PACKAGING AND TRANSPORTATION
OF RADIOACTIVE MATERIAL
(PATRAM '86)

VOLUME 1
(STI/PUB/718)

CORRIGENDUM

On page 445, Table VII, second column, the unit should read (Sv) instead of (mSv).

Page 445, tableau VII, deuxième colonne, l’unité indiquée est (Sv) et non (mSv).
PACKAGING AND TRANSPORTATION
OF RADIOACTIVE MATERIALS
(PATRAM '86)

VOLUME 1
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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

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Printed by the IAEA in Austria
January 1987
PACKAGING AND TRANSPORTATION OF RADIOACTIVE MATERIALS (PATRAM '86)

PROCEEDINGS OF AN INTERNATIONAL SYMPOSIUM ON THE PACKAGING AND TRANSPORTATION OF RADIOACTIVE MATERIALS ORGANIZED BY THE INTERNATIONAL ATOMIC ENERGY AGENCY AND HELD IN DAVOS, 16–20 JUNE 1986

In two volumes

VOLUME 1

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 1987
FOREWORD

The growth in the worldwide applications of radioactive materials and the use of nuclear energy has been accompanied by a corresponding growth in the numbers and types of shipments of radioactive materials. Today many millions of packages are shipped each year, ranging from manufactured items containing very small quantities of radioactive material, such as smoke detectors, to other, more hazardous items, such as gamma radiography sources, radiopharmaceuticals, uranium ores and uranium concentrates, to very hazardous quantities of materials, such as irradiated nuclear fuel. All of these are packaged according to the hazard posed by the radioactive material should it be lost from its packaging and the packages generally can move by all modes of transport within countries, between and through countries and over international waters.

The safe packaging and transport of radioactive materials have long been of concern to regulators, designers, consignors, carriers, consignees and the general public since these activities began decades ago and they have also long been a key part of the International Atomic Energy Agency’s activities, where efforts have been, and continue to be, focused on the Regulations for the Safe Transport of Radioactive Material, known as Safety Series No. 6. This document, which has been revised periodically to reflect changes in technology, practices and needs, serves as a set of regulations for the Agency and for Agency-supported activities and also serves as a model for domestic and international modal regulations throughout the world. The latest comprehensive revision of Safety Series No. 6 was completed and published in English, French, Russian and Spanish during 1985.

Through efforts co-ordinated by the Agency, but supported by many experts from Member States and international organizations, a series of supportive documents have also been developed and are being published, which should assist in the proper application of the Transport Regulations.

A procedure has recently been implemented by the Agency which will provide a rational and continuing basis for keeping the Transport Regulations current with technology and with users’ needs. This procedure includes the issuing every two years of a supplement to Safety Series No. 6 until it undergoes another comprehensive revision. These supplements will provide a mechanism for implementing small corrections and changes in a timely fashion.

The Agency is also involved in many other activities to enhance safety in transport worldwide. For example, additional guidance is being prepared which will serve as a model for competent authority implementation of the Regulations, especially in developing countries. The Agency also assists in the international administration of transport by periodically issuing a list of national competent authorities and a direc-
tory of Approval Certificates issued by the national competent authorities. The Agency, through the assistance of experts in Sweden and the United States of America, has developed a computer code known as INTERTRAN which provides a system for assessing the radiological impact from transporting radioactive materials. Finally, through the Technical Assistance Programme of the Agency, training of regulatory personnel in developing countries has been progressing. In the past three years, for example, training assistance has been provided to ten developing Member States.

Its focus on areas relative to the safe transport of radioactive materials, the significance of the transport of these materials and the need for a consistent and comprehensive regulatory structure were all factors in the Agency's decision to organize this symposium, at the invitation of the Government of Switzerland and with the support of the United States Department of Energy (DOE), in cooperation with the Swiss Federal Institute for Reactor Research (EIR) and the Swiss Federal Department of Energy (BEW). The symposium was scheduled to occur in the period immediately following the publication of the new, 1985 Edition of Safety Series No. 6 so as to facilitate the implementation of this new edition of the Regulations into domestic regulations and international organization regulatory documents. The symposium also served as an effective means of exchanging information on the current level of technology in the field of transport safety and for identifying problems which need to be dealt with in the future.

In his introductory remarks, Dr. E. Kiener (Director of the Swiss Federal Office of Energy) noted that for countries such as his own, the international harmonization of transport regulations was very important, since they depend upon other countries for their nuclear fuel cycle materials. Symposia such as PATRAM'86 were thus very important in enhancing this harmonization. He further noted that the symposium should also help politicians and the public understand the basis of radioactive material transport regulations and lead to more rational bases for decisions in this area in the future.

Expanding on the need for a rational decision making process, Professor M. Cosandy (Chairman of the Board of the Swiss Federal Institute of Technology) expressed, in his opening address, the hope that the symposium would assist in broadening the understanding of the technology behind the transport of radioactive materials so that the proper application of these materials to the betterment of life might be able to proceed on a rational, safe basis.

There were 462 designated participants and 22 observers from 37 Member States and 5 international organizations in attendance at the symposium. In all, 163 papers were presented in oral or poster sessions. The topics dealt with generally included transport regulations — national and international, codes, standards, research and development, experience, planning, administrative issues, emergency response, package design, system design, component design, analysis, testing and materials. Almost half of the papers dealt with the design, analysis and testing for transport, which continue to be the areas where much of the practical application of
the Regulations occurs. A number of papers dealt with the co-operative aspects of adopting and implementing the Regulations on an international basis, which will be needed in the late 1980s and early 1990s if uniformity relative to the 1985 Edition of Safety Series No. 6 is to be achieved and safety is to be assured. These two volumes contain the papers presented at the symposium. In some cases, authors of poster session papers elected to have only their extended synopses published.

The success of the symposium was greatly enhanced by the generosity of the host country — Switzerland — and by the co-operative contributions of the United States Department of Energy. In particular, the efforts of Dr. C. Ospina of the Swiss EIR and Mr. F. Falci of the DOE are gratefully acknowledged. The IAEA, on behalf of all of the participants at the symposium, expresses its sincere gratitude and appreciation to the Government of Switzerland for the excellent arrangements, hospitality and facilities made available at Davos in June 1986.
EDITORIAL NOTE

The papers and discussions have been edited by the editorial staff of the International Atomic Energy Agency to the extent considered necessary for the reader's assistance. The views expressed and the general style adopted remain, however, the responsibility of the named authors or participants. In addition, the views are not necessarily those of the governments of the nominating Member States or of the nominating organizations.

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Switzerland

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IAEA
Keynote Address

NUCLEAR WASTE RESEARCH AND MANAGEMENT AND THEIR TRANSPORT IMPLICATIONS
A Swiss perspective

H. GRÄNICHER
Eidgenössisches Institut für Reaktorforschung,
Würenlingen, Switzerland

1. EARLY DEVELOPMENTS

In taking Switzerland as an example of the general trend, it should be noted that applications of radionuclides were made long before nuclear energy had developed and been utilized, e.g. radium was used in medicine and in industry. Of particular interest for Switzerland was the use of radium for watch dials and hands, though in such applications radium has long been replaced by tritium. In nuclear physics research, artificially produced radionuclides were mostly short-lived and the quantity produced was marginal as compared with current production. These radionuclides were produced by accelerators, such as cyclotrons, which were built in increasing numbers in university physics laboratories after the late 1940s. In particular, I would like to mention the 12 MeV cyclotron built at the Physics Institute of the Eidgenössische Technische Hochschule, Zürich, under Professor Paul Scherrer, and several small linear accelerators both there and at other Swiss university laboratories.

After the middle 1950s, national research laboratories were created which, in most cases, had not only accelerators, but were also setting up research reactors. The production of radioisotopes for medical and industrial applications was one of the important justifications for creating these facilities. These applications, and more extensive nuclear research activities, produced radioactive wastes in increasing quantity. In addition to wastes from such applications and research using naturally radioactive materials, the operations of accelerators and research reactors themselves became sources of radioactive wastes, e.g. activated components. In addition, countries with weapons programmes generated considerably larger amounts of radioactive wastes, all of this at an early stage in the development of nuclear energy.

In parallel with the annual increase in the quantity of radioactive wastes, the nuclear community in general developed an awareness of the requirements of radiation protection and, as a consequence, the need to pay special attention to the safe handling and disposal of radioactive wastes. Thus the responsibility for
investigating and developing methods of waste management in these early days fell, quite naturally, on the major producers of radioactive wastes: the national research centres. The waste handling facilities which they had to build and operate for their own needs became service centres for the radioactive wastes produced by industry, hospitals and from other research activities using radioactive tracers. As legislation and regulations both for the safe handling and transportation of radioactive wastes were developed, the assistance of such waste service centres became indispensable.

Developments in Switzerland have followed the trends described so far. The Eidgenössisches Institut für Reaktorforschung (EIR) (Swiss Federal Institute for Reactor Research), founded in 1955 as Reaktor Ltd., initially stored its wastes in a pit, but by 1963 had built and was operating a temporary storage facility. A special group to deal with such work was created in 1968, which was upgraded into a separate division in 1980. Equipment for low level radioactive waste handling was developed and the following special buildings were erected:

- Radioactive waste laboratory, built in 1970. This is a facility for the sorting of radioactive wastes for pretreatment, such as mechanical compression, and for packing with concrete in steel drums or in concrete containers.
- Incineration plant, built in 1975. Construction was based on developments at the Kernforschungszentrum Karlsruhe, Federal Republic of Germany.
- A new radioactive waste service building, completed in 1983, which includes halls for incoming waste drums and for intermediate storage of conditioned wastes.

Furthermore, Switzerland participated in the first Organisation for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA) sea dumping operation of low level wastes in 1969 and continued to do so until 1982.

2. IMPACT OF NUCLEAR ENERGY PRODUCTION

It is not necessary to dwell on the fact that nuclear energy production has opened up a new dimension to the problem of radioactive wastes for Switzerland, both from the point of view of the quantity and of the quality of wastes generated by power plant operations (Table I).

As long as OECD/NEA sea-dumping operations were possible, EIR carried out the final steps in preparing low level wastes for final disposal, both for the nuclear power plants as well as for such wastes from other national sources. However, in recent years such operations have become politically unacceptable and, as a result, EIR has had to ship LLW back to the (intermediate) storage facilities of the nuclear power plants. For the LLW from medical, industrial and research applications, as well as from government laboratories, the construction of an intermediate storage facility on the EIR site is planned in order to bridge the gap before final geological disposal in a type B facility planned by the Nationale Genossenschaft für die Lagerung Radioaktiver Abfälle (NAGRA).
TABLE I. COMMERCIAL OPERATION OF SWISS POWER PLANTS

<table>
<thead>
<tr>
<th>Power plant</th>
<th>Start of operation</th>
<th>First batch of radioactive wastes conditioned by EIR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Beznau I</td>
<td>1970</td>
<td>1972</td>
</tr>
<tr>
<td>Beznau II</td>
<td>1972</td>
<td>1972</td>
</tr>
<tr>
<td>Mühleberg</td>
<td>1972</td>
<td>1976</td>
</tr>
<tr>
<td>Gösgen</td>
<td>1979</td>
<td>1979</td>
</tr>
<tr>
<td>Leibstadt</td>
<td>1984</td>
<td>1985</td>
</tr>
</tbody>
</table>

As will be described in the paper by F. Dommann [1], spent fuel elements from Swiss reactors have been sent to reprocessing plants abroad. This is also true of EIR's research reactors, with the exception of the last core load of the DIORIT reactor, which was decommissioned in 1977. Since no reprocessing of its fuel elements appeared possible, EIR initiated a dry-storage programme, purchasing the first CASTOR container for commercial use. The results of this programme have proved to be very satisfactory.

The transportation of fuel elements and radioactive wastes has not caused any notable problems in Switzerland, either from the technical or public safety aspects. However, minor contamination of three railroad wagons was detected when Swiss LLW drums were transported to a ship in 1978.

3. WASTE MANAGEMENT RESEARCH AND DEVELOPMENT

In order to cope with developing waste handling requirements, EIR has invested in waste management research and development, focusing on methods to embed LLW, improving and testing concrete containers for final disposal and development of an incineration plant (in operation since 1975), which also has to be adaptable in the future to stricter air pollution control limits [2].

About a decade ago, EIR initiated research activities directed toward safe geological disposal of HLW. This work was expanded into the biggest research project of the Institute when NAGRA started its programme called "Gewähr" ("Guarantee"), to which EIR has made major contributions, particularly in the field of risk analysis. A key item of EIR's research are studies of the propagation of radionuclides through geological formations in the event of the destruction of a final HLW storage facility and contact with water. Thus, assuming that Switzerland has to be prepared to take back HLW coming from the reprocessing of its spent fuel elements, such research and development activities will remain a major and important part of EIR's programme.
4. PUBLIC ACCEPTANCE

Every year several thousand visitors come to EIR and almost all of them are shown our facilities for the treatment of radioactive wastes. They are told the dimensions of the waste problem, that for final disposal Swiss LLW are typically packed in about 1000 tons of concrete per year, though the actual content of radioactive atoms does not exceed one kilogram. These visitors express surprise at the small quantity of radioactive wastes, as compared with the quantity of 'normal' wastes in the country. EIR considers such visits to be a very valuable source of information for the general public on the management of radioactive wastes.

REFERENCES

I consider it a great honour to have the opportunity of welcoming you today to PATRAM 86: The operators of the Swiss nuclear power plants are very pleased that a symposium with the theme 'Packaging and Transport of Radioactive Materials' is being held in Davos. Switzerland is the seat of many international organizations and as a result conferences are often held here. However, symposia with the theme of nuclear energy have been rather few in recent years.

As you know, Switzerland was involved quite early in nuclear research. Our lack of indigenous raw materials was a reason for our interest in this new technology. Today we have four nuclear power plants (five nuclear power reactors) in operation that provide for about 40% of present electricity consumption. Nuclear power plants in Switzerland are also involved in the delivery of process steam (Gösgen) and in district heating (Beznau). The operation of these power plants has been proceeding smoothly for many years, with Swiss industry being able to provide practically all of the components for these plants. It is fair to say that in spite of strong political opposition, especially during the 1970s, nuclear technology has come to be a well-established and safe source of energy in this country.

The packaging and transport of radioactive materials have a long tradition in Switzerland. Until about 1960, such transports were carried out only in connection with applications of radioactive isotopes in research, medicine and industry. Today the overwhelming majority of radioactive material movements still belong to this category. The startup of research reactors at the Eidgenössisches Institut für Reaktorforschung (Swiss Federal Institute for Reactor Research), in Würenlingen, at the end of the 1950s, the pilot power plant Lucens and the operation, since 1969, of the first commercial nuclear power plant in Beznau, have all increased the quantity of wastes and the frequency of transport of fresh and spent fuel. In the future the transport of radioactive wastes arising from the reprocessing of spent fuel will become a more serious issue, since Switzerland has to take over the intermediate and final disposal of these wastes.

The IAEA Transport Regulations (see Ref. [1] for details of the latest edition) have been the basis for all transports of radioactive materials in Switzerland since 1961. Regulations concerning the transport of radioactive materials by rail came into force in 1967 and similar regulations for road transport went into effect in
1972. These regulations were modified over the years, as were the IAEA Regulations. It can generally be said that Swiss experience with such transports has been rather good. Minor incidents were very seldom and major incidents have never occurred.

The most frequent type of transport involving radioactive materials in Switzerland is that which begins and ends in the country. However, in recent years there has been an increasing volume of transport across the borders. The reason for this is the more widespread use of radioactive substances in medicine and industry on the one hand, and increased specialization, for example the reprocessing of spent fuel and the treatment of radioactive wastes in various
European countries, on the other. Unfortunately, the administrative regulation of international transport is very complicated. A unified set of regulations — at least in Europe — could substantially simplify the paperwork and is therefore quite desirable. The transport of fissile material is another, special, problem, though in this case the rules which apply in connection with the Nuclear Non-Proliferation Treaty must of course be taken into account.

Spent fuel has been transported from Switzerland to the reprocessing plants at La Hague, in France, and Sellafield, in the United Kingdom, since the beginning of the 1970s. Different types of flasks, weighing up to 120 t and using both wet and dry systems, have been used. All nuclear power plants in Switzerland, with
FIG. 3. Spent fuel transport cask ready for rail transport at the Gösgen plant.

TABLE I. TRANSPORT OF SPENT FUEL FROM SWISS NUCLEAR POWER PLANTS

<table>
<thead>
<tr>
<th>Nuclear power plant</th>
<th>Operator</th>
<th>Quantity up to end of 1985 (tU)</th>
<th>Flask Type</th>
<th>Beginning of transport</th>
<th>Destination</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mühleberg</td>
<td>BKW</td>
<td>134</td>
<td>NTL-9</td>
<td>1975</td>
<td>La Hague</td>
</tr>
<tr>
<td>Beznau I + II</td>
<td>NOK</td>
<td>135</td>
<td>NTL-4, NTL-5</td>
<td>1971</td>
<td>Sellafield</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>NTL-11</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>100</td>
<td>NTL-11, TN-12</td>
<td>1982</td>
<td>La Hague</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>TN-17, LK-100</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Gösgen</td>
<td>KKG</td>
<td>93</td>
<td>TN-12</td>
<td>1982</td>
<td>La Hague</td>
</tr>
<tr>
<td>Leibstadt</td>
<td>KKL</td>
<td>–</td>
<td>–</td>
<td>–</td>
<td>–</td>
</tr>
</tbody>
</table>
the exception of the one at Mühleberg, have direct rail connections. Mühleberg carries out transport by road. Table I gives a summary of the transport of spent fuel in Switzerland up to the end of 1985.

All transports of spent fuel have up to now been carried out without significant problems. It is especially interesting to note the successful transport of three seriously damaged fuel assemblies from Mühleberg in a special packaging contained in an NTL-9 flask. Figures 1 to 3 depict fuel transportation at the Gösgen nuclear power plant.

The first transport of radioactive wastes from the reprocessing of Swiss fuel at La Hague back to Switzerland is tentatively expected in 1995. All necessary details in connection with the conditioning, packaging and safe transport of these materials have already been discussed with the reprocessing companies.

REFERENCES

TRANSPORT OF RADIOACTIVE MATERIALS IN EUROPE AND THE ROLE OF THE EUROPEAN ECONOMIC COMMUNITY

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Abstract

TRANSPORT OF RADIOACTIVE MATERIALS IN EUROPE AND THE ROLE OF THE EUROPEAN ECONOMIC COMMUNITY.

There are several important dates in the recent history of the transport of dangerous goods in Europe. The first date, 1879, marks the beginning of work in Europe on the drafting of international regulations for the transport of dangerous substances on the Rhine. The paper outlines the nature and origins of European transport organizations and their connections with transport organizations with worldwide responsibilities. Special emphasis is given to the role of the European Economic Community (EEC) and, in particular, the role of the Commission of the European Communities Special Permanent Working Party for the Transport of Radioactive Materials, which was created in 1982. Reference is made to recent studies financed by the EEC supporting the harmonized implementation of transport in Europe.

1. INTRODUCTION

In a serious shipping accident on the Rhine in 1876, a vessel carrying highly toxic arsenic foundered. Serious consideration was given to evacuating the population from the affected area. Just a few years later (in 1879), and as a direct consequence of that accident, regulations for the transport of explosive, flammable, caustic and poisonous substances on the Rhine were issued. These were virtually the first regulations for the transport of dangerous goods in Europe. Independent of this, work was begun in 1878 on an international agreement on rail transport in Europe. Ten European states put this agreement into effect on 1 January 1890. Its Annex 1 contained regulations concerning objects acceptable for transport only under certain conditions. These regulations at that time covered six-and-a-half pages but did not, of course, apply to radioactive materials, which had not yet been discovered.

Demands for greater safety, and the arrival on the scene of new types of carriers (e.g. aircraft and road transport motor vehicles), made it necessary, by
the 1950s, to pay closer attention to the transport of dangerous goods. Whereas only one page of regulations was devoted in 1879 to inland waterway shipping and only six-and-a-half pages in 1893 to rail traffic, today a single country, the Federal Republic of Germany, devotes 3000–4000 pages of regulations just to these forms of transport. Virtually everything that has to do with the transport of a range of dangerous goods, from explosive materials to caustic substances, is subject to regulation.

Dangerous goods are divided into categories according to their properties. For example, explosive materials, which are assigned to danger category 1a, are differentiated from caustic materials, which belong to danger category 8. The breakdown into danger categories is at present based on the recommendations of the United Nations, drawn up by the UN Committee of Experts on the Transport of Dangerous Goods. Category 7 of these danger categories covers radioactive materials, and was added only towards the end of the 1950s. Originally, it was lack of knowledge which caused radioactive materials to be assigned to a subcategory of toxic materials, the reason being that radioactivity was considered to be a type of toxicity. It was only when new information was provided by United Nations experts that radioactive materials were transferred in the mid-1960s to this new category, 7.

2. EUROPEAN TRANSPORT ORGANIZATIONS

Originally, all carriers drew up regulations for their own specific areas of interest, unfortunately often independently of each other, with the result that there were no harmonized regulations that applied equally to all. Listed below are the various transport organizations in Europe.

2.1. CCNR (for inland waterway shipping on the Rhine)

The Rhine became an international waterway in 1868, with all States along the Rhine being required to guarantee free passage for shipping on the river. However, safety also had to be guaranteed for those States themselves. For this purpose, a Central Commission for Navigation on the Rhine (CCNR) was set up in Strasbourg which, in addition to dealing with questions of customs and safety, also dealt with the transport of dangerous goods. These first general regulations for the transport of dangerous goods were established in 1879, after an accident involving poisonous substances and were part of the general police laws. The regulations were continuously developed and amended, but it was not before 1960 that the specific Agreement on the Transport of Dangerous Goods on the Rhine (ADNR) came into effect. These regulations, as developed over the years, today guarantee a high level of safety for the transport of dangerous goods on the Rhine which, it should be said, has the heaviest traffic of any inland waterway in
the world. Though hardly any radioactive materials are carried on the river at present, the transport of high-activity wastes is scheduled to begin in 1988.

2.2. ECE (for European inland waterway traffic)

The Economic Commission for Europe (ECE) of the United Nations, located in Geneva, drew up a recommendation for the transport of dangerous goods on European inland waterways, abbreviated as ADN from its French original title of Accord européen relatif au transport international des marchandises dangereuses par voie de navigation intérieure. This agreement is to apply to all European inland waterways. Work on the ADN was interrupted for some time, but was renewed in 1985 and is still continuing. Whether the ADN will become a European convention or will continue purely as a set of recommendations is not as yet clear.

The CCNR took advantage of the work carried out by the ECE at the end of the 1960s and incorporated almost all of the then existing recommendations into its own regulations on the transport of dangerous goods on the Rhine.

2.3. OCTI (for railway transport)

The first regulations applying to railway transport were put into effect in 1893. They were amended in 1928 to cover six danger categories. It was also in 1928 that the first danger certificates appeared. The designation RID, derived from the French Règlement international concernant le transport des marchandises dangereuses par chemins de fer (International Regulations Concerning the Carriage of Dangerous Goods by Rail) was first applied in 1956. In its present form, RID is implemented by all European countries (with the exception of the Union of Soviet Socialist Republics and Albania) and by several Mediterranean countries. The body responsible for rail transport is the Central Office for International Railway Transport (OCTI), in Bern.

2.4. ECE (for European road transport)

In contrast to European railway transport, transport regulations for dangerous goods transported by road only came into effect in 1957. Use was made of the work already done on the RID, and the RID regulations served as a basis for a European Agreement Concerning the International Carriage of Dangerous Goods by Road, abbreviated to ADR from its original French title of Accord européen relatif au transport international des marchandises dangereuses par route. At the present time, 19 European countries are parties to this agreement.
FIG. 1. The transport of dangerous goods: organizations concerned and relevant provisions [1].
3. TRANSPORT ORGANIZATIONS WITH WORLDWIDE RESPONSIBILITIES (Fig. 1)

3.1. International Maritime Organization (for sea transport)

Sea transport was never a matter for Europe alone, since it is an activity conducted throughout the world. After the International Convention for the Safety of Lives at Sea (SOLAS) was concluded in 1964, the industrialized countries of Europe collaborated in drawing up a code for the transport of dangerous goods in ships, known today as the International Maritime Dangerous Goods (IMDG) Code. Since the Code is only a recommendation, in contrast to RID and ADR, not all European States have introduced it as being binding upon their carriers. Also, there are different ways that the Code has been applied by national legislation. Several countries apply the IMDG Code in its original English version without translating it into their national languages, while other States have been obliged by their laws to translate and implement it in their own languages.

3.2. ICAO (for air transport)

The transport of dangerous goods by air actually first acquired importance with the introduction of wide-body aircraft, although the International Air Transport Association (IATA) was aware very early on of the importance of the transport of such goods and issued appropriate transport regulations. The governments of several industrialized countries themselves became aware of the problem only in the mid-1970s and accordingly drew up their own national regulations. This work was completed by the International Civil Aviation Organization (ICAO), in Montreal, by a special staff of experts and in 1984 these Technical Instructions for the Transport of Dangerous Goods by Air were introduced as binding on all ICAO Member States (over 150). In the meantime, IATA has also adopted the ICAO regulations, so that the regulations issued by both bodies for the transport of dangerous goods by air are, from a technical standpoint, now identical.

4. UNITED NATIONS RECOMMENDATIONS FOR WORLDWIDE IMPLEMENTATION

Mention has already been made of international agreements or recommendations specific to certain types of carriers. Since most experts concentrated for decades solely on 'their' carriers, the regulations naturally diverged. There were — and to some extent still are — insurmountable difficulties in attempting to transpose regulations from one type of carrier to another. As a result, special groups of experts from the Economic and Social Council of the United Nations, the
Group of Experts on the Transport of Dangerous Goods and the Group of Experts on Explosives, were set up to deal with questions concerning the transport of dangerous goods which were of equal significance to all types of carriers. These experts began their work in 1985 and have produced recommendations that are recognized throughout the world as setting the standards for the transport of dangerous goods. As has already been stated, however, they are only 'recommendations' which must still be transformed into agreements and national regulations through legislation. These recommendations are amended every two years and the new versions are issued in a publication with an orange cover, generally called the 'Orange Book'.

5. IAEA REGULATIONS FOR RADIOACTIVE MATERIALS

Only conventional dangerous goods were dealt with initially, since there was hardly any need to transport radioactive materials up to the mid-1950s. However, transport regulations applying to such materials were included in RID as early as 1959. Independent of this, a group of experts from the IAEA was also dealing with the same subject and for the first time drew up regulations, which were issued in 1961 as Safety Series No. 6, Regulations for the Safe Transport of Radioactive Materials. Much new ground had been broken here and, indeed, all international transport organizations adhere to the IAEA Regulations in matters concerning the transport of radioactive materials.

In contrast to everything that had been done so far, the IAEA Group of Experts was entering completely uncharted territory. The conditions laid down for the transport of radioactive materials were justified on scientific grounds. They were based on the principle that transport containers had to be absolutely accident-proof, and this had to be demonstrated by testing. Where packagings that were not accident-proof were used, the contents had to be restricted in such a way that, in the event of a release, it was virtually impossible for persons and the environment to be harmed: this is a unique philosophy which is still applicable and scientifically justified today.

6. NATIONAL REGULATIONS

Although most European countries implement the existing international regulations in the case of international transport, the same cannot be said of transport operations within the countries themselves. It has been found in such cases that each State often applies its own laws.
7. THE COMMISSION OF THE EUROPEAN COMMUNITIES


The Directorate-General for Energy of the Commission of the European Communities (CEC), located in Brussels, organized discussions, in the early 1980s, on the transport of radioactive materials within the Member States of the European Economic Community (EEC). Of particular interest were certain aspects of the nuclear fuel cycle. It was necessary in this connection to ensure that it would be possible, under all circumstances, to transport radioactive materials, for example from a nuclear power station to a reprocessing plant, without difficulty.

In 1982, in response to a resolution of the European Parliament, the CEC set up a special permanent working party with experts from Member States for the purposes of advising it in the preparation of proposals for EEC activities and of studying all of the problems associated with the transport of radioactive materials. The CEC which, moreover, chairs and provides the secretariat services for this special working party, thus now possesses a permanent and official consultation body which has so far met nine times.

The CEC pointed out at the first meeting of the special working party that it did not plan to duplicate the work of the IAEA or to take any measures that fell solely within the competence of the Member States. It stated that, on the contrary, the objectives of its activities were:

1. To speed up the process of bringing the national laws of the EEC Member States into line with IAEA Regulations.
2. To ensure consistency of laws and administrative practices within the EEC in order to make possible the functioning of an effective ‘common market’ in the transport of radioactive materials.
3. To further stimulate, co-ordinate and support the work of the Member States in revising IAEA Regulations, in particular those sectors that are only partly covered or not covered at all by those regulations.

The CEC’s activity has so far taken the following principal forms:


b. Performance of studies, through some 51 contracts concluded with specially qualified bodies and firms, covering certain technical, administrative and regulatory aspects of the transport operations under consideration.

In the first communication mentioned above, transmitted to the European Parliament and the Council in 1984, it emerges that consolidated and substantially uniform requirements concerning the transport of radioactive materials are in force
in all the EEC countries [1]. The common basis of the regulations in force in the Member States of the EEC is Safety Series No. 6, issued by the IAEA. In view of the difficulties encountered in implementing international recommendations at the national level, the results achieved are undoubtedly excellent.

The status of national regulations with respect to IAEA Regulations was examined in a study carried out in 1980 under the sponsorship of the CEC [3]. It appeared — as indicated in the communication — that certain problems, mainly administrative, could be tackled jointly for the benefit of all the countries concerned. The 1985 edition of the IAEA Regulations and the transport regulations based on them, such as ADR, RID, ADNR, etc., will provide for a more general harmonization. The Special Permanent Working Party will continue a close watch on the situation. Moreover, it must be pointed out that there is a substantial measure of agreement between the IAEA Regulations and the provisions of the European Atomic Energy Community Directive on the health protection of the general public and workers against the danger of ionizing radiation.

The above-mentioned communication will be periodically updated by means of supplementary communications describing the main events that have occurred since the previous issue, for example, developments in legislation, the improvement of technical knowledge, comments on any accidents that have occurred, etc. The next communication should appear towards the end of 1986.

With regard to studies, an appropriation of 2 500 000 European currency units granted by the budgetary authorities for the period 1980–86 to fund the CEC’s work on the transport of radioactive materials has made it possible to finance 51 study contracts divided into five categories. The following are the five categories covering the studies during the seven financial years in question (Table I):

- Category I. Work which may possibly lead to the adoption of measures (recommendations, directives, etc.) at EEC level.
- Category II. Work in support of the competent authorities of the Member States involved in the project for revision of the IAEA Regulations.
- Category III. Benchmark work on the verification of calculation methods in the design of transport containers for which design approval is required by the competent authorities.
- Category IV. Work relating to aspects of transport covered only partly by the IAEA Regulations.
- Category V. Work of a general nature requested by the Member States of the EEC.

In the context of the CEC’s activity related to study contracts, 65% of the appropriation has so far been devoted to experiments — tests on various types of packages, with a view to improving the present IAEA Regulations; tests on the stowing of packages, in order to pave the way for new, complementary requirements; reference experiments (benchmark work) on the verification of calculation methods.
### TABLE I. STUDIES FINANCED BY THE CEC DIRECTORATE-GENERAL FOR ENERGY

<table>
<thead>
<tr>
<th>Contract number</th>
<th>Contractor</th>
<th>Topics</th>
<th>Category</th>
<th>Status of the final reports</th>
</tr>
</thead>
<tbody>
<tr>
<td>XVII/322/80/1</td>
<td>TRANSNUCLEAIRE (F)</td>
<td>Gamma shielding benchmark experiment on a package containing irradiated fuel coming from light water reactors.</td>
<td>III</td>
<td>EUR 8017 FR, Nf, 1982</td>
</tr>
<tr>
<td>XVII/322/80/2</td>
<td>CEA-FONTENAY-AUX-ROSES (F)</td>
<td>Study on interactions between crush conditions and fire resistance for type B(U) packages less 500 kg intended for the transport of radioactive materials.</td>
<td>II</td>
<td>available, not published</td>
</tr>
<tr>
<td>XVII/322/80/3</td>
<td>BAM (FRG)</td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>XVII/322/80/4</td>
<td>TRANSNUBEL (B)</td>
<td>Stowing of packages containing radioactive materials on conveyances. Phase 1</td>
<td>IV</td>
<td>EUR 8057 EN, Nf, 1982</td>
</tr>
<tr>
<td>XVII/322/80/5</td>
<td>UKAEA-SRD (UK)</td>
<td>Criticality safety hazards arising from the transport of fissile materials.</td>
<td>II</td>
<td>EUR 8345 EN, Nf, 1983</td>
</tr>
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<td>XVII/322/80/6</td>
<td>RADIOCHEMICAL CENTRE (UK)</td>
<td>The specification and testing of radioactive sources designated as &quot;Special Form&quot; under the IAEA regulations.</td>
<td>II</td>
<td>EUR 8053 EN, Nf, 1982</td>
</tr>
<tr>
<td>XVII/322/80/7</td>
<td>UKAEA-SRD and CEGB (UK)</td>
<td>IAEA 9m drop tests and impact with materials likely to be encountered in an accident.</td>
<td>II</td>
<td>available, not published</td>
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<tr>
<td>XVII/322/80/8</td>
<td>TRANSNUCLEAR (FRG)</td>
<td>Determination of the temperature distribution in spent light water reactor fuel elements during transport and intermediate storage. Phase 1</td>
<td>III</td>
<td>available, not published</td>
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<td>XVII/322/80/9</td>
<td>CNEN (I)</td>
<td>Proposal of standardisation of certificates for transporting radioactive materials.</td>
<td>I</td>
<td>available, not published</td>
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<td>XVII/322/80/10</td>
<td>CNEN (I)</td>
<td>Ability of type A packages to withstand regulatory tests.</td>
<td>II</td>
<td>EUR 8030 EN, Nf, 1983</td>
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<td>XVII/322/80/11</td>
<td>I.I.N. Pisa (I)</td>
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**STUDIES FINANCED IN 1981**

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<tr>
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<th>Description</th>
<th>Code</th>
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<tr>
<td>XVII/AV/81/140</td>
<td>M. SELLING, Ministerie van volksgezondheid (NL)</td>
<td>Mutual emergency assistance in the event of accident during transport of radioactive materials within the Member states of the European Community.</td>
<td>I</td>
<td>EUR 9133 EN, Mf, 1984</td>
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<tr>
<td>XVII/AV/81/141</td>
<td>Mr. G.A. HOLDER (UK)</td>
<td>Analysis of qualifications required and the training of workers taking part in the transport of radioactive materials in the Member States of the European Community.</td>
<td>I</td>
<td>available, not published.</td>
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<tr>
<td>XVII/AV/81/142</td>
<td>CNEN (I)</td>
<td>Risk assessment during the transport of irradiated fuel.</td>
<td>V</td>
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<tr>
<td>XVII/AV/81/309.1</td>
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**STUDIES FINANCED IN 1982**

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Since its inception about five years ago, the CEC's programme of action has produced results, but it must be pointed out that its budget is modest and is determined on a pragmatic basis from year to year, after technical consultations with the industrial and scientific circles concerned and after discussion in the Special Permanent Working Party. Its most significant contribution has been the publication of the general communication mentioned earlier [1]. This communication is an information and reference document for EEC institutions and may, of course, be used freely by the mass media and the sectors concerned. The activity within the programme which had the most impact was the commissioning of studies, the results of which were passed along to the IAEA and were of assistance in preparing the 1985 edition of the IAEA Regulations. The more important studies were on the following subjects:

- dynamic crush test on small, lightweight Type B packagings
- additional requirements for special-form radioactive sources
- the standardization of transport certificates
- the quality assurance programme.

Other studies have focused on:

- Transport analysis from the standpoint of safety (criticality benchmark experiments, gamma and neutron-shielding benchmark experiments, package stowing tests, etc.);
- Transport analysis from the standpoint of health protection (personnel qualifications and emergency services).

As regards the latter subject, the CEC is at present preparing a communication to the Council of Ministers and the European Parliament relating to the following aspects:

- The training and briefing of personnel assigned to the transport of radioactive materials;
- The formulation of criteria for technical and medical assistance in the event of abnormal or accident situations.

The final reports on a considerable number of study contracts have been published in the Euratom collection (EUR reports). A list of these, with their reference numbers, is presented in Table I.

Concerning the rather more distant future, the CEC's attention and activities will be directed in particular toward the following subjects:

- A country-by-country analysis of the administrative and structural aspects of the transport of radioactive materials;
- An analysis of operational problems associated with the transport of radioactive materials, in particular intra-EEC transport.

In 1980, the CEC financed an analysis of Member States' laws and regulations governing the transport of radioactive materials. That document, which
has proved to be extremely useful, will serve as a reference until the national provisions have been amended in line with the 1985 edition of the IAEA Regulations. Generally speaking, this will be done in the Member States of the EEC by 1988. Apart from these regulatory texts, however, there is, in each country, a set of administrative provisions and/or practices which guide the industrial and commercial activities of carriers and are likely to affect the very structure of that sector. Before unjustified barriers to intra-EEC transport can be removed, they must first be identified. The following subjects could be dealt with:

- Authorization to act as a carrier
- Recommendations concerning the construction and equipment of road vehicles for the transport of radioactive materials (stowage platform) which can easily be decontaminated, the possibility of fitting shielding screens, electrical fire-fighting equipment, etc.
- Axle load problems
- Training of personnel assigned to transport
- Formalities in advance of transport (model of declaration for dispatch), notification, approval and execution of transport
- Traffic restrictions
- Transit formalities
- Nuclear insurance.

Whether or not the unjustified barriers to intra-EEC transport of radioactive materials are removed, such transport operations are conducted in accordance with procedures which, apart from the time necessary for the actual movement, require additional time and effort. There are those who believe that some of these procedures are likely to give rise to risk factors over and above those which are inherent in the movement of radioactive materials. Very often, when there is a public controversy on the subject of the cross-frontier transport of radioactive materials, it is the slowness of formalities and the risk factors that are the targets of criticisms generally levelled at questions of transport safety.

