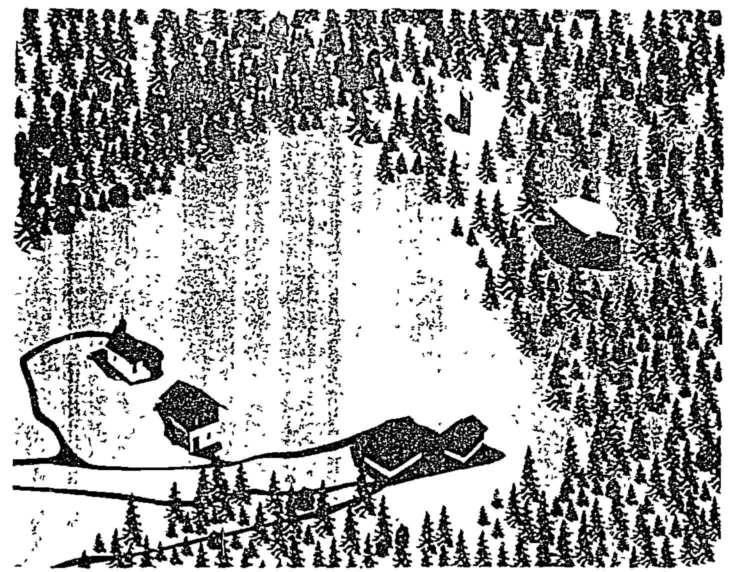

Fig 6. Computer-generated figure of the air intake and exhaust air facilities outside the mountain

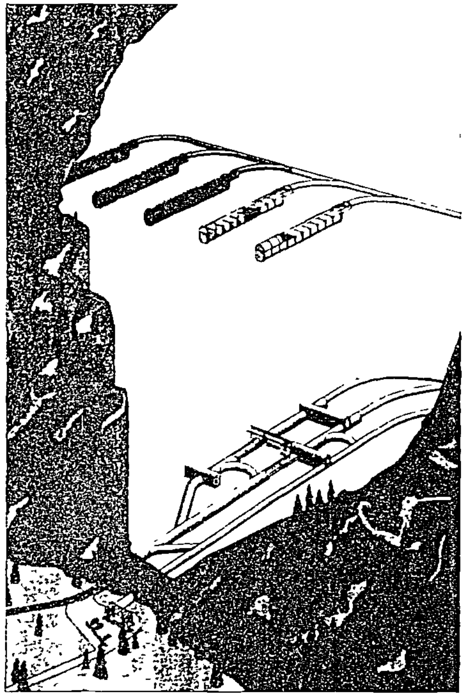

Fig 7. Computer-generated figure of the final repository facilities inside the Wellenberg

Transport by rail will take place from the NPP or interim storage facility via Lucerne to Wellenberg. A Nagra locomotive will haul the railway waggons into the repository. The number of transport operations to the repository will be between 150 and 240 per operating year and operations depend on the number of waggons a train consists of.

In the reception facility there will be a switch from external to internal transport systems. The modes of transport for the packages inside the repository include overhead cranes, air cushion transport for precise manoeuvering in limited spaces and internal rail transport.

The routine transport weight by rail is 56 t and for non-routine transport 80 t (maximum).

MULTIPURPOSE STORAGE/TRANSPORT/ DISPOSAL PACKAGES FOR DOE NUCLEAR LOW-LEVEL WASTES: AN EMERGING NEED AND A REGULATORY CHALLENGE

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Abstract

As the United States embarks upon a major effort to cleanup its nuclear defense facilities, a large quantity of low-level waste (LLW) will be generated. This LLW must be managed and ultimately placed into final disposal. Much of this waste is expected to exceed certain limits defined in U.S. regulations (Title 10, U.S. Code of Federal Regulations, part 61) called Class C. The waste which exceeds Class C, called Greater-than-Class-C (GTCC), poses a major challenge to waste managers. Each GTCC waste form must be placed into costly geologic disposal unless separate approval is obtained from the United States regulator to place it into less costly "near-surface" land burial. Management of GTCC will also require, to some extent, storage and transport prior to its final disposal. A further LLW stream exists in the United States also stemming from the prior operations of United States defense facilities, viz., radioactively contaminated and irradiated scrap metal which has been accumulating over the past forty years. Similarly, as cleanup, decontamination, and decommissioning proceeds, this contaminated scrap metal inventory is expected to grow rapidly. This paper explores the notion of the authors that an opportunity for a synergistic solution to two difficult waste management problems may be available in the United States today, and perhaps may similarly be available in other nuclear countries as well. The possibility exists for fabricating packagings from contaminated scrap metal (which would otherwise be part of the waste inventory) and for using these packagings for storage, transport and disposal of GTCC in near-surface burial facilities without re-opening or re-packing. This approach is appealing and should lead to major safety and cost benefits. An examination of existing regulations with the intent to propose additions, changes, or clarifications that would effectively and beneficially regulate such combined activity is proposed.

1. INTRODUCTION

The U.S. Department of Energy (DOE) is responsible for site restoration and waste management of United States government facilities involved in nuclear defense operations which took place for more than four decades. Within DOE, the Office of Environmental Restoration and Waste Management (EM) is responsible for this major environmental cleanup and management effort.

As these activities progress in the near-term, a very large inventory of low-level waste (LLW) will have to be managed. This waste will result from the restoration of existing facilities, processing of stored waste, reclassifying salvage material as waste, and decontamination and decommissioning (D&D) of facilities. This waste will contain not only radiologically hazardous materials, but in many cases will also contain other hazardous materials. Wastes containing both radiologically hazardous materials and other hazardous materials are called *mixed* wastes.

One classification of LLW that will present a technical challenge to waste managers is Greater-Than-Class-C LLW (GTCC). This waste, as will be discussed, does not fit conveniently into existing disposal regulations. Since there is no identified disposal facility for GTCC, this waste will need to be stored as site restoration activities accelerate and the waste is processed and packaged. This paper will review the regulatory environment for packaging GTCC, discuss the scope of the GTCC LLW inventory in DOE, and will propose a beneficial approach for packaging this waste form for storage, transportation, and disposal.

2. REGULATIONS

Disposal

LLW is defined in the U.S. Code of Federal Regulations, Title 10, part 61 (10CFR61) [1]. 10CFR61 classifies LLW as Class A, B, or C. The classes are defined according to concentrations of certain short-lived and long-lived radionuclides (curies per cubic meter). In general, 10CFR61 allows these classes of wastes to be disposed by near-surface burial (i.e., up to 30 meters below the surface of the earth). However, for wastes that have radionuclide concentrations in excess of the limitations imposed for Class C (therefore called GTCC), 10CFR61.55(a) stipulates that near-surface burial is not generally acceptable.

Further, 10CFR61.55(a)(2)(iv) states that in the absence of specific disposal requirements for GTCC in part 61, "such waste must be disposed of in a geologic repository as defined in part 60 of this chapter unless proposals for disposal of such waste in a disposal site licensed pursuant to this part are approved by the Commission." This means that to comply with these regulations, one must dispose of all GTCC waste in a geologic repository unless one obtains approval of an alternative approach from the U.S. Nuclear Regulatory Commission (NRC).

These regulatory conditions are important to DOE. While DOE is not itself a licensee of NRC and is not required to dispose of LLW generated by the Federal Government in facilities licensed by NRC, it is nevertheless required by the *Low-Level Radioactive Waste Policy Amendments Act of 1985* [2] to dispose of "any other low-level radioactive waste (than that generated by the Federal Government) with concentrations of radionuclides that exceed the limits established by the Commission (NRC) for Class C radioactive waste" that results from activities licensed by NRC (i.e., commercial GTCC LLW) and to do so "in a facility that is licensed by the NRC". Therefore, we assume in this paper that DOE will dispose of all GTCC LLW in full compliance with NRC regulations.

DOE does have the opportunity to propose an alternative to geologic disposal (10CFR60) [3] for approval by the Nuclear Regulatory Commission. This procedure is required for each specific waste form classified in the general category, GTCC, for which an alternative to geologic disposal is being pursued. In many instances, substantial effort will be required to assure NRC that near-surface disposal of a GTCC waste is adequately safe. It should be expected that the regulator will be rightfully cautious about any such approvals and could be expected to reject those proposals which fall short in its judgment of providing adequate safety. As a consequence, the waste manager is faced with acceptance of the costly disposal methods suitable for disposal of high level waste, or must undertake whatever effort is required to plan near-surface disposal in a safe manner, to obtain approval of the regulator that his plan is safe, and to execute the near-surface disposal plan under conditions of the approval. DOE management of this waste form is not completely formulated. The cost/benefits of the disposal options remain to be evaluated.

Regarding packaging design, there are significant differences between packages designed for near-surface burial (pursuant to 10CFR61) and to those designed for geologic disposal (pursuant to 10CFR60) in keeping with the regulations. For near-surface burial, the waste form is required to have substantial physical and chemical stability in addition to volume concentration limits and other conditions. However, no specific design conditions are placed on package design for near-surface disposal. In contrast, for geologic disposal, few restrictions are placed on the waste form and protection of the environment depends strongly on an "engineered barrier system" which includes both the waste package and the underground facility.

Requirements for geologic disposal (pursuant to 10CFR60) are very general in nature and specific packaging design criteria are not established. The DOE is presently in the process of establishing packaging design criteria for geologic disposal. However, this work is far from complete and the final outcome of disposal package design is uncertain.

Storage

There are no specific regulations regarding the storage of GTCC wastes. However, the requirements for storage of spent nuclear fuel and high level waste (pursuant to 10CFR72) [4] could be considered as a conservative model for the storage of GTCC. 10CFR72 establishes requirements, procedures, and criteria for the issuance of licenses to receive, transfer, and possess power reactor spent fuel and other radioactive materials associated with spent fuel storage in an independent spent fuel storage installation (ISFSI). In addition to overall requirements, this regulation provides criteria for assuring nuclear criticality safety and radiological protection, and provides approval requirements for spent fuel storage casks.

The design criteria for storage casks under 10CFR72 are relatively clear when compared to the guidance for disposal package design under 10CFR60. Further, significant precedents have been set for storage cask designs and a significant number of storage casks have been designed, licensed and placed in service for storage of spent fuel at reactor sites.

It is also important to note that 10CFR72 allows for the approval of storage casks that have been certified for transportation under 10CFR71. The approval will rely upon a safety analysis report showing that the cask is suitable for storage under 10CFR72. However, the regulations do apparently contemplate the prospect that a single cask design could satisfy both storage and transport regulations. This clearly suggests the possibility of multipurpose cask designs.

Transportation

Regulations contained in 49CFR171-178 [5] are the U.S. Department of Transportation (DOT) regulations that establish performance criteria for LLW transport packagings. For fissile materials and quantities of radioactivity exceeding Type A levels, the DOT regulations refer to the NRC regulations, 10CFR71, [6] which provide performance criteria for transport packages containing these higher activity levels. It is possible that packages containing GTCC could hold quantities of radioactivity exceeding Type A quantities. Therefore, the transport package containing GTCC could be either a Type A or Type B package under the transport regulations. In general, design criteria pursuant to the transport regulations are relatively clear when compared to available design guidance for packages intended for storage or disposal. This is due in large part to an extended history of radioactive materials (RAM) transport package design and actual shipments in the U.S. that have provided the experience needed to formulate design criteria that will result in safe packagings. Attention is focused on criticality, thermal performance, shielding, and containment.

3. THE CHALLENGE

The prospect of achieving a successful multipurpose package design should be examined immediately to establish whether or not such a package for near-surface burial of GTCC LLW should be introduced into the DOE waste management and environmental strategy. The urgency stems from existing DOE actions and plans to accelerate waste processing, storage, and disposal.

A multipurpose package design for storage, transport, and disposal of GTCC waste must meet the regulatory conditions for all three of these functions. Unfortunately, package design conditions are substantially different for each of the functions: storage, transport, and disposal. While transport packages are designed primarily to contain their radioactive contents when subjected to high levels of trauma and to provide shielding; storage packages are designed to withstand substantially lower levels of trauma without releasing their contents and to provide shielding; and disposal packages (or the waste form) are designed primarily to sustain long-term (300 years or more) physical and chemical stability.

While broad design goals such as those described above can be expressed for the design of a single multipurpose package, the actual design specifications for a single package that would meet the combined set of regulations for storage, transport, and disposal are difficult to establish. The primary difficulty lies in the paucity of experience in the design of storage and disposal packages and the lack of specificity in the regulations themselves for these two areas.

The challenge therefore is to achieve a *rapid evaluation of a multipurpose* package design for storage, transport, and disposal of GTCC in the face of:

[1] Significant differences in packaging design criteria among each of the three functions
[2] Paucity of design experience for storage and disposal packages
[3] Lack of specificity in the regulations for design of storage and disposal packages
[4] A national U.S. commitment for clean-up of existing DOE facilities with a desire for early progress.

4. SCOPE OF THE GTCC CONTAMINATED METAL LLW INVENTORY IN THE DOE

GTCC Low-Level Waste

As DOE site restoration and waste management activities move from evaluation and study to actual clean-up, a growing need for appropriate packagings for storage, transportation, and disposal will emerge. Site restoration includes waste stream identification, waste minimization, waste processing, interim storage, transportation, and disposal.

Two difficulties in assessing the inventory of GTCC waste within the DOE complex are that characterization efforts of the waste sites are still underway and there is no one program responsible for the management of GTCC. However, there are some preliminary estimates which can provide insight into the magnitude of the waste stream. A study conducted by the Idaho National Engineering Laboratory [7] estimates that there is 1,075,000 m^3 of what the DOE designates as special case wastes. These wastes are distinguished by the following descriptors:

- Transuranic waste that cannot meet DOE's Waste Isolation Pilot Plant (WIPP) Waste Acceptance Criteria or the payload compliance plan for the TRUPACT-II shipping container.
- Wastes that exceed the activity concentration limits for near-surface disposal of LLW (pursuant to 10CFR61).
- Nuclear fuel debris used for research purposes.
- Excess nuclear materials that are no longer useful to the present custodians and those that cannot be readily isolated.

Wastes in each of the categories noted above could lie within the definition of GTCC.

Further, for commercially generated GTCC, the Low-Level Radioactive Waste Policy Amendments Act of 1985 requires the DOE to dispose of GTCC waste generated by Licensees of NRC and Agreement States. A second study [8] conducted by EG&G/Idaho estimates the volume of commercial GTCC to be in the range of 4000 m^3. According to the Act, this waste must be disposed of in an NRC licensed facility.

Contaminated Metal Low-Level Waste

One waste stream that poses a significant management challenge to DOE is radioactively-contaminated scrap metal. Contaminated scrap metal in the DOE complex arises from site restoration activities as well as D&D of DOE facilities. The metal scrap may also be surface contaminated with hazardous constituents as well as radioactivity.

The estimated amounts of these metals are enormous. Preliminary estimates have concluded that there are about 1,500,000 tons of contaminated scrap metal stored at various DOE sites [9]. From DOE enrichment plant D&D alone, it is estimated that there will be generated 600,000 tons of contaminated scrap metal [10]. Current contaminated scrap metal generation is about 15,000 tons/year and is expected to increase to 90,000 tons/year as D&D operations accelerate.

This material presents unique packaging problems for storage, transport, and disposal. Much of the metal is in a physical shape that is large and awkward. Packaging such material in standardized containers becomes problematic.

5. A PROPOSAL FOR A MULTIPURPOSE PACKAGING FOR GTCC LLW

Multipurpose Packaging

We propose here a system approach that holds promise for a synergistic solution to two difficult waste management problems. This approach is based on the management of GTCC waste by the use of multipurpose packaging that would be used for storage, transport, and disposal. The system would include the ultimate disposal of the GTCC package by emplacement in a near-surface burial facility pursuant to 10CFR61. For this waste form, the design of the storage and transportation packaging would be relatively straightforward because of the specific requirements for design approval set forth in 10CFR72 and performance criteria set forth in 10CFR71. (A packaging designed to the transportation regulations would normally meet the storage requirement.)