For anyone who is unconnected with the international transport of radioactive materials, it is almost impossible to obtain an objective picture of the situation. For this reason, the CEC is endeavouring, through study contracts, to have such a picture projected by the carriers themselves — or at least by some of them. The idea is to obtain actual and typical 'case histories' concerning various types of transport: fresh and irradiated fuel, plutonium and radioactive wastes, movement by road and rail, by air and sea, through and above the various countries and along their coasts. The description of a case would begin with the first step in the transport operation in question and end with the moment when the carrier is fully relieved of responsibility. This description would cover the types of procedures and formalities and the time needed for their completion, the types of movements and other operations or physical handling required,
including holdups, and the time spent on them. There would, of course, be no
question of attempting to cover all possible cases, but it would be necessary to
compile enough case histories to enable an attentive reader to obtain a realistic
idea of the magnitude of the operational problems that can arise in the
international transport of radioactive materials.

REFERENCES

[3] COMMISSION OF THE EUROPEAN COMMUNITIES, Inventory of the Regulations and
Administrative Procedures Concerning the Transport of Radioactive Materials in the
THE COMMERCIAL WASTE PROGRAMME IN THE UNITED STATES OF AMERICA

A major application of the IAEA Regulations in years to come

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United States Department of Energy,
Washington, D.C.,
United States of America

Abstract

THE COMMERCIAL WASTE PROGRAMME IN THE UNITED STATES OF AMERICA: A MAJOR APPLICATION OF THE IAEA REGULATIONS IN THE YEARS TO COME.

Consistent regulations that are consistently applied are of fundamental importance to a sound regulatory programme, which is the cornerstone for maintaining a safe and efficient transportation system. The Office of Civilian Radioactive Waste Management (OCRWM) within the United States Department of Energy seeks to maintain this attitude toward regulations in preparing to build and operate potentially the largest spent fuel and high level waste cask fleet in the world. Both the technical and institutional elements of the transportation system development programme are directed toward a goal of promoting regulatory consistency. The success of the International Atomic Energy Agency serves as encouragement to the OCRWM.

BACKGROUND

Currently, there are 100 commercial nuclear power reactors licensed to operate in the United States. The Energy Information Administration of the U.S. Department of Energy (DOE) projects that approximately 32 additional plants will start up in the foreseeable future. To date, these plants have generated approximately 12 000 MTU of spent nuclear fuel. It is anticipated that by the year 2020 the spent fuel inventory will reach approximately 106 000 MTU. Since the United States has no facilities for the permanent disposal of spent fuel, most of the material is now stored in pools at the reactor sites. Some commercial spent fuel was reprocessed, and the resultant high-level waste is stored at the reprocessing site at West Valley, New York.

Recognizing that this temporary storage of waste posed a potential health and safety threat and in the long term could cause the shutdown of some reactors because of a lack of
storage capacity, the Federal Government took upon itself the effort to find a solution to what clearly had become a national problem. In December 1982, Congress passed the Nuclear Waste Policy Act of 1982 (NWPA), the first comprehensive legislation for the management of commercial spent fuel and high level nuclear waste to be enacted in the United States. President Reagan signed the NWPA into law on January 7, 1983.

The Act requires the DOE to site, license, construct and operate geologic repositories for spent fuel and high-level waste. In addition, it directs that a detailed study of the need for and feasibility of one or more monitored retrievable storage (MRS) facilities be conducted and proposals for such a facility be submitted to Congress. The Act also provides for a Federally owned and operated system of interim storage facilities. This system, if needed, would have a total capacity of not more than 1,900 MTU of spent fuel. It would be used to store that spent fuel from civilian reactors for which adequate storage could not be reasonably provided by the owners.

The Act added another potential waste stream to the system by requiring the President to evaluate the use of one or more of the civilian repositories for disposal of defense high level waste. This waste is a by-product of the production of nuclear weapons material and of the reprocessing of fuel from naval reactors. In April 1985, the President approved the recommendation of the Secretary of Energy to use the civilian repository for disposal of defense high level waste. Finally, the Act makes the DOE responsible for the transportation of spent fuel and high level waste from its storage location to the Federal facilities operated under the Act.

THE WASTE MANAGEMENT SYSTEM

The Office of Civilian Radioactive Waste Management (OCRWM) was established within the DOE to manage the overall disposal program. One of the first achievements of the office was the publication of a Mission Plan [1] which outlined the strategy for meeting the requirements set forth in the NWPA. Central to this strategy are two approaches to the implementation of the Act: the Authorized Plan and the Improved Performance Plan.

The Authorized Plan entails the construction and operation of a geologic repository (plus a second repository if its construction is authorized by Congress) and development of a transportation system for movement of waste (Fig. 1). If required, Federal interim storage would be added to the system.
The Improved Performance Plan adds an MRS facility into the system (Fig. 2). Analyses have shown that the addition of such a facility has the potential for

- Improving transportation efficiency.
- Increasing reliability and flexibility of the system.
Improving confidence in the DOE's ability to meet acceptance schedules.

Reducing needed at-reactor storage.

Focusing repository licensing efforts on demonstrating long term isolation capabilities.

Common to both approaches is a transportation system to move the spent fuel and high level waste. In the case of the Authorized Plan, the transportation system would be designed to move waste by truck, rail and barge to a repository. Under the Improved Performance Plan, all modes of transportation would be used to ship waste to the MRS facility; the subsequent trip from the MRS facility to the repository is envisioned to be made by dedicated train. No matter which plan is finally implemented, a safe, efficient, cost-effective transportation system is the key to the success of the program.

TRANSPORTATION SYSTEM DEVELOPMENT

The NWPA directs the OCRWM to take title to the commercial spent fuel or high level waste at the reactor or other sites where it is produced or stored. This means that the OCRWM is responsible for providing the shipping casks and transporting the material to a Federal waste management facility. The Act also directs the OCRWM to utilize private industry to the maximum extent possible in providing this transportation function.

The Mission Plan delineates two major tasks which the OCRWM must undertake in order to have a viable transportation system in place at the time a repository or an MRS facility is ready to accept waste. The first of these tasks is the technical development and acquisition of hardware such as casks and related equipment. Also included in this first task is the development of a management concept for the transportation system and the procurement of the required services. The second task is to ensure that institutional concerns are addressed in such a manner so as to allow the system to develop and operate effectively.

The responsibility for accomplishment of these two tasks has been assigned to the Office of Storage and Transportation Systems (OSTS) of the OCRWM. The OSTS is being aided in this effort by three DOE operations offices and their support contractors (which include many of the national laboratories). Several programmatic documents have been issued or are in the process of being written which provide the guidance for accomplishing the tasks. The Transportation Business Plan [2] outlines business strategies for procuring shipping casks and
transportation support services and describes legislation and policies governing transportation under the NWPA. The Transportation Institutional Plan [3] explains how the OCRWM will interact with all interested parties in the development of the transportation system. In 1987, these two plans will be combined with an Operations Plan to form an overall, comprehensive Transportation Plan.

TECHNICAL DEVELOPMENT PROGRAM

The goal of the technical development program is to provide and operate a transportation system which could be comprised of up to 300 casks (for shipping by truck and rail/barge and for storage/transport) over the lifetime of the first repository program. As previously mentioned, this effort is guided by the Transportation Business Plan which divides the system development and acquisition process into two major phases.

Phase I is primarily concerned with the development of cask prototypes and is subdivided into three distinct initiatives. The first initiative is to produce cask designs certified by the Nuclear Regulatory Commission (NRC) and cask prototypes capable of transporting spent fuel from reactors to a repository or to an MRS facility (if such a facility is authorized by Congress). It is expected that multiple contracts for cask designs and prototypes will be let with industry in 1986.

The second initiative will involve the design and development of equipment to move waste from an MRS facility to a repository. Obviously, this task will be undertaken only if the MRS facility is approved.

The third initiative will entail developing casks to handle 'non-standard' spent fuel and non-fuel components. This may be achieved by either modifying existing casks or those developed in the first initiative, or by designing entirely new casks.

In keeping with a Procedural Agreement [4] between the DOE and the NRC, all casks used to transport commercial spent fuel and high level waste from NRC-licensed generation facilities to NRC-licensed waste management facilities will have a certificate of compliance from the NRC. A comprehensive cask-testing program is being established to ensure compliance with all safety requirements. Engineering tests will be required on all casks to provide data to characterize material
performance or the performance of cask components. Design verification testing of scale models or full-scale component sections will be performed to aid in certification. Acceptance tests and operational tests will be performed using prototypes. If deemed appropriate, some cask prototypes may be subjected to a full-scale regulatory test and/or confirmatory demonstrations.

Phase II of the technical program is concerned with developing a management structure for the transportation system, procuring the management services and related technical services required to operate the system and procuring the cask fleet. A number of studies will soon be undertaken to determine how best to manage and operate the system. No matter what management and operational structure is decided upon, private industry will be used to the fullest extent possible. In addition, the OCRWM will rigorously carry out its responsibilities to ensure that the transportation function is conducted safely and efficiently.

INSTITUTIONAL PROGRAM

Although the technical element of the transportation program will provide a significant challenge, the more difficult task may be to create a cooperative institutional environment. The OCRWM recognizes that active participation by all parties interested in the waste management program, regardless of their position on the issues, is essential if the transportation functions supporting the NWPA program are to be carried out successfully. The method of interaction between the OCRWM and the network of interested parties is outlined in the Transportation Institutional Plan. The plan discusses the policy of the OCRWM for establishing the transportation system, discusses the roles of the various participants in the process (the network), suggests mechanisms for interaction between the members of the network and provides a framework for identifying and resolving issues related to development and operation of the transportation system. Appendix A of the plan contains detailed discussion of 16 issues which have been raised to date.

A great deal of progress has been made in attaining the goals of the institutional program. In November 1985, the OCRWM held a national workshop to examine a variety of issues, including cask safety, routing, emergency response, inspection and enforcement and liability. The workshop was attended by participants representing Federal, State, Tribal and local governments; utilities; the transportation industry; cask
manufacturers; and various citizens groups. The workshop provided a clearer understanding of all of the issues and many suggestions on methods to resolve them. The OCRWM is now actively pursuing a program to implement the suggestions emanating from the workshop by working with regional groups representing State and local governments, Tribal groups, and professional organizations having expertise in specific issue areas. While the process may appear to be somewhat cumbersome, it is designed to ensure that all interested parties have the maximum opportunity to participate in the resolution of transportation issues.

APPLICATION OF THE IAEA REGULATIONS

The OCRWM is encouraged by the success of the International Atomic Energy Agency (IAEA) in developing consistently applied regulations that transcend national interests to allow relatively unimpeded shipment of radioactive materials on an international level.

The OCRWM hopes to have similar success, through its institutional program, in fostering consistent regulations and operating requirements across State, Tribal, and local boundaries. The OCRWM recognizes and accepts a basic tenet of the IAEA, found in Paragraph 109 of IAEA Safety Series No. 6, that provides "Except as necessary for solely domestic purposes, such national regulations should not conflict with these (IAEA) Regulations." In many respects, the United States is a microcosm of the world, particularly for transportation-related activities. The United States is a republic of 50 States, each of which shares in the basic responsibility to protect the safety of its citizens. The States occasionally develop regulations that are mutually contradictory or are inconsistent with Federal regulations. Such inconsistency causes potentially severe technical and operational problems within the United States. As previously stated, these institutional problems are the most severe challenge to our program.

Consequently, from an institutional as well as technical basis, the OCRWM reaffirms its belief that consistent regulations and, particularly, consistent application of such regulations, are fundamental to a good and functional regulatory structure. It should be noted, however, that the OCRWM does not have the authority to unilaterally adopt IAEA regulations for the United States. Any such action, which would have to be taken by the appropriate agency within the United States, would require a formal national rulemaking process. Nevertheless, OCRWM intends to voluntarily comply
with IAEA Safety Series No. 6 where possible within the restrictions of national regulations. Such voluntary measures will include compliance with principles for the reduction of radiation exposures to levels "as low as reasonably achievable" (ALARA) and with the increased submergence requirements for spent-fuel casks. As a matter of fact, the entire transportation system is currently being planned according to ALARA principles with a preliminary system-wide ALARA study to be completed prior to 1988. The performance specification of 200 m submergence is included in the request for proposals to build our first set of prototype casks.

In selected cases where the process of achieving compliance with existing regulations is not clearly identified or not consistently defined, the OCRWM will pursue discussions with the appropriate regulatory agency. As previously noted, the DOE already has in place a Procedural Agreement with the NRC. In addition, the OCRWM has developed a Memorandum of Understanding with the U.S. Department of Transportation (DOT) [5]. The agreement with the NRC is specific to technical issues that arise from cask development. The OCRWM has an active technical issue resolution program to pursue general issues related to cask development. The range of issues is wide and includes issues that strike at the heart of inconsistent application of regulations and approaches to proving compliance. The question of how materials can be qualified as structural components in cask construction has a high priority for resolution. The OCRWM recognizes that Certificates of Compliance have been issued in IAEA Member States for casks constructed from advanced materials, but that reciprocal certificates have not been issued to date for the use of these casks within the United States. Through rigorous interaction with the NRC, the OCRWM intends to resolve such issues early in our cask development program.

In some cases, voluntary compliance with IAEA regulations may not be possible because of the institutional concerns of States, Indian Tribes, local governments, and the public. For example, as a result of State and local concerns regarding routing of radioactive materials shipments, a plethora of regulations were generated which generally were not consistent from jurisdiction to jurisdiction. The DOT entered into the fray and issued a regulation for routing of highway shipments of radioactive material; the regulation has withstood legal challenge and provides a uniform routing procedure across the United States. This routing requirement, though not inconsistent with, is contrary to the statement in IAEA Safety Series No. 6 that provides "it is not generally necessary to recommend routing restriction." However, other efforts of the OCRWM to resolve institutional issues may actually promote
compliance with IAEA regulations. The OCRWM's compliance with the IAEA pool fire test requirement could in fact partially resolve an institutional issue regarding the adequacy of the previous thermal test.

Specification of international regulations does add an extra dimension of authority that can be beneficial in resolving institutional as well as technical issues. Nevertheless, consistency remains the greatest benefit of internationally recognized regulations, and the OCRWM is committed to a program that promotes consistency both internationally and nationally.

REFERENCES


Abstract

WORLDWIDE APPLICATION OF IAEA SAFETY SERIES No. 6: REGULATIONS FOR THE SAFE TRANSPORT OF RADIOACTIVE MATERIAL, 1985 EDITION.

The International Atomic Energy Agency has recently issued the 1985 Edition of Safety Series No. 6. The changes made in the Regulations, the plans of Member States and international organizations for adoption and the Agency's own plans for future revisions of these Regulations are reviewed.

INTRODUCTION

Based upon data provided by 52 Member States of the International Atomic Energy Agency (IAEA) in the early 1980s, it is estimated that from 18 million to 38 million package shipments of radioactive material currently occur each year worldwide (see Ref. [1]). These shipments cover all types of materials, in many different forms of packagings and are carried by all modes of transport. When these data are extrapolated over the period of time for which there have been significant shipments of these materials, approximately 400 million to 800 million package shipments of radioactive material have occurred over the past 40 years. Furthermore, the safety record for these shipments has been exceptional. During 1985 the IAEA performed an assessment of the radiological impact of the transport of radioactive materials. This assessment, which was performed with the assistance of experts from many countries, indicated that exposures of most workers and of the public in the normal transport of such materials are low. Only in a very limited number of controlled cases do transport workers receive doses which are a significant fraction of
applicable limits. Finally, it was concluded from the information available that there has never been an accident or incident involving radioactive material transport which has led to the significant exposure of a member of the public.

This excellent record is due primarily to the regulations used by individual countries and international organizations to control the packaging, handling, storage and shipment of such consignments. These regulations are generally based, directly or indirectly, on IAEA Safety Series No. 6: Regulations for the Safe Transport of Radioactive Material. For convenience, Safety Series No. 6 will be referred to hereafter as the Regulations.

This paper will first provide a brief background and history of the Regulations. The latest (1985) Edition will then be overviewed in detail, followed by a review of the method of implementation by Member States, and their plans for adoption, of the 1985 Edition of the Regulations and actions planned by key international organizations. Finally, future plans for continuously reviewing and periodically updating the Regulations will be summarized.

BACKGROUND AND HISTORY

To assure safety during the transport, handling and storage of radioactive materials, it was recognized decades ago that a very strict set of standards — developed and recognized on an international level — would be required. The concept of international acceptance was recognized as being vital, since transport is usually the only aspect of any controlled radioactive material related activity in which the radioactive material itself may directly cross international borders. Indeed, even for transport within a country, international (out-of-country) carriers or packagings can be involved.

Thus, shortly after the formation of the IAEA in 1957, it was given the task of developing safety rules for the transport of radioactive materials covering all modes of transport. The Agency’s activities in this area are currently carried out by the Radiation Protection Section of the Division of Nuclear Safety. Consequently, the focus of these activities is — and always has been — on ensuring safety. Based upon existing good practices in the transport of hazardous goods and the few simple regulations for radioactive materials already in effect, the IAEA began to develop a comprehensive set of rules in 1958.

As a result, and with the assistance of experts from around the world, the first edition of the Regulations was published in 1961. In addition to being applied directly to the Agency’s operations and to Agency-supported activities in Member States, they were (and still are) “recommended to Member States and to International Organizations concerned as a basis for national and international transport regulations”.

Revised editions of the Regulations, which took into account developments in technology and shipping practices, were issued in 1965, 1967 and 1973 (with an
amended 1973 Edition being also issued in 1979), and most recently in 1985. In addition the Agency has, over the years, produced companion documents giving background information on and supporting the adoption and implementation of the Regulations, including Safety Series No. 37, Advisory Material for the Application of the IAEA Transport Regulations (first published in 1973); and a much earlier document, Safety Series No. 7, Notes on Certain Aspects of the Regulations (published in 1961). Both Safety Series No. 7 and Safety Series No. 37 are being revised to reflect the 1985 Edition of the Regulations, where Safety Series No. 7 will have the new title, Explanatory Material for the IAEA Regulations for the Safe Transport of Radioactive Material (1985 Edition).

THE 1985 EDITION OF THE REGULATIONS

The process of producing the latest edition of the Regulations was initiated in 1979 by the IAEA at the request of its special advisory body, the Standing Advisory Group for the Safe Transport of Radioactive Materials (SAGSTRAM). It was recognized that, despite the excellent safety record in the transport of radioactive materials, periodic updating and revision was necessary to allow the Regulations to stay current with advances in technology, with changes in the needs of carriers and regulators, and with the evolving international standards for radiation protection.

This revision was performed over a period of six years with the extensive co-operation of IAEA Member States and the relevant international organizations. It involved the convening at the Agency of a significant number of consultants meetings, technical committee meetings and advisory group meetings. Approximately 150 experts from 22 Member States and 12 international organizations participated. The final draft of the revised Regulations was considered and approved by the IAEA's Board of Governors in September 1984, and the new edition was published in 1985 in English, French, Russian and Spanish.

The process which culminated in the 1985 Edition of the Regulations included a comprehensive and detailed review of the 1973 (As Amended) Edition of the Regulations and of proposed changes in these requirements and the justification for these changes. There were a variety of justifications which were accepted during the revision, ranging from quantified modelling to more subjective arguments, such as practical past experience in applying the Regulations and the need for improved clarity of presentation. The latter consideration resulted in the most apparent changes to the Regulations, recognized by a complete revision of the structure and presentation of Safety Series No. 6.

In order to provide a more 'user-friendly' document, the Regulations were restructured to present the most basic information first and to build on this information to present the complete set of requirements. Consequently, an expanded set of definitions are presented in Section I, basic principles established in Section II, and the activity limiting $A_1/A_2$ values and calculative techniques presented in
Section III. The detailed requirements in Sections IV-VII build on this base of information. Defined terms are presented in bold-faced type for the reader's convenience and to ensure that these terms are only interpreted in accordance with their definition as presented in the Regulations. In several instances long and detailed textual requirements have been presented in tabular form, with a significant improvement in clarity. Finally, a detailed index was added to aid the user.

Radiation Protection

The Agency's Basic Safety Standards for Radiation Protection, Safety Series No. 9 (1982 Edition), are incorporated into the transport regulations. Some principles, such as keeping radiation exposures as low as reasonably achievable, are specifically mentioned and action levels are prescribed for various levels of individual occupational exposure. Additionally, guidance is given for the derivation of segregation distance requirements, since the establishment of specific requirements from a radiation protection standpoint is within the domain of international transport organizations.

Quality Assurance and Compliance Assurance

Increased emphasis is placed on both quality assurance and compliance assurance. Quality assurance is now required to cover the design, manufacture, testing, documentation, use, maintenance and inspection of all packages. Package designs which require competent authority approval now have their approval made contingent on the adequacy of the applicant's quality assurance programme.

The responsibility for maintaining an adequate compliance assurance programme is now placed squarely on the competent authorities. They must ensure that the provisions of the Regulations are being met in actual practice. Both quality assurance and compliance assurance are given visibility and emphasis by their prominent location in Section II of the Regulations. (A detailed guide on quality assurance has been developed and will be included as an appendix in the new edition of Safety Series No. 37. Also, a guide for competent authority implementation of the Regulations is being developed.)

Type B and Fissile Packages

Several significant changes were made to the requirements applicable to Type B and fissile package design and testing. Some of these changes will make it easier for designers to demonstrate compliance with the Regulations, while other changes may require modification or abandonment of some package designs. The most significant changes are:
(1) Dynamic crush test. Type B and fissile packages which are of lightweight (less than 500 kg) and low density (less than 1000 kg/m²) design for transporting normal form contents with activity exceeding 1000 A₂ must be subjected to a crush test. The crush test consists of dropping a 500 kg mild steel plate (1 m by 1 m) onto the package specimen from a height of 9 m. Crush forces have been identified as a potentially significant accident environment and this requirement will ensure a measure of survivability to crush forces for susceptible package designs.

(2) Deep submergence test. Packages designed to transport more than 37 PBq (10⁶ Ci) of irradiated fuel are required to be able to withstand submergence in water at a depth of 200 m without rupture of the containment system. This will help ensure recovery and limit the environmental consequences of a spent fuel flask being sunk in rivers and in lakes or on continental shelf areas of this depth.

(3) Release limits. The allowable activity release limits for Type B(U) and Type B(M) packages following the test conditions were made the same. This will make it easier to demonstrate compliance with the post-accident condition leak rate requirement, while ensuring appropriate control of the package contents.

(4) "Grandfathering" of previous approvals. Packages designed and approved in accordance with the 1967 or 1973 editions of Safety Series No. 6 may continue to be used subject to certain conditions. Packages approved to the 1967 Regulations and, after 1990, packages approved to the 1973 Regulations are subject to multilateral approval.

Radioactive Materials Classification

The requirements governing Type A package content limits and the prescriptions for lower activity materials and objects are significantly revised as follows:

(1) Type A package content limits. A new modelling system has been incorporated for determining individual exposures to Type A package contents following a breach of the packaging. New exposure pathways were built into the model to account for such phenomena as skin contamination. Additionally, the latest International Commission on Radiological Protection metabolic data were incorporated into the model. The changes resulted in some significant modifications of the A₁ and A₂ values, some of which increased and some of which decreased. Most significantly decreased are the values for those radionuclides which when deposited on the skin would result in a significant beta dose to the skin.

(2) Low Specific Activity (LSA) materials are redefined into three groups, LSA-I, LSA-II and LSA-III, while limited hazard contaminated objects are classified as Surface Contaminated Objects (SCO). The previous classification of low level solid has been removed, with the materials it formerly encompassed being divided into LSA-III and SCO categories. The most significant change in this
area is the establishment of a radiation level limit on LSA and SCO materials. The quantity of LSA or SCO in a package must be limited to the extent that the external radiation level at 3 m from the unshielded material is less than 10 mSv/h. This change establishes an upper bound on the radiation hazard these materials can present even if package shielding is lost.

**Industrial Packaging**

The previous packaging category known as ‘strong industrial packaging’ has been redefined. Specific test conditions are assigned to each of the three levels within the new ‘Industrial Package’ category, IP-1, IP-2 and IP-3. The industrial packaging requirements range from only the general requirements for all packaging (IP-1) to essentially a Type A package (IP-3). This finer specification of packaging integrity allows a better alignment between the potential hazards of LSA and SCO and their packaging requirements.

**Fissile Material**

A significant simplification has been made in the classification of fissile material packages and the specification of appropriate degrees of control to be exercised over these packages. The previous categories of Fissile Classes I, II and III have been eliminated. Control of the criticality potential of fissile materials is now based on the transport indexes of the packages containing the material. Certain exceptions are still maintained for small quantities and other extremely criticality-safe configurations, but when control over the loading and storage of packages is required, this is accomplished by transport index summation.

Packages which under the old system would be Fissile Class I are now handled solely on the basis of their transport index as determined by the radiation level. Packages which were Fissile Classes II and III under the old system must meet the old Fissile Class II package criticality criteria and are now generally limited to 50 transport indexes for exclusive use and 100 transport indexes for exclusive use shipments.

**Contamination Limits for Excepted Packages**

Packages which contain small amounts of radioactive materials and instruments or articles which meet certain criteria are now known as “excepted packages”. These packages are excepted from many of the detailed requirements because of their very limited hazard during transport. For example, they are excepted from package labelling and documentation.

The allowable non-fixed external surface contamination on these packages has been set at a level 10 times lower than that for other packages. Since little control is exercised over these packages during transport, it was decided to limit the non-
fixed contamination to values which would be appropriate for the freedom allowed for these packages in the transport system.

Modal Requirements

Generally, the requirements for packages in the Regulations have been mode independent, although modal-specific requirements have been specified previously. Because the international air mode authorities, the International Civil Aviation Organization (ICAO) and the International Air Transport Association (IATA) impose requirements which specifically apply to air carriage, these requirements are included in the 1985 Edition of the Regulations. They relate to maximum accessible package surface temperatures (50°C at an ambient temperature of 38°C); containment integrity requirements for ambient temperatures ranging from -40°C to +55°C; and for packages containing liquid radioactive material, a containment integrity requirement for a pressure differential of 95 kPa.

SI Units of Measurement

The SI system of measurements has been incorporated into Safety Series No. 6 as the legal units of measurement. Customary units are still given in parentheses in order to aid in the transition to the new units. The customary unit values have been rounded so that any error from using them will be on the conservative side.

It is anticipated that some period of transition will be needed to accomplish the changeover to SI units. The international transport organizations will need to work co-operatively to ease this transition, while still striving to meet target dates for the adoption of modal regulations based on the 1985 IAEA transport requirements.

PROCEDURES USED BY MEMBER STATES FOR ADOPTING AND IMPLEMENTING THE REGULATIONS

Member States of the IAEA implement international agreements, regulations and recommendations for regulatory control of the transport of radioactive material in a variety of ways. Each country must act within its own statutory requirements. The IAEA, working with the individual Member States, undertook in 1984 to examine the manner in which domestic, import, export and through-country shipments of radioactive materials are controlled and regulated worldwide. The information to be examined was collected using a questionnaire. By the end of January 1986, completed questionnaires had been received from 52 Member States.

The Member States which responded are: Argentina, Austria, Bangladesh, Belgium, Bolivia, Brazil, Bulgaria, Canada, Chile, China, Colombia, Czechoslovakia, Denmark, Ecuador, Egypt, Finland, France, German Democratic Republic, Federal Republic of Germany, Greece, Hungary, India, Indonesia, Israel,
TABLE I. LIST OF TRANSPORT DOCUMENTS OF INTERNATIONAL ORGANIZATIONS

<table>
<thead>
<tr>
<th>International organizations</th>
</tr>
</thead>
<tbody>
<tr>
<td>Technical Instructions for the Safe Transport of Dangerous Goods by Air (TI), International Civil Aviation Organization (ICAO), Montreal.</td>
</tr>
<tr>
<td>Dangerous Goods Regulations, International Air Transport Association (IATA), Montreal.</td>
</tr>
<tr>
<td>Universal Postal Convention of Rio de Janeiro 1979, Universal Postal Union, Bern.</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Regional international organizations</th>
</tr>
</thead>
<tbody>
<tr>
<td>European Agreement Concerning the International Carriage of Dangerous Goods by Road (ADR) and Protocol of Signature, United Nations Economic Commission for Europe (ECE), Geneva (1957).</td>
</tr>
<tr>
<td>Règlement international concernant le transport des marchandises dangereuses par chemins de fer RID, Convention internationale concernant le transport des marchandises par chemins de fer (CIM), Office central des transports internationaux par chemins de fer (OCTI), Berne.</td>
</tr>
<tr>
<td>Regulations for the Safe Transport of Spent Nuclear Fuel from Nuclear Power Plants of CMEA Member Countries — Transport by Rail, Council for Mutual Economic Assistance (CMEA), Moscow.</td>
</tr>
</tbody>
</table>

Italy, Japan, Malaysia, Mauritius, Mexico, Monaco, Netherlands, New Zealand, Norway, Pakistan, Peru, Philippines, Poland, Portugal, Romania, Singapore, South Africa, Spain, Sweden, Switzerland, Syrian Arab Republic, Tanzania, Turkey, United Kingdom, United States of America, Uruguay, Venezuela and Zambia.

The results of the examination indicate the important role international organizations play in the transport of radioactive materials. All the Member States involved in this examination regulate the transport of radioactive material within their country.
on the basis of international agreements, regulations and recommendations. Safety Series No. 6 is the ultimately controlling document, since it serves as the basis for the radioactive material portions of other international transport documents (Table I) and since it is made directly binding in the regulations of many countries. The data which follow are up to date as of 1 June 1986.

INTERNATIONAL AGREEMENTS, REGULATIONS AND RECOMMENDATIONS AS THE BASES FOR REGULATING TRANSPORT

Domestic transport

Information on regulating domestic transport of radioactive material was received from 52 Member States. In every case the domestic regulations are based on international agreements, regulations and recommendations. The following was concluded:

1. 13.5% of the Member States involved regulate only on the basis of the IAEA Regulations.
2. 21.2% of the Member States involved regulate only on the basis of international documents other than the IAEA Regulations.
3. 65.4% of the Member States involved regulate on the basis of the IAEA Regulations and other international documents.

The relevant international transport documents and the areas of their application are listed in Table II, which also indicates the percentages of Member States using the document, or documents, in question as the basis for their domestic transport regulations.

International transport

Information on regulating international transport of radioactive material was also received from 52 Member States. The following was concluded:

1. 11.5% of the Member States involved regulate only on the basis of the IAEA Regulations.
2. 23.1% of the Member States involved regulate only on the basis of international documents other than the IAEA Regulations.
3. 63.5% of the Member States involved regulate on the basis of the IAEA Regulations and other international documents.
4. 1.9% of the Member States (one country) indicated that it regulates using neither the IAEA Regulations nor any other international documents.

Table II also gives the percentages of the Member States using the listed international document or documents as the basis for regulating their international shipments.
## TABLE II. ADOPTION OF THE INTERNATIONAL DOCUMENTS BY 52 MEMBER STATES (DOMESTIC AND INTERNATIONAL TRANSPORT)

<table>
<thead>
<tr>
<th>International organization and form of document</th>
<th>Area of application</th>
<th>Member States which regulate domestic transport using this document (%)</th>
<th>Member States which regulate international transport using this document (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>IAEA: Safety Series No. 6&lt;sup&gt;a&lt;/sup&gt;</td>
<td>Worldwide/All modes</td>
<td>78.8</td>
<td>75.0</td>
</tr>
<tr>
<td>UN/ECOSOC: Recommendations</td>
<td>Worldwide/All modes</td>
<td>15.4</td>
<td>17.3</td>
</tr>
<tr>
<td>Universal Postal Union: Acts</td>
<td>Worldwide/All modes</td>
<td>40.4</td>
<td>40.4</td>
</tr>
<tr>
<td>ICAO: Technical Instructions&lt;sup&gt;b&lt;/sup&gt;</td>
<td>Worldwide/Air mode</td>
<td>55.8</td>
<td>57.7</td>
</tr>
<tr>
<td>IATA: Regulations&lt;sup&gt;b&lt;/sup&gt;</td>
<td>Worldwide/Air mode</td>
<td>59.6</td>
<td>59.6</td>
</tr>
<tr>
<td>IMO: IMDG Code</td>
<td>Worldwide/Sea mode</td>
<td>63.5</td>
<td>65.4</td>
</tr>
<tr>
<td>ECE: Agreement (ADR)</td>
<td>Regional/Road mode</td>
<td>30.8</td>
<td>44.2</td>
</tr>
<tr>
<td>OCTI: Regulations (RID)</td>
<td>Regional/Rail mode</td>
<td>40.4</td>
<td>44.2</td>
</tr>
<tr>
<td>ECE: Agreement (ADN)</td>
<td>Regional/Inland waterway mode</td>
<td>5.8</td>
<td>5.8</td>
</tr>
<tr>
<td>CCNR: Agreement (ADNR)</td>
<td>Regional/Rhine river</td>
<td>5.8</td>
<td>9.6</td>
</tr>
<tr>
<td>CMEA: Regulations</td>
<td>Regional/Rail mode</td>
<td>3.8</td>
<td>5.8</td>
</tr>
<tr>
<td>OSZhD: Agreement (SMGS)</td>
<td>Regional/Rail mode</td>
<td>3.8</td>
<td>5.8</td>
</tr>
</tbody>
</table>

<sup>a</sup> Regulations for the IAEA and its support activities, recommendations for all other activities.

<sup>b</sup> 80.8% of countries surveyed use either ICAO Technical Instructions, IATA Regulations, or both in regulating the air carriage of radioactive materials.

### ADOPTION OF THE IAEA REGULATIONS AS THE BASIS FOR NATIONAL REGULATIONS

#### Domestic transport

Detailed data on the adoption of the Regulations were received from 44 Member States. Three of these States, however, adopt the Regulations only via other international documents.

Table III indicates which edition of the Regulations is used as a basis for the domestic regulations of the 41 Member States which directly adopt the Regulations.
TABLE III. DIRECT ADOPTION OF THE DIFFERENT EDITIONS OF THE REGULATIONS BY MEMBER STATES, DOMESTIC AND INTERNATIONAL TRANSPORT ORGANIZATIONS

<table>
<thead>
<tr>
<th>Edition of the Regulations</th>
<th>Domestic transport of 41 Member States (%)</th>
<th>International transport of 39 Member States (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1964 Revised Edition</td>
<td>—</td>
<td>—</td>
</tr>
<tr>
<td>1967 Edition</td>
<td>—</td>
<td>2.6</td>
</tr>
<tr>
<td>1973 Revised Edition</td>
<td>34.1</td>
<td>38.5</td>
</tr>
<tr>
<td>(published in 1973)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1973 Revised Edition (As Amended) (published in 1979)</td>
<td>63.4</td>
<td>56.4</td>
</tr>
<tr>
<td>1985 Edition</td>
<td>2.4</td>
<td>2.6</td>
</tr>
</tbody>
</table>

PLANS OF MEMBER STATES FOR ADOPTION OF THE 1985 EDITION OF THE REGULATIONS

Information on plans to adopt the 1985 Edition of the Regulations was received from 42 Member States. Table IV summarizes these plans, from which it can be seen that more than 80% of the Member States plan to adopt the 1985 Edition by 1989. In addition, the data showed that 88.1% planned to adopt the Regulations in the same way as they do now.

TABLE IV. THE PLANNED YEARS OF ADOPTION OF THE 1985 EDITION OF THE REGULATIONS

<table>
<thead>
<tr>
<th>Year</th>
<th>42 States involved (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1986</td>
<td>28.6</td>
</tr>
<tr>
<td>1987</td>
<td>23.8</td>
</tr>
<tr>
<td>1988</td>
<td>19.0</td>
</tr>
<tr>
<td>1989</td>
<td>9.5</td>
</tr>
<tr>
<td>1990 or later</td>
<td>19.0</td>
</tr>
</tbody>
</table>
Some Member States have adopted more than one edition of the Regulations, e.g. different editions for different modes of transport. In these cases the percentages are calculated taking into account the newest edition only. One Member State still utilizes the 1964 Revised Edition and three Member States still utilize the 1967 Edition for part of their regulations.

The procedures used for adopting IAEA Regulations for domestic transport are summarized as follows:

1. 36.6% of Member States involved have adopted the Regulations by direct reference.
2. 46.3% have adopted the Regulations in principle but a different format or different words have been used.
3. 17.1% used other means, e.g. decisions of a competent authority and a combination of the above procedures.

**International transport**

Detailed data on the adoption of the Regulations were received from 41 Member States. Three of these States, however, adopt the Regulations only via other international documents.

Table III also indicates in summary form which edition of the Regulations is used as a basis for the international regulations of the 38 Member States which directly adopt the Regulations.

Some Member States have adopted more than one edition of the IAEA Regulations, e.g. different editions for different modes of transport. In these cases the percentages are calculated taking into account the newest edition only. For international transport, two Member States still utilize the 1967 Edition for part of their regulations.

The procedures used for adoption and for international transport are summarized as follows:

1. 43.6% of Member States involved have adopted the Regulations by direct reference.
2. 41.0% in principle.
3. 15.4% by other means.

**ADOPTION BY INTERNATIONAL ORGANIZATIONS OF THE 1985 EDITION OF THE REGULATIONS**

Since the 1985 Edition of the Regulations was finalized, the IAEA has worked closely with various international organizations to encourage accurate, complete and timely implementation of the Regulations into their regulatory documents (see Table I).
Changes reflecting the 1985 Edition of the Regulations have already been implemented in the fourth revised edition of the United Nations Recommendations on the Transport of Dangerous Goods (ST/SG/AC.10/1/Rev. 4) issued in 1986. Drafts of regulatory documents from ICAO, IMO, ECE and OCTI that reflect the 1985 Edition of the Regulations have been produced. Currently it appears that these four organizations will put the 1985 Edition into force on 1 January 1990 by issuing revised regulatory documents. If these four organizations issue their updated regulatory documents simultaneously, the harmonized international implementation of the Regulations worldwide will certainly be enhanced.