A key element of this proposal is to fabricate the multipurpose packages for GTCC by melting and casting contaminated scrap metal into such packagings. The fabrication of such packagings is discussed in the following subsection.

The difficult challenge would be to design such a packaging that would also meet the uncertain disposal criteria. However, design enhancements (e.g. addition of corrosion allowance material and an evaluation of long-term seals) could be made in order to satisfy the intent of the regulations. Therefore, a packaging design could be undertaken with regard to storage, transport, and disposal criteria. Specifically, the design should focus on the performance capability of the package to maintain integrity for disposal of GTCC in a near-surface burial facility per 10CFR61. This work should be put in the form of a Safety Analysis Report (SAR). Once this work has been instituted, discussions could be opened with regulatory authorities regarding their views. Ultimately, full regulatory approval would be required for all three functions.

Fabricating multipurpose packagings from melted contaminated scrap metal

The DOE is currently conducting several evaluations concerning the melting down of contaminated scrap metal. These studies range from recovery of strategic metals to the melting down of structural steels for beneficial use.

One application of this technology pertinent to packaging of GTCC waste is to melt down DOE-owned contaminated scrap metal and cast multipurpose GTCC Type A packagings. This technology is mature and offers an opportunity for near-term success with relatively low risk. Advantages of applying this technology as described herein include:

1. The DOE contaminated scrap metal inventory will be reduced because it will be used as charge material for manufacturing the multipurpose packagings.

2. The multipurpose packagings will eliminate the need for separate packagings for storage, transport, and disposal for a single payload.

3. Certain low-level metal wastes must be shipped in Type B packages due only to localized hot spots in the metal which could drive a given package contents over Type A levels. The melting and casting of such wastes will produce uniform dispersion of the radioactivity and could result in the redesignation of some waste transport packages from Type B to Type A. This will greatly enhance DOE's ability to transport and dispose of this waste on-site.

4. General radioactive, TRU, and hazardous contaminated metals can be used in the melting process.

5. Much radioactivity precipitates to the slag or flows to the off-gases where it is captured by HEPA filtration systems. Radioactivity remaining in the molten metal, principally Co-60 and Cs-137, becomes volumetrically-fixed. Surface dose rates of the processed metal would be thereby significantly lowered.

6. The melting process may, in fact, reduce the activity volumetric concentrations back to low-level waste Class A, B, or C under 10CFR61, thereby allowing near-surface burial options.

7. Hazardous constituents are volatilized to the off-gases. EPA regulations allow certain hazardous wastes to be treated in this manner. Where not so allowed, this approach would have to be modified.

In the event that a packaging is configured by melting and casting contaminated scrap metal and this packaging is used to carry Type B quantities under the transport regulations, it may be necessary to employ a reusable transport overpack to achieve regulatory transport certification for this package.

Using contaminated scrap metal for the fabrication of GTCC multipurpose packagings that could be used in a near-surface burial facility offers potential advantages to GTCC managers.

6. RECOMMENDATIONS

The following recommendations are identified as a means to evaluate the feasibility of using contaminated scrap metal for fabrication of a multipurpose GTCC packaging and to obtain an early regulatory view regarding disposal of GTCC in near-surface burial facilities according to 10CFR61.

1. Perform a prototype design of a GTCC multipurpose packaging fabricated from contaminated scrap metal. This design should meet all appropriate storage and transportation regulations and should meet the intent of the near-surface burial regulations as put forth in 10CFR61.

2. Prepare a Safety Analysis Report (SAR) based on a packaging that will contain GTCC waste for storage, transportation, and disposal.

3. Open early discussions with regulatory authorities with the prospect of submitting a Topical Report based on the SAR for regulatory review.

4. Concurrent with the above activities, an effort to develop a design specification for multipurpose packages should be initiated. One reason that the adoption of this technology has been slow is that there is no design basis that has been codified. This specification could be developed under the aegis of a non-government consensus standards body with full peer review by technical experts. A public comment period should be built into the review process. Such a specification would lend credibility to the new packaging and would provide assurances to the public.

   An example of a standards body that would be appropriate for such an activity is the American Society for Testing and Materials (ASTM). ASTM committee C-26 on the nuclear fuel cycle, has a subcommittee, C-26.07 on nuclear waste materials. The C-26.07 charter is to "develop appropriate standards and guides for management (including the treatment, transport, handling, storage, and disposal) of the nuclear fuel cycle and other radioactive waste materials that will minimize the environmental impact and associated interactions with man and the biosphere". A task group could be formed under C-26.07 with participants selected from all stakeholder groups, including, private utilities, the manufacturing industry, and of course the regulators and DOE (which has specific obligations under the Low-Level Radioactive Waste Policy Act of 1985). Such a specification could then be used as a basis for developing a similar international specification under the auspices of the IAEA.

5. Because of its obligations under the Low-Level Radioactive Waste Policy Act of 1985, the U.S. Department of Energy should consider these recommendations with a view to taking a leadership role in their pursuit.

7. CONCLUSIONS

Evaluation of packaging options for GTCC waste for storage, transport, and disposal should proceed vigorously. The feasibility of using multipurpose packagings that would meet the intent of the near-surface burial regulations as defined in 10CFR61 should proceed. This work should be documented in the form of an SAR that should subsequently be submitted to regulatory authorities for review and advice. Further, work should commence to develop an appropriate design specifications criteria document through a non-government consensus standards organization. These actions comprise the most expeditious and effective approach to resolving the problem of GTCC waste disposal.

The evaluation of fabricating the multipurpose packagings from contaminated scrap metal will provide radioactive waste managers with an attractive alternative to current disposal options for GTCC and contaminated scrap metal.

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1. Code of Federal Regulations, Title 10, Part 61, "Licensing Requirement for Land Disposal of Radioactive Wastes".

2. Low-Level Radioactive Waste Policy Amendments Act of 1985.

3. Code of Federal Regulations, Title 10, Part 60, "Disposal of High-Level Radioactive Wastes in Geologic Repositories".

4. Code of Federal Regulations, Title 10, Part 72, "Licensing Requirement for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste".

5. Code of Federal Regulations, Title 49, Parts 171 - 178.

6. Code of Federal Regulations, Title 10, Part 71, "Packaging and Transportation of Radioactive Materials".

7. IDAHO NATIONAL ENGINEERING LABORATORY, RADIOACTIVE WASTE TECHNICAL SUPPORT PROGRAM, Department of Energy Special Case Waste Inventory and Characterization Data Report, DOE/LLW-96 (Draft), (1990).

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TRANSPORT OF LOW LEVEL WASTES FROM COGEMA REPROCESSING PLANTS

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Abstract

Wastes generated by reprocessing plants fall into two main categories : high level wastes and low level wastes. Low level wastes are conditioned for disposal in shallow-ground repositories. This paper is devoted to packages which have been developed by COGEMA for transport and final disposal of low level wastes.

Most low level wastes consist of contaminated items that are immobilized in a concrete matrix. The application of IAEA transport regulations concerning the classification of these wastes as LSA materials is unclear, especially concerning the definition of the material to be considered in order to calculate its specific activity, and concerning the method to be used to demonstrate the homogeneity of the material. These two problems concern all operational wastes generated by nuclear facilities. This is why, the French Competent Authority drafted a guide in 1991 in collaboration with the main French radwaste generators (COGEMA, CEA, EDF) for the characterization of wastes as LSA

COGEMA, in applying this guide, has developed packages for shipment of low level wastes. For some wastes, the material may not be classified as LSA. In this case an overpack is used for the shipment to give the package Type B characteristics

1. INTRODUCTION

Radioactive wastes generated by reprocessing plants are of several types·

- products issuing directly from the reprocessed fuel· fission products, hulls, end-fittings and metallic structures of the fuel assemblies,
- products issuing from reprocessing facilities or generated by their operation:
  - technological wastes: cleaning rags, safety clothing, contaminated used equipment, filters, etc.
  - liquid effluents (bituminized sludge)

These wastes are classed according to their radioactive content in three categories [2] .

- type A wastes· β or γ emitters with short- or medium half-lives ($\leq$ about 30 years).
- type B wastes: containing significant amounts of long-lived emitters which are usually α emitters,
- type C wastes high level wastes (mainly consisting of fission products issuing from irradiated fuels or of unreprocessed irradiated fuels)

All products directly issuing from the irradiated fuel and these issuing from liquid effluents are classed among type B and C wastes, and most of the technological wastes are classed as type A wastes

For the time being, only type A wastes are intended for final disposal in shallow-ground repositories

The residues arising from reprocessing are distributed among the different clients of these plants. Residues must be returned to their country of origin as required by the French Law on radioactive wastes adopted in 1991. Those allocated to French clients or to COGEMA are sent to the French shallow-ground repositories managed by ANDRA.(low level waste) or put in interim storage in COGEMA facilities (high and intermediate level waste)

To minimize the risk of spreading the activity present in these wastes, both in its own installations and during their transport, COGEMA directly produces the residues so that they can be accepted for transport and final disposal without reconditioning

The residues produced meet specifications written by COGEMA, approved by the French Competent Authority , and agreed, or on the way of being agreed, by COGEMA's foreign clients. As far as low level wastes are concern, they must be agreed by ANDRA for the acceptance in French shallow-ground disposals

To be transported, these waste packages, alone or with their transport casks, must also meet the applicable radioactive material transport regulations.

For obvious economic reasons, whenever possible, it is preferable for the packages to be of the Industrial Package type (IP2) rather than Type B package. COGEMA therefore tries to condition its low level wastes so that they can be classed as Low Specific Activity (LSA), which can be carried in Industrial Package, while meeting the residues specifications.

2. CLASSIFICATION OF CONDITIONED WASTES FOR TRANSPORT

To meet the requirements applicable to LSA, waste producers were very quickly faced with the inaccuracy of certain regulatory requirements, and especially those concerning leach tests, homogeneity and the calculation of specific activity

In France, a working group, led by IPSN and including French waste producers (COGEMA, EDF, CEA) and ANDRA, was accordingly set up to determine how to apply the requirements applicable for LSA to the low level waste packages

The results of this working group's deliberations [3] were used by COGEMA for the characterization of the residues that it produces and which are sent for disposal in shallow ground repositories.

2.1 Review of regulatory requirements for the transport of low specific activity (LSA) materials [1]

For the transport of LSA, the IAEA regulations, resumed by international regulations and French regulations, distinguishes between the requirements applicable to·
- the material (LSA-2 or LSA-3),
- the package (IP2),
- the vehicle.

<u>LSA-2</u>

To be classed as LSA-2, solid materials must meet the following requirements:
- uniform distribution of radioactivity,
- specific activity $\leq 10^{-4}$ A2/g,
- dose equivalent rate of the bare material $\leq$ 10 mSv at 3 m.

<u>LSA-3</u>

To be classed as LSA-3, solid materials must meet the following requirements:
- uniform distribution of radioactivity,
- specific activity $\leq 10^{-3}$ A2/g,
- dose equivalent rate of the bare material $\leq$ 10 mSv at 3 m,
- activity released by leaching $\leq$ 0.1 A2/week

<u>IP-2</u>

Solid LSA-2 and LSA-3 materials transported under "exclusive use" must be transported in type 2 industrial packages (IP-2). These packages must meet the following requirements:
- pass tests to prove resistance to the following normal transport conditions: free drop test and stacking test, without any loss or dispersion of the radioactive content and/or more than 20% increase in radiation intensity over the entire outer surface of the package,
- meet the dose rate limits of 2 mSv on contact and 0.1 mSv at 1 m,
- meet the following surface contamination limits:
  - β, γ emitters and low toxicity α emitters· 4 Bq/cm^2,
  - other α emitters 0 4 Bq/cm^2.

<u>Transport vehicle</u>

Transport vehicles are subject to the following limits for transport authorization (by road or rail)
- meet the dose rate limits of 2 mSv on contact and 0.1 mSv at 2 m,
- if the material is considered as combustible, the total activity must be limited to 100 A_2.

For waste packages stored in shallow-ground repositories, COGEMA has opted for a classification of the material as LSA-2 Materials that cannot be transported as such are transported as type B packages or, exceptionally, transported by "special arrangement".

Moreover, since some packages might contain combustible materials, the maximum activity per vehicle or wagon is limited to 100 A_2

2.2 Recommendations of the IPSN working group [3]

For the method for evaluating the specific activity of the wastes, the working group made a distinction between two conditioning methods
- bulk conditioning, in which the wastes are placed in bulk in the package and then immobilized by a binder (concrete), which can also serve as radiological shielding,
- conditioning in aggregate, in which the conditioning process produces an intimate mixture between the binder and the wastes

The mass of material that should be taken into account for calculating the specific activity is·
- for bulk conditioning, the mass of wastes before conditioning,
- for aggregate conditioning, the mass of wastes and binder.

As to the low level wastes produced by COGEMA in its reprocessing plants, most of them fall into the first group.

To evaluate the homogeneity of LSA-2, the following rules are recommended:
- the volume considered is the envelope volume of the wastes inside the package,
- the elementary volumes for evaluating local specific activity in the package are $\leq$ 0.2 m^3,
- the specific activity of each elementary volume must not be more than 10 times the activity permitted for LSA-2 or 10^{-3} A_2/g,
- for any waste volume less than 0 2 m^3, the homogeneity rule is considered to be satisfied

COGEMA has therefore set up systems making it possible to guarantee the activity distribution in all packages with an inside volume of more than 0 2 m^3

2.3 Classification of waste packages from reprocessing plants intended for shallow ground repositories

On the basis of the foregoing criteria and the recommendations of the working group, COGEMA checked that most of the packages that it is likely to produce can be classed among LSA-2 low specific activity packages (preparation of data sheets for each type of package) and is setting up methods designed to confirm this for each package produced (control of waste origin, activity measurement, weighing, etc)

3 PREPARATION OF LOW LEVEL WASTE PACKAGES

The wastes thus treated are technological wastes from the different facilities making up the reprocessing plant Depending on the origin of these wastes, the type and/or proportions of the radionuclides they contain may vary considerably Till recently, to facilitate the management of the wastes produced, each plant was subdivided into about ten sectors for which an average radionuclide spectrum was established These average spectra were regularly reassessed over the different reprocessing campaigns. Each of these spectra was associated with transfer functions making it possible to evaluate the activity from a dose rate or neutron dose measurement At the present time, the activity of low level waste is measured

The wastes leaving a facility are usually packed (100 or 120 litre drums or sealed containers). These wastes are identified, weighed, and their activity measured. A waste production form is filled in for each batch of identical wastes removed from a sector This form contains all the data needed to characterize the origin and type of the wastes. The waste conditioning facility can thus control the activity of the residues that it prepares and evaluate the following for each package produced:
- main radionuclides present,
- total activity of the package (in A_2),
- specific activity of the material (in A_2/g),
- homogeneity of the material in connection with its specific activity,
- dose rate at 3 m of the bare material.

The measurement of the dose rates on contact and at 1 m, and of the surface contamination of each package prepared, completes the information needed for transport.

4 *DESCRIPTION OF THE DIFFERENT TYPES OF PACKAGE DEVELOPED BY COGEMA*

Four types of package have been developed :

CO package (not yet produced)
This is a stainless steel container with an volume of about 200 litres. Its maximum weight is 500 kg.
The content of this package consists of 120 litre waste drums compacted, stacked and then immobilized by concrete.

CBF-C1 package
This is a cylindrical fiber concrete container with a volume of about 660 litres. Its maximim weight is 1800 kg
The content of this package consists of 120 litre waste drums, compacted, stacked in a 200 l metal drum, which is itself immobilized by fiber concrete

CBF-C2 package
This is a cylindrical fiber concrete container with a volume of about 1200 litres. Its maximum weight is 4000 kg
The content of this package consists of primary packages of uncompacted wastes, stacked and immobilized by concrete.