PROCEDURES FOR FUTURE REVISIONS OF THE REGULATIONS

New procedures have been developed at the IAEA, with the advice and concurrence of Member States, which will provide for continuous review and periodic revision of Safety Series No. 6. These procedures will lead to the following:

(1) Formal supplements to Safety Series No. 6, incorporating editorial corrections, minor corrections and changes of detail which do not affect the structure of the Regulations will be issued every two years beginning in 1986.

(2) Supplements, beginning in 1988, will also update the supporting documents to Safety Series No. 6, namely Safety Series No. 7 (the explanatory document), Safety Series No. 37 (the advisory document) and the Schedules.

(3) Full revision of these four documents will occur approximately every 10 years (subject to SAGSTRAM guidance), with the next revision expected in about 1994–1996.

Specifically, the process being considered will involve:

(a) Formal, periodic requests by the IAEA to Member States and cognizant international organizations for proposed amendments to the Regulations and its supporting documents.

(b) Receipt by the IAEA, through established official channels, of proposed amendments in a standard format where the proposals would include an identification of the problem, proposed changes, justification and assessment of the impact of the proposed changes and an assessment of the priority which should be given to the proposal.

(c) Circulation by the IAEA of the proposals for comment.

(d) Consideration of proposed changes and comments thereon by regularly scheduled review/revision panels convened by the IAEA.

(e) Oversight of the process and general guidance to the IAEA by periodic meetings of SAGSTRAM.

(f) Promulgation of minor changes and changes in detail, under the authority of the Director General of the IAEA, as Supplements to Safety Series Nos 6, 7 and 37 and to the Schedules once every two years.
Promulgation of major changes to Safety Series No. 6, subject to approval by the IAEA Board of Governors, in the form of new editions to these four documents approximately once every ten years.

Parts of this procedure have already been implemented, and a 1986 Supplement to Safety Series No. 6 has been issued providing minor changes (correcting errors, etc.) and two changes of detail.

It is generally felt that the procedure outlined above will assist in maintaining the Regulations current with needs and for keeping the supporting documents current with the Regulations.

CONCLUSION

The transport of radioactive material has proven over the years to be safe, principally as a result of sound, internationally recognized regulations. The IAEA Regulations for the Safe Transport of Radioactive Material (Safety Series No. 6) serve as the precursor for essentially all other documents and international regulations in this area. With the emergence of the 1985 Edition of Safety Series No. 6, of forthcoming supportive guidance documents and the worldwide adoption and implementation of these principles, the excellent safety record for, and the facilitation of the transport of radioactive material should continue on into the future. Regulators are encouraged to adopt the new edition of the Regulations into their domestic or organizational frameworks by 1990.

Finally, looking to the future, the IAEA, with the assistance of experts from around the world, has developed a procedure for continuing further developments in the Regulations which should meet the needs of regulators, consignors and carriers alike and will thereby provide for the "protection of health and minimization of damage to life and property", as mandated by the IAEA's Statute.

REFERENCE

QUALITY ASSURANCE CONCEPT IN THE FIELD OF LOW AND INTERMEDIATE RADIOACTIVE WASTE PACKAGING

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Würenlingen, Switzerland

Abstract

QUALITY ASSURANCE CONCEPT IN THE FIELD OF LOW AND INTERMEDIATE RADIOACTIVE WASTE PACKAGING.

Using quality assurance measurements, it is possible in connection with radioactive waste packaging to guarantee that a process and the final product fulfill all requirements for the protection of human beings and the environment. The requirements stipulated in regulations, guidelines and standards are included in the specifications for treatment, packaging, transport, handling and storage. Compliance with the requirements is demonstrated by means of type tests. The type test requirements are then the obligatory minimum in the specifications for the manufacture of all standard packages. Compliance with the specifications can be guaranteed by means of an integral quality assurance programme which includes checks of packaging and radioactive waste materials supplied, the treatment and conditioning, the waste matrix and the final product. Quality control is carried out by the manufacturing staff themselves since the responsibility for high quality products always rests with the manufacturer. There can be no delegation of a quality or compliance assurance function to a separate or independent organization, as this would have a demotivating effect on the manufacturing personnel.

1. AIM OF QUALITY ASSURANCE

There are several different definitions of the term 'quality assurance', but all indicate the same purpose: to guarantee that a product fulfills all requirements necessary for the protection of human beings and the environment. Quality assurance (QA) thus covers all measures for assuring the compliance of a process and product with given requirements. The results of these quality controls have to be recorded in quality assurance documentation.

2. TECHNICAL REQUIREMENTS FOR PACKAGES

The technical requirements for packages and packaging of radioactive waste are specified in regulations, guidelines and standards for transport, handling, treatment, packaging, intermediate storage and final disposal. Recommendations for
FIG. 1. Standard type packages for low and intermediate level radioactive waste.
the packaging and transport of radioactive waste can be found in the Regulations for the Safe Transport of Radioactive Material, published as IAEA Safety Series No. 6, and in supporting documents such as IAEA Safety Series Nos 7 and 37.

Special requirements for the products relate to:

- density of waste matrix and container
- leaktightness of the containment
- compactness of the waste matrix (absence of voids)
- leach density of radionuclides
- water and sulphate resistance of the waste matrix
- limitation of significant radionuclides (type and amount)
- limitation of surface contamination and surface dose rate.

3. TYPE TESTS FOR STANDARD PACKAGES

The Waste Treatment Division of the Swiss Federal Institute for Reactor Research also serves as the centre for the collection, treatment, incineration, conditioning, packaging and intermediate storage of all radioactive waste from Swiss industry, hospitals, research institutes, trade and also, in part, nuclear power plants. The Division is developing special standardized and qualified types of packages which fulfil all prescribed requirements (Fig. 1).

Compliance with the regulations is demonstrated by prototype tests. The test stipulations then become the obligatory minimum requirements for the fabrication of all standard packages. The prototype test specifications include all legal requirements, codes, standards and regulations. Additional sections of the specifications define waste matrix mixtures, types of materials, dimensions and tolerances, the preparation procedure, the quality control programme and documentation (Fig. 2).

4. TEST FACILITIES

All standard type tests for waste packages have been carried out internally by the Swiss Federal Institute for Reactor Research. For this reason it has developed all the facilities required for testing and quality control.

4.1. Target for drop tests (in accordance with IAEA Safety Series Nos 6 and 37)

<table>
<thead>
<tr>
<th>Dimensions</th>
<th>4 × 4 × 3 m</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cover plate</td>
<td>steel 25 mm thickness, wet floated</td>
</tr>
<tr>
<td>Total mass</td>
<td>115 t</td>
</tr>
<tr>
<td>Maximum weight of specimen</td>
<td>10 t</td>
</tr>
</tbody>
</table>

The target is illustrated in Fig. 3.
4.2. Pressure vessel for 200 L drum

Maximum pressure 6 bar  
Negative pressure 500 mbar  
Maximum temperature 80°C  
Liquid recirculation system

Application:
  Water immersion test  
  Integrity test  
  Leakage test  
  Leach test
4.3. Core drilling machine

Diameter  54 mm  
Length    400 mm

Samples are taken from the radioactive waste matrix in completed packages for pressure, density and leach testing, as part of the quality assurance programme.

4.4. Tool for penetration test

A special device, made to IAEA specifications, with a guide tube to ensure correct point of impact.

4.5. Thermal test facility

Open, screened pool fire with supports for a specimen of up to 10 t and temperature instrumentation. Unlimited fire duration, with fuel consumption of 0.1 L·m⁻²·s⁻¹.
5. QUALITY CONTROLS

Quality controls apply to the following process steps:

(1) Supply of material and intermediate products: checking for conformity with the specifications upon receipt.

(2) Supply of radioactive waste material: checking, upon receipt, for conformity with the declaration, which includes: origin and type of waste, nuclides and activity, dose rate, outside contamination. Checking by gamma-spectrometry, if necessary.

(3) Treatment and conditioning:
- Conformity with specifications and standard procedures for processing and mixtures
- Visual control of waste, consistency of matrix, segregation of mixtures
- Sampling for laboratory tests
- Control of cement reaction temperature of waste matrix
- Testing of final waste matrix density.

(4) Quality control of final product (packages);
- Leaktightness
- Weight and possibility of void inclusion (for all package types the correct weight is known)
- Density of packaging and concrete liner
- Radiological tests, such as dose rate of the packages and surface contamination
- Analysis of nuclides and activity of the packages by gamma-spectrometry (for individual packages) and comparison with regulatory limits and dates of delivery of waste
- In special cases, taking of samples for density, homogeneity, water and sulphate resistance and leach test measurements.

(5) Administrative control and compliance with regulations and specifications:
- Documentation of each package, including that of QA
- Labelling and marking
- Transport documentation.

6. DOCUMENTATION

All quality control and test results must be recorded in QA reports. Every package has its own documentation. Thus, once the identification number of the package is known, all information about the treatment procedure can be found, from the origin of the waste until packaging for final disposal.
7. QA ORGANIZATION AND RESPONSIBILITY

In principle, quality control should be carried out by the manufacturing staff themselves. In this way the responsibility for high quality products always rests with the manufacturer, and the consciousness of, and desire for, quality in each worker involved will be stimulated.

Delegation of a quality or compliance assurance function to a separate or independent organization will have a negative effect on the motivation of the manufacturing staff, and responsibility will be felt to have been shifted to the assurance personnel. Quality control will be reduced to certain special criteria, which may indeed be fulfilled, but the quality of the complete product itself may no longer have first priority. It is also incorrect to postulate possible treatment worker or shipping error and as a result of this to define additional requirements. This is now being done in transport regulations for the Type A package for liquids, where a leakage is postulated and there is a requirement for absorbent material around the package in addition to the requirement relating to the 9 m drop test.
TIE-DOWN SYSTEMS FOR RADIOACTIVE MATERIAL PACKAGES UNDER ACCIDENT CONDITIONS.

A survey of road transport of goods has demonstrated the paucity of data on accident conditions. Two types of maximum accident have been identified and analysed by calculations and by eight tests with packages weighing 1300 kg. In the first type of accident, the carrier vehicle is subjected to a head-on impact at a speed of 50 km/h, where the primary concern is to keep the package from moving (average deceleration 35g); in the second type, the package is subjected to impact from the side by another vehicle, where the primary concern is to limit the damage to the package to what would be experienced when falling from 1 m onto a punch. There are three mathematical formulations which express what these requirements mean as far as the tie-down systems for the packages are concerned: a balance of the forces in the system of slings and blocks during head-on impact (keeping the package from moving); an energy balance in the striking object-object struck-sling system during impact from the side (freeing of the package); a rigidity criterion in the sling-package system to limit damage to the package during impact from the side. Application of these criteria shows that wedging is essential to prevent packages, particularly big ones (more than 1 t), from moving during head-on impact.

L'ARRIMAGE DE COLIS DE MATIERES RADIOACTIVES EN CONDITIONS ACCIDENTELLES.

Une enquête sur les transport routiers de marchandises a montré que peu de données existaient sur les conditions accidentelles. Celles-ci ont été représentées par deux types d'accident maximum, lesquels ont été analysés par des calculs et par 8 essais avec des colis d'une masse de 1300 kg. Ces accidents sont: d'une part, le choc frontal du véhicule porteur à la vitesse de 50 km/h, l'impératif étant alors de retenir le colis (décélération moyenne 35g); d'autre part, le choc latéral contre le colis par un autre véhicule, l'impératif étant de limiter le dommage au colis à celui de la chute de 1 m sur poinçon. Ces exigences se traduisent, pour les arrimations du colis, par trois formulations mathématiques: un bilan des forces dans le système d'élingues et de cales lors du choc frontal (maintien du colis); un bilan énergétique dans le systèmes heurté-
heurtant-élingues lors du choc latéral (libération du colis); un critère de raideur dans le système élingues-colis pour limiter le dommage au colis dans le choc latéral. L’application de ces critères montre que le calage est indispensable pour retenir surtout les gros colis (supérieurs à une tonne) en choc frontal.

Lors d’une large enquête au sujet des conditions de transport, il s’est avéré que peu de données étaient disponibles concernant les conditions accidentelles. Après la sélection de deux types d’accidents de référence, ceux-ci ont été définis au moyen de calculs et d’essais.

Sur base des résultats obtenus, des méthodes analytiques et des abaques ont été mis au point pour le dimensionnement des arrimages.

Ce dimensionnement est déterminé par deux impératifs:
— pour le choc frontal (décélération de 35g): le maintien du colis sur le véhicule
— pour le choc latéral: la rupture du système d’arrimage.

Ces impératifs sont traduits en trois formulations mathématiques:
— un bilan des forces dans le système d’élingues et de cales lors d’un choc frontal;
— un bilan énergétique dans le système heurté-heurtant-élingues lors du choc latéral;
— un critère de raideur dans le système élingues-colis lors du choc latéral.

En appliquant ces formulations, deux résultats s’en dégagent: d’une part, un système de calage correctement dimensionné doit être utilisé pour maintenir un colis ayant une masse supérieure à une tonne et, d’autre part, il s’avère que l’arrimage est toujours correctement effectué lorsque les trois formulations sont appliquées avec une géométrie libre.

Les valeurs de 2g (longitudinal)–1g (transversal)–1g (verticale) ont été mises en évidence, lors d’une enquête, comme des valeurs le plus couramment proposées pour des arrimages routiers. Par ailleurs, 10g, 5g et 2g sont les valeurs maximales trouvées. Peu d’informations ont été collectées concernant les valeurs de décélération en condition accidentelle de transport.

Les relations entre, d’une part, les types de véhicule, les colis, les forces de décélération sur les colis et, d’autre part, la fréquence et la nature des accidents ont fait l’objet d’une collecte d’informations.

Sur cette base, deux types d’accidents ont été définis:
— un accident avec choc frontal d’un camion contre une barrière fixe et rigide à une vitesse de 50 km/h;
— un accident avec choc latéral avec le même type de camion mais à une vitesse soit de 25 km/h dans le cas où le colis est directement agressé, soit de 35 km/h dans le cas où le chassis du camion est attaqué.

Un modèle de calcul, à éléments finis, a été utilisé afin de pouvoir déterminer l’énergie absorbée par l’emballage et son arrimage. Le code TRICO, développé par le CEA, a été utilisé pour modéliser le comportement de cet ensemble comprenant un colis de 1,3 t lors d’un choc frontal à une vitesse d’impact de 50 km/h.
FIG. 1. Essais en choc frontal à la vitesse de 50 km/h (le but est le maintien du colis sur le véhicule).

FIG. 2. Courbe de décélération moyenne.

FIG. 3. Essais en choc latéral contre l’emballage à la vitesse de 25 km/h (on vise à rompre les arrimage et à provoquer un dommage limité sur le colis).
Des essais, en conditions accidentelles, ont été réalisés sur le site de l'Union technique de l'automobile et du cycle près de Paris. L'objet principal de ces essais a été:

- de préciser entre quelles limites hautes et basses doivent être dimensionnés les systèmes d'arrimage,
- de vérifier les résultats obtenus lors de la modélisation,
- de préparer une proposition de code de dimensionnement des systèmes d'arrimage pour le transport de colis de matières radioactives.

Huit essais ont été réalisés:

- 5 avec choc frontal (fig. 1 et 2),
- 3 avec choc latéral (fig. 3, 4 et 5).

Les résultats obtenus sont décrits ci-dessous.

A) Il est possible de maintenir un colis du type B(U), pesant 1,3 t sur le plateau d'un camion, lors d'un accident avec choc frontal à une vitesse d'impact de 50 km/h,
le colis étant maintenu à l’aide d’élingues et de cales appropriées. La valeur de décélération à considérer lors d’un tel accident (choc frontal) est environ 35g.

B) Le colis arrimé peut subir un dommage supérieur à celui provoqué par sa chute d’un mètre sur un poinçon du type décrit par l’AIEA lorsqu’il est attaqué directement par un véhicule heurtant roulant à une vitesse de 25 km/h et ayant une masse très supérieure à la sienne. La valeur maximale de la décélération mesurée lors des essais en choc latéral a été de 120g.

Le Département des études mécaniques et thermiques du CEA, à Saclay, a entrepris une étude parallèle ayant pour objet de démontrer la faisabilité d’un dimensionnement chiffré d’un arrimage. Cette étude prend en compte les résultats obtenus lors des essais précités. Ce dimensionnement dépend des trois critères énumérés ci-après.

1) Le premier critère se traduit par un bilan des forces. L’arrimage, qui peut comprendre un système d’élingues et un système de cales, doit retenir le colis qui
n'est pas directement sollicité lors d'un choc frontal mais subissant une décélération uniforme de 35g. 

2) Le deuxième critère se traduit par un bilan énergétique du système (heurté-heurtant-élingues). La notion de rupture souhaitée du système d'élingues (on suppose l'absence de cales ne devant jouer aucun rôle au cours d'un choc latéral) est sous-jacente à celle de l'intégrité de l'enveloppe de confinement: le dommage au colis, à partir du seuil d'énergie fixé, est moins grand si les liaisons rompent que si elles maintiennent le colis sur le plateau.

3) Le troisième critère est additionnel au deuxième. Il exprime que la raideur des élingues est inférieure à celle de la paroi du colis. Pour l'appliquer, il est nécessaire de connaître la pénétration de référence suite à la chute du colis sur un poinçon depuis une hauteur de 1 m.

Les trois critères précités ont été traduits par des expressions mathématiques. Des méthodes analytiques et des abaques ont été mis au point en fonction du problème posé et du nombre de critères à appliquer.

Des exemples d'application ont été traités et les deux résultats qui s'en dégagent sont décrits ci-après.

Tout colis de masse supérieure à une tonne doit avoir un système de calage dimensionné. Pour appliquer avec succès les trois critères mathématiques, il est recommandé de postuler un effort à la rupture, \( RC \), égal à au moins les \( 3/4 \) de celui satisfaisant à lui seul au critère de choc frontal:

\[
RC = \frac{3}{4}(35g - M)
\]

où \( M \) est la masse.

Lorsque les trois critères sont appliqués avec une géométrie non imposée (longueur d'élingues, angles, position des points d'ancrage), la méthode aboutit toujours à un résultat. Cette méthode s'impose pour la conception des plateaux de nouveaux véhicules.

L'étude relative au dimensionnement des arrimages a permis de dégager les constatations suivantes:

A) Les moyens d'arrimage employés actuellement et les points d'ancrage sur la plupart des véhicules en service ne répondent pas à ces critères lorsque les colis ont une masse supérieure à une tonne et, en particulier, lorsque la technique de calage n'est pas mise en œuvre.

B) Un système d'élingues résistant est rapidement d'un encombrement et d'un poids importants lorsque le système de calage est insuffisant.

L'ensemble de l'étude a ainsi conduit aux trois recommandations suivantes (fig. 6).

1) Lorsque le colis est protégé des agressions latérales par des structures, le dimensionnement de l'arrimage ne sera calculé qu'avec le critère de choc frontal.

2) Les plateaux de nouveaux véhicules seront conçus pour que l'arrimage des colis respecte les trois critères mathématiques liés aux chocs mentionnés.
3) Les colis et les véhicules existants, non munis de structures latérales, seront soumis aux trois critères. En cas d'impossibilité, le critère de raideur ne sera pas considéré. Dans le cas d'application d'un seul critère, il résulte que la protection sera minimale.
REGULATORY ROLE IN THE QUALITY ASSURANCE INVOLVED IN THE SHIPMENT OF NEW FUEL ASSEMBLIES IN BRAZIL

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Abstract

The paper presents the role of CNEN, the Brazilian competent authority, in the quality assurance activities involved in the packaging and transportation of nuclear fuel, taking as an example the shipment of new fuel assemblies for the first refuelling of the Angra 1 nuclear power plant. A description is given of the activities carried out by CNEN to determine whether all the quality assurance regulatory requirements have been met. Those activities comprise the establishment of regulations, the review and assessment of the submitted documentation, and the conduct of inspections and audits to verify compliance with the regulatory requirements. Since not all the requirements of the regulations had been satisfied, the approval certificate was not granted, but package use under special arrangement was authorized. The results are discussed and the deficiencies found and the actions proposed to correct them or to prevent their repetition are considered.

1. INTRODUCTION

The first Brazilian nuclear power plant, Angra 1, is a 626 MW(e) PWR. According to its turnkey purchase contract, the fuel for the initial core was supplied by the vendor, Westinghouse. However, the subsequent reload fuel will be ordered from a Brazilian nuclear fuel manufacturer, NUCLEBRAS, a state-owned company employing KWU technology.

During the refuelling outage, 40 fuel assemblies, i.e. about one-third of the total core, are to be replaced by new ones.

The NUCLEBRAS fuel manufacture plant is located in Resende, at about 60 km (direct straight line distance) from the Angra site.

As part of the arrangements for the transport of the refuelling fuel assemblies, NUCLEBRAS ordered the manufacture of transport containers from Treu S/A, a tank and pressure vessel manufacturer, in accordance with the design of similar KWU packagings which have been approved in the Federal Republic of Germany by the Federal Physico-Technical Institute PTB. However, the containers manufactured in Brazil had some design features modified, mainly in relation to materials.
To supervise all the fabrication steps, NUCLEBRAS contracted IBQN, a third-party inspection agency which carries out independent inspections for the nuclear industry.

2. PACKAGING DESCRIPTION

The packaging consists basically of an outer, two-part, box shaped, stainless steel container, and an inner, T shaped, carbon steel support bed, which is suspended within the outer container by a set of 16 round rubber mounting cushions. The outer container lid is fastened to the bottom tray by means of 38 nut and bolt sets along a Z shaped flange.

The packaging is 5800 mm long, 990 mm wide and 770 mm high. Its net weight is 1810 kg.

Each packaging can transport one or two fuel assemblies with or without core components. The fuel assemblies are 16 X 16 bundles of fuel rods, 4100 mm long and 590 kg in weight.

Each fuel assembly is braced against the support bed by means of eight equally spaced stainless steel clamps. Two stainless steel spacer blocks support the fuel assembly top and bottom nozzles. The portions of the support bed which are in contact with the fuel assembly spacer grids are lined with 8 mm stainless steel plates.

Each fuel assembly is wrapped with a polyethylene sheet for cleanliness.

The lid is provided with four lifting eyes for handling purposes. The bottom tray is provided with two lifting lugs at the front end and a fulcrum tube at the rear end, where a pivot rod is inserted to form a hinge-like mechanism for tilting the package from horizontal to vertical or vice versa.

The instrumentation on the packaging consists of one accelerometer and two shock indicators, attached to the suspended internals.

3. ROLE AND POSITION OF THE COMPETENT AUTHORITY

3.1. Authority and responsibility

In Brazil, the legal competence for regulating and licensing any shipment of radioactive materials falls on CNEN, the national regulatory body, which is placed directly under the authority of the Ministry of Mines and Energy.

As the competent authority, CNEN is responsible for protecting the public and the environment from undue radiological risks associated with shipments of radioactive materials.

To properly discharge its duties, CNEN has the right of free and prompt access to any facility, document, or employee of the organizations involved in the packaging and transport of radioactive materials.
3.2. Applicable regulations and safety standards

The road transport of new fuel assemblies or of any other kind of radioactive material comes within the scope of the "Regulations for the Transport of Dangerous Goods" issued by DNER, the Brazilian Department of Road Transport. Article 1, paragraph 2 of this regulation states that "explosive products and radioactive materials shall also meet the specific regulations and standards of, respectively, the Army and CNEN".

Through Resolution CNEN No. 05/81, of 27 July 1981, which has superseded Resolution CNEN No. 09/77, CNEN has adopted as mandatory regulations the IAEA Safety Series No. 6 [1], hereinafter referred to as the IAEA Regulations.

Owing to the fact that it is intended to provide guidance, compliance with IAEA Safety Series No. 37 [2] is not required. Nevertheless, CNEN recommends its use, since it provides an acceptable methodology for implementing the requirements of the IAEA Regulations.

As regards specifically quality assurance, CNEN has issued a regulation — CNEN Standard No. NE-1.16 — which is concerned with nuclear power plant items. No regulations for shipping packages of radioactive materials, however, have been issued, such as for example, the US regulation 10 CFR 71 H [3]. Thus, the only applicable requirements are those of Section VIII of the IAEA Regulations.

3.3. Review and assessment of licensing documents

The task of reviewing and assessing the documentation submitted by the applicant is performed by individuals or groups designated from among experts in the different subjects involved (criticality analysis, structural analysis, shielding calculations, acceptance tests, quality assurance, etc.).

As concerns quality assurance, the purpose of the review and assessment process is to determine whether:

(a) The applicant's quality assurance programme supplies all the necessary information about the measures taken to achieve the required levels of quality.

(b) This information is sufficiently complete and accurate to serve as a basis for the granting of the approval certificate.

In the case of Angra 1 new fuel assembly packagings, the review and assessment of the quality assurance programme was not carried out. The reason is described below.

By way of a notification, NUCLEBRAS sent a letter to CNEN, informing them that transport containers for shipping the Angra 1 new fuel assemblies had been ordered and that all manufacturing steps had been surveyed
by KWU and IBQN inspectors. The final acceptance inspection was to take place the following week at the manufacturer's premises. Also requested was an authorization to transport the packages by road from the fuel manufacture plant to the Angra site.

After an exchange of letters, in which CNEN reminded NUCLEBRAS that the IAEA Regulations had to be adhered to and that by ordering the fabrication of the packagings without previous notification to the competent authority, i.e. CNEN, NUCLEBRAS failed to meet the legislation, NUCLEBRAS finally sent to CNEN a "Safety Analysis Report for Unirradiated Fuel Assembly Transport Containers" and requested the Certificate of Approval. The packages were classified as Type A, fissile Class II.

In this Safety Analysis Report, some criticality, structural and other calculations were included to justify design criteria. For economic reasons, the acceptance tests to demonstrate compliance with the performance standards of the IAEA Regulations were replaced by representative calculations and arguments, both for normal and accident conditions of transport. Chapter 8, entitled "Quality Assurance Programme", contained just one page, on which all that was written was: "The fabrication of the containers has been done in accordance with the Quality Assurance Programme of Treu S/A" (the manufacturer of the packagings). This programme was included in the Safety Analysis Report as Annex 5.

It should be emphasized that the review and approval of a manufacturer's quality assurance programme is the responsibility of the applicant, and shall be exercised in the course of the supplier selection and evaluation process.

What should have been submitted to CNEN was a programme describing the overall quality assurance system for (as applicable) design, manufacture, assembly, testing, use and maintenance, including the functions, authorities and responsibilities of all participants, in particular the applicant, which is the organization responsible to CNEN for the quality assurance programme.

The use of an inspection agency, although not required by CNEN, is very helpful, especially when the applicant does not possess an experienced inspection staff, as seemed to be the case.

3.4. Inspections and audits

To verify that the quality assurance programme is implemented effectively and in compliance with the IAEA Regulations, CNEN has established a programme for inspection and audit of all the activities which may affect the quality of the packagings.

During inspections and audits, CNEN's inspectors shall look for evidence that:

(a) The construction methods and materials are in accordance with the approved design specifications and meet the regulatory requirements;
(b) All packagings built to an approved design are periodically inspected and, as necessary, repaired and maintained in good condition so that they continue to comply with all relevant requirements and specifications, even after repeated use.

The activities which may affect the quality of a packaging include: design and engineering; procurement of materials; fabrication, assembly and testing; loading, transport and in-transit operations; receipt, handling and storage; cleaning, maintenance, repair, modification and re-inspection; and disposal of obsolete or faulty packagings.

In view of such a wide variety of quality related activities it is almost impracticable to inspect all relevant items, activities or documents. Therefore, CNEN concentrates its efforts on certain activities deemed critical to the quality of the packagings. For example, during fabrication, the main points of concern are qualification of the manufacturer, qualification of materials, qualification of welding procedures and welders, qualification of non-destructive testing procedures and inspectors, control of measuring and test equipment, identification of materials, and control of non-conforming items.

Moreover, owing to limitations in the resources available to CNEN (mainly manpower), CNEN's inspectors only exceptionally perform independent tests or measurements, although they have adequate knowledge of the inspection areas to which they are assigned. Usually they just verify the work performed by all participants on a spot check basis.

In order to make the necessary inspection arrangements, the competent authority must be informed of the manufacturing schedule in a timely manner by the applicant. In the case under discussion, this was not done, and obviously it would be nonsense to demand an applicant's quality assurance programme covering fabrication after the packagings had been fabricated. For that reason, no inspections were performed during the construction of the packagings, and some important inspection activities could not be carried out, such as:

- verification of the steel sheets to be cut as well as the transfer of their marking
- verification of the edge preparation as well as the alignment and fitup of the parts before welding
- verification of the execution of visual and non-destructive examinations of the inner surfaces of the root passes
- verification of dimensional inspection of weldments as to squareness, straightness, flatness, etc.

To determine whether the packagings had been constructed according to the design specifications, a conformance inspection was conducted, when packaging features such as dimensions, materials, components, assembly, weldments, finishing, etc., were compared to the relevant design specifications, drawings, item lists, etc.
Auditing of the manufacture documentation was conducted with the purpose of verifying, by examination of objective evidence, that all manufacturing stages had been accomplished in accordance with established procedures, instructions, specifications, standards, codes and other applicable documents.

Documents which were verified included material certificates, welder qualification certificates, the welding material control procedure, approved welding plans, certificates of inspector qualifications, dimensional records, measuring and test equipment calibration certificates and records, heat treatment plans and charts, manufacture and inspection sequence plans, test records, non-destructive test procedures, inspection records, non-conformance reports, and final inspection (acceptance) reports and releases.

The inspection/audit results turned out to be very positive as regards the quality assurance measures adopted by the manufacturer and the inspection agency during construction.

Another point learnt during audit is that the design control measures may be accepted provided they are correctly formulated.

As no quality assurance measures had been set up at all for the following phase, i.e. use and maintenance, CNEN demanded from NUCLEBRAS the establishment of procedures to cover the activities associated with that phase, namely, handling, shipping, storage, receiving, cleaning and maintenance (including repairs and modifications).

4. SHIPMENT UNDER SPECIAL ARRANGEMENT

To approve a shipment of radioactive material under special arrangement, CNEN verifies whether the compensatory measures proposed by the applicant to offset the failure to meet the applicable regulatory requirements are adequate to ensure that the overall level of safety in transport is at least equivalent to that which would be provided if all those requirements had been met.

The overall safety level is, of course, affected by the actual safety conditions related to vehicles, roads, weather and physical security. It is necessary to mention that the proposed type of transport vehicle for the Angra 1 packages was an open truck, that the weather in the region is very stormy in summer (when the shipments would take place), and that the roads have a high incidence of landslides.

The information initially supplied in the Safety Analysis Report of NUCLEBRAS was neither sufficient nor clear enough to permit a complete safety evaluation of the packages. Thus, after some rounds of questions involving structural integrity, criticality control and calculational modelling of the acceptance tests, CNEN concluded that compliance with the IAEA Regulations had not been adequately demonstrated. Therefore the Certificate of Approval was not granted.
It should be mentioned that this decision was influenced by other factors besides quality assurance, mainly the failure of the applicant to adequately justify as reliable or conservative some parameters and correlations used in the acceptance test calculation methods, in particular those used to demonstrate that the packages could withstand impact accident conditions, although that matter is beyond the scope of this paper.

However, owing to the urgent need to make arrangements for the shipment of the fuel assemblies, caused by the approach of the refuelling time, there was no time to correct the deficiencies identified.

Nevertheless, there still remained the possibility of utilization of the packagings conditioned to satisfy the requirements laid down in Section VIII of the IAEA Regulations, "Approval of Transport by Special Arrangement". To meet those requirements, NUCLEBRAS proposed a set of measures which included the following operational controls:

(a) Only one package would be transported per vehicle
(b) The shipments would be escorted by a car with health physicists, police cars, a special fire-engine, and a spare truck equipped with lifting gear
(c) The speed of the transport vehicles would be limited to 60 km/h.

To appraise the transport conditions, a simulated shipment was carried out. This consisted of a truck, having the same features as those which would be used in the real shipments, transporting a package of two dummy assemblies along the normal route. The package was instrumented with a triaxial electronic accelerometer for the purpose of continuously recording the accelerations developed during the journey so as to permit an evaluation of the road conditions.

The transducers were mounted on the packaging internals, and the recorder installed in the truck cab. The accelerometer scale ranged from -30 to +50 m/s².

The highest acceleration recorded during the test was about +0.8g, in the longitudinal direction, whereas the design maximum allowable acceleration is 4.0g.

5. CONCLUSION

The shipments took place in March 1986. The fuel assemblies were distributed among 5 consignments, each of them comprising 4 packages.

The shipments were carried out according to a transport plan, duly approved by CNEN, which included a description of the normal route and two alternative routes, a description of the characteristics of the transport vehicles and the communication equipment used, a physical protection plan, and a radiation protection plan, including emergency procedures.

The normal route was the shortest (170 km long). Along it the convoys had to stop twice for cargo inspection (containers, tie-downs, tarpaulins, etc.)
and radiation survey. The shipments were accompanied by CNEN's inspectors and no reportable incidents occurred.

As described in the preceding section, the shipment of the Angra 1 new fuel assemblies has been carried out successfully, thanks to the flexibility of the IAEA Regulations, which permit transportation under special arrangement.

CNEN, of course, has not definitively disapproved the NUCLEBRAS packagings. Their approval may still be reconsidered, provided that the supporting documentation includes more substantial information and is sufficiently accurate as well as reliable or conservative to serve as a sound basis for the granting of the Certificate of Approval.

Anyway, it is necessary to improve the co-ordination of CNEN with all parties concerned (consignors, consignees, carriers, etc.) to avoid a recurrence of problems such as the one described in this paper, but above all, it is imperative that more rigorous enforcement methods be adopted to cope with violations to regulations, including fines, not foreseen within CNEN's legal framework. This should prevent the regulatory requirements from being overlooked again.

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THE RADIOACTIVE TRANSPORT STUDY GROUP

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Abstract

THE RADIOACTIVE TRANSPORT STUDY GROUP.

IAEA Safety Series No. 6: Regulations for the Safe Transport of Radioactive Material, as well as the relevant international modal transport regulations, define a national governmental regulatory body as a competent authority. This regulatory body is assigned certain responsibilities and is also the focal point in terms of receiving prior notification and other information required by the relevant regulations. For the purposes of providing a forum for discussing issues of mutual interest and sharing experience regarding the administrative aspects of implementing international regulations for the safe transport of radioactive materials, some competent authorities have formed an informal association known as the Radioactive Transport Study Group (RTSG). The RTSG also provides advice and consultation, as well as makes recommendations, to the IAEA on necessary changes in the IAEA Transport Regulations. The RTSG was established under the aegis of the IAEA in 1967 and although close contacts with the IAEA are maintained, the Group has no formal links with it. The RTSG is not only an informal association, it is also totally autonomous. The functions of the RTSG can be summarized as follows: (1) To provide a forum for exchanging information, sharing experience and for discussing issues or procedures pertaining to the administrative aspects of implementing the regulations for safe transport. (2) To provide advice and to make recommendations to the IAEA regarding the administrative procedures contained in IAEA Safety Series No. 6.

Introduction

Transporting radioactive materials is essential for many branches of medicine, industry, research and education. Several safety-related and functional requirements apply to the various activities involved in transport. Such requirements can relate to specific operational use of the equipment or package after transport (gamma radiography devices), to commercial considerations (economic value of packaging and/or content) and to safety. In transport regulations concerning radioactive materials, safety is the main and foremost objective; other considerations are accepted only if safety is not infringed.

Actual safety in the transport of radioactive materials is the result of many contributions, e.g. from package designers and manufacturers, consignors,
transport workers, governmental bodies (competent authorities), and the transport regulations defining the safety standards. Frequently co-operation between representatives from such groups is established with the aim of solving common problems, jointly perform development work, or to harmonize certain procedures. This paper is concerned with co-operation between competent authorities.

Functions of the competent authority

National and international transport regulations define a national governmental body as a competent authority. In some countries the full range of competent authority functions is discharged by more than one such national body.

The competent authority is assigned certain responsibilities, including receiving applications for approval (special form radioactive material, package design and shipment), assessing compliance with relevant regulatory requirements and issuing certificates of approval. The competent authority is also the focal point in terms of receiving prior notification of shipments and other information as required by the relevant regulations.

Broadly speaking, the functions of a competent authority can be described as being both of a technical and an administrative nature. In the technical or engineering field (radiation protection, criticality safety, test standards, etc.) international co-operation is managed through a multitude of arrangements, such as IAEA and NEA symposia, seminars and specialists' meetings, as well as various bi- or multilateral agreements involving, among others, R&D and testing organizations and industry. International co-operation in the administrative field is managed by the IAEA, by regional agreements and by the Radioactive Transport Study Group (RTSG).

RTSG - Organization

RTSG was established under the aegis of the IAEA in 1967 by a group of competent authorities. Although close contacts have been, and are, maintained with the IAEA, RTSG has no formal links with the IAEA. RTSG is thus not operating within an IAEA framework as are, for example, SAGSTRAM and SAGSI. There are,
however, links constituted by common interests to assure safety in the transport of radioactive materials through regulatory activities. These interest links are manifested by the inclusion of an IAEA representative in the Group, the membership of which currently includes representatives from twelve countries, Australia, Belgium, Canada, France, Federal Republic of Germany, Italy, Japan, Netherlands, Poland, Sweden, United Kingdom and United States of America.

RTSG is an autonomous and informal association of competent authorities. The composition of the Group is organized such that membership includes countries being engaged to a significant extent in the international transport of radioactive material. The Group being autonomous implies that members can be added only with the approval of the present membership.

RTSG – Purpose and Functions

The main purpose of RTSG is to provide a forum for discussing issues or procedures pertaining to the administrative aspects of implementing radioactive material transport regulations, with the aim of achieving compatibility and harmonization in the application of regulatory requirements. What this really means is that practical situations are highlighted and discussed with the intent to explain the background to a particular national procedure, to enhance international harmonization of the practical implementation of specific regulatory provisions, or to identify areas where additional development of the transport regulations would be needed. Other examples include situations where a specific regulatory provision, albeit clear and unequivocal in terms of what is required, still appears to be difficult to understand because the background, or intent, is not fully realized, or where an operational situation, having developed into something that was not established, or even envisaged, practice when the regulations were issued, can be addressed in the process of applying the transport regulations in more than one way.

The modus operandi used by RTSG is thus partly one of self education and partly one directed towards solving practical problems encountered in the everyday competent authority working environment. In this regard prime emphasis is placed on the administrative
rather than the technical aspects of the transport regulations.

It has been stressed above that RTSG is an informal association of competent authorities. This informal nature of the Group means that it has no executive power. RTSG thus can not in any formal sense make decisions; the members can only arrive at a common understanding and express a consensus view, and through suitable channels make it known. Such suitable channels include communication with the IAEA, where the matter is referred for consideration and appropriate action, and the individual RTSG member. Communication with the IAEA is frequently automatic because of the IAEA's membership and formal communication is limited to particular situations, e.g. RTSG input to a regulations revision process.

The lack of executive powers should not be seen as a drawback. On the contrary, this lack has proved to be of benefit in that it provides for matters to be discussed freely. A consensus view reached on such a basis reflects only the expert opinion of the RTSG members but does not imply formal commitment on behalf of the competent authority an RTSG member represents. The lack of a consensus view on a particular matter does not per se constitute a failure; it reflects the factual situation at the time the matter was discussed, and full knowledge of it has been established.

Depending on the particular circumstances a consensus view may result in, say, a fully developed proposal or recommendation to the IAEA for a change in the transport regulations, or it may be limited to the identification of a problem. It may also result in agreement to adjust certain national procedures in the interest of international harmonization.

Another important aspect of the RTSG's functions is to share experience, to supply information and to provide advice. These activities include reports on compliance assurance activities, the IAEA Transport Safety Programme, changes in national legislation and/or competent authority organization, etc. and practical assistance in the application of a particular regulatory provision.

The practical assistance is significant in that practices and procedures originating in several
countries are highlighted and discussed. Not only do such discussions provide advice to the member asking the question, but they also assist in harmonizing application of the transport regulations.

**RTSG - Perspectives**

RTSG is not the only forum for international co-operation between competent authorities. The most prominent forum is probably the IAEA itself. Through its leading role in the development of transport regulations (Safety Series No. 6), the IAEA is certainly the main forum for formal co-operation. An interesting and important aspect of the IAEA's formal activities - symposia, advisory groups, technical committees - is the informal co-operation and consultations that these meetings make possible. Additional forums include direct bilateral consultations and various regional arrangements.

It is necessary to have this multitude of arrangements and mixture of formal and informal associations and gatherings. Problems and emphasis vary from one environment to the other and co-operative efforts must be tailored taking these variations into account.

For example, competent authorities in the Nordic Countries (Denmark, Finland, Norway and Sweden) have formed the Nordic Transport Group, which operates in a way that is essentially similar to that of the RTSG, although emphasis is sometimes different.

The Nordic Transport Group is also engaged in other activities such as detailed definition of, and active participation in, research projects in the area of transport safety. Through one of its members, contact with developments in RTSG is maintained. The Nordic Transport Group also liaises with other associations and organizations and this range of contacts has proved to be quite useful and functional in that it minimizes staff and travel expenditures, whilst it maximizes the yield. Based on the experience of the Nordic countries, similar suitable regional arrangements, including liaison with RTSG would appear to be generally beneficial.

RTSG membership has been constant over a number of years. The matter of expanding the Group has recently been discussed, with the conclusion that a moderate
growth would be of benefit. Further development in this regard will be subject to interest shown in joining the Group and to considerations on managing membership growth in terms of maintaining efficiency in deliberations. Primarily due to the meeting procedure established (approximately 18-24 months between meetings), changes are likely to come relatively slowly.
IMPLEMENTATION AND ENFORCEMENT OF THE 1985 IAEA TRANSPORT REGULATIONS WITHIN THE UNITED KINGDOM

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Abstract

IMPLEMENTATION AND ENFORCEMENT OF THE 1985 IAEA TRANSPORT REGULATIONS WITHIN THE UNITED KINGDOM.

The implementation and enforcement of the 1985 IAEA Transport Regulations within the United Kingdom is discussed in three parts. Part 1 presents the legislative background and the plans for a two-stage introduction of these Regulations. The arrangements for consultation and advice are outlined and recent national developments are described. Part 2, concerned with radiological protection, describes the recent introduction of the Ionising Radiations Regulations and the initial experience from their operation. In Part 3, the continuing development of the UK compliance assurance programme is discussed, with special regard to the requirements of the 1985 Edition of the IAEA Regulations.

1. GENERAL REGULATORY CONSIDERATIONS

1.1. Introduction and background

Regulations to control the transport of radioactive material (RAM) by road in the United Kingdom were first drawn up in 1970 and were superseded in 1974 by The Radioactive Substances Regulations [1]. These regulations were worded in general terms and were backed up by codes of practice which contained detailed technical requirements based on the 1967, and later the 1973, editions of the IAEA Regulations. Further amendments to the IAEA Regulations have been reflected in the code of practice: the latest edition was published in 1986 [2]. Parallel legislation is in force in Northern Ireland. The Carriage by Road Regulations were complemented by The Radioactive Substances (Road Transport Workers) (Great Britain) Regulations 1970 as amended in 1975.

Transport by air is regulated by the Air Navigation (Dangerous Goods) Regulations 1985 [3], which forbids air transport of dangerous goods unless carried out with the written permission of the Civil Aviation Authority and in accordance

Sea transport is regulated by The Merchant Shipping (Dangerous Goods) Regulations 1981 [4]. These require compliance with the 1978 Report of the Department of Trade's Standing Advisory Committee on the Carriage of Dangerous Goods in Ships (the 'Blue Book') or, where there is no requirement, the International Maritime Dangerous Goods (IMDG) Code and other applicable codes of practice published by the International Maritime Organization (IMO).

Rail transport is regulated by British Rail via the List of Dangerous Goods and Conditions of Acceptance by Freight Train and Passenger Train or Similar Service, BR Publication No. 22426, 1977 edition (LDG). The LDG Conditions are made by the British Railways Board under powers conferred by Section 43 (3) of The Transport Act 1962, and effectively require observance of IAEA Regulations, with some modifications concerning labelling.

Following a reorganization of ministerial responsibilities in June 1983, the Secretary of State for Transport is now responsible as the competent authority for the four principal modes of transport in the United Kingdom — road, air, sea and rail. The responsibility for postal regulations rests with the Post Office under Section 11 of The Post Office Act 1953.

In the interim between the two most recent IAEA regulatory reviews, parallel developments have taken place internationally and nationally in several closely related fields, most notably those of radiation protection and compliance assurance/quality assurance (CA/QA). The regulations concerning the radiological protection of workers and the public in the United Kingdom have been consolidated by the Ionising Radiations Regulations 1985 [5] which, inter alia, supersede and revoke The Road Transport Workers Regulations 1970. The need to develop and enforce CA/QA measures has been clearly identified as being essential to the proper implementation of the 1985 IAEA Regulations. These aspects are developed in Parts 2 and 3 of this paper.

1.2. Plan for implementation of the 1985 IAEA Regulations

Recognizing that the time taken between the conception and the practical realization of some package designs can be considerable, the approach in the UK is to encourage use of the 1985 IAEA Regulations [6] as soon as possible. This approach also recognizes that needs may arise to accept foreign packages approved under 1985 Regulations for carriage in the UK and to approve package designs/shipments to meet foreign requirements. A two-stage implementation is being adopted:

(1) An interim amendment to the Carriage by Road Regulations [7] has been effected to permit the 1985 IAEA Regulations to be used as an alternative to the 1973 Revised (As Amended) Edition. The opportunity has been taken
to amend all quantity references to the equivalent in SI units, in line with UK obligations under EEC Directive 80/181/EEC. A parallel amendment to British Rail’s LDG conditions is expected to be made before the end of 1986.

(2) Legislation to fully implement the 1985 IAEA Regulations for carriage by road has been drafted and will, after a consultative stage, be implemented at a date aligned to the IAEA target date of 1 January 1990. The revision to the Code of Practice [2] is being carried out to complement the new legislation and work to revise other non-mandatory codes, for example certain British Standards, will commence towards the end of 1986.

Since the regulation of carriage by sea and air modes is intimately linked to the relevant international modal codes, national implementation will most probably follow full international modal implementation. Powers exist, however, for the Secretary of State to make regulations earlier if appropriate.

1.3. Consultation and advice on regulatory standards

Expert advice on safety standards is obtained from the National Radiological Protection Board (NRPB), the Safety and Reliability Directorate of the United Kingdom Atomic Energy Authority (SRD) and other public bodies in the UK. The NRPB is currently carrying out, on the Department of Transport’s behalf, a study of radiation doses arising in the sea mode of transport. This work is expected to yield valuable information in support of the revision of the IMDG Code, notably the development of segregation tables and requirements. SRD, sponsored by the Department, has carried out work towards a national risk assessment in the UK. The results have contributed to the IAEA's Programme of Worldwide Risk Assessment.

Arrangements exist to consult users of the Regulations on regulatory matters through standing and ad hoc committees which meet when necessary. Examples are the Department of Transport’s Working Party on the Transport of Dangerous Goods and the Standing Advisory Committee on the Carriage of Dangerous Goods in Ships (Blue Book Committee).

In recognition of the need for continuing vigilance concerning the adequacy of the safety standards, and the public concerns about them, the Secretary of State for Transport decided in 1985 to constitute ACTRAM, the Advisory Committee for the Safe Transport of Radioactive Materials, chaired by Lady Anglesey. The terms of reference of ACTRAM are “To advise the Secretary of State for Transport, or the Health and Safety Commission as appropriate, on major issues relating to the arrangements for the safe transport of civil radioactive materials by any mode normally and in the case of accidents.” The members are drawn from industry, the trades unions, universities and local government.
The Department has been called upon to give evidence at a number of public planning inquiries where transport safety has been under consideration, notably the Sizewell 'B' Inquiry in 1984/1985, the Torness Inquiry, 1985, and the European Demonstration (fast reactor fuel) Reprocessing Plant (EDRP) Inquiry in April 1986. The Reporter at the Inquiry into the provision of a railhead for the transport of spent nuclear fuel from Torness power station concluded that "...under all normal circumstances I consider that no unacceptable risk will arise from the construction and intended use of the proposed railhead, or by the special traffic moving between the rail-head and Torness power station...". The report of the Sizewell Inquiry has yet to be published and the EDRP Inquiry is continuing.

The House of Commons Select Committee on the Environment took into account transport safety aspects during its inquiry into radioactive waste disposal in 1985. In their report [8] it was recommended that:

1. British Rail should reconsider the possibility of timetabling movements of radioactive material so that the remote chance of an accident involving inflammable material in a tunnel can be made even more remote.
2. Wherever possible, radioactive wastes, especially spent fuel, high level wastes and plutonium, should be carried by rail in preference to all other modes of transport. The carriage by air of all except the very lowest levels of radioactive materials should be prohibited.
3. ACTRAM should consider whether changes in practice are needed to ensure greater control and safety of the transport of radioactive material through the English Channel.

2. RADIOLOGICAL PROTECTION

2.1. Introduction

The radiological protection standards in IAEA Safety Series No. 6 are built around the basic safety standards of IAEA Safety Series No. 9 [9] which, in turn, carries into effect the recommendations of ICRP 26 [10].

ICRP 26 was the starting point for a Euratom Directive which laid down basic safety standards for "the health protection of the general public and workers against the dangers of ionising radiations". Within the UK this Directive has been implemented by the Ionising Radiations Regulations 1985 (IR Regulations).

The system of dose restriction is based on the three well-known principles:

1. Justification
2. Optimization
3. Dose limitation.
The Regulations also make provision for classification of workers, medical examination, approval of dosimetry services, record-keeping, hazard assessments, contingency arrangements, preparation of local rules, training and appointment of radiation protection advisers and radiation protection supervisors.

2.2. Application of the Regulations to transport

There are areas of the regulatory provisions that are of specific interest to the transport industry.

2.2.1. Packaging and labelling

There is a requirement that "any person who knowingly causes or permits a radioactive substance to be transported shall ensure, as far as is reasonably practicable, that the substance is kept in a suitable receptacle". The Approved Code of Practice elaborates on what is meant by 'suitable receptacle' in this context by reference to the packaging and labelling requirements of IAEA Safety Series No. 6. Transport has been defined so that a consignment of radioactive material is being 'transported' from the time it is loaded onto a means of transport, until it is unloaded, provided that some part of the journey is through a 'public place'. The reason for transport commencing at the point of loading and ceasing at the unloading point is to enable enforcement of packaging and labelling standards to be carried out at the consignors, trans-shipment points, and at the consignees, enforcement being easier to accomplish at fixed premises.

2.2.2. Designation of controlled areas

The Regulations require controlled areas to be identified. One method that may be adopted by the carrier is to designate the whole area bounded by the 7.5 μSv/h contour. Guidance is being given to enable carriers to determine the extent of any such controlled area by reference to the total "transport index" of the consignment. Clearly, controlled area conditions must not be created in areas where the carrier does not have control, e.g. a public place. In general, the person accompanying the moving source should be regarded as being in a potential controlled or supervised area, but bystanders are not likely to be in that situation. Under normal conditions of transport it is unlikely that a controlled area will exist outside the edges of a conveyance. However, when the conveyance is stationary, it may be necessary to consider demarcating a controlled area around the conveyance. This may be readily accomplished on private premises, but it would be more difficult in a public place and contingency arrangements should cater for such a situation.
2.2.3. Hazard surveys

During the formulation of contingency arrangements, it is necessary to evaluate the problems likely to be encountered following foreseeable accidents and incidents. When the quantity of a radioactive material likely to be carried in any one load exceeds a specified quantity, which is a function of radiotoxicity, there is a requirement that a formal hazard survey be carried out and that a written report be sent to the Health and Safety Executive (HSE).

2.2.4. Contingency arrangements

The nature and scale of contingency arrangements will have to be tailored to factors such as the scale of the organization’s transport operations, the type of radioactive material carried, the mode(s) of the transport, etc. The arrangements should include as many of the following features as are relevant:

1. Managerial arrangements for dealing with the situation;
2. Means of minimizing the effects of the incident and safely restoring normal conditions;
3. Provision of equipment, including personal protective equipment and monitoring facilities;
4. Procedures for liaison with outside bodies.

This list is not exhaustive. It will need to be modified in accordance with the precise nature of the organization’s transport operation. It is essential that the procedures, equipment and facilities necessary for the effective operation of the contingency arrangements are kept under review and the ‘local rules’ should contain an up-to-date record of the arrangements and the responsibilities of individuals.

3. COMPLIANCE ASSURANCE ARRANGEMENTS

3.1. Introduction

Compliance assurance measures currently undertaken directly by the Department of Transport include the following:

1. Design and criticality assessment, including a detailed study of the applicant’s safety report, supporting test results, text and drawings.
2. Witnessing of regulatory proof testing.
3. Witnessing of packaging manufacture.
4. Assessment of QA programmes for manufacture.
5. Issuance of relevant approval certificates.
(6) Observing selected transport operations, including despatch, carriage and receipt.

(7) Issuance of guidance notes concerning RAM transport to industry.

The 1973 Edition of the IAEA Regulations (para. 150) was somewhat vague in defining responsibilities and prescribing who should carry out CA activities. However, the 1985 Regulations now define CA more clearly (paras 117 and 210) and assign responsibility directly to the competent authority. Clear parameters or areas of interest are now stated where suitable CA measures should apply as part of a competent authority's programme.

3.2. Compliance assurance programmes

Recognizing the change of emphasis concerning CA, the UK competent authority embarked upon the development of a suitable programme which would encompass all the necessary CA activities. The development of the programme was carefully considered because, not only should assurance of compliance be achieved, but also evidence of such assurance needs to be provided. Bearing in mind the IAEA target date for full implementation of the 1985 Regulations and the UK legislative system, it was decided to develop the programme, as shown in Fig. 1, so that total CA would be possible by the target date.

3.3. Compliance assurance/quality assurance relationship

The Department of Transport recognized that the industry was required to establish relevant QA programmes for all aspects of RAM transport in order to ensure compliance with the IAEA requirements and that a fundamental aspect of any QA programme is the carrying out of a self-audit and review of that programme. It was therefore decided that an effective way to assure many aspects of compliance would be for the Department to examine those QA programmes operated by industry and the associated audit/review activities in particular.

As QA measures, techniques and standards are widely applied and understood in the UK with several national and internationally accepted QA standards, e.g. BS-5750, BS-5882, DEF-STAN 05-21, IAEA 50-C-QA, being employed, it was decided not to develop another QA standard for the transport of RAM, but to permit existing standards to be utilized in the development of the required QA programmes. The relevant draft appendices of the recently revised IAEA Advisory Material have been made available to interested organizations for information and for comparison with their own developing or established QA programmes.
1 Determine existing level of operating QA programmes
2 Relate 1 to national and international QA standards
3 Determine what independent auditing or assessment occurs (3rd Parties such as B.S.I.)
4 Identify the variety and types of organisations involved
5 Identify peculiar Departmental requirements and other Government Departments’ interests
6 Develop Departmental Compliance/Quality Assurance programme
7 Obtain DT support for the programme
8 Advise all involved organisations of Department’s plans
9 Implement audits of existing QA programmes as required by 1973 edition of Regulations
10 Implement 1985 edition of IAEA Regulations
11 Build up Departmental record of audits to 1985 Regulations
12 Monitor corrective actions
13 Review/Confirm Departmental Compliance/Quality Assurance programme
14 Witnessing of Maintenance and Use operations
15 Witnessing of Manufacturing operations
16 Witnessing of Testing
17 Witness Transport operations
18 Design Approval Assessment including Specific Quality plans
19 Regulations, Incidents involving RAM Transport
21 Continued periodic auditing using information gleaned from:
22 Long term continual review of Departmental Compliance QA programme involving:
23 Design Assessment
24 Audits of QA programmes
25 Maintenance and Servicing
26 Witnessing of testing
27 Maintenance of Department’s Compliance / QA programme
28 Witnessing Manufacture
29 Monitoring of Transport operations
30 Regulatory bodies input/feedback

**FIG. 1.** The compliance assurance programme.
3.4. Implementation of compliance assurance programmes

The implementation of the CA programme is a gradual process incorporating consultation with other interested Government bodies and industry in order to achieve a controlled, effective transition to compliance with the 1985 IAEA Regulations. The monitoring and auditing to existing arrangements promulgated under the provisions of the previous editions of the IAEA Regulations have been placed on a more formal basis and as applications for approval against 1985 IAEA Regulations are processed, full compliance with those requirements is verified and recorded. The Department of Transport has developed a procedure for the approval and auditing of applicants and other organizations’ QA programmes which, over a period of time, will generate the necessary evidence to demonstrate compliance.

A significant part of the initial effort expended by the Department in implementing its CA programme has been to inform and educate all involved in what exactly is meant and required to enable the competent authority to further assure compliance with the 1985 IAEA Regulations.

3.5. Review

In the same way that the review process is an important part of any good QA programme, the CA programme must be reviewed for its continuing effectiveness and objectives. The Department of Transport sees the review process as a way of maintaining a ‘live’ and reactive programme, which also provides continued confidence in both the Regulations and the competent authority’s activities and role.

3.6. Enforcement

Enforcement in the UK has been achieved primarily as follows:

(1) Enforcement through the certificate approval process (Department of Transport only).
(2) Enforcement of regulatory testing (Department of Transport only).
(3) Examination of transport operations, including random inspections (all modal authorities).
(4) Observations of manufacturing operations (Department of Transport only).
(5) Investigation of incidents involving RAM transport and follow-up action (Department of Transport and others, as appropriate).

The Department’s CA programme provides for these activities to continue in a more orderly and planned manner in order to generate the necessary evidence of compliance. To assist in a uniform integrated approach to enforcement within the UK, a joint Department of Transport/HSE Enforcement Liaison
Committee (ELCTRAM) has been formed in which all the modal authorities are represented. It is anticipated that this committee will provide a valuable forum for discussion and determination of enforcement strategy.

3.7. The future

It is expected that the UK’s CA programme will demonstrate that the 1985 IAEA Regulations are being complied with, enabling other national competent authorities to have a high level of confidence in UK competent authority activities. The recent IAEA initiative to develop CA guidance material should greatly assist in achieving and maintaining a common understanding of the subject.

REFERENCES

COMPLIANCE ASSURANCE IN THE FEDERAL REPUBLIC OF GERMANY

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Abstract

COMPLIANCE ASSURANCE IN THE FEDERAL REPUBLIC OF GERMANY.

The 1985 Edition of the IAEA Transport Regulations require the establishment and execution of a programme by the competent authorities to assure compliance with the Regulations. In the Federal Republic of Germany, this task is in part already practised by approving/licensing of sealed sources, Type B and fissile material packages and shipments even of smaller quantities of radioactive material. The manufacture and use of approved packages are subject to inspection. All legal provisions for inspection and enforcement actions for radioactive material transport are already available in parallel in two sets of legislation — under the act on the transport of dangerous goods and the atomic energy act. The splitting of competence between Federal and Local State authorities needs careful adaptation to avoid overlapping and gaps. Whereas the Federal authorities already discharge their responsibility by adequate actions, the Local State authorities are still engaged in completing their possibilities. The paper is an inventory of those responsibilities and activities, including their legal basis.

1. RESPONSIBILITIES

Relative to the former text on compliance assurance in the 1973 Revised Edition of IAEA Safety Series No.6, Para. 210 of the 1985 Edition clearly states that:

"The competent authority is responsible for assuring compliance with these Regulations" and that

"Means to discharge this responsibility include the establishment and execution of a programme for monitoring the design, manufacture, testing, inspection and maintenance of packaging, and the preparation, documentation, handling and stowage of packages by consignors and carriers."

The legal basis in the Federal Republic of Germany for the responsibilities of ‘the competent authority’ for the national and international transport of radioactive material is given in both transport and atomic energy legislation. Whereas in general the transport legislation covers mainly safety aspects, the atomic energy legislation deals with administrative matters, safeguards and insurance questions. However, under compliance assurance (CA) aspects both legislations must be observed, because both contain relevant requirements.

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1.1. Transport legislation

The Transport of Dangerous Goods Act [1] is the cover under which all the regulations for the transport of dangerous goods for all modes of national and international transport are valid. This Act authorizes (§ 3) the Federal Government (delegation to the Department of Transport is possible) to provide regulations and general regulatory guides with respect to (inter alia):

- licensing of dangerous goods, their packaging, packing and stowage
- licensing of shipments
- safety provisions for drivers under normal transport and accident conditions
- training and medical surveillance of drivers and other responsible people.

In the Act, § 5 indicates those areas where the Federal State has general competence and thus becomes responsible for compliance with the regulations. These areas are:

- Federal rail transport
- air transport
- transport on sea and Federal inland waterways including Federal harbours.

Special provisions are allowed for the army, police, etc. Implicitly, all the remaining areas become subjects for the Local States (Bundesländer). These are:

- road transport
- transport on non-Federal waterways/harbours
- transport by non-Federal rail
- transport underground (mines).

The most important paragraph with respect to CA (§ 9) requires supervision ("Überwachung") and provides the competent authority with the right of access to any information, conveyance, site/plant, samples of dangerous goods and packagings for official inspection, including the right to inspect the manufacture of packagings, tanks, containers and conveyances.

The possibility of civil penalties in the event of defined violations is provided for in § 10.

The following responsible institutions are denoted in the mode specific transport regulations for road, rail, sea, air and inland waterways, specifically for class 7:

- BAM (Federal Institute for Materials Testing):
  - approval for special form radioactive materials

- PTB (Physical-Technical Federal Institute):
  - approval for Type B and fissile materials packages
  - licensing of shipments (Type B, fissile materials and special arrangement).

In the case of road transport, each Local State regulates the competence of its authorities independently, on the basis of general legislation on safety and order. As a result, it is mainly the State police which is responsible for compliance assurance for transport on public roads and the so-called "Gewerbeaufsichtsämter" for compliance assurance in plants/sites and areas outside public roads.
However, as far as radioactive material transport is concerned, the last mentioned competences, valid for the transport of dangerous goods in general, are overruled by the prescriptions of the atomic energy legislation.

1.2. Atomic energy legislation

With respect to the safe transport and compliance assurance (other than non-proliferation, safeguards and insurances) the Atomic Energy Act [2] and the Radiation Protection Regulation [3] are relevant.


Paragraph 19 of [2] stipulates official supervision ("Staatliche Aufsicht", which has the same meaning as "überwachung", the term used in the transport legislation) and provides the competent authority with the right of access to any site/plant or information, and the right to perform the necessary inspections and tests.

In § 23 of [2], the PTB is denoted as responsible (among others) for the licensing of shipments of fissile materials and large sources (in the sense of the 1967 Edition of the IAEA Regulations).
Paragraph 24 of [2] ascribes the function of licensing of shipments of non-fissile materials and the supervision of these transports to the Local States, with the exception of Federal rail transport and transport by the army.

Paragraphs 46 of [2] and 81 of [3] provide for the possibility of civil penalties in the event of defined violations.

Under these paragraphs each Local State regulates the competence of its authorities independently. As a result, there are different authorities for the shipment of fissile and non-fissile materials.

Figure 1 provides a description of the legal background while Fig. 2 summarizes the general and specific competences of Federal and Local State authorities.

2. ACTIVITIES

2.1. Compliance assurance programme

Paragraph 210 of the 1985 Edition of the IAEA Regulations requires the establishment and execution of a programme to cover all aspects of compliance assurance. To become clear about the scope of such a programme it is helpful,
especially with respect to international harmony, to take into account the advisory and explanatory material [4, 5] elaborated by the IAEA (Safety Series Nos 7 and 37). On the basis of these guides it can easily be judged how far such a programme is already covered by the activities of the authorities concerned.

2.2. Status of activities

2.2.1. Review and assessment activities

All of the legal possibilities for approving and licensing described above are applied by the responsible authorities. This is especially true for the approval of special form radioactive material, Type B and fissile materials packages, where the competent authorities PTB and BAM, partly using the manpower of the TUV (Technical Supervision Institutions, which are non-profit organizations), are themselves performing capsule and Type B tests and systematically inspect the manufacture of approved packages and packages in use. Great emphasis is laid on the approval of quality assurance programmes for the manufacture of approved package designs, for which a special guide has been issued [6]. Full descriptions of these activities can be found in [7]. All of the prescribed possibilities of licensing shipments of 'large sources', fissile material and smaller quantities are used by the authorities on the basis of a review of the application, which shall provide all necessary data on contents, packaging, emergency response and so on.

2.2.2. Inspection and enforcement

Whereas all major steps in the testing, manufacture and use of approved package designs are inspected systematically by the competent (Federal) authorities, no systematic actions are established for the Local States to prove compliance with the technical provisions for non-approved packages (industrial, Type A packages), though Refs [2] and [3] require (inter alia) compliance with the transport regulations as a necessary presupposition for the shipment licences.

No systematic inspections by the competent authorities are performed to verify the prescriptions for packing and shipment as for example selection of proper packagings, compliance with admissible contents, dose rates (package surface, 1 m rate, conveyance surface, driver seat) surface contamination, labelling, stowage and compliance with special provisions as laid down in shipment licences. Systematic controls (at least of surface contamination) are performed in those cases where packages are to leave the controlled areas of nuclear facilities. These are checked by those in the facilities responsible for radiation protection, as with any item leaving the areas.
2.2.3. Radiation protection

The IAEA Regulations assume transport workers to be members of the public, not subject to regular radiation monitoring. Economic effectiveness considerations, however, can lead to deviations in the doses admissible to the public which would require those workers to be monitored officially. The decision is up to the responsible officials, who shall be aware of the actual doses received by transport workers in transport operations. With the exception of the Federal rail system, where systematic surveillance of personal doses is applied and thus detailed records are available though none of the workers are classified to be supervised officially, no systematic official surveillance actions for non-classified workers are known for other modes of transport.

The evaluation of radiation doses to members of the public has been prepared by the Government within the framework of the research programme PSE [8], finalized at the end of 1984, which was planned to elaborate all necessary methods and procedures to determine the radiological impact of radioactive material transport on workers and members of the public.

2.2.4. Accident provisions

The dangerous goods regulation for road transport, for example, requires that drivers are informed about the nature and hazards of the dangerous goods and that instructions in writing (“schriftliche Weisungen”) are handed over to them (prior to each transport). These instructions, prepared by the manufacturer or sender for each dangerous substance or class of dangerous substances, shall specify concisely:

- “The nature of the danger inherent in the dangerous substances carried, and the safety measures that need to be taken to avert it;
- The action to be taken and treatment to be given in the event of persons coming into contact with the goods carried or with any substances which might escape therefrom;
- The measures to be taken in the event of fire and, in particular, the firefighting appliances or equipment not to be used;
- The measures to be taken in the event of breakage or deterioration of packagings or of the dangerous substances carried, particularly where such dangerous substances have spilled over the road.”

Except for the transport of radiogammagraphic cameras, containing sealed sources [9], there is no generally accepted pattern of such instructions nor official guidance on how to specify the contents for the main applications (such as for example LSA, UF₆, spent fuel or fissile material) which would assure that the instructions correspond to the safety philosophy of the IAEA Regulations.
3. CONCLUSIONS

The 1985 Edition of the IAEA Transport Regulations are planned to be introduced into the European and Federal German transport legislation at the beginning of the nineties. As a result of negotiations on the RID/ADR level so far, it is not planned to include all of the provisions of IAEA Safety Series No.6 in the regulatory text. This is especially true for the above mentioned Para. 210 on compliance assurance! A responsible treatment of the impact of the transport of radioactive material on public safety and opinion will lead, whether this is prescribed or not, to the same steps by the competent authority to assure compliance with the regulations. The status analysis reveals that already the major points of interest are covered by adequate actions of the authorities; other issues still need systematic action, especially by the Local State authorities. The establishment of an overall programme, as recommended, would indeed provide for greater reliability in compliance with the transport regulations.

REFERENCES


TRANSPORT OF RADIOACTIVE MATERIALS AS PART OF THE TRANSPORT OF DANGEROUS GOODS

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Abstract

TRANSPORT OF RADIOACTIVE MATERIALS AS PART OF THE TRANSPORT OF DANGEROUS GOODS.

The transport of radioactive materials (RAM) can be seen as being a part of the transport of dangerous goods. The United Nations has provided recommendations for the classes of dangerous goods 1 to 6 and 8 to 9, while class 7 requirements have been developed by the IAEA. Both UN and IAEA rules cover all modes of transport, and their work is being followed by other transport-mode related bodies. In all classes of dangerous goods, except class 7, the safe containment of substances under normal transport conditions is required. Some safety margins are provided, for example by safety factors in relation to working loads and specific material requirements. Class 7, however, defines an accident-safe packaging, the so-called Type B(U) package, that is designed, tested and approved at a much higher level of safety than other types of packagings, which, in any case, are only for small quantities of RAM and which in this respect are comparable to the UN packagings mentioned above. The marking of the packagings with the UN symbol and with the Type B(U) plate should guarantee unhindered shipment. In general, the UN and IAEA bases are accepted all over the world.

1. INTRODUCTION

The transport of radioactive materials (RAM) can be considered to be a part of the transport of dangerous goods. In general, the United Nations Recommendations for the Transport of Dangerous Goods [1] present the internationally accepted level of safety. Since the same set of requirements are used all over the world, these recommendations are an important component of the worldwide safety system.

The United Nations provides recommendations for all classes of dangerous goods, except for class 7 radioactive materials. While the UN Recommendations are valid for all transport modes for classes 1 (explosives), 2 (gases), 3 (flammable liquids), 4 (flammable solids), 5 (oxidizing substances), 6 (poisonous substances), 8 (corrosives) and 9 (miscellaneous dangerous goods), class 7 requirements have been developed by the IAEA on behalf of the UN and cover UN and all other modes of transport. It is expected that the IAEA will continue to follow general UN guidelines in order that its regulations fit into the whole system of rules [2].
United Nations and IAEA work is followed by the 'mode-related' bodies (Fig. 1). The International Civil Aviation Organization (ICAO) has derived a set of regulations, based on UN and IAEA work, using a set of rules for air transport entitled ICAO Technical Instructions [3]. The International Maritime Organization (IMO) has done the same for the sea mode with its International Maritime Dangerous Goods (IMDG) Code [4], which governs the transport of dangerous goods on the sea. There are no worldwide accepted regulations for road and rail, but there are rules for Europe. The UN Economic Commission for Europe (ECE) has been the leading organization in the preparation of the Accord européen relatif au transport international des marchandises dangereuses par route (ADR) [5]. Closely linked with the UN through the 'Joint Meeting' is the Office central des transports internationaux par chemins de fer (OCTI), which prepared the Règlement international concernant le transport des marchandises dangereuses par chemin de fer (RID) [6]. Both ADR and RID are conventions valid in their member states.

2. UN AND IAEA ACTIVITIES

The goals of the UN and the IAEA are to ensure the safety of people, property and the environment. United Nations Recommendations give both the principles of classification of the substances to be transported and the definition of the classes. The main areas are:

- Listing of dangerous goods by UN number
- General packaging requirements
- Testing procedures
- Marking
- Labelling and placarding
- Shipping documents.
All of these items are necessary to eliminate risk during transport, or to reduce it to a generally acceptable minimum. Uniformity at the world level for all modes of transport is a key safety element.

Dangerous goods are listed in three groups, the so-called packaging groups I, II and III. Substances in packaging group II are dangerous, group III contains substances less dangerous than normal (group II) and substances in group I are more dangerous than normal. The overall principle is to keep the substances contained during normal shipping conditions. For example, packaging group II requires a drop height of 1.2 m, packagings of group III are dropped from 0.8 m and packagings for group I substances are tested by a free drop from 1.8 m. Additionally, the principle is not that the tests should represent real shipping conditions or real incident conditions, but should instead provide the same degree of deformation and damage that would be caused by incident conditions.

While the capacity for packages ranges up to 400 kg (450 L), that of the intermediate bulk container (IBC) has gone up, to 3000 L. Both packagings and IBCs will bear UN markings, guaranteeing safe containment. Prior to the application of the UN markings, tests and approvals by the competent authorities are performed.

In addition to packagings and IBCs, the list of UN-approved forms of containment is completed by tank containers (TCs). These start from 450 L and reach a level of about 30 000 L. Normally the tanks are fitted with an ISO specification framework of approximately 6 m (20 ft) size. Unfortunately, the UN TC is not in service. While the mode-related bodies (ICAO, IMO, ECE, OCTI) have taken over the responsibility for packagings and IBCs, their specifications deviate for TCs, even in the design requirements. Thus, the TC accepted worldwide is an IMO portable tank.

Since the IAEA has ruled that all radioactive substances are RAM (even those which have a secondary risk higher than the radioactive one), every radioactive substance is subject to its regulations. The RAM can be shipped in strong industrial packagings, or in Type A and Type B packagings. Their basis is the so-called $A_1/A_2$ figure, with $A_1$ and $A_2$ coming from the maximum permissible intake (MPI). As a guideline, $10^3$ MPI can be shipped in a normal packaging, and $10^6$ MPI in a Type A packaging. Both types of packagings have to withstand tests which would cause damage such as that which would occur under normal shipping or incident conditions. In this respect, the relationship to the UN system of regulations is clear. Furthermore, they do not need official approval, except if the contents are fissile. Thus, in this case, a registration procedure is lacking.

When shipping more than $10^6 A_1/A_2$, a Type B packaging is required. This is an accident-safe packaging. The same principles cover this procedure, but at a much higher level of safety. Here, the intention is not to simulate accidents or to reproduce in a synthesis the outcome of risk studies, but to provide, by using higher levels of specifications, the same degree of damage as would be caused under accident conditions. The result is to be seen in the so-called Type B(U) packaging, the U standing for unilateral. Here again we find a similarity to UN markings for packagings of all
classes other than class 7. Type B(U) guarantees unhindered international shipment, with testing and approval by a competent authority as the necessary precaution.

3. IAEA PACKAGINGS

3.1. General remarks

The packagings described by the IAEA do not at present fit into the UN system. There is no limited range of weight or volume comparable to the limits of 400 kg or 450 L, no comparable testing and approval procedure for industrial and Type A packagings. The very complex behaviour of RAM may be one reason for this deviation, though a second reason is to be found in the working procedures of the UN and IAEA, which are characterized by a lack of information exchange. It would be a great advantage if the IAEA and the UN were to work together more closely to improve this communications flow.

In any future review process between the UN and IAEA, it should be stated that industrial and Type A packagings, as well as UF₆ cylinders, should be treated by both in a co-ordinated manner, especially UF₆. It would be helpful if the UN were to produce detailed requirements for the transport of pressure vessels in general to complete the system for packagings. On that basis, the IAEA could work out the special provisions for UF₆ cylinders even when the radioactive risk is lower than the chemical one.

3.2. Type B(U) packagings

What are the reasons for the unhindered international transport of large quantities of RAM in Type B(U) packagings? The very high level of design requirements that have lead to an accident-safe cask and the detailed work of experts at the IAEA once a first impetus is given. Problems and special tasks are handled by consultants and different technical committees to cover areas in the necessary depth. While a Co-ordinated Research Programme is usually initiated and continued, it is the panels that have really developed the internationally accepted Safety Series Nos 6 and 37 [2, 7]. The work is focused by the scientific secretariat of the IAEA, which provides both administrative and scientific guidance.

Though every Member State is assured that the results of IAEA efforts are a very high level system of regulation, the IAEA is anxious to hold to these levels. Thus if it were found that the 9 m drop test did not represent a heavy mechanical load for lightweight packagings the crush test could be introduced into the new regulations [2, 8].

Another advantage of a system of higher levels of safety is that these levels are so conservative as to allow different engineering approaches not requiring only one prescribed procedure. Therefore the regulations do not contain specific material
requirements or one specified procedure. Equivalent evaluations will always achieve the safety goal in a tolerable range (Fig. 2).