CBF-K package
This is a cubic fiber concrete container with a volume of about 5 m³ . Its maximum weight is 15000 kg
The content of this package consists of miscellaneous wastes in bulk and immobilized by fiber concrete.

5 QUALIFICATION OF PACKAGES DEVELOPED BY COGEMA

The waste packages produced in COGEMA's reprocessing plants are prepared by following qualified procedures and by observing the quality assurance requirements. In particular, this concerns the content and the containment of the content of the packages, as well as the processing of the materials (mainly concrete).

Specimens of each of the different types of package described above have been subjected to the tests set for type 2 industrial packages:

- free drop tests (§ 622 of the IAEA Regulations for the Safe Transport of Radioactive Materials)
- stacking tests (§ 623 of the IAEA Regulations for the Safe Transport of Radioactive Materials)

The damage caused by these tests to the packages is very slight. The deformed or burst volumes amount to a few litres for initial volumes ranging between 213 litres and several cubic metres, and the containment of the material is unaffected.

6 TRANSPORT SYSTEMS

Apart from the large cubic packages, all the other types of package are transported in large transport containers, specially designed, and meeting the definition given in § 130 of the IAEA transport regulations.

Large packages are transported without special arrangements.

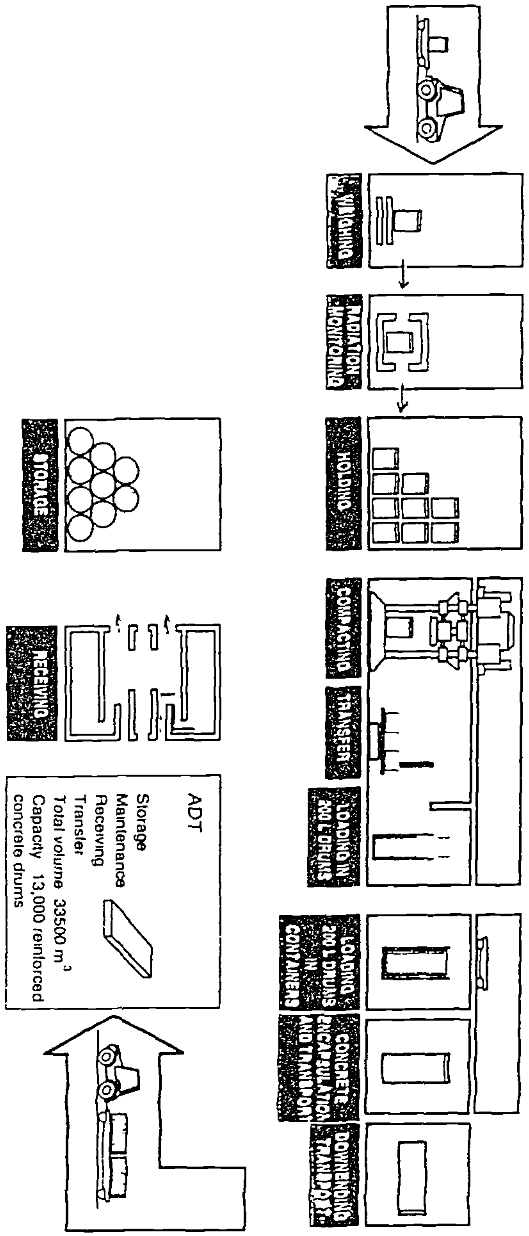
All shipments are made by road and rail.

6.1 DV-78 transport container

This container has the standard ISO dimensions of a 20-feet container, with an opening roof and back doors. It has the following main characteristics:

- capacity: 12 CBFC-1 or 6 CBFC-2
- overall length: 6058 mm

TECHNOLOGICAL WASTE PROCESSING



EXAMPLE OF RADIOACTIVE SPECTRUM (LA HAGUE)

S1	ZONE ORIGINE	CAMPAGNE DE RETRAITEMENT ANNEE 1992
	0 1	

RAPPORT EN ACTIVITE

α/β = 1,07E-2

Aa(t=300)/Aa(t=0)= ,8403

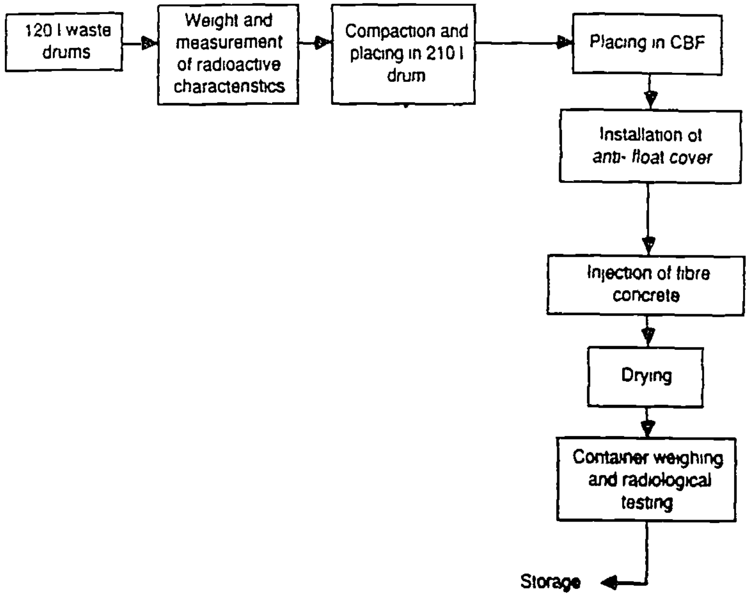
U/B = 8,48E-8 Aa(t=0)/Aa(t=300)= 1,1901

RADIOELEMENT	%en ACTIVITE	RADIOELMENT	%en ACTIVITE	RADIOELMENT	%en ACTIVITE
Mn 54	9,00	Np 237	6,10E-3	U 232	1,05
Co 60	83,00	Pu 238	40,02	U 233	
Ru 106		Pu 239	8,27	U 234	70,94
Sb 125		Pu 240	11,35	U 235	,84
Cs 134	1,00	Pu 241	5,71E-2	U 236	9,98
Cs 137	3,00	Pu 242	4,00E-2	U 238	17,18
Ce 144		Am 241	25,48		
Pm 147	2,00	Am 243	,23		
Eu 154		Cm 242			
Sr 90	2,00	Cm 244	14,57		

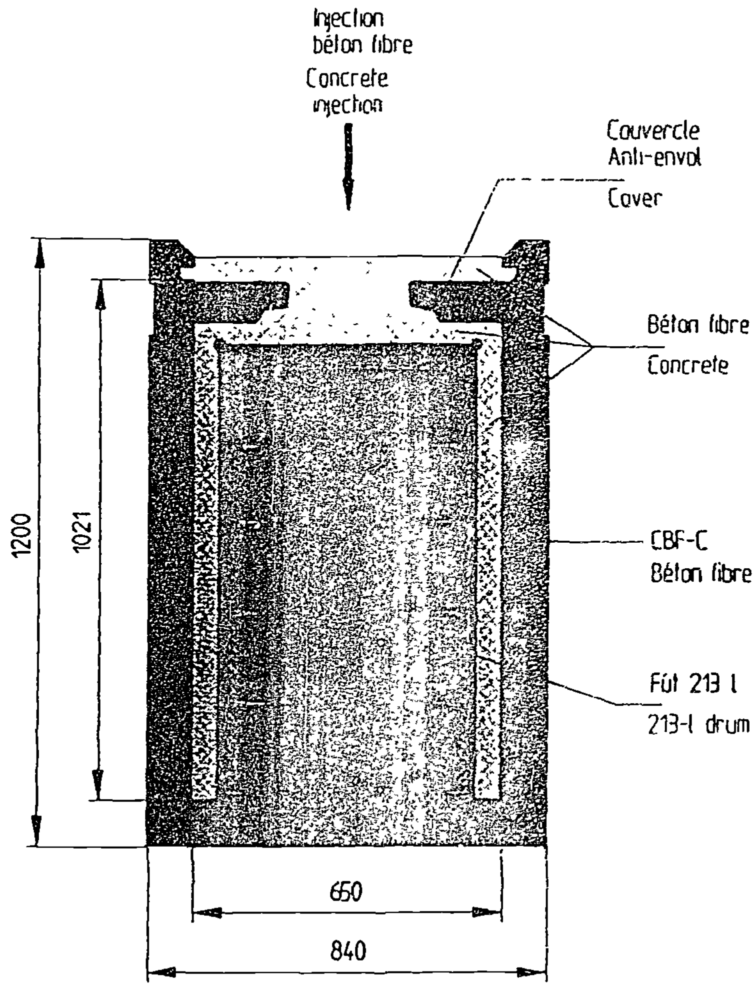
- overall width: 2438 mm
- overall height: 2591 mm
- maximum loaded weight. 24,000 kg

This container will be used to ship wastes to the ANDRA repository.

CONDITIONING PROCESS FOR CBF-C1 (LA HAGUE)



CBF-C1 WASTE PACKAGE (LA HAGUE)



6.2 DV-79 transport container

This container has the standard ISO dimensions of a 20-feet container, with an opening roof and back doors, and reinforced front and sides. It has the following main characteristics:

- capacity. 50 packages of types C0, 4P or 4Q,
- overall length: 6058 mm
- overall width: 2438 mm
- overall height· 2591 mm
- maximum loaded weight: 24,000 kg

This container will be used to ship wastes to the ANDRA repository.

REFERENCES

[1]: IAEA Safety Series No. 6 - Regulations for the Safe Transport of Radioactive Material, 1985 Edition (As Amended 1990)

[2]: Ministère de l'Industrie, des Postes et Télécommunications et du Commerce Extérieur, DSIKN, Rapport d'Activité 1992.

[3]: IPSN, results of "radioactive waste management group" ; DSMR/SSTR/CR/MAD/92-86, 30/01/1992.

DESIGN FEATURES OF SHIPPING CONTAINERS FOR LOW AND INTERMEDIATE LEVEL WASTES

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Abstract

Immobilisation of Intermediate Level Waste and Low Level Waste concentrates in cement matrices has been accepted as a conditioning process and is widely used in India since long. To keep pace with the increasing throughputs from recent solidification plants, shipping casks with improved designs are being adopted to transport single and multiple product drums from plant to storage/disposal sites. A salient feature of the vertical bottom loading shipping containers is the adoption of modular concept incorporating the loading/unloading platforms, the transportation unit and the lifting system as separable modules. Emphasis has been mainly on making the transportation units compact and devoid of any drives, controls and sensors which are prone to damages during transit which have been provided on the stationary loading/unloading platforms. The product entry into the cask is through a single bottom door in place of the split door design practised earlier thereby avoiding bulky compensatory shielding around the opening. The lifting system accommodates the lifting tackle for the cask as well as the hoisting system for the product drum. The casks are designed to employ a fail safe and load positive pneumatic grapple where pneumatic actuation is utilised for releasing the grapple once the load is stably supported. While the intermediate level shipping container is meant for a single drum, the low level shipping container has been designed to accommodate multiple drums inside the housing with provision of a rugged mechanised conveyor segment for handling the drums within the cask. Limitation of capacity of the design has been only availability of space and handling systems at the existing facilities. The cask designs have been evaluated analytically to determine their compliance with the applicable regulations governing containers in which radioactive materials are transported.

1. INTRODUCTION :

The Waste Immobilisation Plant Trombay is in the final stages of completion. The plant is meant to cater to the conditioning and packaging of various liquid waste streams of reprocessing plant origin. While glass has been selected as the matrix for solidification of high level waste, the intermediate waste will be incorporated in a cementatious matrix. The low level waste streams would undergo concentration by either evaporation or reverse osmosis before they are incorporated in cement. In order to cater to the increased quantities of waste to be processed as well as for consistency of product quality, and also to make the technology amenable to incell remote operation, it was decided to employ batch mixers for the cementation plants in place of in-drum mixing technique presently being followed. The expected throughputs from cementation plant per day will be within the range of 40-50 drums of 200 litre capacity. To make their handling and transportation to the disposal site cost effective, design of the shipping containers were conceptualised. The scheme is based on multi drum shipping containers with built-in handling systems.

2. OPERATIONAL REQUIREMENTS :

In the processing plant the homogenised cement-waste mix from the batch mixer is received in mild steel drums of following specifications:

Diameter	: 575 mm
Height	: 880 mm
Construction	: seam welded shell welded top and bottom rims
Closure	: bolt ring type amenable to be closed remotely
Weight	: approximately 400 kg when filled

The filled drums after initial setting are closed remotely and transported to the disposal site in a shielded container. The processing facility and the disposal site are located within BARC premises. Movement of the container is limited to a restricted area only. and the distance to be covered is in the range of two to five kilometers by road. In the processing plant, an electric overhead crane would handle the container while at the disposal site a mobile crane would be employed for its handling. Transportation of the container to the disposal site will be by a truck of suitable capacity. Design of the shipping container was carried out under the above perspective.

3. SHIPPING CONTAINER FOR LOW LEVEL WASTE PRODUCT :

Bottom loading concept was chosen for the design of the shipping container. A vertical design of all-steel construction has been adopted for carrying the low level solidified waste drums. The container is designed to house four drums at a time, the limiting factors being the availability of handling facilities and space at the processing plant.

The design takes care of the regulatory requirements laid down by IAEA [1] in respect of structural and radiological aspects. In arriving at the structural requirements, analytical methods were employed and no test on scale model or prototype has been carried out. Commercially available software using finite element techniques was employed for stress analysis. Based on the analysis, impact limiters [2] were incorporated wherever required.

A modular concept has been adopted where a shipping unit devoid of drives, sensors, limit switches etc. is employed in conjunction with transfer platforms and the lifting module which are separable from the shipping unit.

The technical specifications of the above transportation system are as below:

a. Shipping Container:

Length	: 3000 mm
Width	: 1000 mm
Height	: 1385 mm
Shielding	: 50 mm mild steel all around
Weight	: approximately 6,500 kg with full load
Conveyor Segment:	consists of rollers of 2" NB sch 40 pipes supported on 20 mm dia shaft using antifriction roller bearings and spaced in such a way that four rollers support each product drum.
Tie downs	: 4 numbers, 35 mm dia, both end supported pins for tie down using 30 mm dia wire rope.

b. Transfer Platforms :

Shielding	: 100 mm mild steel around product removal port
Port opening	: circular, 750 mm diameter
Door size	: 790 mm x 905 mm
Door drive	: pneumatic cylinder, 75 mm bore and 800mm stroke working on 6 bar pressure for the platform at loading end and hydraulic cylinder,75 mm bore and 800 mm stroke working on 15 bar pressure for the platform at unloading end.
Engagement with: cask door	mechanical coupling using 50 mm dia taper pin
Overall dimension	: 3160 mm L x 1200 mm W x 150 mm H

c. Lifting module :

Drum hoisting	: using 500 kg capacity, 10 meter lift, single fall electric chain hoist
Pneumatic hose reel	: 1/4" size with 10 meter fall
Main lifting beams	: box type construction of the size 150mm L x 75mm W x 6 mm thickness
Lifting trunions:	4 numbers, both end supported

By employing this modular concept, weight and size of the shipping cask has been kept to the minimum . All drive actuators, sensors and limit switches have been separated from the shipping cask and provided on transfer platforms for loading and unloading which are kept stationary at concerned sites. A lifting module equipped with built in hoisting system can be used at both sites by transporting it from the plant to the disposal site. But, it is not desirable to transport the same every time with the container and hence two separate and similar modules are provided to avoid any damage to the sub-systems and controls during transit.

A distinctive feature of the design is the geometry of the container and the drum configuration within it. One of the options was a circular design, incorporating indexing arrangements with spiders. Alternatively, a rectangular geometry with a single row configuration could be chosen. A detailed analysis of these options revealed that the former had certain limitations with respect to optimal utilisation of the cask volume and design/construction problems with respect to transfer door at the bottom. Accordingly, rectangular geometry was chosen.