Thus, IAEA safety philosophy allows for the use of the approval procedure, the use of analogy, calculations, model testing and prototype testing to meet the requirements. For instance, brittle fracture as a material datum only has to be taken into consideration, but no special verification procedure is required.

The criteria for an approved packaging are given by the limitation of the surface dose rate, the leaktightness and the subcriticality of fissile material, i.e. primary conditions for meeting the requirements. The IAEA has renounced so-called secondary conditions derived from primary ones.

4. EXPERIENCE WITH UN/IAEA REGULATIONS

Packagings for dangerous goods having UN markings should in theory be transported unhindered between all Member States, since it has been internationally understood that these markings would guarantee adherence to UN standards. The intention here was to allow the free flow of packagings, not only for transit shipments but also for domestic transport of packagings coming from abroad.

Up until now no rejections have been observed except by the United States of America, where UN marked packagings can arrive at a port or airport, but then need additional US Department of Transport (DOT) approval, generally issued by an
institution/company registered at the DOT. The DOT regulations describe the packaging required for the transport of dangerous goods. It has been decided that as long as the DOT does not accept UN markings as guaranteeing safety against normal conditions of transport, the Ministry of Transport of the Federal Republic of Germany will also not accept packagings with only DOT labels for domestic shipments. It would be helpful if the DOT adopted UN Recommendations into their national laws. However, in the interim, it would be useful if a system of reciprocal allowances could be devised.

In contrast, Type B(U) packagings can be used internationally and for domestic transport in foreign countries. All IAEA Member States have accepted the Type B(U) concept as being an internationally accepted safety standard, without any reservations. The act of validation therefore should normally only be a formal administrative one, i.e. no additional technical evaluation is necessary. The situation in this regard has been quite satisfactory for many years, especially after the creation of the Radioactive Transport Study Group (RTSG), where the competent authorities involved work together. With this basis of confidence, it should be possible to ship RAM in Type B(U) packagings and it should be self-evident that necessary information must be given on a voluntary basis. Many Type B(U) packagings are designed only for domestic transport, while others are used internationally and are revalidated by the countries concerned. All of this information is continuously registered at the IAEA (where a list of current certificates is in preparation Ref. [9]). All validations of this list have been issued without additional technical evaluations.

However, this process is far from being self-evident. The following is given as an example of the problem. For the shipment of spent fuel elements, a new generation of casks has been designed using, for the first time, modular cast iron as cask material. The use of this material is in line with IAEA Regulations, which do not prescribe certain types of material. Nevertheless, extended testings and evaluations have been carried out, including full-scale drop tests at -40°C. These have been performed for the first time for casks having a total mass of up to 100 t [10]. The approval procedure has successfully been completed and has led to the issue of Type B(U) certificates of approval, validated in the meantime by the competent authorities by normal administrative procedures.

Some difficulties arose, however, during the validation process for domestic transport purposes in France and for shipments to and in the USA. Both countries insisted on new evaluation procedures of their own. While France accepted ADR/RID shipments and only required specific investigations concerning domestic use, the US DOT did not (and continues not to) allow shipment to the USA. France, which originally would have chosen another engineering approach for its approval procedure, has finally accepted that the original approach is only one of the possible satisfactory evaluations as determined by the IAEA, since it takes into account detailed information presented by the country of origin. The USA, however, does not appear to accept the philosophy of different, but equivalent, engineering approaches, insisting instead on its own evaluation system even for incoming shipments.
Thus, it would be better, in the first instance, to start from a general basis of confidence between well-known competent authorities to allow shipments based on IAEA guidelines. If there are any concerns left that the IAEA Regulations do not cover, it would be better to submit these problems to the IAEA itself for consideration within the normal revision process.

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LEAKAGE TEST METHODS DEMONSTRATING INTEGRITY OF TRANSPORT PACKAGINGS, SEALED RADIOACTIVE SOURCES AND SPECIAL FORM RADIOACTIVE MATERIAL

A review of standardization efforts and requirements

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Abstract

LEAKAGE TEST METHODS DEMONSTRATING INTEGRITY OF TRANSPORT PACKAGINGS, SEALED RADIOACTIVE SOURCES AND SPECIAL FORM RADIOACTIVE MATERIAL: A REVIEW OF STANDARDIZATION EFFORTS AND REQUIREMENTS.

The paper gives suggestions and ideas on ways to bring together IAEA requirements, ISO standardization efforts and practical experience. A number of items are considered:

(1) Comments on activity release limits which are derived either from practical detection limits or by radiological assessment. (2) Graphical representation of a simple model of gas flow through narrow gaps which is complementary to the capillary model and which is, for example, more appropriate for leaks caused by accidental impacts on sealing regions. This approach is useful for any assessment of potential powder leakage. (3) A range of permeation leakage per sealing length as a function of temperature for the most important elastomer types and based on an evaluation of literature and on practical measurements, including a graph for practical design requirements, is presented. As a result of suggestions made jointly by several interested experts, International Organization for Standardization/Technical Committee 85 (ISO/TC85) decided recently to consider leakage tests on packages for shipment of radioactive material as a potential topic of future international standardization work. Balloting showed that 20 out of 22 P-Members (i.e. members willing to participate actively) of TC85 were in favour of this proposal. A relevant American National Standards Institute (ANSI) standard, N14.5 (issued in 1977), is closely related to IAEA Safety Series No. 6. At present this standard is under review and may prove to be a very useful basis for further work in this field. It also describes a number of test methods which in practice are more suited to small, sealed radioactive sources than to large packages. A close relationship is maintained with the work of ISO/TC85/SC2/WG11, on leak testing of sealed sources, which is currently engaged in developing ISO Technical Report 4826 into an ISO Standard.
1. LIMITS FOR ACTIVITIES AND LEAKAGE RATES

IAEA Transport Regulations are based on clear and sound radiological principles:

(1) Maximum activity contents for different types of packagings are based on radiological assessments of potential accidents.

- Dose equivalent $H = \Gamma \times A_1 \times 0.5 \text{ h/(1 m)}^2 = 0.05 \text{ Sv}$.
- $10^{-6} A_2 \rightarrow$ annual limit on intake (ALI) = 0.05 Sv.

This detailed, though somewhat complex, system could offer many advantages in cases of accidental impacts of packagings and sealed sources not caused during transport as compared with some regulatory requirements that are based on the more simplified system of exemption limits based on toxicity groups. Table I shows the ranges of both systems.

(2) Activity release limits are derived from the $A_2$ values. It is suggested that there were three main reasons for increasing the allowable release rates for Type B(U) packages under accident conditions from $10^{-3}$ A$_2$ per week to $A_2$ per week.

(i) The relatively small difference in relation to the allowed release rate during normal transport, $10^{-6}$ per hour being about $1/6$ of $10^{-3}$ A$_2$ per week.

(ii) A miscalculation by the same factor of $10^3$ during one of the first meetings of a group of experts [1].

(iii) A new scenario assuming that all persons in the immediate vicinity of the damaged package would be rapidly evacuated to distances of at least 50 to 200 m, or would be working under health physics control [2]. It has been shown that an activity release of $10^{-3} A_2$ per week would also take into account a scenario in which any person staying at a distance of some metres from the damaged package would not receive more than the ALI activity [3]. However, from practical experience, it is concluded here that an activity release of between $10^{-3} A_2$ and $A_2$ would not occur very often, thus in practice not adversely affecting radiation protection. However, it appears that the 'as low as reasonably achievable' (ALARA) principle has not been taken into account. In the Federal Republic of Germany discussions are about to be initiated with fire-fighting services on radiation risks during fire accidents. It would have been very convincing to offer the former argument, that even without any radiation protection precautions, the risk from Type B packagings involved in a fire would be acceptable. Because of the new factor of 1000 this argument can no longer be used.
<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>$A_2$ (S.S. 6)$^a$ (TBq)</th>
<th>Minimum Ali by inhalation (Bq)</th>
<th>$\text{ALI} \times 10^6$ (TBq)</th>
<th>Exemption limit (Bq)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ac-227</td>
<td>$2 \times 10^{-5}$</td>
<td>$2 \times 10^1$</td>
<td>$2 \times 10^{-5}$</td>
<td>$5 \times 10^{3}$</td>
</tr>
<tr>
<td>Pu-239</td>
<td>$2 \times 10^{-4}$</td>
<td>$2 \times 10^2$</td>
<td>$2 \times 10^{-4}$</td>
<td>$5 \times 10^{3}$</td>
</tr>
<tr>
<td>Sr-90</td>
<td>0.1</td>
<td>$1 \times 10^5$</td>
<td>0.1</td>
<td>$5 \times 10^4$</td>
</tr>
<tr>
<td>Cs-137</td>
<td>0.5</td>
<td>$6 \times 10^6$</td>
<td>6</td>
<td>$5 \times 10^5$</td>
</tr>
<tr>
<td>H°3 (water)</td>
<td>40$^b$</td>
<td>$3 \times 10^9$</td>
<td>$10^3$</td>
<td>$5 \times 10^6$</td>
</tr>
<tr>
<td>I-129</td>
<td>$\infty$</td>
<td>$3 \times 10^5$</td>
<td>0.3</td>
<td>$5 \times 10^6$</td>
</tr>
</tbody>
</table>

$^a$ IAEA Safety Series No. 6.

$^b$ Ratio: $2 \times 10^6$.

$^c$ Ratio: $1 \times 10^3$.

(3) According to the American National Standards Institute (ANSI) standard NL4.5, a package is deemed leaktight if the standard air leakage rate is equal to or less than $10^{-7}$ mbar-L·s$^{-1}$ [4]. While this is acceptable, it should be noted that such small leakage rates can only be measured using helium and the equivalent air leakage rate then has to be calculated. In the case of molecular flow, the leakage rate is inversely proportional to the square root of the average molecular weight. Therefore, 2.7 times more helium atoms than air molecules flow through such a leak. Indeed, some manufacturers rate their helium leak detectors in air leakage equivalents because the figure is lower, making the leak detector appear more sensitive. It is the contention here that a factor of 2.7 is not very significant in practice, though the alternative factor of 1.96 is also not acceptable, being based on the arbitrary model of a capillary 1 cm in length.

(4) An activity leakage limit of 2 kBq after accident testing of special form radioactive material seems to be acceptable. A limit of 0.2 kBq for leakage testing of sealed sources can be justified because of the less severe tests simulating normal use. But for the future it is suggested that only one limit be set.

2. GAS FLOW MODELS

Leakage and permeation rates, $q$, depend on the pressure difference (upstream and downstream pressure) across the leak or material according to the relationship

$$q = f (p_u^x - p_d^y)^z$$

where $f$ describes the leak geometry, the fluid and the material. The parameters $x$, $y$ and $z$ are explained in Table II.
TABLE II. DEPENDENCE OF FLOW RATES ON PRESSURE

| Permeation of diatomic molecules through metals | 0.5 | 0.5 | 1 |
| Permeation of molecules through elastomers | 1 | 1 | 1 |
| Leakage of gases, molecular flow | 1 | 1 | 1 |
| Leakage of gases, viscous flow | 2 | 2 | 1 |
| Leakage of gases, turbulent flow | 2 | 2 | 4/7 |
| Leakage of gases, choked flow | 1 | $-\infty$ | 1 |
| Leakage of liquids, viscous flow | 1 | 1 | 1 |

a Both flow regimes together described by Knudsen's law

Draft ANSI N14.5 (1985) now includes many calculations based on a straight circular tube 1 cm in length that models the leakage path. Unfortunately, the condition for the occurrence of choked flow is based on a false assumption (applicable to an aperture only), thus invalidating some of the calculations. Choked (adiabatic) or sonic flow occurs under certain conditions of pressure and leak geometry. Assuming that there is a passage in the form of an aperture, and that the upstream pressure is kept constant, if the downstream pressure is gradually lowered, then the velocity through the aperture will increase until it reaches the speed of sound. For longer openings, such as capillaries and gaps, the maximum leakage rate of air at 298 K in the case of a choked flow is then $q = 20 \times A_{\text{min}} \times p_u$ (q being in mbar-L-s$^{-1}$, $A_{\text{min}}$ being the minimum cross-section within the leak path, in cm$^2$, $p_u$ being expressed in mbar). In Table B2 of draft ANSI N14.5, choked flow can be assumed only for hole diameters $>10^{-2}$ cm. Figure 1 shows the correlations between leakage rates and capillary diameter and gap width. Such a graph should be included in a new standard in addition to the relevant equations.

Some practical applications of these issues are given below.

(1) In an experiment, very high leakage rates were measured immediately before and after a drop test at 40°C of a cask model sealed using standard Viton O-rings. After ambient temperature was reached, the cask once more became leaktight. In other experiments, an O-ring of 16 mm $\times$ 5 mm size was mounted on a copper tube that was dipped into liquid nitrogen, the nitrogen being evacuated with a helium leak detector so that the O-ring was surrounded by helium. With a cooling rate of about $-5$ K per minute, the Viton gaskets showed a sharp increase in helium leakage at $-22$°C. At this temperature, standard Viton becomes inelastic. When lowering the temperature further, the elastomers shrank...
approximately ten times as much as the surrounding metal parts. The difference between the coefficients of linear thermal expansion for copper and Viton was about $1.5 \times 10^{-4}$ $\text{K}^{-1}$ which, related to a seal width of 4 mm, resulted in a gap of about 0.6 $\mu\text{m}$ width per K and a corresponding standard gas leakage rate of about $10^{-2}$ mbar$\cdot$s$^{-1}$. This effect is reversible, and would probably not affect any Type B(U) qualification. In practice, any radioactive contents would produce enough heat, ensuring that the O-ring would not be cooled to the minimum ambient temperature of $-40^\circ\text{C}$.

(2) To avoid corrosion of the metallic gaskets used for sealing dry spent fuel storage casks, the width of the gap between the cask body and the first lid is limited to 10 $\mu\text{m}$, thus preventing any significant mass transfer of corrosive fission products. This gap width is checked by performing a pressure rise test with a specified air leakage rate of some $10^2$ mbar$\cdot$s$^{-1}$.

(3) If a cask has been leaktight before any tests and shows a leakage rate after the tests, the possibility that a capillary has been formed can in most cases be excluded. A leakage evaluation based on a gap model might often result in a very small gap width, preventing any release of powder particles of radiological significance (most dangerous in the range of 0.1–10 $\mu\text{m}$ diameter).

Figure 2 shows one example of permeation leakage rate ranges per sealing length for some commonly used types of elastomers (some experimental results from the Bundesanstalt für Materialprüfung (BAM) are included [5]. Similar figures can be derived from data published by other manufacturers [6], such data providing some basic and typical information for designers on useful temperature ranges and the dependence of permeation on temperature.
3. NEED FOR TECHNICAL STANDARDS

Regarding leakage assessment, it is suggested that the interested parties concentrate their efforts on one item at a time which should be co-ordinated by the ISO and which should result in an internationally recognized standard and guide. In recent years some activities have been initiated in attempts to develop national standards, or better or more simple guides and comments. It would seem that ANSI N14.5 and parts of the old British Standards BS-3895 from 1976 [7] might be the most useful points of departure for future joint work in this field. However, ISO 2855 [8] seems to have no practical significance.

Two other items need clarification:

(1) Contamination measurement and assessment. These are sometimes also used as leakage detection methods. A very useful ISO draft is about to enter the stage of final voting [9].

(2) Dose rate measurement and assessment. Some radiation level limits are related to 'any point' of the surface. This requirement might move authorities to insist on Monte Carlo calculations of the dose rate. Discussions at various IAEA meetings have resulted in a consensus view that large instruments should not be deliberately selected to artificially reduce the apparent radiation level. From observations and from literature (e.g. measurements of radioisotope cardiac pacemakers [10]), it is concluded that there is a need for a relevant standard. One

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*FIG. 2. Temperature dependence of helium permeation through seals made of commonly used types of elastomers.*
alternative would be to adopt a more practical and reasonable requirement from some radiation protection standards, such as measurements at distances of 5 cm and 1 m from the external surface related to the centre of the sensitive volume of the detector and taking into account the mean reading for an area of 10 and 100 cm$^2$, respectively. It is useful to remember that radiation level limits have primarily been derived to avoid film exposures in excess of 0.1 mGy and that, conversely, film dosimetry is currently the most widely used method in radiation protection surveillance.

4. SEALED RADIOACTIVE SOURCES

It is unfortunate that many regulations, standards and guides use different terms to denote the same concepts, although the meanings are the same. There is a similar problem with terms such as 'transport' or 'shipment', 'wipe' or 'smear'. It would be useful to agree on the main terms and to keep equivalent terms in brackets — if needed.

Some very useful work on testing and verifying the safety of sealed radioactive sources has been performed in the United States of America. After having investigated 57 plants representing 18 industrial categories and using approximately 1200 sealed sources, the Battelle Memorial Institute published a report entitled Industrial Environmental Conditions of Sealed Radiation Sources [11]. Oak Ridge National Laboratory subsequently developed a system of test procedures simulating normal conditions of use and also investigated the safety margins of some source types under industrial accident conditions [12]. Parallel to this, a group composed of source and device manufacturers, and experts from authorities and national laboratories, developed N 5.10-1968, Classification of Sealed Radioactive Sources, which later became ISO 2919 (1980) and which contained some minor changes in the test conditions already accepted by ANSI.

Figure 3 shows some data on thermal testing. The upper curve describes the heating rate around or near the specimen — this being a package or a device containing, for example, a sealed source — and the lower curve describes the minimum allowed heating rate of the specimen, for example a sealed source itself. Sometimes these different test requirements are mixed up. Experimental heating rates of ionization-chamber smoke detectors and special form radioactive material during prototype testing by BAM lie well between the two curves.

Table III shows that there are already available nearly complete systems of regulations, standards and guides dealing with the integrity of sealed radioactive sources during manufacture, use and accidents. ISO/TC85/SC2/WG11, dealing with sealed sources, intends to prepare a single standard covering all requirements on sealed sources. Some further research work has been done to harmonize ISO and IAEA test requirements [13], in the process adding further experimental findings on leakage test methods to the basic work performed at ORNL [14].
<table>
<thead>
<tr>
<th>Object</th>
<th>ISO/IAEA</th>
<th>DIN</th>
<th>ANSI N.542-1977</th>
</tr>
</thead>
<tbody>
<tr>
<td>General requirements</td>
<td>1677 (1977/82)</td>
<td>25426 Part 1 (1977/86)</td>
<td>Main part</td>
</tr>
<tr>
<td>Resistance to normal conditions of use</td>
<td>2919 (1980/85)</td>
<td>25426 Part 2 (1979)</td>
<td>Appendix B</td>
</tr>
<tr>
<td>Quality assurance</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Resistance to transport accidents</td>
<td>S.S.6 (1979/85)</td>
<td>25426, Parts 1 and 2; Bavarian regulatory guide (1984)</td>
<td>Appendix E</td>
</tr>
<tr>
<td>Resistance to building fires</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Leakage tests:</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Methods after prototype tests</td>
<td>TR 4826 (1979/-)</td>
<td>25426 Part 3 (1982)</td>
<td>Appendix A</td>
</tr>
<tr>
<td>Methods for quality control</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Methods for current inspections</td>
<td></td>
<td>25426 Part 4 (1977/86)</td>
<td></td>
</tr>
<tr>
<td>Regular intervals for current inspections</td>
<td></td>
<td>Federal regulatory guide (1979) (Ö-Norm 5222/3 (1982))</td>
<td></td>
</tr>
<tr>
<td>Special applications (industry, medicine)</td>
<td></td>
<td>Some DIN standards</td>
<td>Appendices C + D</td>
</tr>
</tbody>
</table>
FIG. 3. Heating conditions for thermal testing of packagings, devices and sealed sources.

Mention must be made of an example of improper use of technical standards. In Bavaria recently, authorities published a Regulatory Guide for Radiation Protection of Firemen. It stipulates that users can handle, for example, any volatile radioactive substances with activities up to $10^4$ times the exemption limit (i.e. up to between 50 MBq and 50 GBq (Table I)) without any special fire protection precautions. However, to handle sealed sources with activities up to $10^7$ times the exemption limit, they must meet ISO Classification 66646 (including a hydraulic pressure test at 1700 bar) with a maximum activity release less than 0.2 kBq caused by each of these tests. Such sealed sources are not yet available in the Federal Republic of Germany (FRG). Thus, it seems in this case that the authorities have simply chosen the most severe test requirements printed anywhere. Even special form radioactive material specifications have been disregarded because of the very short heating time (10 min at 800°C) (Fig. 3), though in other cases in the FRG (e.g. radiography and medical therapy), inspection procedures are reduced if special form radioactive material is used. Because of some recent difficulties, the suggestion is made here that it would be convenient to establish an international register of special form radioactive material approvals and to have an exchange of experiences, of typical certificates and useful periods of validity (in the FRG now five years).

Since the main concern in this paper is that test requirements be well-founded and reasonable, the author is happy that the IAEA has succeeded in producing an integrated system of publications offering information on the ‘what’ (Safety Series No. 6), the ‘how’ (Safety Series No. 37) and last, but not least, the ‘why’ (Safety Series No. 7) of radioactive material transport.
REFERENCES

THE ANSI N14 STANDARDS COMMITTEE: PACKAGING AND TRANSPORTATION OF RADIOACTIVE MATERIALS

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Professional Analysis, Inc.*, Oak Ridge, Tennessee

M. WELCH
International Energy Associates Ltd., Washington, D.C.

United States of America

Presented by R.T. Haelsig

Abstract

THE ANSI N14 STANDARDS COMMITTEE: PACKAGING AND TRANSPORTATION OF RADIOACTIVE MATERIALS.

The American National Standards Institute (ANSI) N14 Committee has responsibility for developing national standards in the United States of America for the packaging and transportation of radioactive materials, but not including movement or handling during processing and manufacturing operations. The paper describes the organizational structure of ANSI N14 and the status of the N14 Standards.

1. INTRODUCTION

The ANSI N14 Committee operates as an accredited committee under the American National Standards Institute (ANSI). ANSI N14 has the responsibility of developing national standards for packaging and transportation of radioactive materials in the United States of America, but not including movement or handling during processing and manufacturing operations. ANSI provides the formal structure to ensure that standards that are designated 'American National Standards' have a national consensus of the various affected interests.

* Formerly associated with JBF Associates, Knoxville, TN, USA.
1. Secretary mails scope and ballot concerning need to full N14
2. Secretary tabulates results, copy to Chairman
3. Secretary mails statement to NSNB, copy to Chairman
4. Secretary receives approval, copy to Chairman
5. Co-ordinator establishes communication lines
6. Writing Group correspondence, copy to Chairman
7. Draft 2, Outside “expert” comments resolved, Working Group Draft 3
8. Draft 3 sent to Co-ordinator, who forwards to Secretary, copy to Chairman
FIG. 1. The ANSI N4 function chart.
<table>
<thead>
<tr>
<th>Standard designation</th>
<th>Title</th>
<th>N14 co-ordinator</th>
<th>Working Group chairman</th>
<th>Draft</th>
<th>Status/date ballot</th>
<th>BSR</th>
<th>Comments</th>
</tr>
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<tr>
<td>N14.1</td>
<td>Packaging of Uranium Hexafluoride for Transport</td>
<td>Arendt</td>
<td>Reynolds</td>
<td>X</td>
<td>3/85</td>
<td>2/86</td>
<td></td>
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<tr>
<td>N14.2</td>
<td>Tiedowns for Transport of Fissile Lee and Radioactive Containers</td>
<td>Towell</td>
<td></td>
<td>Under</td>
<td></td>
<td></td>
<td>Revised as result of negative ballot; expect reballot 2/86</td>
</tr>
<tr>
<td></td>
<td>Greater than 1 t in Truck Transport</td>
<td></td>
<td></td>
<td>revision</td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>N14.3</td>
<td>Packaging and Transportation of Radioactively Contaminated Biological Materials</td>
<td>Welch</td>
<td>Walker</td>
<td></td>
<td></td>
<td></td>
<td>N14 will ballot on scope currently being developed</td>
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<tr>
<td>N14.4</td>
<td>Quality Assurance in the Fabrication, Use and Maintenance of Shipping Containers for Radioactive Materials</td>
<td>Welch</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>N14 will vote on dropping N14.4, per ad hoc recommendation</td>
</tr>
<tr>
<td>N14.5</td>
<td>Leakage Tests on Packages for Shipment of Radioactive Materials</td>
<td>Arendt</td>
<td>Fischer</td>
<td>X</td>
<td>11/85</td>
<td></td>
<td>One negative vote to be resolved</td>
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<td>N14.6</td>
<td>Special Lifting Devices for Containers Weighing 10 000 lb (4500 kg) or More for Nuclear Materials</td>
<td>Lee</td>
<td>Townes</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>Awaiting ANSI approval</td>
</tr>
<tr>
<td>N14.7</td>
<td>Guide to the Design and Use of Shipping Packages for Type A Quantities of Radioactive Materials</td>
<td>Lee</td>
<td>Edling</td>
<td>X</td>
<td></td>
<td></td>
<td>Expected soon for N14 balloting</td>
</tr>
<tr>
<td>N14.8</td>
<td>Fabricating, Testing and Inspection of Shielded Shipping Casks for Irradiated Reactor Fuel Elements</td>
<td>Eggers</td>
<td>Dawson</td>
<td>Committee recommends suspension until Peer Review Panel recommends new standard; DOE will be sole user</td>
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<td></td>
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<tr>
<td>N14.92</td>
<td>Packaging of Nuclear Power Plant Tarnuzzer Radioactive Processed Wastes for Transportation</td>
<td>Mayo</td>
<td>X</td>
<td>X</td>
<td>Changes as result of negative ballot; will be reballed by N14</td>
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<tr>
<td>N14.10</td>
<td>Guide for Liability and Property Insurance in Shipping Nuclear Materials</td>
<td>Tarnuzzer</td>
<td>Quattrocchi</td>
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<td>N14.10.1</td>
<td>Administrative Guide for Packaging and Transporting Radioactive Materials</td>
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<td>Expect draft in July 1986</td>
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<td>N14.19</td>
<td>Ancillary Features of Irradiated Fuel Shipping Casks</td>
<td>Eggers</td>
<td>Goldman</td>
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<td>N14.20</td>
<td>Control of Contamination of Transport Vehicles</td>
<td>Lee</td>
<td>Jackson</td>
<td>Revised draft by March 1986</td>
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<td>N14.23</td>
<td>Design Basis for Resistance to Shock and Vibration of Radioactive Material Packages Greater than 1 t in Truck Transport</td>
<td>Eggers</td>
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<td>X</td>
<td>One negative ballot; review change and determine if reballot is needed</td>
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<td>N14.24</td>
<td>Barge Transport of Radioactive Materials</td>
<td>Eggers</td>
<td>Wilmot</td>
<td>X</td>
<td>X</td>
<td>Published</td>
<td></td>
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<tr>
<td>N14.25</td>
<td>Tiedowns for Rail Transport of Fissile and Radioactive Materials</td>
<td>Lee</td>
<td>Towell</td>
<td>Awaiting completion of N14.2</td>
<td></td>
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<tr>
<td>N14.26</td>
<td>Inspection and Preventive Maintenance of Packaging for Radioactive Materials</td>
<td>Arendt</td>
<td>—</td>
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<td>Title</td>
<td>N14 co-ordinator</td>
<td>Working Group chairman</td>
<td>Draft</td>
<td>Status/date ballot</td>
<td>BSR</td>
<td>Comments</td>
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<td>N14.27</td>
<td>Carrier and Shipper Responsibilities and Emergency Response Procedures for Highway Transportation Accidents Involving Truckload Quantities of Radioactive Materials</td>
<td>Lee</td>
<td>McCreery</td>
<td>X</td>
<td>X</td>
<td>2/86</td>
<td>Ready to be submitted</td>
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<td>N14.28</td>
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<td>Tarnuzer</td>
<td>Wawraszek</td>
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<td>New chairman still forming committee</td>
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<td>N14.29</td>
<td>Guide for Writing Operating Manuals for Radioactive Materials Packaging</td>
<td>Arendt</td>
<td>Waite</td>
<td>X</td>
<td>X</td>
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<td>Negative ballots being resolved</td>
</tr>
</tbody>
</table>
2. N14 COMMITTEE

The Institute of Nuclear Materials Management (INMM) serves as the secretariat for ANSI N14 (and N15). The membership of N14 consists of organizations (preferably national in scope), companies, government agencies and individuals that have a direct and material interest in the activities of the Committee. New members are evaluated according to their expertise, area of interest (producer, user, general interest) and the current balance of interest groups on the Committee. The officers of N14 have been liberal in accepting members for the Committee. Reflecting ANSI guidance, the Committee has looked for (1) organizations demonstrating substantial concern and competence within the scope of N14; (2) individuals having expert knowledge. A large membership has resulted. The N14 Committee currently consists of 50 members (27 representing organizations and 23 individuals). These members review and ballot proposed standards toward a consensus, where consensus is more than a simple majority but probably not unanimity.

3. N14 MANAGEMENT COMMITTEE

The N14 Committee has formed a ‘Management Committee’ to perform administrative functions, such as co-ordinating activities with the writing groups, soliciting new writing group chairpersons and organizing the annual meeting. The Management Committee is also available to evaluate and recommend solutions to problems encountered by the writing groups. For example, the Management Committee has been consulted in resolving negative comments on draft standards and has reviewed the scope of standards that may conflict with other standards committees. The Management Committee is purely advisory and does not intrude upon the consensus-making responsibilities of the N14 Committee. The Management Committee membership is as follows: J.W. Arendt (Professional Analysis, Inc.), P.E. Eggers (Eggers Ridihalgh Partners, Inc.), J.W. Lee (Transportation Consultant), R.T. Haelsig (Nuclear Packaging, Inc.), C. MacDonald (US Nuclear Regulatory Commission), M. Welch (International Energy Associates Limited), E.C. Tarnuzzer (Yankee Atomic Electric Company).

4. N14 WRITING GROUPS

Standards Writing Groups are standing units of N14 and are responsible for the preparation of the proposed standards. Specifically, a Writing Group is responsible for (1) the technical adequacy and quality of the document; (2) obtaining reviews; (3) demonstrating, in co-operation with the N14 officers, that a consensus has been achieved; (4) maintaining the standard after it has been approved. While each Writing Group is responsible for its own staffing and the conduct of its business, both
ANSI and N14 provide guidance and direction. Figure 1 shows the steps involved in developing a standard and the associated written communication and documentation. ANSI publishes formal guidance via its 'Style Manual' and a series of guides addressing practices and procedures. In addition, the ANSI Standing Nuclear Standards Board also works with the N14 Committee to provide (a) a numerical designation for the project of a specific Writing Group, (b) a title for the proposed standard and (c) a description in the form of a charter.

The status of N14 standards and standards development is given in Table I, ANSI N14 Status Report, 1 February 1986.
A STANDARD FOR BARGE TRANSPORT
OF TYPE B QUANTITIES
OF RADIOACTIVE MATERIALS

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United States of America

Abstract

A STANDARD FOR BARGE TRANSPORT OF TYPE B QUANTITIES OF RADIOACTIVE MATERIALS.

Although barge transportation plays an important role in the general commerce of the United States of America, only two radioactive material shipments have been completed. Despite the underutilization of the past, barge transportation is being considered by the Office of Civilian Radioactive Waste Management of the United States Department of Energy (DOE) as a viable option to augment land modes for the transport of commercial spent fuel and high level wastes. Barges appear to have sufficient potential as a competitive mode for several circumstances to warrant further investigation. The DOE will be relying on a standard for barge transportation of radioactive material which has been issued recently by the American National Standards Institute. The Standard is written particularly for large amounts of radioactivity that will be carried in Type B packagings. Its issuance prior to an immediate need will aid in fostering public confidence that radioactive material shipments by barge can be accomplished in a safe and efficient manner.

INTRODUCTION

In 1825, the Erie Canal was officially opened providing barge service between the Great Lakes and New York City. This event ushered in an era in which barge transportation in the United States has grown from a few simple horse-drawn canal vessels to modern ocean-going barges that carry complete chemical or refining plants pulled by 9000 horsepower tugs. The barges and towboats which ply the inland waterways, Great Lakes and intracoastal and domestic ocean routes are a vital part of the U.S. transportation network, ranking second behind the railroads in the carriage of a broad spectrum of commodities on a total tonnage basis.
There is, however, an interesting commodity distinction between barge transport and the others: with two exceptions, radioactive materials (RAM) have not been transported in the U.S. by barge. This paper first briefly discusses the potential role of barge transportation in the nuclear field and then addresses a major step in the utilization of this mode.

BARGE TRANSPORT NEED AND APPLICATION

A balanced transportation network is one that is flexible enough to permit the shipper to choose among alternatives based on their service, safety and economics. Nuclear shippers in the U.S., particularly those of irradiated fuel, have encountered impediments to the routine movement of these materials by land-based modes. A great deal of effort is currently going into the resolution of these issues which negatively affect the normal land movement of RAM. Though there is optimism over achieving acceptable solutions, the introduction of barges as an alternative transport mode will expand the options of RAM shippers.

The positive attributes of rail and highway transportation of RAM are well known, but those of a barge need some highlighting. Barge transport has several distinct attributes, the most predominant of which is an extremely large carrying capacity. A large deck-cargo barge of dimensions 30.5 meters (100 feet) wide by 122 meters (400 feet) long by 7.6 meters (25 feet) deep can carry approximately 16 300 metric tons (18 000 short tons), which is far greater than the carrying capacity of a number of railroad cars or highway trailers that would occupy that area of barge deck. Another advantageous characteristic is that this mode is essentially unimpeded during transit. Both trucks and trains move at speeds that are much greater than that of barges. However, these land modes are often disrupted by such things as switchyard operations, terminals, traffic and state and local regulations. Barges move steadily along at about 8 to 11 kph (5 to 7 mph), day and night. Barge disruptions are few: waiting at locks on inland waterways or slow running in congested areas. On some routes or in some regions this unimpeded service could be to the shipper's advantage.

Currently only a few agencies, primarily the U.S. Coast Guard, have jurisdiction over water traffic, and the rules and regulations are consistent on a nationwide basis. Thus, requirements that are imposed by state or local jurisdictions
such as routing, overweight shipment restrictions, frost laws, wheel loadings and bridge laws and that are notably non-uniform along routes have no counterpart on the water.

Objectivity dictates mentioning some of the disadvantages of barge shipping. Perhaps the most obvious shortcoming is that the number of locations potentially served by barge is quite limited when compared to those with highway access. Another is that barges must be used in conjunction with some other land-based mode. The land segments may be on-site or in the public domain; they may be short or could involve long distances. In any event, all barge shipments are intermodal involving two carrier transfers per one-way shipment. Under the assumption that RAM shipments by barge would be handled by specialized carriers, the cost of barge shipments is high compared to the alternative modes. This is largely due to the number of transfer steps and the low current demand for such service. If barge transport were employed to a reasonable extent in the future, the cost of such service may decline.

How can barge shipping be applied to the nuclear business? The prospects of moving large spent fuel casks bring into focus nuclear reactor sites even though interim storage facilities and final repositories should also be included. Further, irradiated fuel is only one RAM form which is suitable for barge service. There are a total of 77 plant sites, 50 of which have one or more operating units. Of the 77 sites, 55 (71%) have access to navigable waters either on site or a short distance away, the farthest removed being 10 miles from a water course. This is comparable to the 59 sites (77%) with direct rail access. Also, of the 18 sites without direct rail access, 16 have direct access to a navigable waterway. Of course, all sites have highway access.

Could the barge mode be a direct substitute for the land-based modes? The answer to this is a qualified no. As discussed earlier, barges require land-mode service to make them useful. Next, there are a large number of RAM commodities that are not shipped in the quantities or package sizes that would justify the application of this mode. Further, there are regional circumstances where impediments to land transport are not sufficiently restrictive to impact on transit times. And lastly, there are proposed receiving sites that are far removed from a navigable waterway.

Barge service would appear to be applicable to the following commodities:
Low level waste in shielded transport casks,
- Spent fuel or high level waste in transport casks,
- Spent fuel in dry storage containments,
- Contaminated equipment in transport packages,
- Decommissioning wastes, hardware and components with or without transport packaging.

The last category could be an extremely productive application of barge transport. The two barge shipments of RAM in the U.S. were (1) a failed steam generator which was shipped from the East Coast to Hanford, Washington via the Panama Canal and (2) a reactor compartment from a dismantled nuclear submarine which was also shipped to Hanford. These successful shipments demonstrated the ability to move very large, contaminated objects by waterborne means.

This brief introductory discussion has not addressed the details of completing a shipping campaign using barge transport. This is where the American National Standard N14.24 is significant.

ANSI N14.24 STANDARD

The notion of domestic barge shipping of spent fuel was first raised in the early 1970's by the then developing U.S. irradiated fuel reprocessing business. Several of the proposed or under-construction plants were located close to navigable waterways. In 1974, the American National Standards Institute (ANSI) under the N14 Committee formed the N552 Subcommittee on Water Transportation of Irradiated Fuel. The charter of this group was to write a Standard or Guide for the shipper of this material to permit an easy integration of the nuclear, regulatory and maritime requirements; it was to be a 'how-to-do-it' manual on spent fuel shipping by waterborne means. A draft document was produced for N14 approval, however, the effort was suspended until 1982. At that time, the Subcommittee was activated, reconstituted and renumbered N14.24.

The N14.24 Subcommittee changed the focus of the effort somewhat from that of the old N552 group. The commodity carried was broadened from just spent fuel to highway route controlled quantities of RAM. The transport vessels considered were narrowed from barges and self-powered craft to only barges. After several rounds of minor adjustments, including a title change, the N14 Committee accepted the
document [1]. It was approved for publication on July 23, 1985 and is now available.

Scope

The Scope of N14.24 reads

This standard identifies the organizations, equipment, operations, and documentation that are involved in domestic (i.e. between U.S. ports) barge shipments of highway route controlled quantities of radioactive material (RAM) on inland waterways and in coastwise and ocean service. This standard is expected to be most useful to shippers of radioactive material and is written in such a format that it provides them with sufficient information to prepare for, initiate, and complete one or more shipments. Included are requirements pertaining to: selection of barge, towing vessel, and packaging; preparation of certificates and documents; radiological and nonradiological operations; emergency planning; insurance; keeping records; and physical protection of the package.