Another feature of the unit is the incorporation of a rugged roller conveyor segment attached to the sliding door of the container which enables automatic positioning of the drums inside the container while loading. At the time of unloading of the drums an external push rod is employed to position each drum correctly below the grapple . The sequence of loading/unloading operations is inherently rendered safe by design. The mechanisms provided to achieve this are rugged and simple and require minimum maintenance.

Lead screw operated positioning and locking pads have been used for keeping the drum in position inside the container during transit. Side railings have also been provided to avoid transverse movement of the drum. Inadvertent release of these locking pads is prevented since they are operated only when the lifting module in position.

In place of split door design practised earlier single drum cask, a single bottom entry door design has been chosen. This has enabled reduction of container weight by obviating need for bulky shielding around the door. Risk of non sychronised movement of two pieces of the door has also been eliminated. Besides, drive arrangement for the door has also become simpler and maintenance free.

The double door transfer mode requires that the container and transfer platform doors remain open together during the transfer operations. Mechanical couplings between the container and platform doors have enabled employment of a single pneumatic/hydraulic cylinder mounted on the platform to actuate the doors. Cylinders with end cushioning are employed to avoid probable impacts at travel ends. Availability of compressed air as a utility plant service in the processing facility has enabled employment of pneumatic cylinder in the transfer platform at loading end. The transfer platform at disposal end is provided with a hydraulic cylinder actuated by a dedicated power pack.

Employment of a single drive for the two doors has ensured that container door cannot be opened unless it is well positioned over the transfer platform. Provision has also been made for manual opening of the two doors independent of each other for purpose of maintenance/repair of any of the component like inside of the container, grapple, door guides, drum positioners, locking pads etc.

The beams provided on the lifting module consists of lift beams designed for handling three times the weight of the container and the module put together. Guides have been provided on this module to locate it precisely over the cask. The lifting module is also equipped with an electrically operated chain hoist to lift the product drum for loading into the cask and to lower it for disposal. A pneumatically operated grapple for the drum has been provided for use within the container. The grapple has minimum radial dimensions for its easy passage through ports on the container and transfer platforms. The fingers of the grapple are of positive holding type and can be pneumatically actuated for their release. Air for actuation of the fingers is conveyed through an air hose reel suitable for 10 M lift. Provision has also been made to prevent inadvertent release of the fingers. The grapple has been tested for satisfactory performance in repetitive operations. Connection/disconnection of the grapple from the lifting module is achieved through a specially designed connector attached to the end of the hoisting chain. This connector ensures a positive mechanical connection between the chain and the grapple in addition to providing a leak tight connection between the hose reel and the grapple. The connector is operated through a worm and gear arrangement. The grapple is normally locked with the lid of the container and gets released only when the lifting module is positioned over the shipping container.

The door of the shipping container has been provided with two separate locking arrangements. One is operated manually while the other gets released only when the container is positioned over one of the transfer platform. Further, coupling between the door of the container and the transfer platform cannot be effective unless the two modules are positioned properly one over the other. These features ensure that accidental opening of the door of the container is totally avoided.

Electric power and compressed air to the system are received from the main connectors on the transfer platforms. Jumpers for power, control and air connection between the transfer platform and lifting module are housed on the shipping container. This arrangement ensures that the operation of the hoisting unit and the grapple could be carried out only when the three modules are positioned one over the other in the properly.

4. SHIPPING CONTAINERS FOR INTERMEDIATE LEVEL WASTE PRODUCT :

Operational requirements, design features and safety interlocks of the transportation system for intermediate level waste product drums are similar to those described above for low level waste and adopts the modular concept employing separate assemblies like transportation unit, lifting module and transfer platforms. However, as against an all-steel design in the case of low level shipping container, a lead filled container has been chosen in view of higher radiation levels encountered. Due to enhanced shielding requirements, the doors of the container and the transfer platforms have become heavier necessitating employment of separate drives for their operation. Lead screws driven by air motors are utilised for door movement. Components and sub-assemblies like grapple, connector, hose reel, chain hoist etc. are of similar design as those used for low level waste. The present design is for handling only one drum at a time on account of the heavy shielding involved and consequent overall weight. Containers housing multiple drums can also be adopted by augmentation of handling facilities at the loading and unloading sites.

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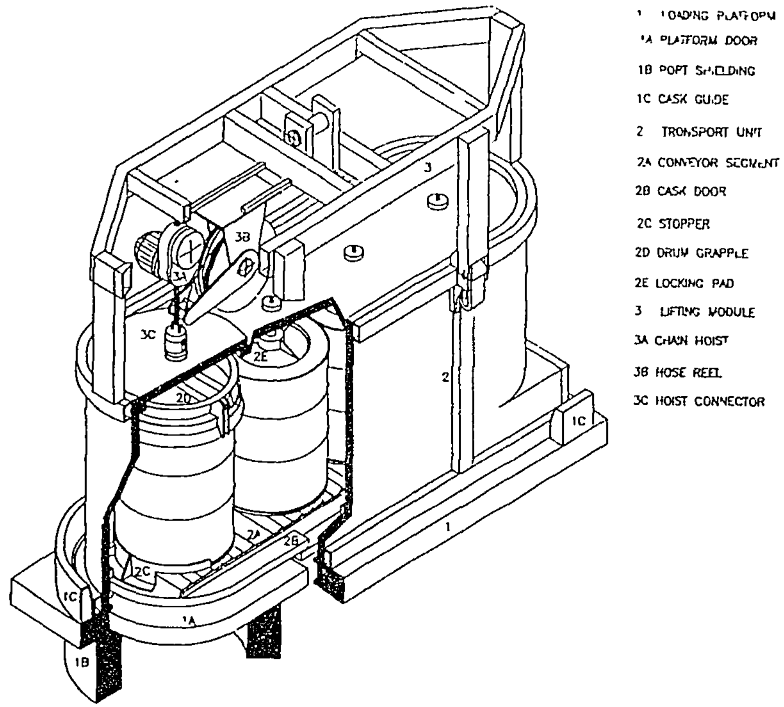
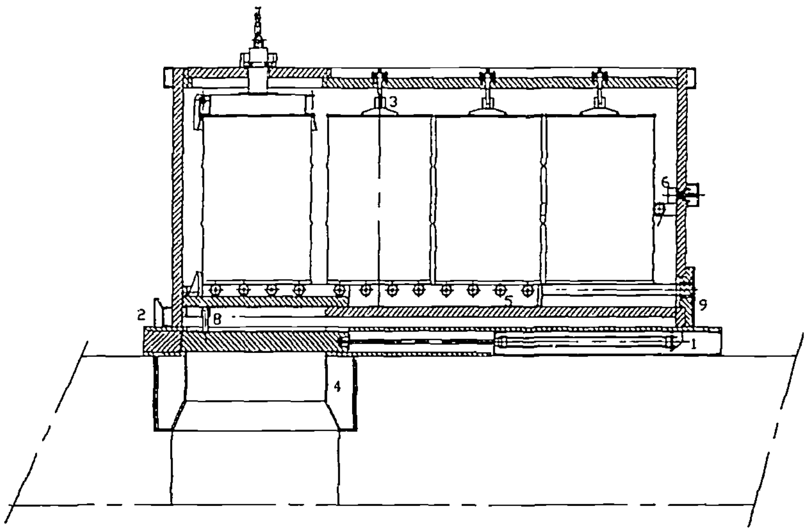
LOW LEVEL WASTE TRANSPORTATION SYSTEM

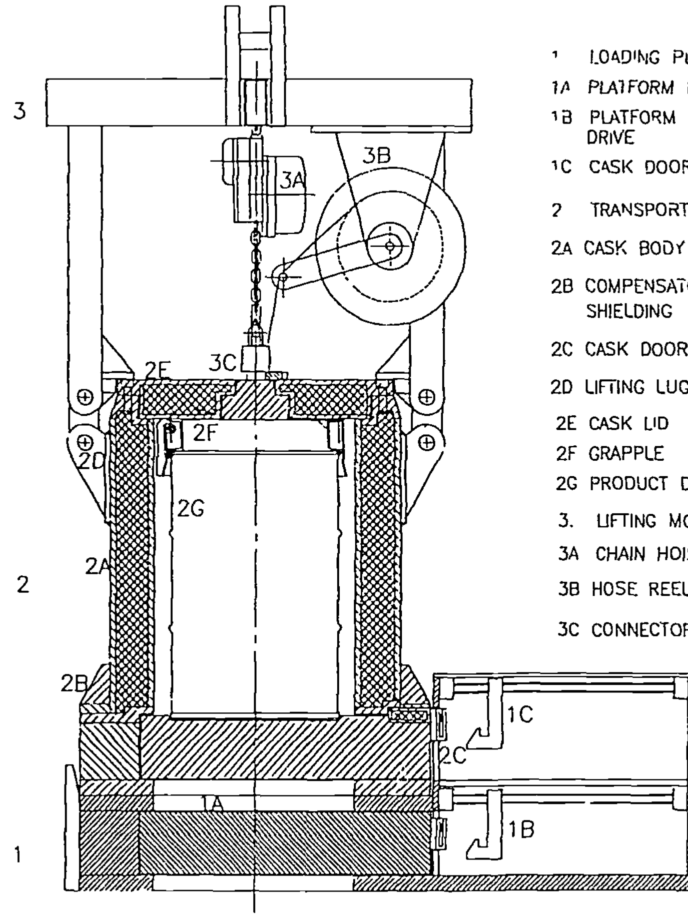


SECTIONAL ELEVATION OF SHIPPING CONTAINER FOR LOW LEVEL WASTE



1 DOOR DRIVE
2 GUIDE FOR CASK
3 LOCKING PAD
4. PORT SHIELDING
5. CONVEYOR SEGMENT
6 DRUM PUSHER
7 RAILINGS
8. DOOR COUPLER
9 DOOR LOCK

SHIPPING CONTAINER FOR INTERMEDIATE LEVEL WASTE



1 LOADING PLATFORM
1A PLATFORM DOOR
1B PLATFORM DOOR DRIVE
1C CASK DOOR DRIVE
2 TRANSPORT UNIT
2A CASK BODY
2B COMPENSATORY SHIELDING
2C CASK DOOR
2D LIFTING LUG
2E CASK LID
2F GRAPPLE
2G PRODUCT DRUM
3. LIFTING MODULE
3A CHAIN HOIST
3B HOSE REEL
3C CONNECTOR

XA9540102

DEVELOPMENT AND EVALUATION OF CONTAINERS FOR RADIOACTIVE WASTE DISPOSAL

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Abstract

The design and safety requirements for containers to be used for the final disposal and the interim storage of non-heat-generating waste are based in Germany on the criteria which are defined for the "Konrad" repository and for representative interim storage facilities, respectively.

These requirements will be compared with those defined by the transportation regulations for shipping casks used for the transport of low-active radioactive waste. A proposal will be submitted for package design requirements related to categories that cover all aspects of transport, interim storage and final disposal. Results of tests with a cubically shaped waste disposal container ("Konrad" Type VI) performed in 1991 and 1993 are presented in this context.

Because of the considerable administrative differences for waste disposal containers on one the hand and for transport containers on the other hand and in order to apply quality assurance properly, a competent authority design approval or, at least, a manufactor's design approval to be registered by that competent authority who is responsible for quality assurance and control is proposed for IP2, IP3 and Type A packages. Such a requirement should be implemented in the IAEA Regulations for the Safe Transport of Radioactive Materials.

1. INTRODUCTION

Two sites are selected in Germany for the final disposal of non-heat-generating radioactive waste, the former salt-mine Morsleben (Saxony-Anhalt) and the former iron-mine "Konrad" in Lower Saxony. Whereas the salt-mine in Morsleben has been already used in the former GDR as a repository for low-active radioactive waste, the "Konrad" repository is not yet licensed up to now.

The regulatory basis for the technical system of waste disposal in the "Konrad" repository and for the containers to be used in that system are guidelines from the German competent authorities (especially of the Federal Office for Radiation Protection - BfS), which are based on the more general instructions of the Atomic Energy Law /1/ and the Radiation Protection Regulation /2/.

The German requirements for waste disposal containers, especially the conditions for testing of those containers, are derived from the "Vorlaufige Endlagerbedingungen" (Preliminary conditions for final disposal) /3/ and put into design evaluation requirements in an administrative agreement between BfS and BAM on testing of containers for the final disposal of radioactive waste /4/. Special concerns of that agreement are, among others, the requirements for testing and quality assurance. Test conditions and responsible persons or organisations, respectively, for the tests are defined for two classes of containers (class I and class II). The test reports and certificates have to be handed in to the BfS, the federal office responsible for the construction and operation of the repository according to the Atomic Energy Law. The BAM is the responsible expert organisation for design evaluation.

Representative sites for the interim storage of non-heat-generating radioactive waste in Germany are the Waste Storage Facility in Gorleben and the EVU Storage Hall in Mitterteich. The requirements of containers to be stored in these facilities (/5/,/6/) are similar to those of the "Konrad" repository but there are defined three waste container categories. The categories I and II are similar to class I and class II containers of the "Konrad" repository. The category III has to fullfil the criteria of mechanical integrity in extreme accident situations like airplane crashs.

The requirements for transport containers for radioactive waste and the testing and administrative procedures for these containers are regulated by the transportation regulations which base in Germany on the IAEA Regulations for the Safe Transport of Radioactive Material /7/. According to the transportation regulations (i.e., the GGVS /8/, e.g., for road transports), packagings designed for the transport of low-active waste have to fullfil testing and administrative requirements which differ from those to be considered for interim storage and waste disposal containers.

The packagings to be used for the transport of low-specific-activity (LSA) materials are of the types IP (Industrial Packagings) and A. These types of packagings do not require a design approval by a competent authority and the testing requirements are lower in specific cases than those for containers used for final waste disposal. The inconsistency of the administrative and safety requirements will be especially obvious because the same containers will be used usually for the waste transport to the repository as well as for the interim storage and for the final waste disposal.

2. CONCEPTION OF THE INTERIM STORAGE AND FINAL DISPOSAL OF RADIOACTIVE WASTE IN GERMANY

Up to now, most of the non-heat-generating radioactive waste of the German nuclear power plants has been stored within storage capacities of these plants or other nuclear installations. Owing to the increasing waste quantities and the temporal uncertainty as getting a final disposal facility, it became necessary to provide external capacities for the interim storage of waste. The interim waste storage facility in Gorleben got its operating license in 1983 with supplements in 1987 and 1989. During the last years, the storage site company "BLG" has applied for an extension of the operating license of that storage facility. The application has been examined by the competent authority and their experts. The approval of additional waste and container types analogous to the conditions of the "Konrad" repository was an important aspect in this context.

The former iron-mine "Konrad" and the former salt-mine in Morsleben are provided for the final disposal of low-active waste. The Morsleben mine has been operated in the former GDR up to German reunification. The operating licence of this storage facility is now renewed by a legal decision of the competent Administrative Court of Justice. Because of that special situation, the final storage facility in Morsleben did hardly not influence the requirements for the waste disposal container design in Germany.

According to the present conception, "Konrad" will be used for about 40 years for the disposal of about 95 % of the radioactive waste produced in Germany. The overall volume of the depository is about 1.1 million m³. Ten storage fields are planned, each of the storage galleries in these fields has a cross section of about 40 m² with a width of the floor of about 7 m and a height of about 6 m.

The conception of the waste disposal containers to be used in the "Konrad" repository was primarily determined by the objective to create a system of standardized disposal containers which meet the safety and operational requirements of the repository , i.e., which allow a safe and frictionless handling.

Three basic container types were designed with a total number of 11 subtypes:

- o cylindric concrete containers, type I-II; gross volume 1.2-1.3 m³ ,
- o cylindric cast iron containers, type I-III; gross volume 0.7-1.3 m³,
- o cubical shaped containers, type I-VI; gross volume 3.9-10.9 m³.

The maximum gross weight including the waste is 20,000 kg for each container type. The safety requirements for the waste disposal containers and the status of performance testing of a cubical shaped container will be dealt with in section 3.3.

3. DESIGN AND SAFETY REQUIREMENTS FOR SHIPPING CASKS AND CONTAINERS FOR INTERIM STORAGE AND DISPOSAL FOR LOW-ACTIVE WASTE

3.1 DESIGN CRITERIA AND SAFETY REQUIREMENTS FOR TRANSPORT CONTAINERS

The IAEA Regulations for the Safe Transport of Radioactive Materials /7/ are the general basis for the German transportation regulations which have to be taken into consideration for low-active waste transports. These regulations define not only the design and testing criteria but also the qualitity assurance requirements. The basic requirements to be fulfilled by Industrial Packages (IP) are described in paras. 519 (IP-2) and 520 (IP-3) of the IAEA Regulations. If a freight container shall be used as an Industrial Package, the requirements of para. 523 have to be taken into account. The requirements of paras.524 to 540 have to be met for Type A packages.

The requirements for quality assurance for packages are defined in para. 209 of the IAEA Regulations in a general form. According to paras. 728, a quality assurance programme is a necessary precondition for a shipment approval certification by the competent authority which is necessary for Type IP and Type A packages.

The following test requirements shall be met by Industrial Packages and by Type A-Packages according to the IAEA and to the national transportation regulations (e.g., the GGVS /8/):

- o Water spray test: simulation of a rainfall of approximately 5 cm/h for min. 1 hour;
  required for IP-3 and Type A packages,

- o Free drop test: drop distance between 1.2 m (package mass < 5,000 kg) and 0.3 m (package mass > 15,000 kg);
  required for IP-2, IP-3 and Type A packages (additional tests have to be carried out for packages containing fissile materials or for packages with small masses),

- o Stacking test: compressive load of max ($5*m_p$, $1.3kPa*A_p$)
  with m_p = mass of package
  A_p = vertical projected area of package;
  required for IP-2, IP-3 and Type A packages,

- o Penetration test: 1 m drop of a bar of 3 cm in diameter and a mass of 6 kg;
  required for IP-3 and Type A packages.

For IP-3 and Type A packages, the water spray test is a preceding test succeeded by each of the free drop test, the stacking test and the penetration test. Paras. 619 and 620 of the IAEA Regulations describe the special conditions for the water spray test and for the test combinations.

The tests prescribed in ISO 1496/1-1978 /9/ are required if a freight container shall be used as an IP-2 or IP-3 packaging.

The test criteria for all tests and test combinations are the prevention of the loss or dispersal of the contents of the packages and the prevention of an increase of the radiation level of more than 20 % by a loss of the shielding integrity.

Quality assurance is regulated in Germany by national guidelines for packagings which require a competent authority design approval (TRV 006 /10/) as well as for packagings which only have to fulfill the test requirements of the IAEA Regulations, i.e. Type IP ard Type A packages (TRV 001 /11/). The TRV 001 applies exclusively to the quality control during the manufacture of packagings. Quality control during operation as well as periodic inspections fixed in the quality assurance programmes will be carried out in the responsibility of the users themselves.

3.2 DESIGN AND SAFETY REQUIREMENTS FOR INTERIM STORAGE CONTAINERS

The design and safety requirements for interim storage containers for the representative facility in Gorleben are very similar to those for the final disposal containers designed for the "Konrad" repository. It means that the Gorleben container categories I and II are widely equivalent to the "Konrad" container classes I and II, which are described in more detail in section 3.3. Only the Gorleben container category III defines extensive requirements for an accident safe container design. Those containers have, e.g., additionally to guarantee mechanical integrity after extreme accident events like airplane crashs.

3.3 DESIGN AND SAFETY REQUIREMENTS FOR WASTE DISPOSAL CONTAINERS

3.3.1 DESIGN AND SAFETY REQUIREMENTS

The actual design and safety requirements for final disposal containers are specified in the BfS-Report /3/ and an administrative agreement between BfS and BAM /4/, the responsible authorities for licensing and testing of such container designs in Germany.

First of all, any container design has to fullfil basic requirements relating to the dimensions, volume and mass because of the fundamental limitations arising from the repository conditions. Additionally, any container has to guarantee leaktightness and integrity during handling and stacking up to 6 m. Moreover, basic requirements exist relating to corrosion protection.

Each final disposal container can be related to one of the waste container classes I and II which must fullfil specific requirements depending on safety aspects as to the individual activity limits of the specified waste. The mechanical integrity of class I containers must be demonstrated up to an impact velocity of 4 m/s with the additional condition of a following accident fire at 800 °C for a period of 1 h. Those scenarios may not lead to an open burn down of the waste. For containers relating to class II must be demonstrated their mechanical integrity and a leakage rate $< 10^{-4}$ Pa.m^3/s after a 5 m drop onto a target representing the real facility foundation. Additionally, the container has to withstand an 1 h accident fire at 800 °C which may not lead to a critical activity release.

The proof of conformity of the container design with the safety and test requirements can be performed by

- o analogical reflections with similar problems,
- o testing of representative models,
- o calculations by verified analytical and numerical methods

or by testing of a prototype or test container manufactured according to the requirements of a quality assurance programme and loaded with a simulate of its radioactive contents. The following tests are provided:

- o 5 min stacking test with a test load of 15 or 30 t (depending on the type of container and the load conditions).

- o Lifting test with a stepwise increase of the load from the maximum gross mass of the container with its contents up to the twofold of that mass.

- o Drop test
  - for class II containers:
    5 m drop onto a real target specified by the requirements for the concrete to be used (These requirements shall consider the mechanical properties of the geological formations of the mine),
  - for class I containers:
    0.8 m drop onto the same target as for class II containers.

  The drop position of the container shall produce maximum forces and stresses in the components of the container (walls, weldings, screws, seals, etc.).

- o 1 h thermal test at a temperature of 800°C followed by a cooling period in air without forced flow conditions.

- o Leakage test with usual procedures.

The results of these tests will be summarized in a test report which have to be handed in to the competent authority.

The quality assurance measures for the waste disposal containers include:

- o a quality assurance manual which contains requirements of the organisation, execution and documentation of the quality assurance measures as well as the responsibilities,

- o a quality assurance programme distiguishing between two grades:
  - grade 1: class II container with specified leakage rate and components of the containment,
  - grade 2: all other components.

A final inspection after manufacturing and assembling of the containers is part of the quality control. Its results will be summarized in a test certificate. Moreover, the quality assurance programme includes measures of quality control during the operational phase up to the intercalation of the container in a repository.

The competent authority, the Bundesamt fur Strahlenschutz (BfS; Federal Office for Radiation Protection), states the applicability of the containment design for the final disposal of defined radioactive waste in a certificate (similar to an approval certificate) based upon the design evaluation expert's report issued by the BAM.

3.3.2 RESULTS OF PERFORMANCE TESTING OF CONTAINERS FOR FINAL DISPOSAL OF LOW-ACTIVE RADIOACTIVE WASTE

In the past, BAM performed several drop tests with a cubically shaped monolithic ductile cast iron (DCI; GGG 40) container "Konrad Type VI". The body of that container has wall thicknesses of 150 mm and 240 mm (at the lid side), respectively, and is sealed by a thick internal DCI lid and a thin external lid of mild steel. The lids are srewed on (the inner lid by 24 stainless steel bolts M36*110) and sealed with elastomere gaskets. The outer overall dimensions of the container are H*W*L = 1695*1595*1995 mm. The weights of the empty and filled container amount to 18,320 and 20,000 kg, respectively. For its handling, the container has in-casted ISO-corner fittings.

The following 5m drop tests had been performed in 1991 in a first test series:

o flat drop onto the bottom,
o flat drop onto a small container side,
o drop onto a small container edge.

The rigidity of the real underground in the repository was simulated in the tests by a concrete target of B45 quality /14/. Figure 1 gives an overall view about the most important test conditions.

Important results of the tests performed in 1991 are summarized in Table 1. More detailed information about the test conditions and results one can find in the PATRAM paper of Droste et al. /12/. Generally, the drop tests did not lead to a damage of the container structure and to a loss of its leaktightness. However, the analysis of the strain measurement during the impact duration had shown highly dynamic effects with vibrations of the container walls and locally high strain and stress levels up to nearly 87 % of linear-elastic yield strength. Because of these problems, BAM conducted in 1993 5 additional drop tests with another Type VI DCI Container of the same design. All these test were executed flat onto the container bottom but with drop distances of 0.8 m (2 tests), 3 m (1 test) and 5 m (2 tests). The drop test orientation had been identified before as the most critical one with respect to the real accident conditions. The instrumentation for measurement of strain levels had been improved considering the test results of the first test series. These drop tests demonstrated again the mechanical integrity of the container. The maximum measured tensile strain was about 2200 µm/m. Little plastic deformations occured in the highest stressed areas.

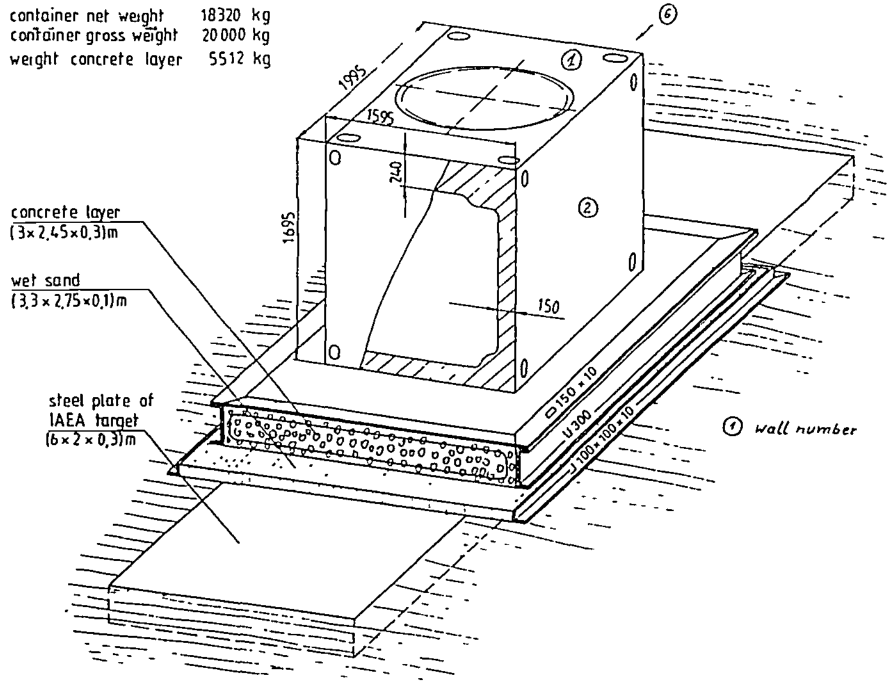

Figure 1: DCI Container and Concrete Target Used for Flat Drops

The problems resulting from the highly dynamic behaviour of the containers under drop test conditions had been identified and analysed reproducable in detail. In order to finalize the safety analysis, a concept considering aspects of a brittle fracture safe design according to IAEA-TECDOC 717 /15/ with a specification of fracture toughness values and criteria for non-destructive testing, or construction features to reduce the high stresses will be necessary.

4. PROPOSALS FOR THE HARMONIZATION OF THE SAFETY REQUIREMENTS AND ADMINISTRATIVE PROCEDURES FOR INTERIM STORAGE, WASTE DISPOSAL AND TRANSPORT CONTAINERS

The comparison of the design and safety requirements for transport, interim storage and final disposal containers for non-heat-generating waste (i.e. in terms of the transportation regulations: LSA materials) show considerable differences. In other words, different safety requirements have to be taken into account for packagings with the same radioactive contents.

Table 1: Drop Tests with a Cubically Shaped DCI Container, Type VI; Test Results

drop conditions	flat onto bottom	flat onto wall 6	onto the edge between walls 2 and 6
(1) primary impact duration, ms	4.5	5.0	15.0
(2) maximum deceleration, g	1260	1210	250
(3) maximum tension strain, $\mu m/m$)[1]	1050	1200	380
(4) maximum tension strain in screws, $\mu m/m$	4000)[2]		500
(5) concrete penetration, mm	about 3	about 3	50...70
(6) leakage rate after the drop, mbar.l/s	$<1.2*10^{-7}$	$<6.3*10^{-8}$	$<5*10^{-6}$

)[1] linear-elastic yield strain is about 1600 $\mu m/m$
)[2] plastic deformation

Generally, the considered accident impacts for containers used for interim storage and final disposal are higher than those for pure transport containers.

Moreover, the comparison shows

- o Design approval certificates issued by competent authorities are obligatory for interim storage and waste disposal containers. Contrary to it, Type IP and Type A packages for the transport of low-active materials do not require any design approval certificate (neither by the manufacturer nor by a competent authority).
- o The requirements for indepedently approved quality assurance programmes are more detailled for interim storage and waste disposal containers than for transport containers. The certificate of applicability issued in those cases by the competent authority includes concrete terms of quality assurance measures for manufacture and operation. Contrary to it, quality assurance programmes for Type IP and Type A packagings are indeed necessary to get a shipment approval certificate but limited in Germany to the manufacturer's responsibility.

(The language of the IAEA Regulations /7/ is: "Where competent authority approval for .... shipment is required, such an approval shall take into account and be contingent upon the adaquacy of a quality assurance programme.")

Because of all differences in the test requirements and licensing procedures for transport packagings and interim storage and final disposal containers used for low-active waste, in Germany are deliberated possibilities to harmonize the requirements in order to avoid an unnecessary repetition of design testing procedures. The actual proposal defines four different Design Groups with different combinations of requirements summarized in Table 2.

The testing procedures for any Design Group shall be elaborated in a way that the strictest requirements of the intended purposes will be considered. That implies a design approval also for Type IP2, IP3 and Type A packages.

Germany has therefore proposed in the frame of the revision process for the IAEA transport regulations /7/ the introduction of the design approval of Industrial Packages (IP2, IP3) and Type A packages. One reason for the proposal is that the control of the conformity of the design of those packages with the regulations and the control of the inforcement of quality assurance programmes by the competent authorities is only a permissive provision up to now but not a procedure prescribed imperatively by the regulations. Moreover, the contents of quality assurance programmes are not clear enough defined for package designs which must not be approved by a competent authority.

In order to improve the situation and taking into account the problems arising from the different design and safety requirements as well as from the requirements of quality assurance for transport, interim storage and final disposal containers, the design approval for Type IP2-, IP3-, and Type A packages by a competent authority or, at least, a manufacturers design approval to be registered by the competent authority seems to be a reasonable basis to apply quality assurance in a more proper form. Such an approval would also facilitate the certification process for containers which shall be used for interim storage and/or the final disposal of low-active waste.

Table 2: Proposal for Harmonized Design Groups

Harmonized Design Group	Tranportation: Package Type	Interim Storage: Container Category	Final Disposal: Container Class
1	IP1, IP2	I	I
2	Type A	II	II
3	Type B(U)	II	II
4	Type B(U)	III	II

An appropriate guideline for design safety reports and design approval for Type IP and Type A packages is given in /13/.

5. SUMMARY, CONCLUSIONS

The paper gives an overall view about the design and safety requirements for packagings used for transport, interim storage and for final disposal of low-active, i.e. non-heat-generating waste. The presented requirements for interim storage and final disposal containers are primarily those which are derived in Germany for planned interim storage facilities and repositories.

A proposal has been elaborated in Germany for the harmonization of the design and safety requirements of packagings used for transport, interim storage and final disposal, respectively. The IAEA should give some guidance in developping harmonized packaging requirements to cover safety aspects of transport, interim storage and final disposal in order to prevent troublesome national experiences with the existing discrepancies.

As an urgent need for compliance assurance in order to apply quality assurance for packagings of the Types IP2, IP3 and A more properly, a competent authority design approval or, at least a manufacturer's design approval that has to be registered by the competent authority for quality assurance and control should be implemented in the IAEA transport regulations.