As can be seen, the N14.24 Standard covers a large number of important topics.

Interface Matrices

Cask/barge operations are divided into four phases: planning, pre-operational, operational and postoperational. Matrices are presented which 1) identify the organizations that participate in each phase, 2) define the role of each organization and 3) assign to each organization a measure of the degree of its responsibility. Organizations include shipper, carriers, regulators and supporting services.

Loading/Unloading

In recognition of the need to perform an intermodal transfer twice in each one-way trip, the Standard discusses requirements for both roll-on/roll-off and lift-on/lift-off operations. Subjects include mooring, tug attendance, inspection of tiedowns and lifting gear, weather restrictions, transfer bridges, rolling equipment specifications and inspections and physical protection. Transfer safety is stressed.
Critical Requirements

Critical requirements are established for the barge, the tow, the package and the tie-downs. Barges for service under this Standard must be designed and constructed to recognized maritime classification society rules, codes and standards. The barges must meet intact stability as well as damage stability requirements which state that, following damage to any one compartment of the barge, the vessel must still meet specified margins of safety and not sink or capsize. The minimum acceptable barge length is 38 meters (125 feet). Minimum set-backs for collision protection are defined for locating the radioactive material package(s) on board the barge. These set-backs are stated in terms of the beam and length of the barge and apply to both inboard and fore/aft stowage locations. Barge construction and materials are also addressed. An adequately equipped barge will be required to carry certain ancillary equipment primarily for accident mitigation measures: emergency position indicating radio beacon, radar reflector, emergency towing wire, air identification marker and a sonic signaling device are specified in the Standard.

Minimum requirements are also established for the towing vessel. The vessel must be at least twin screw and have at least two propulsion engines, each capable of powering the tow. If in ocean or coastwise service, the tow must have a current Load Line Certificate. Also, for this type of service, the following navigational aids must be provided: radar, LORAN C or SATNAV, magnetic compass and gyrocompass. Towing vessels shall have appropriate and redundant communications equipment.

The Standard recognizes that current U.S. packaging testing conditions do not require submergence to depths greater than 15 meters. The Standard recommends that the route to be traversed by the barge be evaluated for the depth of water to be encountered. The package should then be evaluated to the maximum depth expected during transit. Such a recommendation could potentially exceed U.S. requirements.

A major development within the Standard is the specification of accelerations to which tie-downs or sea fastenings must be designed. The tie-down conditions are subdivided into two categories of dynamic loadings: (1) collision and (2) wave motion. The wave motion accelerations specified are applicable only in the case of ocean service and are not superimposed onto collision accelerations. Collision accelerations are based on existing U.S. Coast Guard requirements for independent tank barges. Two load
cases are considered for longitudinal and transverse directions. The tie-downs for the package and any accompanying vehicle shall be designed not to exceed limits specified in the Standard when subjected to 1.5g acceleration along the longitudinal axis or to 0.5g along the transverse axis. All accelerations are through the center-of-gravity (CG) of the package or package/vehicle combination.

Wave motion accelerations were derived from a detailed sea-state analysis. Two dominant load cases are established for extreme accelerations: 1) head seas and 2) beam seas. Tie-downs must be evaluated for accelerations acting in a vertical longitudinal plane through the package or package/vehicle CG with the following magnitude in head seas:

- **Longitudinal acceleration**: +1.4g
- **Vertical Acceleration**:
  - Dynamic portion = +2g
  - Static portion = 1g (gravity)
  - Yaw acceleration = 0.1 radians/sec/sec

For beam seas, the analogous requirements are established as:

- **Transverse acceleration**: +1.6g
- **Vertical acceleration**:
  - Dynamic portion = +2g
  - Static portion = 1g (gravity)
  - Yaw acceleration = 0.1 radians/sec/sec

Detailed tie-down structure stress limits and criteria are given in the Standard.

Other Considerations

The Standard addresses the importance of proper documentation with emphasis on the shipping plan. The shipping plan details the operation in terms of origin and destination, equipment, departure and arrival times, route and alternatives, water depths, weather limitations, communication frequencies, contacts and special instructions on the cargo. This plan is supplemental to the usual shipping papers.

Emergency response and emergency action recommendations are also in the Standard, as is a discussion of insurance
requirements, both nuclear and non-nuclear. Some of the text is devoted to discussing personnel requirements for manning the vessel, as well as providing radiological monitoring and physical protection, where required.

A last important consideration addressed by the Standard is postoperations activities. These activities include the retention of the records of the shipment(s) as well as any required restoration of the vessel. The records are those of the actual material transfer operation plus those of radiation exposure and physical protection. Vessel restoration involves radiation surveys and, as required, decontamination. Other restoration actions could include tie-down or rail hardware removal from the deck or repair of minor damage.

CONCLUSION

Although virtually untried for the shipment of RAM, the credentials of and capabilities of barge transportation for certain radioactive materials are too good not to be considered. The existence of the ANSI N14.24 Standard should give prospective nuclear shippers much more confidence in their ability to successfully execute a shipping campaign. Although there are no near-term plans for a major shipping campaign where barges might be used, this mode is being integrated into the long range planning of the U.S. Department of Energy for spent fuel disposal under the Nuclear Waste Policy Act. Having a Standard in place in advance of the need will do much to pave the way for the inclusion of barge transport in the overall transportation scheme. The Standard will also aid in the application of this mode to future decommissioning activities. Water transportation deserves a place in the nuclear transportation arena.

REFERENCE

CONTROL OF THE TRANSPORTATION OF RADIOACTIVE MATERIALS IN THE GERMAN DEMOCRATIC REPUBLIC
A view of the competent authority

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Abstract

CONTROL OF THE TRANSPORTATION OF RADIOACTIVE MATERIALS IN THE GERMAN DEMOCRATIC REPUBLIC: A VIEW OF THE COMPETENT AUTHORITY.

The paper presents an overview of the regulations, practices and experiences, as well as the systems of surveillance, in the transportation of radioactive materials in the German Democratic Republic (GDR), including considerations on the relationship of this issue to overall aspects of the transport of dangerous goods. The national transport regulations in the GDR are discussed, as are their development and relation to the IAEA Regulations and to international agreements, the administrative system for approvals and surveillance in the GDR and the transport experience in this field.

1. INTRODUCTION

In the German Democratic Republic (GDR), the National Board for Atomic Safety and Radiation Protection (Staatliches Amt für Atomsicherheit und Strahlenschutz (SAAS)), acts as the competent governmental authority responsible for protecting against the risks from the use of atomic energy [1, 2]. This function includes regulation, licensing and surveillance, and comprises radiation protection, nuclear safety, environmental protection, nuclear material safeguards and physical protection as applied to nuclear facilities and radiation devices, as well as to the handling of radioactive materials. It thus also includes the transport of radioactive materials and nuclear fuels. The transport of these materials is, however, integrated into the broader field of the transport of dangerous goods and is handled in co-operation with the Ministry of Public Transport and other national authorities [3].

In acting as the competent authority for the transport of radioactive materials, the major tasks of the SAAS can be stated to be:

1. Establishing and revising safety regulations;
2. Issuing approvals for the design of packages and special form radioactive material and for shipments subject to authorization;
(3) Controlling compliance with regulations for transport and packaging;
(4) Advising users on the implementation of regulations, especially in the
manufacture of packagings subject to authorization.

2. REGULATIONS

In the GDR, the legal basis for the transport of radioactive materials was
established in the mid-1960s, after international experience had first been gained
with the IAEA Regulations for the Safe Transport of Radioactive Material (Safety
Series No. 6), and their incorporation into international agreements for the carriage
of dangerous goods. Because of the novel and specific character of the subject,
and also because of the relatively large number of these regulations, it appeared
appropriate at that time to issue national regulations, rather than to refer to inter­
national documents. Table I gives an overview of the different national regulations
now in force in this field and their links to international recommendations and
regulations.

The first, Order on the Transport of Radioactive Materials (ATRS), was pre­
pared by the SAAS and published in the GDR in 1967. ATRS also contained
provisions for package testing and was in compliance with IAEA Safety Series No. 6,
1961 Edition. So far, ATRS has been revised twice, in 1971 and 1978, taking
into account national and international developments and experience, as well as
experience from the transport of other dangerous goods. The 1978 Edition, still
in force at present [4], is based on the 1973 Revised Edition of Safety Series No. 6
[5]. It is planned to incorporate the regulations of the 1985 Edition of Safety
Series No. 6 into the national regulations and to have them take effect in 1990.
It should, however, be mentioned that the question of whether it is better to
issue comprehensive national regulations or to stipulate mandatory reference to
the IAEA Regulations has not yet been resolved.

In parallel to the ATRS, there are a number of other national orders in the
GDR regulating the carriage of dangerous goods by the various different modes
of transport (road, rail, air, sea and mail):

Order on the Transport of Dangerous Goods by Rail, Road and Inland Waterways
(TOG);
Order on the Air Transport of Dangerous Goods (OLTG);
Order on the Sea Transport and Handling of Dangerous Goods in Port (OSHG);
Order on Postal Service.

Within these regulations, specific requirements concerning the different modes
of transport, as well as some related additional requirements on the packaging, are
contained in the rules for class 7 substances (Radioactive Materials). ATRS (in the
cases of transport by road or rail) and international regulations developed by such
organizations as the International Air Transport Association or the International
TABLE I. RELATION BETWEEN INTERNATIONAL AND NATIONAL REGULATIONS FOR THE TRANSPORT OF RADIOACTIVE MATERIALS

<table>
<thead>
<tr>
<th>Mode of Transport</th>
<th>International Regulations</th>
<th>National Regulations and Orders</th>
</tr>
</thead>
<tbody>
<tr>
<td>Rail</td>
<td>RID and SMGS</td>
<td>T06</td>
</tr>
<tr>
<td>Road</td>
<td>ADR</td>
<td>DLTG</td>
</tr>
<tr>
<td>Air</td>
<td>IATA DG Regulations</td>
<td>OSHG</td>
</tr>
<tr>
<td>Sea</td>
<td>IMDG Code</td>
<td>ATRS</td>
</tr>
</tbody>
</table>

Maritime Organization cover the basic requirements for all modes of transport (e.g. regulations for test and approval procedures).

The various national orders mentioned above are prepared by the Commission of the Transport of Dangerous Goods, set up by the Ministry of Public Transport. Representatives from the SAAS participate in this commission and are responsible for drawing up and revising the class 7 provisions. In addition, contributions have been made to the revision of various international regulations for the transport of dangerous goods, such as the Regulations Concerning the Carriage of Dangerous Goods by Rail (RID), the European Agreement Concerning the International Carriage of Dangerous Goods by Road (ADR), the regulations of the Council for Mutual Economic Assistance (CMEA) and the CMEA Agreement on the International Transport of Goods by Rail (SMGS). Within the framework of the CMEA, regulations for the transport by rail of spent nuclear fuel from nuclear power plants of the member states have been elaborated [6] taking into account the IAEA Regulations, while for sea transport the rules are still under preparation.

3. STATE CONTROL OF RADIOACTIVE MATERIAL TRANSPORT

The licensing activities of the SAAS in the field of radioactive material transport extend to examining applications for approval and issuing package design approvals for Type B packages and packages under nuclear safety classes I–III, as
well as design approvals for special form materials and approvals for shipments subject to authorization (including approvals for exceptional cases where the regulations for the packaging cannot fully be observed).

The first package design approval for a Type B container carrying high level $^{60}$Co sources was issued by the SAAS in 1971. This year the twentieth will be issued. Up to now ten classifications into one of the three nuclear safety classes have been made.

To permit a detailed assessment of packagings designed and constructed by the user, the SAAS has entrusted the Fuel Institute, in Freiberg, with the construction of a Drop Testing Facility [7]. There transport packagings for radioactive materials, or their models, weighing up to 5 t can be exposed to a 9 m or 1 m drop test. This facility is described in IAEA-TECDOC-295 [8]. Besides experimental test methods for mechanical and thermal stresses, relevant computer programs have been developed. Furthermore, since 1972, the Fuel Institute has published, and steadily updated, a catalogue which in schedule form gives guidance on transport packagings for radioactive materials [9] and compiles requirements and methods for the design and assessment of packagings. If Type A packagings for radioactive material transports are not tested by the user, such tests can be carried out in the GDR by the Board for Standardization, Metrology and Quality Control.

Control of compliance with the regulations for radioactive material transport is ensured by the clearance of packagings subject to authorization, by inspections of selected shipments and, in general, of packages approved by the SAAS, as well as by random checks of users, manufacturers and suppliers of radioactive materials (in connection with inspections related to the licence given to these institutions to handle radioactive materials) and at junctions with increased trans-shipments of packages containing radioactive material.

4. TRANSPORT PRACTICE

In the GDR, radioactive material transports occur as a result of activity in the following sectors:

(1) Radioactive materials originating from the manufacture, distribution and application of radionuclides and radiation sources. This includes a variety of radionuclides of different activity levels, mostly requiring the use of Type A containers. At present there are about 50 000 shipments per year, a large part of which is carried by road in combination with air and rail transport. This is mostly undertaken by a special company (Isocommerz GmbH) established for centralizing the trade with radioactive materials in the GDR.

(2) Fissile materials to provide nuclear power plants and research reactors with fresh fuel from the Soviet Union.

(3) Redelivery of spent fuel from power plants in the GDR to the Soviet Union (transport of high level and fissile materials).
(4) Radioactive wastes from the waste producer to the central repository of the GDR.

Between 1972 and 1978, a container transport system was developed for wastes from nuclear power plants and radionuclide production and use. It was examined by the SAAS and was incorporated into the licensing procedure for central collection and disposal of radioactive wastes in the GDR. In accordance with the provisions for materials of low specific activity (LSA), low and medium level wastes are packed in 200 L drums and in other transport containers for liquid and solid wastes designed and constructed in the GDR. They are carried as full loads in large containers in a combined rail and road transport system from the waste supplier to the central repository for radioactive wastes of the GDR [10].

Spent fuel from the Rheinsberg power plant was transported to the USSR in a heavy container weighing about 80 t and mounted on a railway wagon. This device was developed especially for this purpose and was made complete by the addition of two escort carriages to form a special train. The transport container — which was designed and constructed before the first set of IAEA Regulations for the transport of radioactive material were published — was later reviewed and approved in accordance with the procedures given in the IAEA Regulations. It was approved as a special arrangement and has been repeatedly employed.

For the reshipment of spent fuels from nuclear power plants with WWER-440 reactors, the USSR makes available a similar, but further developed, special transport train [11]. The container and the special train as a whole, as well as the shipments, were subject to multilateral approval (see also Ref. [6]).

5. RADIATION PROTECTION AND THE REQUIREMENTS OF THE ATRS

When the limits for surface contamination fixed in the radiation protection legislation of the GDR for objects which are allowed to be removed from radiation protection areas for further use (for $\beta$-emitters: 5 kBq/m$^2$, for $\alpha$-emitters: 0.5 kBq/m$^2$) are compared with the hitherto permissible limits for surface contamination of packages (including excepted packages), means of transport and cargo rooms, it can be seen that the values of the ATRS are about ten times higher. In the framework of the rail transport regulations of CMEA countries, this discrepancy was resolved for the transport of mixed cargo as early as 1974, when the limits for surface contamination for $\beta$-emitters were set at 4 kBq/m$^2$ and for $\alpha$-emitters at 0.4 kBq/m$^2$. The new IAEA Regulations now require the low limits only for “excepted packages”.

As regards the permissible external dose rate on the surface of packages and means of transport, it appears from the practice in the GDR that the limit of 2 mSv/h at the surface can so far be considered not to be too high. An analysis of the radiation exposure of transport workers showed that during the last ten
years the group of workers most exposed to external radiation (drivers at Iso-
commerz GmbH) received doses of up to 8.5 mSv/year (on an average 3.5 mSv/
year). A rough assessment of the radiation exposure of the general public along
the transport routes yielded a value of \( \leq 10 \, \mu \text{Sv/year} \), i.e. a very low one.

On the other hand, taking into account the radiological situation when
handling packages very frequently, or when using large transport containers
repeatedly within a facility, it became necessary to provide for additional shielding
and instructions to limit the radiation exposure of personnel. In these cases, the
designer or the user cannot automatically base the design or the operational
instructions on the permissible limits of the ATRS.

As regards unusual events that happened during the transport of radioactive
materials (e.g. transport accidents, damage to packages, loss or recovery of radio-
active materials) over the last 20 years, the analysis showed, on average, three
events/year. In no case, however, did impermissible exposures of persons or
releases into the environment occur. Seventy per cent of incidents were due to
man-made errors and only approximately 30% were due to technical failures.

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TYPE A PACKAGING COMPLIANCE

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Abstract

TYPE A PACKAGING COMPLIANCE.

Recent changes in US transport regulations involved updating requirements relative to Type A packages in order to be consistent with the 1973 Edition of IAEA Safety Series No. 6. As a result, a study was carried out to assess the compliance of existing packages with the new requirements. General areas of consideration dealt with in the study included (1) the basic concept of a 'performance package', (2) the 'authorized contents' of packagings and the effects that different types of contents can have on the ability of a package to meet the required performance tests, (3) the quality control/quality assurance requirements necessary to assure compliance, particularly with respect to reuse. Specific questions dealt with included the following. (a) How does one simulate, as closely as practicable, the expected normal radioactive contents? (b) What is 'loss of contents'? (c) How many packagings of any one type need to be tested? (d) How does one evaluate the prevention of any significant increase in the radiation levels recorded at the external surfaces? (e) How does one evaluate the Type A test for a filtered packaging?

1. INTRODUCTION

On June 30, 1983, a change in the regulations of the United States Department of Transportation (DOT) for the transport of radioactive materials became effective. These regulatory changes were part of an overall effort to bring US regulations into closer harmony with those of the International Atomic Energy Agency (IAEA) [1]. One significant change was the US's adoption of a major portion of the IAEA's Type A packaging requirements. A two-year transition period was provided for US shippers to bring their Type A packaging into compliance with the new requirements. Thus, on 1 July 1985, the new Type A packaging requirements became effective.

In 1976, a Type A Packaging Certification Document was published as a result of a United States Department of Energy (DOE) funded study. In this 1976 study, conducted by MRC-Mound, packaging hardware was tested against the (then existing) Type A packaging requirements. The hardware that met the requirements was then certified as Type A packaging. The purpose of this 1976 document was to eliminate redundant testing of the same packagings by numerous shippers.

In preparation for the 1 July 1985 transition, a follow-up study was funded by the DOE's Security Evaluations, DP-4. MRC-Mound again conducted the testing and analysis of the packaging hardware. Not only had there been a change in the regulations, but, as one would expect, there were also changes in the approach to regulatory compliance with respect to Type A packagings. One major source of information was the experience base gained in the 10 years of US shippers utilizing the first (1976) Type A Packaging Certification Document. This experience showed that several areas needed to be strengthened in order to achieve effective regulatory compliance, specifically the shippers' understanding of:

1. The basic concept of a 'performance package'.
2. The 'authorized contents' of packagings and the effects that different types of contents can have on the ability of a package to meet the required performance tests.
3. The quality control/quality assurance requirements necessary to assure compliance, particularly with respect to reuse.

2. SPECIFIC QUESTIONS

In addition to these, and other general areas of consideration, several specific questions were identified as requiring answers before even starting the certification programme. These are:

1. How does one simulate as closely as practicable, the expected normal radioactive contents?
2. What is 'loss of contents'?
3. How many packagings of any one type need to be tested?
4. How does one evaluate the prevention of any significant increase in the radiation levels recorded at the external surfaces?
5. How does one evaluate the Type A test results for a filtered packaging?
6. Exactly how should paragraphs 201–208 (49CFR173.411, Ref. [2]) (General Design Requirements) and paragraphs 210–227 (49CFR173.412) (Additional Design Requirements) of the IAEA Regulations [1] be applied with respect to Type A packaging evaluation?

The response to those questions provided a basis for the beginning of the testing/analysis and documentation portion of the programme.
TABLE I. MATERIAL FORM CATEGORIES FOR SIMULATING RADIOACTIVE CONTENTS

<table>
<thead>
<tr>
<th>Form</th>
<th>Contents used in testing</th>
<th>Typical contents authorized in certified Type A packagings</th>
</tr>
</thead>
<tbody>
<tr>
<td>No. 1</td>
<td>Flour and fluorescein</td>
<td>Solids: all materials including granular and powdered forms. Thus, a packaging certified for Form No. 1 materials is expected to contain radioactive contents of any solid form.</td>
</tr>
<tr>
<td>No. 2</td>
<td>Sand (normal cement grade)</td>
<td>Solids: materials of a large particle size, i.e. sand, soils and concrete construction debris</td>
</tr>
<tr>
<td>No. 3</td>
<td>Lead ingots and/or sheets</td>
<td>Objects/materials with no significant dispersible or removable contamination</td>
</tr>
</tbody>
</table>

2.1. Question No. 1. Simulating the expected normal radioactive contents

In the 1976 study, sand was the typical material used to represent the contents and the pass/fail criterion then was whether there was a release of sand from the packaging. For 1985 test purposes, this approach was deemed inadequate for simulating finely divided/small particle size radioactive materials. But what should be used? One thought was to use water and inspect visually; another was to use a leak-test gas and monitor for loss of contents using sophisticated gas detection equipment. Both of these were deemed too conservative and thus normal baking grade flour was finally chosen as a readily perceived ‘finely divided material’. Fluorescein, a (finely divided) material that responds to ultraviolet light was added to the flour to enhance detection. This then was the answer to simulating finely divided materials. But what if one were just shipping sand or solid bulk objects with no removable or dispersible contamination? Flour/fluorescein was deemed unnecessarily conservative in simulating these contents. In response to these questions and thoughts, three material form categories were established for testing, analysis and certification (Table I). The intent behind these three forms becomes clearer with the discussion of Question No. 2, below.

2.2. Question No. 2. What is meant by ‘loss of contents’?

The first premise was that an exact quantification of package contents released was not required. However, one must be able to reasonably substantiate that there has been no loss or dispersal of the contents. No loss or dispersal was defined as follows for different forms of materials:
(a) Form No. 1: No visible evidence of the loss of flour or of fluorescein under ultraviolet light (or methods of equal or greater effectiveness).

(b) Form No. 2: No visible evidence of the loss of sand (or equivalent test contents).

(c) Form No. 3: No relocation of the contents from inside to outside. Openings of seams and punctures of walls are not necessarily cause for failure, since there is no removable or dispersible contamination by definition.

2.3. Question No. 3. How many packagings of any one type need to be tested?

There was no easy answer to this question. The approach used was to consider the usage, availability of packaging and cost of an individual package and then establish the number of packagings to be tested. Then, after the test sequence on all specimens of a specific package design was completed (i.e. five tests were planned and completed, consisting of 1.3 m (4 ft) drop tests on a bolt of a DOT specification 17H 55 gal steel drum), the 'degree of pass' was assessed. The degree of pass was a non-quantitative assessment of how close the packaging came to failure. In the case of the 55 gal drums mentioned above, even if all had passed, it would have been necessary to assess factors indicating that for one reason or another the drums might have been close to failure. Should there be less leniency in assessing the degree of pass, then more testing would be appropriate. For packagings with high-use factors (i.e. DOT Specification 17H and 17C 55 gal drums), three to five drums were tested in each specific test: top (approximately a 45° angle), bottom (approximately a 45° angle) and side, etc.

For packages with low-use factors, and depending again on the materials and methods of construction and the actual test results, only one package was tested.

2.4. Question No. 4. Evaluating the prevention of any significant increase in the radiation levels recorded at the external surfaces

In this case, a 20% increase was deemed significant, according to the 1985 IAEA Regulations. Data based on the test results were provided to allow the user/shipper to conduct this evaluation for each package. For example, the deformation of the bottom of the 17H 55 gal steel drum following the 1.3 m drop test was measured for each package and was provided in the Certification Document.

2.5. Question No. 5. Evaluating Type A test results for a filtered packaging

Both filtered steel drums and filtered steel boxes were offered for evaluation as Specification 7A Type A packages. There was resistance by some shippers in the

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1 gal (US) = 3.785 x 10³ m³.
United States to accept a filtered Type A package. The question of pass/fail was basically the same — no loss of contents and no significant increase in the radiation level.

The filter utilized was a carbon composite filter (NucFil) threaded into a 17C steel drum lid and a variety of steel boxes. The filtered drums were dropped from a height of 1.3 m and impacted with a penetration bar dropped from 40 in, pressurized (up to 11.2 lbf/in²) and subjected to the water spray test. In all tests, flour and fluorescein were used as the contents and in all cases there was no release of contents.

2.6. Question No. 6. How should IAEA Regulations on General Design Requirements and Additional Design Requirements for Type A packages be applied with respect to Type A packaging evaluation?

Concurrent with this test programme, an American National Standards Institute (ANSI) standard was being prepared [3]. As part of the need to provide guidance on the application of the regulations to ANSI standards readers, and for use in the Type A test programme, written guidance was prepared on each paragraph of the major sections governing Type A package certification. Two examples of these analyses are given here.

3. GENERAL DESIGN REQUIREMENTS

3.1. 49CFR173.411 (General design requirements)

Except for a package that contains a limited quantity or excepted instrument or article under 49CFR173.421 through 173.424, each package used for shipment of radioactive materials shall be designed so that:

— Paragraph (a) "The package can be easily handled and properly secured in or on a conveyance during transport".

Guidance for application. This does not necessarily mean that handles are required. The intent is to maintain a package size which permits conformity with paragraph (a) (a package 4 ft × 6 ft × 4 ft cannot be easily handled by hand). For such a package, other handling methods would be desirable, as is the case also with packages with sharp edges or components, which are not easily handled manually. Package stability is also required during transport. A round-shaped package, a top-heavy package, or any similar package which

\[1 \text{ inch} = 25.4 \text{ mm.}\]
\[1 \text{ lbf/in}^2 (g) = 6.895 \times 10^3 \text{ Pa.}\]
\[1 \text{ ft} = 0.3048 \text{ m.}\]
would not maintain its initial orientation and load position under conditions incidental to routine transportation would not meet these requirements. Tie-downs are required for such packagings and exclusive-use shipments or documented special provisions with the carrier are also required.

Paragraph (e) "The external surfaces, as far as practical, may be easily decontaminated".

Guidance for application. Plywood, although not normally thought of as being easily decontaminated, is acceptable. The intent is to provide surfaces which are relatively smooth and which minimize cracks and crevices in which surface contamination could reside. Consideration should be given to such actions as sanding or painting the surfaces.

Owing to limitations on space, only these examples can be given, but a complete presentation is available in Ref. [4] and in Ref. [3].

3.2. Reduced pressure

Besides the Type A tests, the other significant test sequence conducted was for reduced pressure. These results are summarized below for steel drums and steel boxes.

(1) Steel Drums. The reduced pressure (49CFR173.412(i)) test was one of the more severe ones from the standpoint of steel drums failing a test. The drums were pressurized to 75 kPa (0.75 atm) and monitored for leakage. Many steel drums did not hold pressure and added measures were utilized to achieve a 'passing' condition for these drums. For those that failed, RTV (the brand name of a self-vulcanizing rubber) on the gasket was tried and proved successful as a modification which allowed the packaging to pass this test. Thus, in the Certification Document (MLM-3245), for most steel drums RTV is required. Where RTV is used, the need for a 4 mil plastic liner (as discussed in the free-drop section for steel drums) is eliminated. Tests of steel drums with RTV on the gaskets demonstrated that Form 1 contents (flour and fluorescein) were successfully retained in the 1.3 m drop test.

(2) Steel Boxes. The steel boxes previously utilized in the USA typically did not meet the reduced pressure test. Thus, manufacturers had the choice of structurally reinforcing the boxes or filtering the boxes to allow the equilibration of pressure while containing the particulates. As expected, different manufacturers chose different concepts and so a variety of sizes of steel boxes — filtered and non-filtered — successfully passed the reduced pressure test.

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4 1 mil = 0.0254 mm.
4. SHIPPERS' RESPONSIBILITIES

As mentioned at the beginning of this paper, one of the general problems encountered in the USA was a lack of understanding on the part of shippers of what a performance packaging is and what the shippers' responsibilities are concerning use of Specification 7A Type A packaging. To deal with this need, an explanation of responsibilities was developed to highlight certain requirements.

4.1. General

There are a number of specific functions which must be performed in order to ensure that a Specification 7A package is properly prepared for safe transport and is in full compliance with the applicable regulations. The affiliation or the specific arrangements between the designer, manufacturer and user should be adequately described and understood. In some cases, all of these responsibilities will be charged to a single organization, such as a shipper who designs and tests his own package. In other instances, the responsibilities may be spread among organizations, such as when the design is separate from the manufacture which is separate from the user. Where these responsibilities are separated, it is necessary to have effective quality assurance to ensure that each function is performed and that documentation and communication of this performance are complete. With regard to the preparation of a Specification 7A package, the specific functions can be separated into six categories:

1. Use
2. Reuse and maintenance
3. Design
4. Fabrication of the packaging
5. Testing, analysis and certification
6. Quality assurance and documentation.

4.1.1. Use of the package design

It is the shipper, the user of the package, who has the overall responsibility to ensure that the package offered for transportation is in full compliance with DOT regulations. While the shipper may not perform each and every one of the above functions, he does have the responsibility of ensuring that each function is properly carried out. This overall responsibility also includes ensuring that the communication of these functions leads to a fully evaluated and documented Specification 7A package design which is used properly.

There is a fundamental relationship between the contents and the packaging, which has a direct bearing on the ability of the package to pass the required tests.
This relationship can be illustrated, for example, by the mass and physical configuration of the contents. A particular packaging may be entirely adequate for lightweight, relatively 'soft', solid contents. The same packaging, however, may be incapable of passing the required drop test when filled with high-density 'hard' contents.

4.1.2. Reuse and maintenance

Specification 7A packages may be designed for either single or multitrip use. The shipper has the responsibility of ensuring that each time a 7A package is reused and offered for transportation, it meets the design specifications. This ensures that the package is just as capable of meeting the test requirements as the originally tested (or analysed) prototype of the design. Each time a shipper offers a particular package for transportation, he must certify that the package is in proper condition for transportation. This incorporates certifying that the reused packaging is as capable of meeting the Specification 7A requirements as an unused package.

4.1.3. Design of the package

The designer must specify in detail the actual hardware and assembly methods which will be used to make up the package. It is the responsibility of the designer to describe the package design in sufficient detail so that each packaging is constructed and assembled in accordance with the design and will perform in a consistent manner when subjected to the Type A tests. This specification of hardware must include all necessary dimensions, materials of construction and methods of fabrication. Assembly instructions must adequately convey the procedures necessary to load the contents and prepare the package for transport.

4.1.4. Fabrication of the packaging

The person who manufactures or fabricates the packaging for a Type A package is responsible for ensuring that the hardware produced complies with the appropriate material and design specifications.

4.1.5. Testing, analysis and certification of the design

The testing of a proposed Type A packaging involves subjecting the prototype with simulated radioactive contents to the prescribed test. The tester has responsibility for ensuring that the hardware tested complies with the design specifications and that the simulated contents are such that maximum stress is imposed on the item being tested. The test results must be carefully described and documented for each test.
4.1.6. Quality assurance and documentation

While the shipper has the overall responsibility for ensuring that an adequate quality assurance programme exists and is followed, each person contributing to the make-up of the package is responsible for the quality of his/her performance. This includes performance in accordance with the quality assurance programme and documentation of these actions. Only through strict compliance with such a programme can satisfactory results be expected.

A similar section on quality assurance was written for the Certification Document [4]. In addition, a specific section on quality assurance was prepared for each packaging identifying aspects which were critical and major with respect to package compliance and further identifying characteristics with which to measure compliance. This section could not be included because of space limitations, though information is included in Ref. [4].

5. 49CFR173.465 TYPE A PACKAGING TESTS

The packaging hardware tested for certification as 7A packaging can be categorized as:

1. Steel drums
2. Steel boxes
3. Wooden boxes
4. Fibreboard packages
5. UF₆ cylinders
6. Packagings for liquids and gases
7. Miscellaneous packagings.

5.1. Steel drums

5.1.1. Free-drop test

One surprising result of the 1.3 m (4 ft) free-drop test was that most steel drums failed with flour and fluorescein as contents when dropped (flat on their side and on the top at an approximately 45° angle) with the bolt at the point of impact. This change was obviously the result of using a fine powder (flour and fluorescein) to represent the contents, since testing with sand produced passing results based on no detectable loss of contents. Preliminary side-drop tests indicated loss of contents comparable to the angle drops on the bolt, so all subsequent test sequences included side drops. The side drop resulted in much less damage to the package than the angle drop on the bottom (largest dent) and the angle drop on the top.
This raises the question of what needs to be done to enable the drums to meet this requirement if they fail the drop test. A number of options were considered and two were chosen for testing. One was to place a bead of room-temperature vulcanizing material (RTV) on the gasket to enhance its sealing ability. The second was to line the drum (enclosing the test contents) with a 4 mil polyethylene bag. In 41 separate drop tests, both of these methods worked very well, whereas in 23 separate 1.3 m drop tests without RTV or a 4 mil liner, 20 resulted in failure (loss of contents). All metal drums certified for Form No. 1 materials require either RTV or a 4 mil polyethylene bag in order to pass the free-drop test.

As a continuation of this study, numerous tests have been, and are being, conducted with scrap/debris contained in the steel drums. These tests, although still in progress, demonstrate that in the 1.3 m drop test, items such as iron pipes, valves, steel plates, etc. will readily puncture the sides of 16 and 18 gauge steel drums at gross weights of less than 227 kg (500 lb). Even broken concrete blocks penetrated 16 gauge steel walls in the 1.3 m drop test at gross weights near 227 kg. This demonstrates the need to secure these items within a package or to add additional penetration barriers. At the Mound facility, metal, fibreboard, plastic and fibre liners for these drums and contents have been tested. Several individual items and some combinations have proven effective. These tests are not yet complete, but they highlight the need for shippers to evaluate the behaviour of the contents with respect to overall Specification 7A Type A package performance.

5.1.2. Water spray test

Twenty-six different steel drums (nine different styles/sizes) were given the water spray test and, as expected, there was no reduction in the ability of the package to meet the subsequent Type A tests and there was no in-leakage of water.

5.1.3. Penetration test

The only item of significance here was the addition of an impact targeted to be near the closure ring. In the past, it was always assumed that the centre, seemingly the most unsupported area of the top of a drum, was the most vulnerable area. For the smaller drums, it was found that the penetration test near the bolt closure ring tended to pull the sealing surfaces apart (body, lid and gasket), whereas the test in the centre just left a dent.

5.2. Wooden boxes

5.2.1. Water spray test

The water spray test was conducted on 14 different types of boxes (24 different tests). Eight of these tests were to demonstrate that there was no reduction in packag-
ing integrity that would affect the ability of the package to pass the 1.3 m free-drop test, something confirmed by the drop test results. The water spray test was also conducted to assess any potential in-leakage of water which could result in the leaching or dispersal of contents. Five standard boxes, of 3/4 in plywood, and nailed construction, with all seams glued and with internal bracing were subjected (while empty) to the water spray test. Two of the five passed, as indicated by the fact that there was no evidence of water on blotter paper attached to the inside of the box. The in-leakage of water was minimal (equivalent to a few drops in most cases) and seemed to originate from several sources — seams, nails, imperfections in the plywood, etc. RTV was then applied to the inside seams and the test was conducted again. This time all five failed. A water sealant (Geocel (brushable sealant — one coat)) was used to coat the same boxes that had failed the first test (but not with RTV added). All five boxes now passed. Further studies are in progress to gather data on the long-term effectiveness of this and other coatings.

No wooden boxes by themselves were certified for Form 1 materials, though some were certified for Forms 2 and 3 and some were certified only for Form 3 materials. Since there was an in-leakage of water in most cases, and thus a potential for transfer of radioactive materials from the package, restrictions were placed on the use of wooden boxes. These restrictions were that the contents had to be wrapped in plastic (or equivalent) to keep the water from coming in contact with the radioactive material or demonstrating that (principally for Form 3 materials) there would be no transfer of radioactive material to the water in the event of in-leakage and contact.

6. CONCLUSION

The use of flour/fluorescein provided a more sensitive leak detection method and resulted in modifications to standard packaging practices. Filtered packages were certified, meaning in part that filters were authorized as a method of meeting the reduced pressure test. Content evaluation is a major responsibility that a shipper has to understand and fulfil. Tests with a variety of contents are part of an ongoing test programme and will be dealt with in greater detail in the future. Finally, the lessons learned from the use of the first study in the USA were invaluable in structuring this most recent Type A test programme.

These test data have been shared with the Department of Energy and its contractors and MRC–Mound has received funding to co-ordinate all DOE testing, certification and use of Specification 7A Type A packagings. Through a comprehensive communications and education programme, MRC–Mound will advise DOE contractors concerning shipper regulatory compliance responsibilities. This focal-point concept will greatly enhance consistency in testing/analysis, preparation of certification documents and overall use of Specification 7A Type A packaging within the DOE and in the United States of America.
REFERENCES


CURRENT STATUS OF SAFETY REGULATIONS AND EMERGENCY RESPONSES FOR NUCLEAR FUEL TRANSPORTATION IN JAPAN

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Abstract

CURRENT STATUS OF SAFETY REGULATIONS AND EMERGENCY RESPONSES FOR NUCLEAR FUEL TRANSPORTATION IN JAPAN.

Japan has 48 nuclear power plants with total electric capacity of 40 694 MW, including those in operation, under construction and in preparation. The amount of nuclear fuel materials transported in connection with the operation of these power plants in 1984 was 730 ton U metal of UF₆, 889 ton U metal of UO₂ powder, 1094 ton U metal of fuel assembly and 396 ton U metal of spent fuel. The nuclear fuel transports are regulated at the package design, packaging approval (registration) and shipment confirmation phases by national competent authorities such as the Science and Technology Agency and/or Ministry of Transport using Japanese regulations based on the 1973 IAEA Transport Regulations. By the end of 1989, the 1985 IAEA Transport Regulations will be introduced into the regulations. Concerning emergency responses for nuclear fuel transport, the consignor concerned is obliged to make the necessary arrangements, with the competent authorities concerned also making arrangements. The authorities are currently developing guidelines for a manual on safe transport.