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SOME ASPECTS REGARDING THE QUALIFICATIONS TESTS OF PACKAGES USED FOR TRANSPORT AND STORAGE OF RADIOACTIVE WASTE (LOW ACTIVITY) IN INR PITESTI

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Abstract

SOME ASPECTS REGARDING THE QUALIFICATIONS TESTS OF PACKAGES USED FOR TRANSPORT AND STORAGE OF RADIOACTIVE WASTE (LOW ACTIVITY) IN INR PITESTI.

Radioactive wastes generated by TRIGA INR research reactor are packaged according to the national and international standards and to the IAEA Regulations for the Safe Transport of Radioactive Materials and Advisory Material for the Application of the IAEA Transport Regulations.

The technology for packaging and treatment of Radioactive wastes used in our institute can be applied, in perspective, also at the Nuclear Power Plant Cernavoda, after commissioning.

This paper describes the qualifications tests (type tests) for packages used for transport and storage (for a long period of about 30 years) of radioactive wastes (low activity, up to 0.5068×10^{10} Bq/drum, 0.164 Ci/drum, respectively).

The package used is a drum (see Fig. 1 and 1a), manufactured by Romanian industry (according to the national standard 7683-79) of 1 mm thick mild steel with the following dimensions :
height: 915 $\pm$ 10 mm, diameter: 600 $\pm$ 5 mm, volume: 220 liters, approximately.

There are presented the type tests carried-out, e.g. compression, penetration, free fall, lixiviation, safety in *utilizing (biological protection), checking* of *chemical* and mechanical characteristics and the effect of the product on the environment, the results, interpretation and conclusions. The performing of the above mentioned tests and other additional ones, the results obtained, prove that our technology for treatment and packaging of radioactive wastes is in accordance with IAEA Regulations in the field.

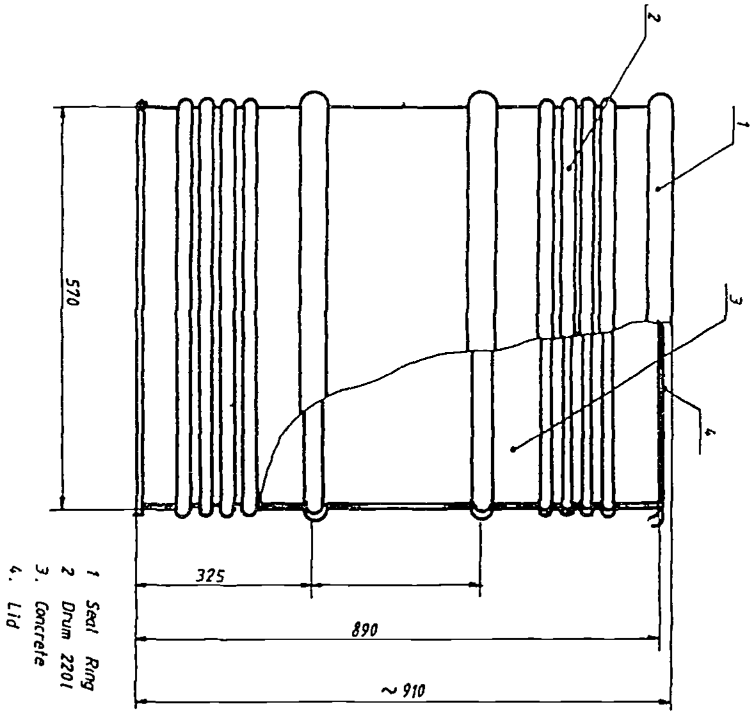

FIG.1. The drum used for radioactive wastes transport

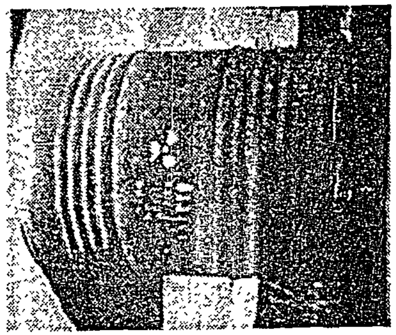

FIG. 1a. The drum used for radioactive wastes transport and storage

2. INTRODUCTION

In order to see how good are our packages for storage and transport of radioactive wastes (low activity) and to meet the requirements of IAEA Regulations and our standards in the field, we developed laboratory type tests (for type A packages) which provided us with a means of predicting the field performance of the package. Having no other experience in the field, our choice was to reproduce accurately the representative tests provided by our standards and IAEA Regulations, which allowed us to measure the capability of the package to withstand the various environmental stresses to which it is likely to be subjected, taken into consideration that the packages of radioactive materials must have the quality of retaining their contents under the routine or accidental transport conditions of being thrown around, beaten, rattled, poked, kicked, squashed and soaked. The radioactive wastes packages must guarantee all requirements for the protection of human beings and the environment. Compliance with the requirements is demonstrated by means of type tests. The type test requirements are then compulsory minimum in the specifications for the manufacture of all standards.

We tried to prove that there is a direct correlation between the result of laboratory type tests carried out and the accidental storage and transport radioactive wastes conditions, including the normal everyday environment.

The wastes to be packaged are generated by our TRIGA research reactor, our post-irradiation laboratory and by radiochemistry activities, such as: metallic pieces, protection equipments, filters and glasses which cannot be decontaminated, plastics, individual protection equipment, used ion exchangers from TRIGA, organic liquid radioactive wastes and used filters, etc.

Shortly, the treatment technology(e.g. for used-ion exchangers wastes) consists of the following: wastes are prepared and treated after a special technological process and included into a resin mixed with bitumen (into a 2:3 ratio). This mixture is introduced in a small drum, of about 80 liters capacity, (see Fig. 2); this mixture is introduced into the 220 liters drum, having on the bottom a layer of approximately 150 mm concrete (see Fig. 3). According to the treating technology, the package so prepared, is filled with concrete. The drum is closed with a lid and a seal ring.

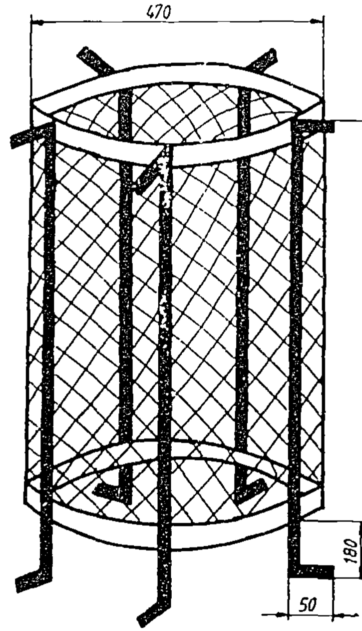

FIG.2. The 80 l drum used for wastes matrix

In this way the radioactive waste package prototype is ready to be subjected to laboratory qualifications (type) tests which are described in the following paragraph.

3. QUALIFICATIONS (TYPE) TESTS OF PACKAGES USED FOR TRANSPORT AND STORAGE OF RADIOACTIVE WASTES (LOW ACTIVITY) IN INR PITESTI

For carrying out the qualifications tests for radioactive waste package, in accordance with Technical Specifications (Standard 130/1990) and meeting the IAEA Regulations concerning the number of specimens subjected to the tests (taking into consideration the usage, availability of packaging and cost of an individual package, the materials and methods of construction and the actual test results together with the low-use factor), only one package (drum) was tested for every kind of radioactive waste.

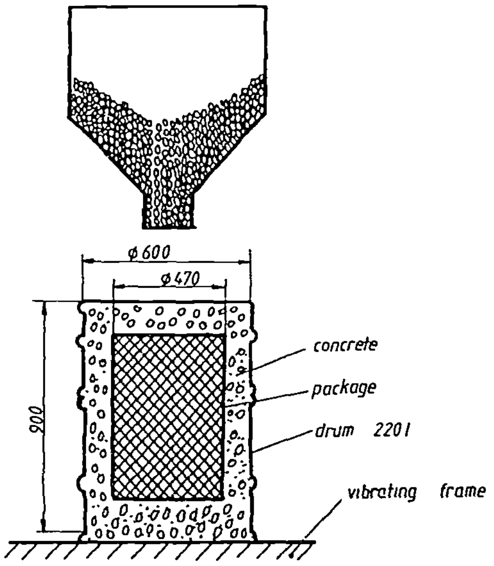

FIG.3. The 220 l drum filling with concrete

It is to be noted that the contents of specimen for testing is the real radioactive waste intended to be transported and stored, so the contents is not simulated (filled with sand, water, or others).

Before testing, the specimen (package) has been inspected and examined and we did not find or recorded any faults or any damages due to defects in construction or during preparing for testing (according to the treating technology), corrosion, accidental deteriorations or other distortion of features or divergences from specifications or the drawings. Referring to the supply of material and intermediate products, the checking for conformity with the specifications upon receipt has been made. The protection against corrosion (painting, oil films, etc.) has been checked.

Also, before the beginning of the qualifications tests, the specimen has been examined from the point of view of non-fixed radioactive contamination on external surface, based on the wipe method through swabbing, with alcohol. The measured activity was smaller than 1 Bq (allowed limit: smaller than 185 Bq).

3.1. Test facilities

All standard type tests for waste packages have been carried out internally by Reliability and Testing Laboratory from the Institute for Nuclear Research, Pitesti. For this reason it has developed all the facilities for testing and quality control.

Devices used for carrying out the type tests are provided by Standard 130/1990 and not only.

3.2. Penetration test

The specimen under qualification testing (the drum) was placed on a rigid surface (concrete), perfectly horizontal and with a negligible movement during testing.

Before beginning testing, a water-spray test has been done, lasting about 1 hr. The amount of water per unit of ground area was about equivalent to a rainfall determined rate of 5.1 cm per hour (5 cm per hour provided by IAEA Regulations), at an angle of approximately 45^{o} from horizontal and uniformly distributed, as in a rainfall, to simulate the most severe conditions for the features under investigation.

The penetration test was performed two hours after the water spray test was carried-out and not before the second examination of the specimen, as a result of the water spray test.

The tool used to perform the test was a 6 kgs. bar, made of steel and having a hemispherical end of 3.2 cm in diameter. The drop height was 1 m (according to our Standard 130/1990 and to the IAEA Regulations).

After the test, the specimen was visually inspected and no serious damages have been recorded (see Fig. 4), no perforation of mild steel was observed, neither any deformation of the bar used for the test.

The test was considered passed.

3.3 Compression test

Before performing the compression test, according to the Standard 130/1990 and IAEA Regulations, the water spray test was performed, with 1 hr. duration, and, after 2 hr., the compression test was performed. It is to be noted that the weight of the drum containing the treated radioactive waste, is

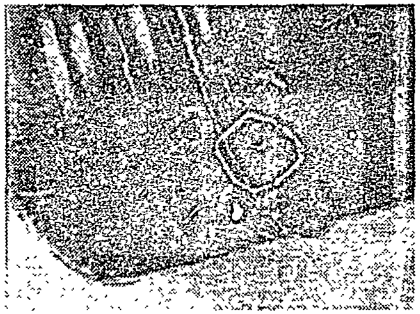

FIG. 4. The penetration test

between 450 - 500 Kg. Thus, for 24 hrs., the load for compression was 5 times with respect to the weight of the specimen, 2300 Kg, in our case. The load was applied uniformly, on the opposite sides of the specimen under testing (see Fig. 5), one of the sides being the base on which the drum is normally standing.

After 24 hrs, a visual inspection has been made and no visible deformations have been observed. The integrity of the radioactive waste content has been maintained.

The test was considered successfully passed.

FIG. 5. The compression test

3.4. Free drop test

This test was performed after the 1 hr. water spray test. Two hours from the end of the water spray test, the specimen was inspected (visual inspection) and no modifications have been observed.

The free drop test was performed according to the IAEA Regulations and to the Standard 130/1990 statements. The drop height was 1.2 meters, being measured from the lowest part of the specimen to the upper surface of the target (falling surface). See Fig. 6.

The drum (the specimen under testing) was subject to a visual inspection and no serious damages have been observed, that might affect the integrity of the package and the content.

We consider that the test was passed in a successfully way.

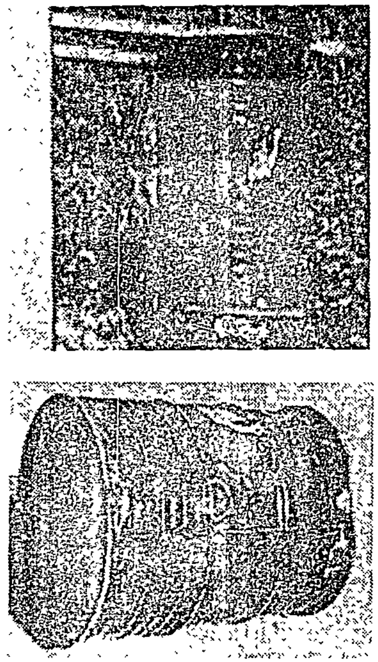

FIG. 6. Free drop test

3.5. *The verification of* the compression resistance of the concrete included in the content of the drum

This test was performed as per Standard 1275/70. According to our calculations, the compression resistance of the concrete must be at least 5 MPa (50 Kgf/cm^2).

The mechanical resistance at compression determined was 35.2 MPa, more than the above pass/fail criterion.

This test was considered successfully passed.

3.6. The checking of biological protection

According to our standard, the equivalent dose rate at the wall of the drum must be of maximum 2 mSv/h (max. 200 mRem/h). Using a radiation debitmeter we measured the dose rate, in different phases of qualifications testing period. The determined equivalent dose rate was 0.5 mSv/h (50 mrem/h). We consider that the biological protection has been proved and there is no danger in package handling from this point of view.

3.7 Lixiviation test

In order to check the conditions regarding the influence of the package containing radioactive waste on the environment, we performed the lixiviation test based on the method presented in ISO/TC - 35/6961 and in accordance with the IAEA Recommendations.

In the following I present the lixiviation test made on used ion exchangers from our TRIGA reactor.

The leaching tests have been carried out on the samples of bitumen-used ion exchangers, taking into consideration that the layer of concrete contains no radioactive material. On the other hand, one must keep in mind that in extreme cases accidents or degradations are possible to occur on a long term prospect.

A set of samples was subjected to lixiviation test for a long period (620 days) in order to analyze the resistance vs. time at the action of water on the block bitumen-used ion exchangers. Another set of samples, taken during technological treatment of radioactive waste was also subjected to the leaching test, but for a short time (35 days) in order to determine the loss of radionuclides in real conditions related to the conditioning process of radioactive waste. The leaching laboratory tests have been carried out in controlled conditions, according to the ISO/TC-85/6961.

The analysis of radioisotopes *from* the waters in which the samples were introduced, have been done by means of a MCA Canberra chain. The values of leaching rates (for the samples subjected to the long test) ranged between the following limits:

- 10^{-5} up to 10^{-6} g/cm^2 per day for Cs - 137;
- 10^{-6} up to 10^{-8} g/cm^2 per day for Co - 60 ;
- 10^{-6} up to 10^{-8} g/cm^2 per day for the global rate.

These values *show a* high resistance of the bitumen-ion exchangers block vs. water action.

The standard 130/1990 provides a leaching rate smaller than 10^{-3} cm/day (approximately 10^{-3} g/cm^2 per day).

The values of leaching rates for the samples subjected to the 35 days test are :

- 10^{-5} up to 10^{-7} g/cm^2 per day for Co-60
- 10^{-5} up to 10^{-7} g/cm^2 per day for Cs-137

It is to be noted that many times the loss of radionuclides was under the detection limit of the MCA Canberra chain.

The obtained results confirm that the specimen met the requirements of Standard 130/190 and IAEA Regulations in the field. There is no risk for contamination of the environment, even in the case of serious damage of the packages containing low active wastes, neither during transport or storage.

The density determinations have been made also in order to determine the homogeneity of the bitumen-used ion exchangers block.

The results obtained show a very good homogeneity of the product, respectively the mixture bitumen-used ion changes.

4. Conclusions

Based on our experience related to the qualifications of type A radioactive wastes packages, we tried to clarify some of the specific questions which were identified during carried-out the type tests, before starting the certification programme. These questions were regarding of the followings: which is the loss of contents, how many packages of any type need to be testes, prevention of any significant increase in the radiation levels recorded at the external surfaces, how should IAEA

Regulations on General Design Requirement and Additional Design Requirement for type A packages be applied with respect type A packaging evaluation.

For the moment, because only one package has been tested for every type of waste, we think that our result are satisfactorily, even that these met the requirements of national standard and IAEA Rgulations For The Safe Transport of Radioactive Materials.

In the same time we need to complete the type tests with others, based on the experience of other Member States of IAEA Vienna(e.g. reduced pressure) in the field.

We consider that we shall complete the experience related to the type test for radioactive wastes packages in the nearly future when we shall test type B packages.

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PACKAGING DESIGN AND QUALIFICATION: THE EXPERIENCE OF THE CENTRO DE DESENVOLVIMENTO DA TECNOLOGIA NUCLEAR/ COMISSÃO NACIONAL DE ENERGIA NUCLEAR

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Abstract

Since 1982 the Centro de Desenvolvimento da Tecnologia Nuclear, the Nuclear Technology Development Center, has been designing, testing and qualifying radioactive materials packagings These packagings are used for the transport of radioisotopes and disposal of spent sealed sources, wastes generated in the nuclear fuel cycle and the wastes produced in the radiological accident occurred in the city of Goiânia For radioactive tracers and medical/industrial radioisotopes, the used packagings are cardboard and wood boxes, while the spent sealed sources are conditioned preferably in metal drums containing lead shielding and a gas absorber material To condition and transport the wastes from the various nuclear cycle activities, metal drums and boxes are used in Brazil For the higher active wastes from the nuclear power plant Angra I, a metallic drum in a concrete overpack is used. The wastes generated in the accident were first conditioned in the readily available packagings, like commercial drums, square boxes and large shipping containers Later on, more appropriate packagings were designed by the CDTN staff a metal cylindrical container for conditioning the broken ^{137}Cs source, a concrete overpack for 14 drums and a metal cylindrical box for 14 drums In order to evaluate the durability of commercial drums used for waste conditioning, CDTN has performed a program since 1983 In the first part of this study two drum types, with different internal/external coating, were stored inside a hall and in the open After a period of 8 years, one of them had a failure in the lid, thus allowing water penetration In the second phase the drums were sectioned and representative sections of their body, in contact with pure grout or with cemented simulated wastes were stored in a laboratory and in the open The results obtained point out that the drums are not adequate for an outdoor storage and that their internal coating has a poor resistance to the cemented wastes

1. Introduction

The low and intermediate radioactive wastes produced in Brazil comprises spent sealed sources for medical and industrial uses, residues from the nuclear power plant Angra I and from the other nuclear fuel cycle facilities and the wastes arisen from the radiological accident in Goiânia

The Centro de Desenvolvimento da Tecnologia Nuclear (Nuclear Technology Development Center) - CDTN -, the Brazilian official packaging testing institute, since 1982 has been designing, testing and qualifying low and intermediate level material packagings The available facilities and equipments are· a platform and a 3 ton crane for the drop test, small angle structures for water spray and penetration tests, concrete blocks for the stacking test, a hydraulic/pneumatic circuit for the containment test and radiation monitors for the shielding test

The CDTN experience is presented here The tested packagings are classified into three groups medical and industrial radioisotopes packagings, packagings used in the nuclear fuel cycle and packagings for Goiânia's radiological accident wastes.

The drum corrosion studies being performed at CDTN are discussed too. The program's main purpose is to evaluate the long-term behavior of the commercial drums under different storage conditions and the influence of the cemented waste upon the drum's internal coating

2. Medical and industrial radioisotopes packagings

Four different types of packagings for radioisotopes transportation were tested: cardboard and wood boxes, metallic drums and lead cylinders

The cardboard boxes are designed for the transportation of radioisotopes for medical use, as Tc-99 and In-113, with activities up to 1 Ci With few variations, these packagings consist of an outer cardboard box, styrofoam (polystyrene foam) spacers and a lead shielding containing the solution flask with connections

As type A packagings, they were submitted to the prescribed tests water spray, free drop, stacking and penetration test, followed by containment and shielding verification. Due to their relatively low specific weight, neither the drop test nor the stacking test induced significative damages. The water spray test, by wetting the cardboard, and the penetration test, by striking the packaging on its supposed weakest point were the critical ones.

In order to approve these packagings in the tests, slight modifications were introduced: application of impermeable adhesive tapes on the box edges, containment system improvement

Type A wood packagings for tracers transportation were also tested. They consist of a steel frame reinforced square wooden box enclosing a cylindrical lead shielding which, in turn, houses a screw taped Plexiglas flask with the radioactive material. As these packagings are very robust, they were approved in the prescribed tests

The third radioisotope packaging group consists of metallic drums for spent Ra-needles transportation and disposal, available in two versions.

In the small one the outer unit is a 20 l drum surrounding a lead shielding ($\Phi 160x200mm$), which acts also as the containment system The shielding is held in position by means of wooden pieces The packaging weight is about 57 kg

The other version consists of a 200 l drum surrounding a $\Phi 200x610mm$ cylindrical shielding, which is fixed by fiberboard disks The space between the shielding and the outer drum is filled with an activated carbon/granulated styrofoam mixture which acts as Radon absorber The total weight is about 135 kg

Both packagings were approved since their containment system and shielding were not damaged in the tests

The outer drums (20 l and 200 l) suffered only minor dents at the drop and penetration tests impacting point

The last radioisotope packaging group is a type A packaging for a ^{137}Cs sealed source used for industrial gamagraphy

It consists basically of

- a cylindrical lead shielding covered with a stainless steel plate with a central hole;
- a source housing inserted in the shielding cavity;
- a radiation obturator,
- a trigger to activate the obturator

This packaging was presented in three different sizes weighing respectively 24 kg, 44 kg and 88 kg.

In view of the test results, the packaging was approved with the recommendation that the applicant establishes a Quality Assurance Program encompassing all packaging manufacturing steps, in order to ensure conformity to the normalization requirements.

3. Packaging used in the nuclear fuel cycle

As a result of the operation of the different fuel cycle facilities, a great amount of nuclear materials and radioactive wastes has to be transported and disposed. In order to get adequate and safe packaging, many facilities operators asked CDTN to test their packagings.

FURNAS Centrais Elétricas S/A, the operator of Angra I - the Brazilian nuclear power plant - uses a reinforced 200 l metal drum to condition low and intermediate wastes (ion-exchange resins, filter cartridges and evaporator concentrates). The drum for resins has an internal 5 cm concrete shielding. It contains a perforated metal cage internally covered by a wire net, the intermediate space being filled with a cement/sand mixture. The resin is poured into the cage and the water contained in it drains through the holes and reacts with the mixture

For the filter drum, a concrete block is internally moulded with a central cavity to hold the filter A concrete disk is used between the block and the drum lid as shielding and to fix the cartridge.

The drum for evaporator concentrates has an injector positioned along its symmetry axis and is completely filled with a cement/vermiculite mixture The waste is fed through the injector and reacts with the mixture

As a maximum allowable content dispersion is not defined quantitatively neither in the IAEA's Regulations for the Safe Transport of Radioactive Material nor in the Brazilian transport regulations, a leakage was permitted in such an amount that would not be hazardous from the radiological point of view. Thus, the escape of a fraction of 10^{-3} of the content was allowed, corresponding to the maximum activity for excepted packagings transporting solid material

Another tested nuclear power plant waste packaging was a 1 3 m^3 square metal box intended for incompressible wastes. The wastes - surface contaminated metallic pieces - are immobilized in an expanded grout matrix The gross weight is around 1,200 kg. This packaging was classified as Industrial Package Type 2 (IP-2) being thus submitted to the free drop and stacking tests

To condition higher active wastes, FURNAS intends to use a metallic drum inside a concrete overpack The packaging dimensions are 1 07 m diameter and 1.5 m height and its weight (loaded) is about 3t. It was observed that only the drop test causes some damage in the packaging cracks along its lateral surface, bottom and top Although some cracks spread out until the internal cavity the drum integrity was not affected - as verified through concrete boring

The operator of the Brazilian uranium mining near the city of Poços de Caldas - in the Southeastern region of Brazil - uses 200 l drums to condition and transport the yellow cake exported to nuclear fuel producing countries. Through the tests performed at CDTN it was verified that the use of a resistant plastic bag enclosing the yellow cake increases the safety during the transportation without significant extra costs.

The drums used to condition low level compressible wastes (paper, clothes, etc.) produced during the fuel element assembling - average weight 130 kg - were successfully submitted to the drop and the stacking tests, suffering merely minor dents at the impact region - the closure bolt

4. Packagings for Goiânia's accident wastes

In September 1987 a severe radiological accident took place in Goiânia, a state capital in the central Brazilian highlands. A radiotherapy equipment was stolen and opened, its radioactive content - 100 g of ^{137}Cs - being widely spread over several points of the city by means of human and biotic transportation. As a consequence, four persons died due to ingestion or exposure to the radioactive material and several suffered physical damages As a preventing measure, a great amount of contaminated domestic animals had to be scarified and some houses - hopelessly contaminated - demolished The resulting wastes have an approximate volume of 3,500 m^3, among debris, furniture, animal carcasses, hospital material, contaminated soil and trees, etc The packagings used to condition these wastes are described below

Early during the accident readily available packagings were used, like commercial drums, metal square boxes as used by FURNAS Centrais Elétricas (described above) and large shipping containers For higher active wastes, one-drum concrete overpack was used

Later on, in view of the decision to dispose the wastes at a trench and at a near surface facility - to be built in a near future -, other packagings were designed by the CDTN staff

- a 4 m^3 metal cylindrical container for conditioning the broken ^{137}Cs source;
- a concrete overpack for 14 drums;
- a metal cylindrical box for 14 drums.

These packagings are shown in Figures 1 to 4

The recovered portion of the ^{137}Cs source was immobilized in the interior of concrete canalization pipes The resulting block was conditioned in a cylindrical metal container with the following characteristics.

- internal diameter 1 8 m,
- height 1 6m;
- lateral plate width 3/16" (4.76 mm);
- bottom plate width 9/16" (14.28 mm);
- net weight 700 kg;
- filling mortar weight 3,350 kg;
- total weight 9,500 kg

The lifting system consists of two holders diametrically laid. The lid is attached to the mortar prior to its curing process by means of anchor hooks

According to the established waste management program the drums were classified in three groups. Those containing wastes with zero-year decay time will remain in the interim storage site and in the future disposed in a trench. Those containing wastes with decay time up to 150 years were reconditioned directly in concrete overpacks, while the others were first put into metal containers and these set in a concrete liner.

The concrete overpacks were designed according to the following assumptions:

- conditioning of wastes with decay time up to 150 years,
- 14 drums capacity,
- a four-year period in the open prior to transfer to the final repository;
- a two-level maximum stacking;
- filling material: mortar with bentonite (specific weight 1.7 g/cm^3),
- packaging data·
    - external diameter. 2 34 m;
    - internal diameter. 2.04 m;
    - external height· 2.20 m;
    - cavity height: 1.90 m;
    - net weight. 7,200 kg;
    - number of lifting holders: 4 (four);
    - lid and bottom with structural feature;
    - impermeable internal coating,
    - material: reinforced concrete with additive

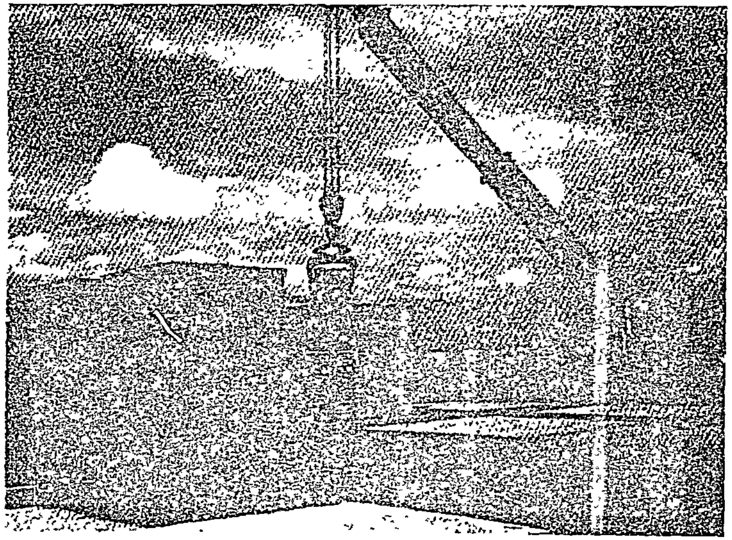

FIG 1. Metal drums and boxes for Goiânia's wastes

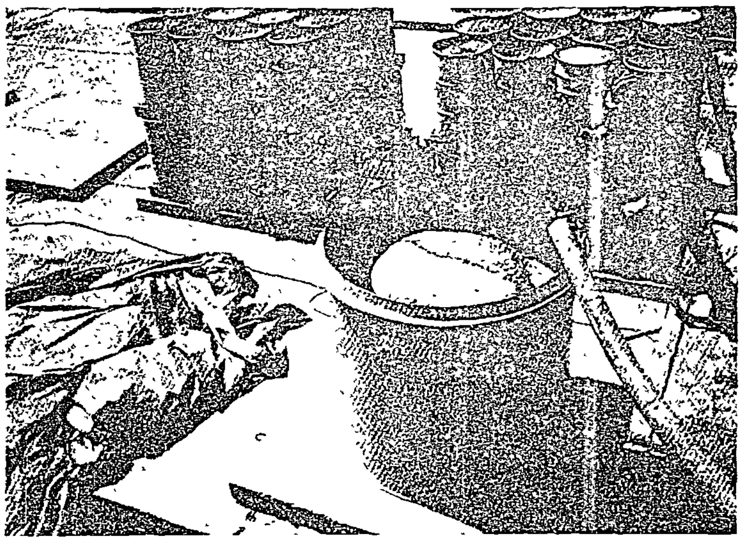

FIG 2. Packaging for the recovered portion of the ^{137}Cs source

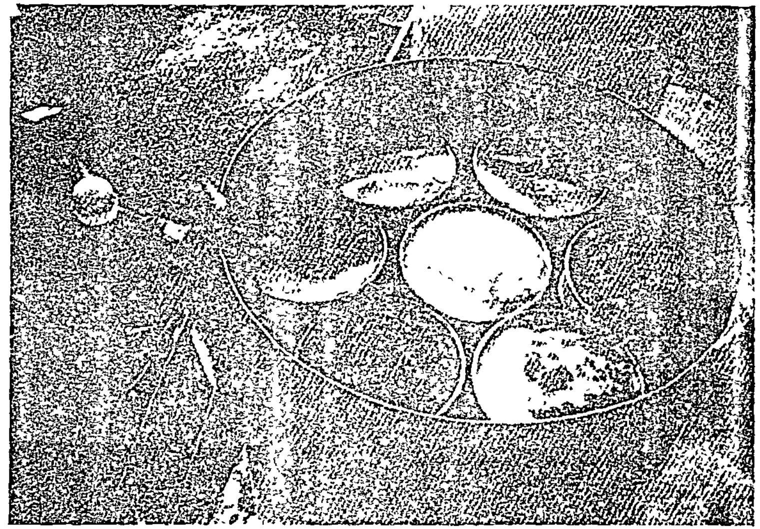

FIG 3 Metal boxes for 14 drums

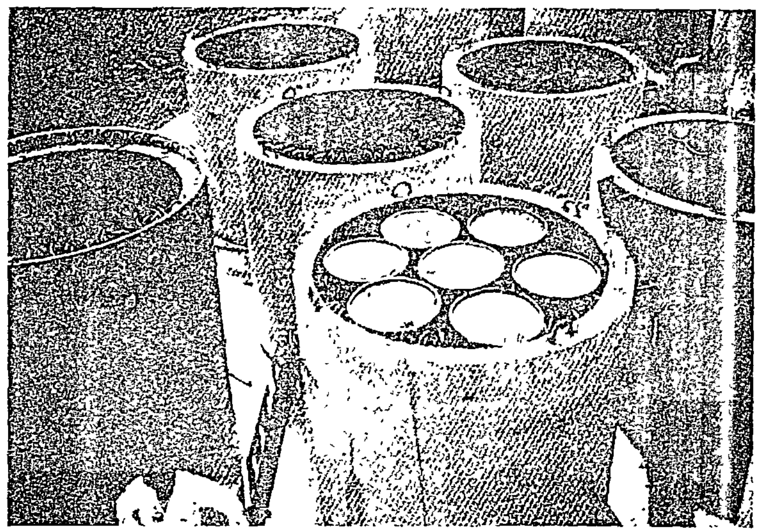

FIG 4 Concrete overpack for 14 drums

The packaging data for the wastes with decay time longer than 150 years are

- metal container
  - external diameter 1 9 m,
  - height· 2 m,
  - lateral plate thickness· 1/4" (6 35 mm);
  - bottom plate thickness 9/16" (14 28 mm);
  - material ASTM A-36 steel,
  - net weight 1,030 kg;
  - gross weight. 10,800 kg

- concrete liner·
  - internal diameter· 2 06 m;
  - internal height. 2.69 m.