1. INTRODUCTION

Approximately thirty years have passed since the development (and subsequent use) of nuclear energy in Japan first began. In the past ten years, the electric capacity of these nuclear power plants has grown to six times that of the previous ten-year period.
TABLE I. NUCLEAR FUEL MATERIAL TRANSPORTS FOR NUCLEAR POWER PLANTS

<table>
<thead>
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<th></th>
<th></th>
<th></th>
<th></th>
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<td>Total Quantity (ton U)</td>
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<td>24</td>
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<td>315</td>
<td>27</td>
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</tbody>
</table>

1 Data which STA confirmed for the application based upon the Regulations
2 Excluding GCR fuel
FIG. 1. Transportation routes for nuclear fuel materials and locations of nuclear facilities in Japan.
FIG. 2. Latest jurisdiction scheme and the competent authorities.
Such an increase in nuclear power has increased dramatically the frequency and quantity of nuclear fuel transports. A plan for the construction of nuclear fuel cycle facilities, consisting of enrichment, reprocessing and radioactive waste storage facilities, is being developed. As a result, it is expected that a variety of transport modes will also develop.

One of the basic principles is to protect workers and the public from the exposure "as low as reasonably achievable" (ALARA) in order to assure safe transport of nuclear fuel.

Japan has regulated package design and shipment based on this principle of safety and, as a result, there has not been any accident reported during nuclear fuel transport. However, regulations have been formulated for the contents a consignor can transport in an emergency and meetings have been arranged for other types of emergency responses, such as sending experts and equipment for rendering assistance. Currently, the Government of Japan is establishing guidelines for a manual on safe transport which will be available to consignors so that they can also develop a system of emergency response.

2. CURRENT STATUS OF SAFETY REGULATIONS FOR NUCLEAR FUEL TRANSPORT

2.1. Nuclear power generation and nuclear fuel transport in Japan

As of October 1985, there were 31 nuclear power plants in operation (14 PWR, 16 BWR, 1 GCR, with a total of 23 631 MW), while 11 plants were under construction (10 780 MW). In addition, there are six nuclear power plants (6275 MW) which are under consideration for construction, making a total of 48 power plants with a capacity of 40 694 MW.

The number of nuclear material transports in relation to the power plants during 1980–1984 are shown in Table I. The transport of nuclear fuel and materials among the various facilities is shown in Fig. 1. For example, if each nuclear fuel cycle facility at Shimokita in northern Japan, as indicated in the figure, is constructed and begins operation, the number of transports which start or finish there will greatly increase in quantity and frequency.

2.2. Current status of safety regulations

Regulations relating to nuclear fuel transport in Japan are governed by the Law for Regulation of Reactors, by Maritime Safety Law and by Aviation Law. The technical standards for their transport are stipulated in ministerial ordinances and notifications based on these laws. The contents of these technical standards are mainly based on the 1973 IAEA Transport Regulations. The jurisdiction scheme and authorizations related to nuclear fuel transport are shown in Fig. 2.
As far as practical operation of these regulations is concerned, safety confirmation based on the law is required when transporting either fissile or Type B(M) and B(U) packages. The safety confirmation procedure consists of three steps: approval of the design of the nuclear fuel package, approval (registration) of the packaging and confirmation of the safety of the shipment (confirmation of the package and confirmation of transport method; the latter is required only for class 3 fissile packages).

2.2.1. Approval of the package design

The competent authorities, the Science and Technology Agency (STA) and the Ministry of Transport (MOT), carry out the safety examinations, with the help of advisory committees (consisting of experts in such fields as structure, heat conduction, containment, shielding and criticality), related to safety analysis of the package. The term of validity of a certification for the package design is three years. The certification can be extended by following the appropriate procedures.

2.2.2. Packaging approval (registration)

The applicant must obtain the approval of the competent authorities, who examine the results of inspections carried out according to the inspection manual, defined in advance, which consists of an inspection method and the relevant criteria. The competent authorities carry out inspections as necessary during manufacture and at the time of the completion of the packaging. The term of validity of a certification for packaging approval (registration) is three years. This certification can also be extended.

2.2.3. Confirmation of shipment

(a) Confirmation of the safety of a package.

The competent authorities examine, prior to each shipment, whether the contents meet the approved design specification and are in the approved packaging. Before the shipment, inspections (as necessary) such as measurements of surface dose rate and sealing performance are carried out by inspectors from the competent authorities. Only then is the confirmation certificate issued for the package.

(b) Confirmation of the safety of a transportation method.

In addition to the procedure detailed in (a), the MOT carries out a safety inspection of the method of fastening the package on the conveyance being used (vehicle, vessel, aircraft, etc.), of the structure and equipment of the conveyance and the method of radiation exposure control. The confirmation certificate is then issued for the relevant transport method.
2.3. Incorporation of the 1985 IAEA Transport Regulations into Japanese regulations

The technical aspects of Japanese regulations related to the transport of radioactive materials are based on the 1973 IAEA Transport Regulations. Measures are now being taken to incorporate the 1985 IAEA Transport Regulations. This action will be realized first in terms of the standards for the transport of radioactive materials, to be laid down by the Nuclear Safety Commission of Japan based on the 1985 IAEA Regulations. Second, competent authorities will legislate these safety standards into law. The new regulations are to be enacted by the end of 1989.

3. EMERGENCY RESPONSES FOR NUCLEAR FUEL TRANSPORT

Safety confirmation of Type B and fissile packages, as well as of the method of transport, for all transport modes, is required by the competent authorities (STA and/or MOT), as is notification of the Prefectural Public Safety Commission in the case of land transport. In the case of sea transport, notification to the Maritime Safety Agency (MSA) is required (Fig. 3). In the interests of greater safety, escort cars are frequently assigned in front of and behind truck convoys when transporting nuclear fuel packages by road (Fig. 4).

In the case of transport of spent fuel by sea, an exclusive-use vessel which conforms to MOT regulations (e.g. with such features as anticollision and antistranding construction and double-hull construction) is used. In the event of an accident, the consignor taking part in the transport is obliged to notify the police, the maritime officer and/or the fire-station, as well as to take applicable safety measures. A detailed plan is usually prepared before shipment, including such particulars as the type of communication system, the emergency response planned in the event of an accident, and the level of education and training of the transport workers.

3.1. The Nuclear Safety Commission of Japan

The Nuclear Safety Commission is currently considering the establishment of guidelines for the preparation of safe transport manuals for every kind of nuclear fuel package. These guidelines can then be referred to by consignors to draw up their own manuals and to establish their own emergency responses.

In examining the question of emergency response, it should be noted that Type B and/or fissile packages which withstand accident conditions (i.e. suffer severe damage) during transport are required not to release their contents. In an emergency, however, other measures to prevent the release of contents are required. As for packages which may release their contents in an accident, i.e. Type A and/or non-fissile packages, it is necessary to take the quantity of release into account, with appropriate measures for an emergency being required. Consequently,
FIG. 3. Safety review procedure for transport of nuclear materials (excluding transport by vehicle of a Type B package with high radioactivity).
the above guidelines will include measures for Type A, Type B, non-fissile and fissile packages.

3.2. Consideration at the administrative level

The relevant government offices concerned (MOT, STA, MSA, the National Policy Agency, the Fire Defense Agency, etc.) have come to an agreement regarding the safety measures to be taken in the event of an accident during the transport of radioactive materials.

3.3. Survey and studies related to emergency response

Various kinds of studies are being carried out by the government offices concerned with the object of examining emergency responses. Examples of the studies are:

(1) A study on the behaviour of UF₆ packages.
   (a) Investigation of the behaviour of a packaging when it is submerged in the sea.
   (b) Investigation of the reaction between UF₆ and brine during submersion.
   (c) Investigation of the effects on the surrounding environment.
   (d) Emergency responses to be taken with regard to nearby residents. The purpose here was to provide information on the emergency response needed to ensure the safety of nearby residents if a UF₆ package, such as a 30B cylinder, were to be immersed in the sea. The following conclusions were obtained:
   (i) The 30B-type packaging exhibits elastic behaviour, even when submitted to hydrostatic pressures of 1.5 kgf/cm² gauge for 8 h, as prescribed by regulations, and 20 kgf/cm² gauge for 1 h.¹

¹ kgf/cm² = 9.807 × 10⁴ Pa.
Moreover, the soundness of the seals for all other parts, including valves and blank plugs, remains unchanged. At approximately 60 kgf/cm² gauge of hydrostatic pressure, buckling of the cylinder occurred, as predicted by calculation.

(ii) As a result of the reaction experiment, it was observed that white smoke (caused by reaction products) was created when brine got into the simulated cylinder. However, when the interior of the cylinder became flooded, the reaction speed became very slow. As a result, an explosive reaction did not occur.

(iii) As for the effects of uranium and hydrofluoric acid on the human body in the event of submersion of a UF₆ package, they are presumed to be negligible even in the worst-case scenario.

(iv) The following would be reflected in an emergency response:
— Countermeasures taken immediately after the accident.
— Countermeasures taken until the container is retrieved.
— Retrieval of the container.
— Countermeasures taken if the container cannot be retrieved.

(2) Study on the behaviour of nuclear fuel packages when the truck is crashed into on the side.

In crash studies, it was estimated that the accident would result in a fall and fire. It is thus important that the behaviour of the package be accurately predicted.

The simulation considered three crash possibilities, front, rear and side. It was concluded that the effects of front and rear impacts were already well understood. However, there were few data for the side crash. Since it is important to understand the effects on packages when a truck loaded with nuclear fuel is hit on the side by another large vehicle, such as a truck, further studies are now being carried out:

— 1984 (Japanese fiscal year: JFY).
  Fundamental study of phenomena in a side crash.
— 1985 JFY. Experiment to estimate behaviour of the ‘crashing’ truck, the crashed truck and the loaded goods.
— 1986 JFY. Experiment to confirm that the package will keep its integrity.

4. CONCLUSION

In spite of an increase in the transport of nuclear fuels in Japan, no accidents have been reported. The relevant competent authority regulates nuclear fuel materials transport according to their type. Currently, the technical standards for regulating transport are based on the 1973 IAEA Transport Regulations. The incorporation of the 1985 Transport Regulations is planned. On the question of emergency response,
though the arrangement to be used is already established, guidelines for the prepara-
tion of safe transport manuals are being developed and other studies on emergency
response, such as studies on the behaviour of UF$_6$ and nuclear fuel packages in a
side crash of a truck, are being carried out.
MIXING OF PACKAGE DESIGNS: NUCLEAR CRITICALITY SAFETY*

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Abstract

MIXING OF PACKAGE DESIGNS: NUCLEAR CRITICALITY SAFETY.

International transport regulations incorporate rules for mixing package designs during transport, including transit storage. Permissible combinations of packages are based on transport indexes. In the paper, two main types of flaws in the rules for mixing package designs have been defined and investigated. (1) Reduced leakage of neutrons from a group of packages. A package which has a high reactivity value in the centre of an array may have a low value on the outside. For another type of package, the opposite may be true. A special case is a non-fissile package. (2) Increased neutron coupling between packages. Sometimes safety depends on the extent of neutron absorption in the packaging materials. Similar packages can be almost entirely isolated from each other. Mixing of such packages with others can reduce the neutron-absorbing effect of the packaging. Flaws are the result of physical relations that can be understood without the need for complicated calculations. However, calculations of a few examples have been made. The findings of the study are that the rules for the mixing of package designs are questionable. General application of the rules may lead to a considerable deterioration in safety. Another conclusion is that a thick layer of concrete, lead or iron on one side of a cube-shaped configuration of packages may provide a lower level of safety than water on all six sides. In the long term, a change in the transport regulations is recommended in order to give due consideration to those cases in which a mixing of package designs would not provide an adequate level of safety.

1. INTRODUCTION

International transport regulations for the transport of fissile materials (based on the Recommendations of the IAEA [1]) permit mixing of package designs. Permissible combinations of packages are based on transport indexes. These indexes are determined for each individual package design. Similar rules are also applied on a national level in connection with the storage and handling of fissile materials.

In 1981, the IAEA took up the question of possible flaws in the rules for mixing of package designs. It was recommended that the studies continue even if no clear indications of such flaws were found [2]. Following a request from the IAEA, the question was dealt with by a working group under the direction of the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development (OECD/NEA). Some calculations were made, but a restricted time schedule meant that the matter was left open for further consideration [3].

* Work supported by the Swedish Nuclear Power Inspectorate.
2. RULES FOR MIXING OF PACKAGE DESIGNS

For each package design intended for fissile materials, an 'allowable number' must be determined. This includes analyses of different combinations of mechanical, fire and water immersion tests. To a certain extent, administrative shortcomings that lead to the number of packages increasing above the allowable number must be considered. In addition, each configuration of packages shall be 'reflected' on all sides by water. The transport index for each package design is 50 divided by the allowable number (radiation outside the package sometimes leads to the transport index being raised). The rules for mixing of package designs are the same as if all packages were of the same design. The accumulated transport index for all packages must not exceed 50 or, in the case of a full load, 100.

3. MAIN TYPES OF FLAWS IN THE RULES FOR MIXING

At the beginning of the study, two main types of flaws in the rules for mixing were identified. In this section, the flaws are explained, while examples and calculations follow in Sections 5 and 6.

3.1. Reduced neutron leakage

For package designs with a limited allowable number, the leakage of neutrons from a configuration of fissile packages is of importance with regard to safety. This leakage is influenced by various materials outside the configuration of fissile packages in that there is a certain degree of neutron reflection. If there are different types of fissile packages, the leakage is affected by how the packages are positioned in the configuration. In order to demonstrate the importance of leakage and reflection, three configurations are shown (Fig. 1). Each square symbolizes the cross-section of a long package.

Figure 1(a) shows a $5 \times 5$ array of package design 1. In an infinite number this package design would give high neutron multiplication. In a limited number, safety requirements are met through considerable leakage of neutrons, even if some are reflected by the water surrounding the array. The water-reflected $5 \times 5$ array is safe. Figure 1(b) shows a mixed $5 \times 5$ array of package designs 1 and 2. To what extent the safety is affected by mixing depends on the quantity (and to a certain extent the energy spectrum) of neutrons returned to the central part of the array by the outer layer of packages, together with the water reflector. It is obvious that mixing of this type can reduce the degree of safety. In Fig. 1(c), package design 3 has been introduced into the configuration shown in Fig. 1(a). The transport index for package design 3 is 0. Consequently, the number of these packages that can be mixed with packages of design 1 is unlimited. In this case it is even more obvious that the mixing can reduce the degree of safety.
3.2. Increased neutron coupling between packages

The neutron absorption effect of a packaging wall depends on the material type and thickness. For packages of the same design, the fissile materials in different packages are separated by two wall thicknesses.

Figure 2(a) shows a simplified model of an infinite number of slabs (assumed to be infinite in two dimensions). The only two materials are fissile material and water. The water slab is so 'thick' that the neutron coupling between the fissile slabs is very limited. Owing to this, the neutron multiplication will be safe for an infinite number of slabs, arranged as in Fig. 4. This model can be used to design a fissile package with water representing a packaging material. Two alternatives can be constructed (see Figs 2(b) and 2(c)). In Fig. 2(b), the packaging walls go through the middle of the water slabs shown in Fig. 2(a), whereas in Fig. 2(c) they go through the middle of the fissile slabs. Figure 2(d) shows how a mixture of the two alternatives can result. The water separation is halved between some of the fissile contents of a mixed configuration of package designs. It is again clear that the mixing can lead to reduced safety.
4. CALCULATIONS

For calculations use has been made of the computer codes KENO-V and NITAWL, as well as the neutron cross-sections in 27 energy groups from SCALE [4]. The calculation method, including the codes and cross-sections, has been validated through calculations of critical experiments. Some of those calculations have been carried out for a working group directed by the NEA [3, 5]. The materials used in the examples below have also been taken from the working group's studies. The effective neutron multiplication factor, \( k_{\text{eff}} \), is calculated with KENO-V. A standard deviation, \( \sigma \), is also reported for each \( k_{\text{eff}} \).

It should be noted that the calculation models were chosen to demonstrate the flaws in the mixing rules. The calculated \( k_{\text{eff}} \)'s are, in many cases, too high to be acceptable in any real application. However, the \( k_{\text{eff}} \)'s for the mixed-package designs are significantly higher than the \( k_{\text{eff}} \)'s for unmixed designs.

5. EXAMPLES: REDUCED NEUTRON LEAKAGE

A number of examples have been developed and calculated. The examples have been selected to illustrate the previous discussion. Some of the cases may seem unrealistic, but real package designs can give similar physical effects (even if they are not so extensive).

5.1. Large number of \(^{235}\text{U}\) metal and non-fissile packages

A model from the NEA's working group is taken as a reference case [3]. The package design consists of a cube-shaped package containing a sphere with metallic \(^{235}\text{U}\). There is no packaging material. In order to secure a safe margin to criticality, the radius of the sphere is slightly decreased, i.e. from 6.242 to 6.1 cm.

![Diagram showing reduced leakage](image)

**FIG. 3.** Reduced leakage (no packaging material except water).
### TABLE I. FISSION PACKAGES SURROUNDED BY NON-FISSION PACKAGES

<table>
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<th>CASE NO.</th>
<th>NO. OF PACKAGES</th>
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<td>2x2x2</td>
<td>2.4</td>
<td>LEAD</td>
<td>40</td>
<td>0.937</td>
<td>0.0071</td>
</tr>
<tr>
<td>10</td>
<td>216</td>
<td>6x6x6</td>
<td>2.4</td>
<td>LEAD</td>
<td>20</td>
<td>1.054</td>
<td>0.0082</td>
</tr>
</tbody>
</table>

If the allowable number is to be 100 (transport index 0.5), 500 undamaged packages and 200 'damaged' packages will be subcritical in a cubic configuration reflected by water. A damaged package is assumed to have an unchanged geometry, but with a 2.4 cm thick water layer (shell) outside the sphere (Fig. 3(a)). Since packages containing concrete, lead, steel or natural uranium are not given a transport index, they can be placed outside the configuration during transit storage or transport. The calculation model is similar to Fig. 1(c) without the water reflector.

Table I shows the results of a number of calculations. The first four calculations refer to cases that are required according to the regulations. These are followed by six cases which show the effect of lead packages outside the central configuration of fissile packages. The results show that safety may be reduced substantially if fissile and non-fissile packages are mixed.

#### 5.2. $^{235}$U metal and UNH packages (transport index 0)

The $^{235}$U metal package design is the same as in the previous example (Fig. 3(a) without water). The uranium nitrate solution (UNH) packages consist of large, flat packages of UNH and water, with the uranium consisting of 100% $^{235}$U. The material data for UNH are also taken from Ref. [3].

In order for a package design, as shown in Fig. 3(b), to meet the requirements of the regulations for a transport index of 0, an infinite number of packages must be safe. The limiting case is obtained if the UNH layers are turned towards each other. According to the NEA's studies, water between the UNH layers does not lead to any significant increase in $k_{eff}$. For an infinite number of packages, $k_{eff}$ is 0.934 ($\sigma = 0.0085$).

Figure 3(c) shows the calculation model for $^{235}$U metal packages without water layers surrounded by UNH packages. For this mixed array $k_{eff}$ is 1.056
TABLE II. FISSION AND NON-FISSION PACKAGES (NEUTRON-ABSORBING WALLS)

<table>
<thead>
<tr>
<th>CASE NO.</th>
<th>NO. OF PACKAGES</th>
<th>ARRAY SHAPE</th>
<th>OUTSIDE PACKAGE MATERIAL</th>
<th>THICKNESS (CM)</th>
<th>$K_{EFF}$</th>
<th>$\sigma^-$</th>
</tr>
</thead>
<tbody>
<tr>
<td>11</td>
<td>512</td>
<td>8\times8\times8</td>
<td>WATER</td>
<td>20</td>
<td>0.967</td>
<td>0.0072</td>
</tr>
<tr>
<td>12</td>
<td>100</td>
<td>5\times5\times4</td>
<td>LEAD</td>
<td>40</td>
<td>1.106</td>
<td>0.0078</td>
</tr>
<tr>
<td>13</td>
<td>100</td>
<td>5\times5\times4</td>
<td>STEEL</td>
<td>40</td>
<td>1.036</td>
<td>0.0071</td>
</tr>
</tbody>
</table>

(a) U-235 METAL PACKAGE WITH STEEL/CD WALLS  
(b) TWO CYLINDRICAL U-235 PACKAGES  
(c) ONE FISSION AND TWO STEEL PACKAGES

FIG. 4. Reduced leakage (fiissile packages including packaging materials).

($\sigma = 0.0084$). In other words, the addition of packages with a transport index of 0 may give a lower level of safety.

5.3. $^{235}$U metal/cadmium and non-fissile packages

Consideration of the packaging materials does not change the conclusions from earlier examples, but rather reinforces them. On the basis of earlier $^{235}$U metal packages, wall materials of steel and cadmium (Cd) (as in Fig. 4(a)) are added. The package design does not allow in-leakage of water during testing. The radius of the sphere can be increased to 6.65 cm.

Three different cases are calculated. The first refers to 512 ($8 \times 8 \times 8$) $^{235}$U metal packages with water reflection. This meets the requirements with regard to undamaged packages and an allowable number of 100. The other two cases concern 100 $^{235}$U metal packages surrounded by a 40 cm thick layer of lead or steel packages, respectively. The results are shown in Table II.
Comparison with the example in Section 5.1 shows that the packaging material can reinforce the effects of mixing. This is only to be expected since water produces moderation, which increases the absorption in the packaging material. Such moderation is not obtained from materials with low contents of hydrogen.

5.4. Mixing a cylindrical $^{235}$U/Cd package with two steel-packages

Figure 4(b) shows a model of two ‘damaged’ cylindrical packages with water reflectors on two sides (symmetry used). The $^{235}$U, Cd and steel materials are the same as previously. Since the $k_{\text{eff}}$ is calculated to be 0.940 ($\sigma = 0.0075$), this can be regarded as a package design with a transport index of 50. In the example shown in Fig. 4(c), a single fissile package is mixed with two 40 cm thick steel packages, one on each side. Here, the $k_{\text{eff}}$ is calculated to be 1.005 ($\sigma = 0.0061$).

6. EXAMPLE: INCREASED NEUTRON COUPLING BETWEEN PACKAGES

6.1. Package designs with UNH or $^{235}$U metal in slab shape

Figures 2(b) and 2(c) give examples of two different package designs which, in infinite numbers, are similar to the example in Fig. 2(a). The fissile material is UNH, its total width in both package designs being 7.0 cm. The total water width is 14.0 cm.

A modified package design is also introduced for $^{235}$U metal instead of UNH. Since water between two $^{235}$U metal slabs would increase the $k_{\text{eff}}$, only the example in Fig. 2(b) was used as a basis for a $^{235}$U metal package design. The $^{235}$U width is 0.4 cm, while the total water width is still 14.0 cm. The results are shown in Table III. It can be seen that despite the fact that all three package designs have a transport index of 0, criticality cannot be excluded.

<table>
<thead>
<tr>
<th>CASE NO.</th>
<th>PACKAGE MATERIAL</th>
<th>DESIGN 1 FIGURE</th>
<th>PACKAGE MATERIAL</th>
<th>DESIGN 2 FIGURE</th>
<th>$k_{\text{eff}}$</th>
<th>$\sigma$</th>
</tr>
</thead>
<tbody>
<tr>
<td>17</td>
<td>UNH</td>
<td>2 (B)</td>
<td>UNH</td>
<td>2 (B)</td>
<td>0.959</td>
<td>0.0073</td>
</tr>
<tr>
<td>18</td>
<td>UNH</td>
<td>2 (B)</td>
<td>UNH</td>
<td>2 (C)</td>
<td>1.040</td>
<td>0.0067</td>
</tr>
<tr>
<td>19</td>
<td>U-235</td>
<td>2 (B)</td>
<td>U-235</td>
<td>2 (B)</td>
<td>0.944</td>
<td>0.0072</td>
</tr>
<tr>
<td>20</td>
<td>U-235</td>
<td>2 (B)</td>
<td>UNH</td>
<td>2 (B)</td>
<td>0.957</td>
<td>0.0073</td>
</tr>
<tr>
<td>21</td>
<td>U-235</td>
<td>2 (B)</td>
<td>UNH</td>
<td>2 (C)</td>
<td>1.056</td>
<td>0.0071</td>
</tr>
</tbody>
</table>
6.2. Two packages of different designs with transport indexes of 0

Consider two package designs which, in damaged condition, are similar to the examples in Figs 5(a) and 5(b). The materials are water and UNH, as in earlier examples. For an infinite number of packages, in accordance with Fig. 5(a), and with the thin layers of water facing each other, $k_{\text{eff}}$ is calculated to be 0.974 ($\sigma = 0.0075$). This is somewhat high, but is used here only for the sake of comparison. The fissile materials in different packages are separated by at least 12.0 cm of water. An infinite number of packages, according to Fig. 5(b) gives a lower $k_{\text{eff}}$. If two packages, one of each design, are placed together, according to Fig. 5(c), the separation between the fissile materials is only 6.0 cm of water. For this case, with only two packages with transport indexes of 0, $k_{\text{eff}}$ is 1.043 ($\sigma = 0.0084$). This is clearly higher than for an infinite number of the one-package design.

7. CONCLUSIONS AND RECOMMENDATIONS

The studies have shown that mixing package designs during transport can result in a lower level of safety. It cannot be excluded that criticality occurs even if all rules are adhered to and the packages are not damaged. For example, reflection by concrete, lead or iron on one side of a configuration of packages can result in a lower level of safety than water on all sides of the configuration. A package design often includes various options for both contents and packagings. The results for the mixing of package designs can be extended to the mixing of different options or different 'damages' of the same package design.

In the long term, it is recommended that the rules for mixing package designs should be changed. In addition, it is recommended that the rules for neutron reflection should be changed so that the realistic effects of concrete and other materials can be accounted for.
REFERENCES


QUALITY ASSURANCE ASPECTS OF THE FABRICATION OF THE AGN 1 CASK AND ITS TRANSPORT WITH SPENT FUEL

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Abstract

QUALITY ASSURANCE ASPECTS OF THE FABRICATION OF THE AGN 1 CASK AND ITS TRANSPORT WITH SPENT FUEL.

The AGN 1 is the first cask for LWR spent fuel entirely designed and fabricated in Italy. The cask, made with a thick steel forging, has a maximum capacity of 7 elements; two ellipsoidal shock absorbers are connected to the ends of the body. The fuel elements are spaced and supported by a stainless steel basket provided with boron poisoned steel channels for criticality control. A finned water jacket is placed around the cylindrical surface to provide neutron shielding. Some of these aspects may be considered a completely new concept and design. The basic design of the cask was developed by AGIP and extended, through the detailed mechanical and thermal design to the stage of fabrication documents by the Nuovo Pignone Company, which is also the manufacturer. The most significant activities performed by Nuovo Pignone, with particular reference to quality assurance, are outlined in the paper. Since 1984, the AGN 1 has been in continuous use for the removal of spent fuel from the Trino Vercellese and Garigliano power plants to the storage pool at Avogadro. The transport campaign from Garigliano to Avogadro is performed by Borghi Nucleare, which was requested also to be responsible for the supply under proper quality assurance of all related services and engineering support. The AGN 1 non-stop transport from Garigliano to Avogadro, in fact, owing to the long distance to be covered, is one of the most demanding in Europe. In the paper, some of the most significant transport quality assurance aspects are summarized.
1. QUALITY ASSURANCE PROGRAMME FOR CASK FABRICATION

The design and fabrication of the AGN 1 transport cask obviously had to be subject to the quality assurance (QA) requirements set down in the IAEA Regulations and those indicated in the "Guida tecnica n° 8" issued by the Italian Nuclear Authority CNEN (now ENEA) and mandatory for all nuclear activities. In addition, the cask also had to comply with the regulations of the Italian Authority for Pressure Vessels ANCC (now ISPESL).

In this connection, the preliminary activities of the manufacturer Nuovo Pignone were devoted to establishing a job quality assurance programme in order to define tasks and responsibilities for the various quality related activities and to establish the number and type of technical documents to be produced to fulfil all governing requirements.

It was decided to extend the full QA requirements not only to the container itself but also to all its accessories, allowing a reduced quality level only for very few secondary components. This approach was possible since Nuovo Pignone could rely on an already active quality organization, but it raised a number of specific problems, mainly in the selection of new subsuppliers and the close supervision of these, and the need for radically new fabrication methods.

AGIP, and also the final user of the cask (ENEL, the Italian National Agency for Energy), together with the Competent Authorities were involved even at these early stages in the auditing of the Nuovo Pignone QA programme and facilities.

2. ENGINEERING

The basic design of the AGN 1 cask was carried out by AGIP. The detailed design was then done by Nuovo Pignone, still in close co-operation with AGIP. In particular, Nuovo Pignone developed the mechanical design of the containing system in accordance with ASME Code Section III and also with the ANCC Regulations.

At this stage, the involvement of the Quality Assurance Department of the Nuovo Pignone Massa plant, where the fabrication took place, greatly increased. Certain design arrangements or limitations, imposed by manufacturing techniques or by non-destructive test requirements, became evident during the preliminary joint review of workshop drawings, in particular those for the special jacket.

Naturally, all purchasing, fabrication and test activities needed to be covered by relevant documents, so that over 150 workshop drawings, 35 material purchasing specifications, 40 inspection/tests and fabrication procedures, 50 welding procedure specifications and 25 fabrication/inspection plants have been produced and reviewed by the QA Department.

Particularly in this early stage, interface activities with the QA Departments of AGIP and ENEL and with the appointed services of ENEA and ISPESL, in order to obtain their approval for the detailed project, were very important.
3. PURCHASING ACTIVITIES

The evaluation of the subsuppliers, within the Nuovo Pignone organization, is the responsibility of the QA Department alone.

For AGN 1 it was necessary to select new supply sources, as a result of the particular kind of materials to be purchased.

For the cask body, selection and audit were made of a steel manufacturer capable of producing a high integrity hollow forging, having a wall thickness of about 350 mm and a total length of over 4.5 m, inclusive of the extra length at both ends for destructive tests and additional sections to be utilized as a weld procedure qualification test sample at the Nuovo Pignone plant. This European manufacturer also had to be approved by ISPESL. The supplier's QA Programme was then positively reviewed and the supply was performed under the close control of Nuovo Pignone and with the participation of AGIP and the other involved parties at the established hold points.

Again, for the stainless steel plates (105 mm thick) for the jacket, ISPESL had to approve the supplier, and all activities were performed under the co-ordination of the Nuovo Pignone QA Department.

The choice of the supplier for the boron poisoned stainless steel plates for the basket channel fabrication was practically imposed, being restricted to only one company at the time, but anyway a stringent QA Programme was required and implemented.

The minor forgings and the bolting and carbon steel plates were purchased, still under the QA system, from habitual suppliers already approved by Nuovo Pignone.

The involvement of the Nuovo Pignone QA Department was very considerable in these purchasing activities, not only during the preliminary audits but also in the review and approval of the technical documents submitted by the suppliers, their approval by AGIP, ISPESL, ENEL and ENEA and, finally, the participation, direct and co-ordinated with all other parties in the most significant fabrication, inspection and tests stages at the supplier's facilities.

During these activities, about one hundred further technical documents (procedures, specifications, quality control plans, etc.) were reviewed.

4. SPECIAL PROCESS QUALIFICATION

Before the start of fabrication, qualification tests of special processes, specifically of the weld procedures, were carried out.

The most important weld in the project is the circumferential joint between the cylindrical body and the head. Nuovo Pignone already had experience with heavy wall butt welds but this was, and remains, the maximum thickness ever welded. For the qualification of this procedure it was decided to use a full scale
test sample from the actual material, obtained from the same body forging. The bevel preparation adopted for the joint, that is a narrow gap with nearly parallel edges, was aimed at reducing the amount of filler metal and making it possible to weld almost completely from the outside in order to minimize difficulties resulting from reduced internal accessibility and increased by the preheating conditions. The weld was carried out by the multipass submerged arc procedure using a special weld machine head specifically developed in order to assure the correct positioning of the wire inside the groove.

The test sample was then subjected to the same non-destructive examination (NDE), stress relieving heat treatment, and final NDE as provided for the production weld joint. All destructive tests in accordance with ANCC Regulations and AGIP additional requirements, for a total of 120 specimens, were carried out with widely satisfactory results.

The other type of welds, even if Nuovo Pignone already had experience of them, were requalified for the job.

5. MAIN FABRICATION STAGES AND RELATED PROBLEMS

The maximum care and surveyance were exercised during the welding of the main joint in the body. This required over 115 hours of uninterrupted work in the presence of representatives of all participating parties in continuous shifts. A good weld result was essential, especially to avoid repairs that would be very critical.

The machining to obtain the finning of the jacket from the barrels fabricated by rolled plates was done in the Nuovo Pignone Florence plant which is equipped with very accurate machine tools (being a well known manufacturer of turbines, compressors and pumps).

The machining of the basket baffles was also performed there.

In order to solve certain problems related to the tight tolerances imposed by the project it was necessary to plan an adequate sequence of positioning of the pieces on the machines and to provide special fixtures in order to control the deformations during the machining itself — not only those due to the natural elasticity of stainless steel but also those due to the amount of removed material.

As an example, for obtaining the holes for fuel channels and for the tie-rods in the basket baffles, it was necessary to remove about 80% of the material. In this case, the need to respect the necessary out-of-plane tolerance by adopting adequate sequence made it necessary in view of the shape of the pieces and the surface finishing conditions to utilize different kinds of machine with different positionings.

Specific problems occurred in the production of the seven basket channels fabricated with stainless steel boron poisoned plates 6 mm thick. The three central channels were made of four plate strips welded together at the corners to form a
square hollow parallelepiped. The remaining peripheral channels were made by a strip bent to form two sides, and welded to another two flat strips to form a pseudo-square section parallelepiped.

The special design of the basket, which was made by mechanical assembly of all the components without any welded joints in order to allow easy eventual replacement, required the channels to comply with very tight geometrical and dimensional tolerances. The distortion during both bending and welding had therefore to be limited as much as possible.

The first problem was solved by utilizing oleodynamic equipment especially designed to balance the bending effect during forming. The second problem was solved by using pulsed-arc TIG (tungsten inert gas) welding with filler metal and by providing inside the channel a special cooling fixture for quick heat removal.

A simulated test of inserting the fuel elements into each channel was performed using a mock-up reproducing the size and weight of one element connected with a dynamometer with a digital indicator.

The assembling of the finned jacket, made up of four barrels, closed by two forged rings at each end and supported by intermediate stiffening rings, required a particular sequence to reduce the deformations for weld shrinkage.
All the main weld joints in the jacket are butt welds and were fully radio-
graphed.

Both cask cavity and neutron shield have passed the hydrostatic tests
required by ANCC regulations. The tightness of all gaskets was checked by a
helium leak test with a mass spectrometer in the plant.

In order to measure the maximum steady state temperature reached during
transport by the accessible outer surface and to verify that it does not exceed the
value imposed by the IAEA, a thermal test was conducted on the cask.

The dimensioning of the fins had already been verified by the Nuovo Pignone
Research Department by tests on a full scale mock-up, and the same Department
also carried out a thermal test on the completed cask. The thermal test was
performed by heating up the cask with two electric resistances having a total
thermal power of about 22 kW inserted into the cavity. The cavity and the jacket
chamber were partially filled with water and the power was progressively increased
up to the maximum value. The total time of the test was over three weeks,
during which continuous readings of temperature, as measured by several
thermocouples, were made.

At the end of fabrication, all the functional and handling tests were carried
out. These, included positioning of the cask on its transport saddle, clamping and
moving out of shock absorbers, rotation and lifting of cask (see Fig. 1).

In early 1984 the cask was consigned to the customer and the user, together
with the relevant records of all quality related activities, as a natural consequence
of the QA programme implementation.

6. GARIGLIANO TRANSPORT CAMPAIGN

The AGN 1 is now employed in an intensive way for the transport of
Garigliano irradiated fuel elements (see Fig. 2). The programme for decommission-
ing of the plant foresees that all the irradiated fuel elements (more than 320) must
be transferred from Garigliano, 200 km south of Rome, to the storage plant at
Avogadro in the north of Italy. The ENEL entrusted Borghi Nucleare SpA, a
qualified Italian firm operating in the field of radioactive material transport for
more than 20 years, with the task, including the supply of all the related services
and engineering supports in accordance with the QA requirements. The transport
campaign perhaps is one of the most arduous ever undertaken in Europe owing
do the number of transports to be carried out without stops, the distance to be
covered (more than 900 km each time) in all weather conditions and the rhythm
of the campaign. The ENEL requested that QA procedures and methods be used
for the following reasons:

(1) The IAEA Regulations for the transport of radioactive material are
orientated towards the idea of managing this activity, as well as all nuclear ones,
by applying QA;
(2) It is necessary to carry out a considerable number of transports without stops and to guarantee always that the high safety standards requested be met.

The Borghi's convoys for the transport of the loaded cask include: a special bogie hauled by a Fiat IVECO truck, an identical Fiat IVECO truck as escort, a workshop vehicle and a laboratory vehicle. The laboratory has a full set of tools and instruments for monitoring and measurements, and radioprotection and health physics interventions in emergency cases.

Borghi began to introduce QA in its own activities in 1976 and therefore had a QA programme in operation before the Garigliano campaign. However, the campaign required the implementation of special QA activities in some particular areas, namely: staff training, management procedures, the supply of vehicles and safety and intervention devices in the event of an emergency, and the controlled management of vehicles, devices and all pertinent spare parts. The transport engineering was based on three criteria:

(1) The loaded AGN 1 cask must be transported from Garigliano to Avogadro without stops, including those which could be due to anomalous occurrences.
(mechanical failures, accidents along the way, slowing down and queue formation as a result of road works, bad weather);

(2) The main and subsidiary transport systems and devices must be dimensioned to be reliable and able to be used in hypothetical anomalous and serious occurrences.