The drum capacity, intermediate period and maximum stacking are the same as described above.

5. Corrosion studies in metallic drums

In order to evaluate the durability of commercial drums used for low and intermediate level wastes conditioning, CDTN has performed a program since 1983.

In the first part of this study, unsectioned drums containing both compactable or cemented simulated wastes were stored inside a hall and in the open. Two drum types produced by the major Brazilian manufacturers were used:

- type 1: external coating - synthetic enamel
  internal surface - epoxi-phenolic coating
- type 2: external coating - synthetic enamel
  internal surface - phosphated, without coating

A total of sixteen drums were tested.

In order to approach the corrosion phenomena under a quantitative perspective, coating thickness measurements and adhesion by tape test measuring were performed Besides, drums sections were submitted to the following tests· exposure to 100 percent relative humidity, exposure to SO_2 atmosphere, salt spray testing and test for abrasion resistance of coating

This phase begun in 1983 with a predicted duration of five years. After this period it was verified that the drums stored outside presented large corrosion areas, mainly at the lid surface, due to its design that allows rain water collection. On the other hand, the drums inside the hall were in good conditions Three of them were opened and submitted to the mentioned metallurgical tests, with similar results as before.

In a visual inspection performed in 1992 the existence of holes at the lid surface was detected through which air bubbles from the drum interior emerged through accumulated rain water layer

In view of this, this part of the corrosion program was concluded, under the consideration that its main purpose had been reached The results obtained show that the tested drums are not suitable for an open storage, since they failed after an 8-years storage period. As the failure occurred in the lid, a possible solution would be the replacement of the original lid by a "mushroom" type one, as used, for example, in Germany

In the second phase of the study only the type 1 drum was tested. The purpose is to evaluate the long-term influence of both the environment - externally - and the waste - internally - upon the drums Unlike the previous study the drums were sectioned Representative samples of their body - upper and lower borders, lateral wall, welding region and reinforcement hoop - with their internal side in contact with pure grout or with cemented simulated wastes were stored in a CDTN laboratory and in the open. A solution of boric acid is used to simulate the waste, as this substance is present in the PWR wastes

A total of 150 samples were prepared and photographed As the program duration is 5 years, 30 samples are being taken out per year and visually inspected. The performing of metallurgical tests is not feasible due to the severe degradation of the coating.

Two sets of samples have already been taken out in March/92 and March/93. The results point out that the internal painting (epoxi-phenolic) is not adequate as cemented waste packaging coating due to its poor resistance to this type of waste.

AN INDUSTRY STANDARD FOR INDUSTRIAL PACKAGES

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Abstract

This paper proposes a safe, consistent and cost-effective approach to the qualification and testing of industrial packages in accordance with the IAEA Regulations. It represents the views of the United Kingdom nuclear industry.

1. INTRODUCTION

There has been some difficulty across the Nuclear Industry with a consistent approach to the qualification and testing of Industrial Packages. This paper tries to set out an Industry accepted standard.

2 QUALIFYING PACKAGES AS INDUSTRIAL PACKAGES

The IAEA Regulations (Safety Series 6) give specific requirements for packages to qualify as Industrial Packages. The relevant requirements with paragraph references are given in Table 1.

The following alternatives to this requirement are allowed.

(a) Drummed Material

Drums tested specifically to Package Group III in the "Recommendations in the Transport of Dangerous Goods" prepared by the United Nations Committee of Experts on the Transport of Dangerous Goods may be used as IP2 packages.

The drums, when tested to this requirement, should prevent:

(i) loss or dispersal of the radioactive contents

Table 1

REQUIREMENTS FOR INDUSTRIAL PACKAGES

IP1	IP2	IP3	Brief Description
505	505	505	Ease of handling and securing
506	506	506	Lifting attachments
507	507	507	" "
508	508	508	Protruding features and decontamination
509	509	509	Prevent collection/retention water
510	510	510	Features not to reduce safety
511	511	511	Acceleration/Vibration
512	512	512	Chemical compatibility
513	513	513	Valve protection
514	514	514	Other dangerous properties of contents
515	515	515	Maximum surface temperature - air only
516	516	516	Ambient temperature design -40°C - +55°C - air only
517	517	517	Liquids - Pressure differential requirements - air only
525	525	525	Minimum package size
		526	Package seal
		527	Tie-downs
		528	-40°C - +70°C
		529	Design, fabrication, manufacture acceptable standards
		530	Containment system fastening
		531	Special Form material inclusion
		532	Containment system fastening independent of package
		533	Radiolytic decomposition
		534	Reduce ambient pressure requirement
		535	Valve covers
		536	Radiation Shield fastening
		537	No loss or dispersal/20% radiation increase
		538	Ullage requirements
		619	Precedence of testing
		620	Time intervals of testing
		621	Water Spray test
	622	622	Free drop test - (table xiv)
	623	623	Stacking test
		624	Penetration test

(ii) loss of shielding integrity which would result in more than a 20% increase in radiation level at any external surface of the package.

The Department of Transport has issued a note "Guidance on the application of UN Tested Steel Drums to the Carriage of Class 7 Radioactive Material as IP2 (see Appendix 1)

(b) ISO Freight Containers

ISO Freight containers may be used as IP2/IP3 packaging if they conform to the requirements prescribed by the International Organisation for Standardisation document ISO 1496/1-1978 "Series I Freight Containers - Specification and Testing - Part 1. General Cargo Containers".

The ISO containers when tested to the requirement should prevent·

(i) loss or dispersal of the radioactive contents.

(ii) loss of shielding integrity which would result in more than a 20% increase in radiation level at any external surface of the package.

(c) Tank Containers

Tank Containers can be used as IP2/IP3 packaging if they are designed to conform to the standards prescribed in Chapter 12 of the "Recommendations on the Transport of Dangerous Goods" prepared by the United Nations Committee of Experts on the Transport of Dangerous Goods, or other requirements at least equivalent to those standards and are capable of withstanding a test pressure of 265kPa

They also should be designed so that any additional shielding which is provided shall be capable of withstanding the static and dynamic stresses resulting from normal handling and routine conditions of transport and of preventing a loss of shielding which would result in more than a 20% increase in the radiation level at any external surface of the tank container.

Tanks other than tank containers may be used provided they conform to standards equivalent to those above

The alternatives above still require the packages to meet the requirements of IP1 (518)

3 INTERPRETATION OF "PREVENT LOSS OR DISPERSAL"

The criteria for successful testing of some packages (IP and A) has used the phrase "would prevent loss or dispersal". (519, 523)

The maximum allowable leakage rate for normal transport of Industrial and Type A packages has never been defined quantitatively in the Regulations but has been required in a practical sense The intent of the statement "prevent loss or dispersal" is to ensure that under normal transport conditions the radioactive content of the package cannot escape in sufficient quantities to create a radiological or contamination hazard. (A-537.2)

The IAEA Advisory Notes go on to say that it is very difficult practically to advise on a single test method that could satisfactorily incorporate the vast array of packagings and their contents A qualitative approach dependent upon the packaging under consideration and its radioactive contents, may be employed (A-517 3)

In paras A-537 4 and A-537 5 the Advisory Notes go on to suggest ways of carrying out the tests but in many instances the appropriate method of detection is a visual examination to confirm that the contents have not escaped

(a) Application to UN Tested Drums

The Department of Transport have issued a statement (see Appendix 1) on the use of UN tested drums as IP2 Packagings The main condition is that the drum test must be carried out with material which has similar characteristics to the radioactive material to be carried The main characteristics are particle size, particle density, and total contents weight.

The method of detection should be a visual examination to confirm that the contents have not escaped.

(b) Application to Freight Containers

Where off-the-shelf freight containers are used it is recommended that LSA material should be packaged in inner containers (such as drums; these inner containers need not be tested to UN requirements). The inner containers should not be degraded by normal transport conditions of vibration, acceleration etc Simple wrapped material may be carried provided no separate item has minimum dimensions of less than 100mm The external surface of any internal package must not have loose contamination of more than that permitted on external packaging.

Where non-standard freight containers are used (ie half height ISO containers for Drigg waste) unpackaged waste may be carried as long as the lid/door seals are adequate.

(c) Tank Containers

There is no loss or dispersal requirement for tank containers.

4. SHIELDING REQUIREMENTS

It is acknowledged that using the UN and the ISO tests as a qualification only applies to those packages where no shielding is being claimed for the packaging itself.

5. QUALITY ASSURANCE

In using the alternatives of the UN Packaging Group III or the ISO 1496/1 - 1978 tests, it is the responsibility of the organisation issuing the Approval Certificate, the purchaser, and the user, to ensure that Quality Assurance has been applied by the manufacturer of the packaging.

6 PACKAGE REGISTER

It is intended that the Industry will have a national register of Industrial Package approvals The way this will work will be that an organisation wishing to use a particular package make-up would issue an IP Certificate and then register it on a database administered under the auspices of RAMTUC. If another organisation wishes to use that package they may do so by calling up the relevant certificate from the register. The certificate would include operating and maintenance requirements as appropriate and QA documentation requirements

7. CONCLUSION

The UK radioactive material transport industry wishes to remain a highly safe industry particularly since its long term future depends upon it However at the same time its future could be jeopardised by the introduction of over expensive and elaborate transport package designs for low hazard materials. Reviews of the standards of industrial and Type A packaged carried out by the US NRC/DOT following accidents and by the IAEA as part of the continuous review process have unanimously rejected the need for an increased standard of packaging or the need for quantitative leak criteria This paper is an attempt to generate a uniform and safe application of the IAEA Regulations in the UK in the area of Industrial Packages.

APPENDIX I

DEPARTMENT OF TRANSPORT
WORKING PARTY ON THE TRANSPORT OF DANGEROUS GOODS

GUIDANCE ON THE APPLICATION OF UN TESTED STEEL DRUMS TO THE CARRIAGE OF CLASS 7 RADIOACTIVE MATERIAL AS INDUSTRIAL PACKAGES TYPE 2 (IP-2)

INTRODUCTION

The Department has previously issued guidance to those concerned with the transport of radioactive material, the effect of which has been to require that the transport of all packages of radioactive material within the United Kingdom which do not require Department of Transport approval (eg Type A packages, Excepted packages and Industrial packages), shall be carried out in accordance with the "Regulations for the Safe Transport of Radioactive Material 1985 Edition (As Amended 1990)" published by the International Atomic Energy Agency, 1990. Those Regulations, inter alia, prescribe design and performance requirements and test methods for Industrial Packages Type 2 (IP-2), other than freight-containers or tanks, in IAEA paragraph 519 This paragraph permits, as an alternative to the tests of paras 622 and 623, that tests be done as specified for packaging group III in the "Recommendations on the Transport of Dangerous Goods", prepared by the United Nations Committee of Experts on the Transport of Dangerous Goods, and specifies pass criteria for those tests. This guidance note has been prepared in response to enquiries received by the Department concerning the use of steel drums, tested according to Chapter 9 of the UN Recommendations (the Orange Book), for the carriage of Class 7 (Radioactive) material. The note will be of interest to consignors and users of drums, drum manufacturers and NAMAS accredited test stations.

1 Recommendation 9.1 2 (a) of the Orange Book (Rev 7) is not meant to be circular (it is expected that this paragraph will be amended in the next revision of the Recommendations)· the use of a UN approved package as an IP-2 is permitted (see IAEA para 519) for certain types of radioactive material, viz certain low specific activity materials and surface contaminated objects (according to TABLE V of IAEA) but all other pertinent requirements of IAEA still apply, in particular the criteria for passing tests are those of IAEA para 519, not those of the UN Orange Book. However, the remaining exclusions from applicability of the UN, in paras 9 1 2 (b), (c) and (d) of the Orange Book, must also be seen as applicable to Class 7, as no package containing more than 400kg/450 litres could be certified under the UN Recommendations

2. Given the different pass criteria for Class 7, the scope for using "off the shelf" drum designs, already tested to the UN Recommendations for some different contents, will be limited by the following considerations·-

i) The "loss of shielding" pass criterion is not one which will have been applied in previous UN testing, nor is it one which the NAMAS accredited test stations could be expected to apply in the future, by any form of "live" testing with active material Thus use of designs already tested and approved would necessarily be limited to those cases where the inherent self shielding of the proposed contents and/or the drum wall is such that satisfaction of the pass criterion can be demonstrated by some other method than live testing, based on the recorded test evidence with surrogate material. Again the responsibility for demonstrating this is unlikely to fall to the NAMAS test station, who may wish to endorse their test reports accordingly, and the user will be responsible for satisfying himself (and, on request, the Department) that the criterion is met. Pira International is the Department's certification body for UN package testing in relation to the provisions of UN Chapter 9; Pira has no authority to endorse certificates to the effect that a package complies with the provisions of Safety Series 6.

ii) Not only the total content mass, but also the form, density and homogeneity of the contents are important factors in considering the performance of a package when tested. Most pre-existing designs will have been tested with essentially homogeneous contents such as powders, granules, pellets or liquids etc. Uses for Class 7 material are likely to range from, for example, high density but homogeneous materials - ores and concentrates - to low density though inhomogeneous uncompacted waste materials. This will limit the applicability of existing test data to those cases where close similarity of all pertinent factors (or conservatism of those factors) may be demonstrated, and will necessitate further testing in cases where this cannot be shown.

iii) Not all existing UN approved designs are necessarily suitable for Class 7 contents and each case should be considered on the merits of both packaging and proposed contents. For example UN 9.7.3.5 2 permits a package to pass the test, even if it does not remain sift-proof, provided "the entire contents are retained by an inner packaging or inner receptacle (eg a plastics bag)". An inner plastics bag is unlikely to consistently provide this duty for contents consisting of a mixture including sharp objects together with fine material.

3. The IAEA Regulations (para 209) require the application of quality assurance at all appropriate phases during the transport of radioactive material (including the design, manufacture, testing, documentation, use, maintenance and inspection of packages amongst other things). The responsibilities for carrying out these measures is shared among those concerned ie designers, manufacturers, consignors, carriers etc as appropriate. This remains the case where UN tested packages are used in lieu of IAEA tested packages, but the user, particularly the consignor, of any packaging will have to satisfy himself as to the quality of design, testing and manufacture and the fitness-for-purpose of any packagings "bought in" to transport his particular contents and to demonstrate this to the Department on request.

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