(3) The convoy must be independent in the event of anomalous occurrences.

It has been necessary to draw up a manual on management procedures for intervention in the event of anomalous occurrences and this was submitted to the competent authorities. The document also deals with the possibility of deviating the transport along pre-arranged alternative intersecting routings, so to avoid stops and slowing down due to any reason.

The itineraries were chosen with due consideration given to the loads caused by the transport vehicle (about 140 t) on the structures along the way and the desirability of avoiding inhabited areas and roads carrying heavy traffic. The vehicles were manufactured especially for the Garigliano campaign with appropriate QA applied. In particular, the transport bogie is the first example in Italy of a vehicle mode with qualified welding procedures, qualified welders and also with materials certified after qualification of the manufacturer by Borghi.

All the structural welds have been tested by magnetic examinations and special impact tests have been requested on the materials.

Finally, the use of the vehicles, the equipment, the spare parts and the radio-protection apparatus is in compliance with duly drawn up procedures.

Proper forms and checklists filled in by qualified staff at the beginning of and during each transport, allow the resolution of any non-conformity which may occur.

7. CONCLUSIONS

It is difficult to deny that some time delay in the design, fabrication and testing of a spent fuel transport cask such as the AGN 1 is unavoidable if QA procedures are followed at each phase. There is also no doubt that some increase in costs occurs. In addition, more people and more organizations are involved in reviewing, checking and testing everything, and putting them together at the proper time and place is often a hard job for the QA service and the manufacturer. Delivery time and costs seem to suffer too from the application of QA requirements.

Another consequence of strict QA procedures is an increase in the amount of paper!

On the other side, one could ask what actions required by QA criteria could be omitted in such a safety-related job. The answer to that question is probably that the design, fabrication and transport work would certainly require the application of QA, even if we called it by another name.
LEGAL, POLITICAL AND OPERATIONAL ELEMENTS INVOLVED IN THE TRANSPORT OF BROOKHAVEN SPENT NUCLEAR FUEL IN THE USA

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Abstract

LEGAL, POLITICAL AND OPERATIONAL ELEMENTS INVOLVED IN THE TRANSPORT OF BROOKHAVEN SPENT NUCLEAR FUEL IN THE USA.

In the United States of America, as in many other countries, the shipping of radioactive materials is a difficult task. One of the major problems is the issue of perception versus reality, with significant emotional and political overtones attached to it. Such was the case with a United States Department of Energy (DOE) fuel movement project for which nine years were needed in order to resolve complex legal issues. This included State, local and Federal regulatory issues, problems with package certification, risk assessments and a host of political and administrative issues. Although the DOE had been moving fuel from Brookhaven National Laboratory since 1962, the City of New York was successful in stopping these shipments from 1976 to 1985. Resolution of the problem included lawsuits, rulemaking by the Department of Transportation (DOT) and, finally, a final review by the Supreme Court of the United States of America.

In 1976, the City of New York amended its Health Code to effectively bar the transportation of high-level radioactive materials through the City. This ban impacted directly on the transportation of spent nuclear fuel elements from the High Flux Beam Reactor (HFBR) located at Brookhaven National Laboratory.

This paper is a summary of the events that took place between January 15, 1976, the date of the action by the City, and the time the shipments were resumed in January 1985. It took 9 years to solve the legal problems and to overturn the ban.
These 9 years were spent on the following:

(1) Prolonged litigation that ended with an appeal to the Supreme Court of the United States.

(2) Regulatory issues about the Federal preemption of State or City laws.

(3) Risk assessments where the City attempted to prove alternate routing was preferable.

(4) Certification of casks and discussions about the shipping cask safety analysis.

(5) Political and administrative issues with countless pressures from elected public officials trying to prevent any shipments from happening.

As noted, the enacting of the ban impacted shipments originating at the HFBR, and only on those shipments. No one else has to ship spent fuel through New York City. However, in implementing the ban, the City stated their goal was stopping not only the 10 shipments per year from Brookhaven, but also prohibiting future shipments from the Shoreham Nuclear Power Station which was then under construction on Long Island.

These were estimated as 50 shipments per year. Up until the time the ban was implemented, the Laboratory had transported the casks by truck through the City on the way to a chemical reprocessing plant.

A total of 330 shipments of spent fuel had been made from the Laboratory prior to 1976 when the City imposed the ban. All shipments went through the City.

The HFBR is a 60 MW D\textsubscript{2}O-moderated and cooled PWR-type research reactor which has been in continuous operation since 1965. The HFBR is a DOE-owned facility located at Brookhaven National Laboratory on Long Island in the State of New York. The reactor is fueled with 93.5 percent fully enriched uranium contained in aluminum plate-type elements. The core consists of 28 of these elements, with an annual use of 154 elements. Elements discharged from the reactor were stored at the reactor facility for 1 year before shipping. The fuel storage canal had an original capacity of about 280 elements. During the 9-year ban on shipments, the reactor continued to operate and the canal storage capacity was increased. A maximum of 980 elements were stored in the canal in January 1985, when shipping resumed.

Immediately after the ban, the Laboratory attempted to develop an alternate shipping route by sending the shipments east on Long Island, then by ferry to the City of New London in
Connecticut, then by truck to the reprocessing plant. Six of these shipments were made before the City of New London implemented a similar ban.

The authors had the opportunity to discuss with New London officials the intent of their ban. It was solely to prevent the City of New York from routing hazardous materials through New London rather than through New York. They felt, without any doubt, the City of New York was attempting to shift the burden of transport from its own backyard to that of its neighbors. They noted this was a problem that demanded a solution at the national level and only Federal intervention would solve it.

A week after the action by New York City to ban these shipments, the U.S. Justice Department brought suit in the U.S. District Court in New York seeking a permanent injunction against New York City's preventing the transportation of radioactive materials in and through its borders. The requested injunction was not granted. The Government had not been able to show irreparable harm, the reactor could store the fuel. Instead, the judge suggested that the DOT find a solution to this problem. In February 1977, Associated Universities, Inc., the corporation that operates the Laboratory, then made an application to the DOT for an administrative ruling that the amendment to the New York City Health Code was inconsistent with the U.S. Hazardous Materials Transportation Act. The DOT held a public hearing on this application and in April 1978 announced that the New York City amendment was a routing requirement and since the DOT had not yet issued routing regulations, there was no inconsistency. It was obvious that a set of Federal routing regulations were required. Over 2 years had passed and little progress had been made.

This prompted the DOT to proceed immediately in a rulemaking process which ended in January 1981, when it issued a final rule. This new rule entitled Radioactive Materials, Routing and Driver Training Requirements, commonly known as HM-164, provides that carriers of "large quantities" of radioactive materials (such as spent fuel elements) are required to use "preferred routes" for the shipments. Preferred routes are defined as (1) interstate system highways or bypasses, or (2) alternative highway routes designated by a State routing agency. The rationale of this rule is when Federal rules are complied with, spent nuclear fuel can be transported over any interstate highway and most other comparable routes with a confident level of safety.

This new rule was to take effect on February 1, 1982. However, in March 1981, New York City started an action in Federal District Court seeking to invalidate DOT's rule. In an opinion filed in February 1982, this court held that DOT had failed to follow proper procedures in its rulemaking process and declared DOT's rule invalid. In August 1983, the U.S. Court of Appeals for the Second Circuit reversed that District Court decision. After the City was unsuccessful in having the Second Circuit stay its
decision, the U.S. District Court, in November 1983, decreed that the DOT rule was valid. This ruling upheld HM-164 and its preemptive effect over State and local transportation bans. On February 27, 1984, the U.S. Supreme Court refused to hear an appeal to this decision ending the legal battle. It had taken 8 years.

The clock had apparently run out for the City. With the Supreme Court denial to review the case, it appeared the shipments could begin. The City, however, made another appeal to the Federal government requesting a 6-month delay in shipments to allow them time to have a new risk analysis prepared. The City was aware that the canal storage facility at the HFBR was nearly filled. To get agreement for the 6-month delay required to complete this study, the City of New York offered to pay the additional storage costs at Brookhaven. Since the fuel discharged during that 6-month period would have completely filled the storage racks in the canal, the City agreed to pay for an additional storage rack. The final cost paid by the City for this rack was $20,000. The analysis attempted to demonstrate that lower risks to the public would be involved if alternate shipping routes were used. Obvious alternate routes would involve some form of water transport and both barging and ferry routes were evaluated. The study should demonstrate that alternate routing provides at least an equivalent level of safety and its requirement would not unreasonably burden commerce.

The analysis failed to show that the shipments through New York City, under the conditions dictated by HM-164, would have higher risks than alternate methods (that the level of safety would be significantly improved if an alternate route was used).

The analysis did show significant increases in costs for all alternative routes. A decision was made that no further delays were allowable and on January 1, 1985, permission was given for shipments to begin.

The Laboratory had, for some months, been discussing with New York officials the detailed procedures to be used if and when shipments began:

(1) A route was selected.

(2) It was agreed that the City would be notified as to the dates of each shipment.

(3) It was agreed that the City would escort the shipments.

(4) The shipments would transit the City only during early morning hours.

In addition, it was also agreed that the State of New York would escort the shipments, so similar procedures were worked out
with State officials. When the decision was made to start shipping, these procedures were implemented and the first shipment was made on January 22, 1985.

This story would be incomplete without a comment concerning the problems with cask certifications. Since the HFBR is a DOE facility, the certificate of compliance authorizing the use of a specific shipping cask for the fuel is issued by DOE rather than the Nuclear Regulatory Commission (NRC). However, the DOE had recently decided to request NRC review of the shipping casks that were being used for these shipments so they could be used by NRC licensees.

As a result of this review, in May 1985 the NRC raised a number of questions about the safety analysis report for the cask and the DOE cancelled the certificate of compliance. All shipments stopped. A different cask, one with an NRC certificate of compliance, was made available and shipments resumed 2 months later. This shipping campaign continued from January 22, 1985, into April of 1986. A total of 32 shipments were made moving 763 elements.

Plans are being made to make another series of shipments in the fall of 1986. The backlog of stored fuel has been successfully reduced and future campaigns will be about seven shipments per year. Although there still are occasional statements by local politicians and rallies by anti-nuclear groups calling for a halt to these shipments, it appears the rules and procedures established by HM-164 are prevailing and these shipments will continue.

There is a footnote on the DOE compliance certification process. As mentioned, DOE regulates its own internal activities. Until recently, eight different DOE field office managers could issue a certificate of compliance. This arrangement led to some inconsistencies and has now been changed.

In January 1986, DOE established certification authority with a single certification official in Washington D.C. This official has set up a process nearly identical to the NRC. Consistent and rigorous reviews of packages will be assured. In this process, the DOE is fully committed to the high standards set by the International Atomic Energy Agency regulations.
EXPERIENCE IN REGULATING THE TRANSPORT OF RADIOACTIVE MATERIALS IN INDIA

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Abstract

EXPERIENCE IN REGULATING THE TRANSPORT OF RADIOACTIVE MATERIALS IN INDIA.

Formal legislation specific to the transport of radioactive materials is to be brought out soon in India. In the absence of such legislation, radioactive shipments have so far been regulated by the Division of Radiological Protection (DRP), Bhabha Atomic Research Centre (BARC), Bombay. The safety standards prescribed by DRP are in general conformity with the 1973 IAEA Regulations (As Amended). Design approval is necessary for both Type A and Type B packagings, even if the former is intended for non-fissile materials. Approval is necessary for each non-routine shipment. Consignors comply with the requirements since only qualified, trained, users are authorized to handle radioactive materials. Interactions with carriers reveal their awareness of the need to ensure that the prescribed standards conform to international regulations. The safety record shows that there have been very few incidents involving radioactive consignments and none have resulted in significant exposure to transport workers or to the public.

Preamble

Transport of radioactive materials has been gaining importance over the years. This may be attributed to two circumstances. (A) It is during this operation that members of the public as represented by the cargo handlers are likely to be exposed to relatively high levels of radiation. In the population group, it is the transport workers who are exposed to the highest dose rates [1]. (B) This operation involves leaving packages containing radioactive materials in the common streams of transport, for the natural currents of these streams to carry the packages to their destinations. It therefore becomes necessary to regulate the transport of radioactive materials.

India's experience in the transportation of radioactive materials spans a period of over a quarter century. The volume of the radioactive shipments has been on the increase over the years (Table I).
TABLE I. TRANSPORT OF PACKAGES CONTAINING RADIOACTIVE MATERIALS DURING THE PERIOD 1981-1985

<table>
<thead>
<tr>
<th>Years</th>
<th>Number of packages*</th>
</tr>
</thead>
<tbody>
<tr>
<td>1981</td>
<td>12 423</td>
</tr>
<tr>
<td>1982</td>
<td>13 952</td>
</tr>
<tr>
<td>1983</td>
<td>15 252</td>
</tr>
<tr>
<td>1984</td>
<td>16 328</td>
</tr>
<tr>
<td>1985</td>
<td>17 604</td>
</tr>
</tbody>
</table>

* Packages containing sources supplied by the Isotope Group, BARC to medical, industrial and research institutions in India

The shipments connected with the nuclear fuel cycle are many and engendered with potential radiological consequences of varying severity whereas radioactive shipments outside the nuclear fuel cycle have provided varied experiences in the regulation of transportation.

Radiation Protection Legislation in India

The Atomic Energy Act, 1962 requires the Central Government to exercise control over the production and use of atomic energy. The responsibility of making rules to ensure safety in the use of radiation sources lies with the Central Government. The Division of Radiological Protection (DRP) of the Bhabha Atomic Research Centre (BARC) was notified as the competent authority for enforcing the Radiation Protection Rules, 1971 (RPR 1971) which were issued under the Act. These rules are concerned with safety in radiation installations and licensing for the procurement and handling of radiation sources. Currently, the Atomic Energy Regulatory Board is the competent authority for framing rules under the Act.

Rules specific to the transport of radioactive materials are to be issued by the Government of India.
shortly. Despite the absence of legislation in this regard the executive power vested in the competent authority under the Act is adequate to exercise control over transport of radioactive materials in the country.

Guidelines and codes of practice for ensuring radiological safety in the transport of a wide range of radioactive shipments have been prepared by DRP. The standards of safety and operational control prescribed in these codes provide the technical basis for exercising control over transport of radioactive materials. These standards are in general conformity with those prescribed in the IAEA Regulations [2].

The draft rules for the safe transport of radioactive materials which are due to be issued are based on the current edition of the IAEA Safety Series No. 6 [3]. The deviations from the IAEA Regulations are deemed necessary in the context of the local conditions and have been kept to a minimum.

**Transport Operations**

All transport operations connected with the nuclear fuel cycle are carried out by the Department of Atomic Energy. Hence there has been no difficulty in ensuring compliance with the stipulated safety standards.

The following are instances of transport of radioactive materials outside the nuclear fuel cycle in India:

1. Supply of sources by BARC to medical, industrial and research institutions in India.
2. Return of decayed sealed sources by the users to the supplier after use.
3. Movement of industrial radiography sources (particularly Iridium-192) from one site to another during use.
4. Occasionally, transport of radioactive waste such as contaminated objects and decayed sources from users to BARC for disposal.

**Scope of Control**

The radioactive consignments which are regulated vary from a few MBq of radiopharmaceuticals to several PBq of mixed fission products in spent fuel shipments.

The control machinery has a wide range of activities.
- Design approval of special form radioactive material
- Design approval of packagings
- Specific approval of non-routine shipments.

A break-down of the different types of shipments is given in Table II.

### TABLE II. TYPICAL BREAKDOWN OF SHIPMENTS

<table>
<thead>
<tr>
<th>Type of consignment</th>
<th>Percentage</th>
</tr>
</thead>
<tbody>
<tr>
<td>Type A</td>
<td>78</td>
</tr>
<tr>
<td>Type B</td>
<td>1</td>
</tr>
<tr>
<td>Exempt</td>
<td>21</td>
</tr>
</tbody>
</table>

**Design Approval**

A packaging, whether Type A or Type B, may be deployed for the transport of radioactive materials in India only after specific type approval is obtained from DRP.

All Type A packages are required to be actually subjected to the prescribed tests. Design approval for Type A packagings is required even if the intended contents are not fissile. As for Type B packages, theoretical assessments of package response to test conditions are examined prior to approval. The design approval certificates for Type B packages are subject to review once in five years. Most of the Type A and Type B packagings in use in India are designed by the Isotope Group, BARC [4]. So far type approval has been issued for 14 Type A and 4 Type B packagings of Indian origin. These are deployed for most of the shipments in India. Further, essentially 7 Type B packagings of foreign design approved by their national competent authorities are used in India for shipments of teletherapy sources and remotely operated type radiography cameras.

**Shipment Approval**

For each non-routine radioactive shipment in India specific approval from DRP is necessary. The current
practice ensures that only persons (or institutions) who are authorised under the RPR 1971 to handle radioactive materials can be consignors of radioactive shipment. Only qualified and trained persons are authorised to handle sources. These users are instructed under the terms and conditions of authorisation not to move the radioactive materials from one institution to another without prior permission from DRP. The following checks are made by DRP before issuance of shipment approval:

a) The need for transport
b) The particulars regarding the content
c) Whether the packaging in which the radioactive material is to be transported is of approved design
d) The measures to be taken to ensure safety during transport.
e) Whether the consignee is authorised and ready to receive the radioactive material.

The decayed radiography sources are returned to BARC in the same packaging in which replacement sources are supplied. Under these conditions individual shipment approval is not required. All such authorised users are instructed regarding the safe transport procedures.

Interactions with Carriers

In the procedures prescribed for the transport of radioactive materials, the carriers' role is kept to a minimum, but all carriers are not alike. Road transport organizations do not hesitate to carry radioactive cargo, but delivery of packages at the destination is delayed, frequently due to intermediate off-loading and reloading operations necessitated by some vehicles not having permits to ply over the relevant regions.

The railways are wary about accepting radioactive cargo, particularly from private consignors. This circumspection is the result of an instance of delay in the delivery of a radioactive consignment which was earlier feared lost. A series of meetings were held with the railway authorities familiarising them with the design/performance standards of packages and the control procedure aimed at ensuring radiological safety of cargo handlers, passengers and public. This has brought about a moderate change in their outlook.

Similar hesitation to carry radioactive cargo on the part of the domestic airlines entailed an extended
dialogue with the concerned authorities. A training course was conducted by DRP for the managerial staff of the airlines. They required to be convinced that the relevant safety standards prescribed in India are in conformity with the IATA Regulations. Currently, the airlines carry the bulk of the radioactive cargo in India (Table III).

### TABLE III. TRANSPORT MODE WISE BREAKDOWN OF RADIOACTIVE CONSIGNMENTS

<table>
<thead>
<tr>
<th>Mode</th>
<th>Percentage</th>
</tr>
</thead>
<tbody>
<tr>
<td>Road</td>
<td>25.5</td>
</tr>
<tr>
<td>Air</td>
<td>74.0</td>
</tr>
<tr>
<td>Post</td>
<td>0.5</td>
</tr>
</tbody>
</table>

Maritime carriers are generally hesitant to book radioactive consignments. However, upon demonstration that the relevant safety standards are in conformity with the IMO Regulations, they accept the cargo.

**Education and Research**

The effectiveness of control over transport of radioactive materials improves if the efforts at regulating are backed up by an exercise in educating consignors, carriers and other interested parties. This is particularly true in the absence of legislation where informed caution has to operate rather than blind faith. Pursuing this belief DRP brought out a series of documents—Radiation Protection Guides—each a booklet relating to various aspects of transport of radioactive materials and each booklet complete in itself (e.g. *Maximum Permissible Activities in Radioactive Consignments*, *Emergency Response Planning for Transport Accidents involving Radioactive Consignments*, *Safe Transport of Low Specific Activity Materials* and *Safe Transport of Fissile Materials*).
These documents were made available to the users. The users' response to these documents is a clear indication of their utility. Education imparts care as well as conviction. Conviction assures compliance.

The adequacy of the package design requirements and the efficacy of the education imparted to consignors and carriers should be reviewed. For this purpose, a study was conducted under a Co-ordinated Research Programme of the International Atomic Energy Agency to assess the radiation dose received by cargo handlers at the Bombay airport. The study revealed that the individual worker would receive less than 1 mSv per year [5]. Our efforts will be aimed at keeping the dose level low even with an increase in the number of consignments handled.

Safety Record

In the absence of formal legislation, difficulties are encountered in controlling the transport of radioactive materials. Yet our safety record has been good. Over the past 25 years there were only two accidents involving trucks carrying Type B packages. On both occasions the packages came out unscathed. There have been seven incidents in which small Type A packages were damaged during handling. In none of the incidents was there any release of the radioactive contents. There have been four instances of Type A packages reaching the consignee with the sealed radioactive sources having partially come out of the shielding but lying within the outer container and a solitary case of the package being tampered with en route.

In none of these instances was estimated dose to public significant.

At the Destination

The system of control is established and it works adequately. The experience gained in the process would no doubt prove useful in implementing the rules when they are promulgated.

The safety record relating to transport of radioactive materials has set a high standard. Therefore, with the increase in the volume of transport of radioactive materials, it is necessary to maintain these safety standards in the years to come.
Acknowledgements

Our thanks are due to Mr. S.D. Soman, Associate Director Radiological Group, BARC for the discussions and his helpful suggestions. We acknowledge the co-operation extended by Mr. R.G. Deshpande, Associate Director, Isotope Group, BARC and Mr. S.R.K. Iyer, Isotope Group, BARC for providing us with the required data.

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QUALITY ASSURANCE APPLIED TO THE USE OF TYPE B FLASKS

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Abstract

QUALITY ASSURANCE APPLIED TO THE USE OF TYPE B FLASKS.

In the context of spent nuclear fuel shipments from European LWRs to the reprocessing plant, Cogéma and NTL (Nukleare Transportleistungen) have developed a quality assurance programme which covers transport and loading operations of Type B flasks. The paper describes on the one hand the present operating practice, and on the other hand principles to be used for setting up a quality assurance system applicable to the management of fuel consignments.

1. PRESENT OPERATING PRACTICE

In order to follow all operations during a complete transport cycle, from the departure of an empty flask until its return (loaded with irradiated fuel) to the reprocessing plant, a so-called transport dossier is built up step by step as the actions are carried out. This dossier represents the documented evidence that the transport is executed in accordance with the contractual and regulatory requirements. In order to explain the way in which it has been elaborated, we will analyse:

— the role and responsibilities of the various bodies involved (see Fig. 1);
— the arrangements made and the methods developed by the carrier, who co-ordinates the execution of the actions during the transport cycle.
1.1. Role and responsibilities of the different intervening bodies

1.1.1. Carrier and its subcontractors

First of all, before starting the transport cycle, the carrier ensures that he is in possession of the documents listed below.

(a) Documents related to fuel qualification:
   — Package approval certificate transmitted by the flask owner
   — Document certifying compatibility between flask and fuel, produced by the carrier
   — Document of fuel acceptance, transmitted by the reprocessor.
(b) Maintenance certificates for flask and vehicles  
(c) Official authorizations 
(d) Flask operating documents  
(e) Consignor’s and reprocessor’s agreements on the transport programme. 

Furthermore, the carrier is responsible for the supervision of operations from the preparation of the empty flask, including handling on its vehicle, up to the return to the reprocessor of the loaded flask.

1.1.2. Consignor

Basically, these are the electrical utilities, owners or users of the fuel to be evacuated. Their roles and responsibilities include:
- Establishing contracts with the reprocessor 
- Giving to the reprocessor the fuel data for the evaluation and acceptance of the fuel 
- When necessary, providing the fuel owner’s agreement for reprocessing 
- Being responsible for applying the flask operating documents transmitted by the flask owner through the reprocessor 
- Transferring their consignor responsibilities to the reprocessor, who in turn delegates to the carrier signature of the declaration sheet in the ‘transport document’.

1.1.3. Reprocessor

The reprocessor:
- Approves the list of fuel to be evacuated, as proposed by the consignor 
- Subcontracts to the carrier the fuel evacuation.

1.1.4. Flask owner

The flask owner:
- Is responsible for establishing and distributing the flask operating documents 
- In conjunction with the designer is in charge of obtaining the package approval certificate and the necessary validations from the competent authorities 
- Is responsible for the maintenance status of the flasks and vehicles.

1.1.5. Competent authorities

These are the official bodies delegated by the different governmental ministries of the countries through which the transport passes. Their responsibilities are to:
- Issue the package approval certificates and the necessary validations 
- Issue the different transport authorizations required by the appropriate regulations and particular laws (security bodies, customs, insurances, etc.).
1.2. Skeleton of the procedures used by the carriers

Nukleare Transportleistungen (NTL) and Cogéma have set up the methods and associated documents which support the responsibilities of the carrier as described in paragraph 1.1.1. These are listed below.

1.2.1. Fuel homologation sheet

It is this document which demonstrates the compatibility of the flask with the fuel to be evacuated, on the basis of the parameters defined in the package approval.

1.2.2. Maintenance inspection certificates of flasks and vehicles

The carrier is in possession of the certificates of the last periodic maintenance performed on the equipment in the maintenance workshops selected by each owner. During each cycle it records the condition of the utilized equipment in the relevant 'transport document' sheets.

1.2.3. Administrative formalities

The carrier establishes a check-list which makes it possible to verify the presence of the necessary official authorizations.

1.2.4. Flask operating documents at the reactor site

Flask handling at the reactor site is supervised by the carrier, who gives technical advice to the reactor operator. The carrier certifies through check-lists for each type of flask that the nuclear power plants use the procedures approved by the owners.

If an incident arises during flask handling, the carrier applies the instructions for intervention in the event of an emergency, as defined by the flask owner and approved by the consignee.

An incident report describes the actions taken.

1.2.5. Establishment of fuel loading plans

From the administrative procedures transmitted by the electrical utilities, the carrier establishes the loading plans which allow verification of fuel identification and of the position of the fuel in the flask cavity. A procedure describes the method employed by the carrier in order to eliminate any source of mistakes.
1.2.6. *Shipment supervision*

Telephone or automatic computer connections from the carrier's base office make it possible to monitor the position of the shipment at any moment. In the event of an accident, the freight contractor informs the carrier, who ensures the link with the competent authorities.

The carrier, in co-operation with the flask owner, gives the necessary practical instructions to the local security bodies and to those of the consignee country.

2. PRINCIPLES FOR THE ESTABLISHMENT OF A QUALITY ASSURANCE SYSTEM

The proposed principles, based on the practice described above, originate from the recommendations usually adopted in quality assurance guides applicable to product manufacturing. In the present situation Cogéma and NTL will transpose these recommendations to their respective service company activities.

The following associated principles could be considered as parameters of a quality assurance system plan applicable to a carrier.

2.1. Scope of service activities

The interfaces and the associated document links with the other activities will be clearly defined and the contractual clauses between the nuclear material carrier and the consignee will be made clear.

2.2. Organization

The roles of the intervening bodies will be detailed with regard to the carrier's responsibilities.

Staffing and training must be implemented within the different bodies, careful attention being paid to the specialities of their personnel.

2.3. Document control

The procedural support between the intervening bodies will be defined in such a manner as to demonstrate that the information circuit during the whole transport cycle is under control, that documented evidence is shown and that distribution of applicable documents is made amongst the different contracting parties.
2.4. Fuel data analysis

The fuel user must give to the carrier and to the consignee all data necessary to allow demonstration of conformity with the package approval and with storage or reprocessing requirements.

2.5. Shipment control

Provision must be made to demonstrate that the flasks and vehicles are in proper condition prior to and during shipment and that the equipment's location at any moment is monitored during the complete transport cycle.

2.6. Special process control

All safety and security related processes utilized for transport and loading operations must be qualified according to recognized standards. Procedures will detail the associated process performances.

2.7. Test and inspection control

All data sheets related to special safety or regulatory measurements performed during shipment and loading operations must be recorded and signed by the responsible organizations. They will be part of the quality assurance dossier established for each transport cycle.

2.8. Non-conformance control and corrective actions

A deviation from the applied procedures will lead to a non-conformance treatment if a physical defect in the equipment or a procedural discrepancy are detected. In any case, remedial actions will be taken to prevent repetition and these will be clearly written in the associated non-conformance report. The consignee approves each step of the non-conformance treatment.

2.9. Storage of records

Lifetime and temporary storage have to be defined according to the different types of documents dealing with the execution of a complete transport cycle, and to the particular body concerned. Storage conditions have to be settled and must comply with normal document preservation criteria.
2.10. Audits

The different parties involved in the execution of the transport cycle have to be audited periodically by the respective contracting organization. Audit questionnaires will be based on the commitments written in the relevant applicable documents. The organization responsible for quality assurance will establish the audit questionnaire and designate internally the competent auditor.

The present communication circuit (Fig. 1), as determined from the transport contracts which concern separately NTL and Cogéma, gives satisfactory results in the quality assurance management as described in this paper.

The carriers' constant aim is to improve the existing support documentation, especially from a security point of view.
THE LIMITS OF THE LEGAL SCOPE OF UNILATERAL APPROVAL.

Unilateral approval is simple in theory. However, its practical application raises one or two problems not only with regard to international transport, where its interpretation differs somewhat between countries, but also with regard to domestic transport, where certain countries would like to see it applied. An attempt is made here to identify the aspects which give rise to misunderstandings in this area and to propose a way of overcoming those misunderstandings, which create difficulties for the transport of radioactive materials.

LES LIMITES DU DOMAINE JURIDIQUE DES AGREMENTS UNILATERAUX.

L'agrément unilatéral est simple dans son principe. Son application dans la pratique pose cependant quelques problèmes, non seulement en transport international, où son interprétation diffère quelque peu entre les pays, mais aussi en transport domestique, où certains pays veulent le faire intervenir. On essaie ici, d'une part, de bien déterminer quels sont les points qui donnent jour à des malentendus dans ce domaine et, d'autre part, de proposer une voie pour essayer de résoudre ces malentendus, qui ne facilitent pas le transport des matières radioactives.

INTRODUCTION

Lorsque, dans l'édition de 1973 du n°6 de la Collection Sécurité, l'AIEA a introduit explicitement la notion d'agrément unilatéral, un grand pas a été fait pour institutionnaliser une facilité importante donnée au transport international des matières radioactives.

Rappelons que ceci veut dire que l'agrément d'un modèle de colis de matières radioactives, ou de matière radioactive sous forme spéciale, donné par l'autorité compétente du pays dit «d'origine» est valable dans un autre pays.

En ce qui concerne les colis, cet agrément unilatéral couvrait les modèles dits B(U) et une partie des colis de matières fissiles.

En 1985, l'agrément unilatéral des colis de matières fissiles, dont le domaine était d'ailleurs très limité, a été abandonné.

Cependant, la pratique de ces dernières années, qui a succédé à cette introduction, a montré que l'application de cette notion posait quelques problèmes, en particulier dans la définition de ses limites.
1. LES PRINCIPES

1.1. Au plan international

On sait que les recommandations de l’AIEA donnent leur substance technique aux textes normatifs, ce qui permet d’avoir une uniformité à l’intérieur des modes de transport et entre ces modes, au moins pour l’essentiel.

Ces textes sont:
A) Les réglementations et recommandations établies dans le cadre de conventions mondiales, telles les Instructions techniques de l’Organisation de l’aviation civile internationale (OACI) pour le transport aérien de matières dangereuses et le Code maritime international de transport des matières dangereuses de l’Organisation maritime internationale (OMI).
B) Les règles établies dans le cadre de conventions régionales telles que, par exemple, l’ADR, le RID et l’ADNR pour l’Europe et les pays voisins.

Le principe est donc que tout pays partie à ces conventions se doit d’admettre les agréments unilatéraux des autres pays parties, dans le domaine couvert par ces conventions, c’est-à-dire les agréments unilatéraux émis par l’autorité compétente du pays d’origine se référent explicitement à ces conventions internationales, dans le cadre du transport couvert, c’est-à-dire le transport international entre les pays concernés, suivant le mode concerné.

Une exception est constituée par les transports pour le compte de l’AIEA, pour lesquels le texte de la Collection Sécurité n° 6, une fois édité par l’Agence, est la règle applicable à la sûreté de ces transports.

1.2. Au plan national

La question n’a d’intérêt que dans le cas d’un agrément unilatéral, émis par une autorité compétente étrangère, d’un modèle de colis qui pourrait être utilisé lors d’un transport domestique. Dans la mesure où les recommandations de l’AIEA ne font l’objet d’aucune convention qui en prévoit l’application directe, et où les conventions existantes ne couvrent que le transport international, il est bien évident, en principe, que tout transport interne à un pays est exclu de l’application directe de l’agrément unilatéral étranger.

Nous allons voir cependant que la pratique est assez loin de s’aligner de manière parfaite sur les principes implicites que nous venons d’exposer.

2. LA PRATIQUE ACTUELLE

2.1. Dans le transport international

Certains pays comme les États-Unis appliquent directement les recommandations de l’AIEA au transport international à destination de leur pays. Il faut
cependant valider formellement un certificat étranger et en émettre une version américaine.

D'autres comme le Royaume-Uni appliquent la notion d'agrément unilatéral (B(U)) sans réserve.

La France étend depuis le 1er janvier 1985 les règles de l'OACI et de l'OMI, qui concernent l'emballage et l'étiquetage, aux parcours terrestres initiaux et terminaux.

Cette dernière règle revient à appliquer les recommandations de l'AIEA aux transports internationaux quels qu'ils soient, mais par l'intermédiaire de conventions modales. Il en est de même en République fédérale d'Allemagne.

Même pour les pays qui reconnaissent l'agrément unilatéral B(U), on voit déjà dans la considération de ces exemples que la pratique varie; elle va de l'application directe des règles de l'AIEA, sans validation formelle, à une application stricte des principes exposés plus haut. Ces variations peuvent, entre autres, paraître liées aux problèmes de langue; en effet, comment, dans un pays, prendre en compte un certificat rédigé dans une langue étrangère, pas forcément familière aux organismes de contrôle?

Par ailleurs, qu'appelle-t-on pays d'origine? Est-ce celui du modèle d'emballage, du modèle du colis (emballage + contenu) qui peut être différent, de l'expédition qui peut être encore différente? Le Radioactive Transport Study Group (RTSG) a déjà eu l'occasion de se pencher sur ces questions. La réponse n'est pas toujours évidente et mène presque inévitablement à l'intervention, dans l'agrément, des pays concernés pour le modèle de colis ou pour le contenu, c'est-à-dire à un caractère multilatéral partiel, comme cela avait déjà été souligné aux réunions antérieures du PATRAM. Mais ces différences d'appréciation sont finalement résolues dans la pratique du transport international, bien qu'elles occasionnent souvent des retards dans la réalisation de ce transport.

Par contre, c'est dans le domaine du transport domestique que l'agrément B(U) mène aux difficultés les plus grandes dans son application.

2.2. Dans le transport national

La plupart des pays appliquent avec rigueur les principes que nous avons exposés plus haut et limitent l'application du B(U) d'origine étrangère au transport international; ils gardent donc leur totale souveraineté sur la sûreté d'un transport strictement intérieur.

Ceci n'exclut pas que les textes réglementaires ou légaux appliqués puissent être analogues ou même strictement identiques en transport international et en transport national. Dans ce dernier cas, il est évident que ne se pose un problème que dans les cas où l'intervention d'une autorité compétente est nécessaire (agrément, approbation des modalités).

Mais l'identité technique des critères réglementaires ne retire rien à l'existence de domaines de souveraineté distincts pour les autorités compétentes nationales.
Dans le cas du transport domestique, n'intervient pas l'abandon de souveraineté en faveur de l'autorité du pays d'origine que suppose l'agrément unilatéral B(U) en transport international.

Certains pays au contraire semblent estimer que l'application de règles techniquement identiques par une autorité compétente étrangère conduit à un agrément automatique, même dans un transport domestique. C'est là un des points qui conduisent à certains malentendus.

L'une des questions est de savoir si la notion d'agrément unilatéral s'applique à toute l'existence d'un modèle de colis ou bien si elle est destinée à couvrir une expédition particulière. L'AIEA, en 1985, dans son paragraphe 114 a tranché semble-t-il dans la première direction: «Par agrément unilatéral, on entend l'agrément d'un modèle qui doit être donné seulement par l'autorité compétente du pays d'origine du modèle».

L'ancien paragraphe 142 de l'édition de 1973 («Par approbation unilatérale, on entend l'approbation donnée seulement par l'autorité compétente du pays d'origine») ne disait rien de la nature de cette origine et pouvait paraître ambigu. On a pu penser qu'une solution possible était de définir l'autorité compétente d'origine, apte à donner l'agrément unilatéral, comme celle de l'expédition.

Comme il s'agit de l'agrément d'un modèle, c'est alors limiter l'agrément unilatéral au cas où il y a identité entre le pays d'expédition et le pays de conception du colis, ce qui a l'avantage d'éliminer automatiquement le cas de l'utilisation d'un agrément unilatéral étranger en transport domestique, mais aussi l'inconvénient de restreindre considérablement son utilisation en transport international.

Une autre solution consisterait à bien spécifier que l'agrément unilatéral, dans la mesure où son but est de faciliter le transport international, est limité à ce dernier domaine. Toute autre interprétation pourrait être considérée comme une facilité accordée, bien plus au commerce international des emballages, qu'au transport international des colis de matières radioactives.

3. CONCLUSIONS

Il est certain que l'application des règles de l'AIEA dans un contexte très internationalisé, non seulement au niveau du transport proprement dit, mais aussi dans les domaines de la conception, de la fabrication, de l'utilisation, et même dans certains cas de la maintenance, oblige à des échanges à tous les niveaux. Plusieurs exemples en ont été donnés au cours des dernières années.

Le concept de B(U) facilite considérablement le transport international dans les cas simples. Il doit dans ce cas être appliqué avec rigueur, afin de ne pas être vidé de son sens et, surtout, garder un caractère de réciprocité qui est la condition même de son existence.

Cependant, le caractère international de certains des aspects que nous avons cités s'étend aussi aux transports intérieurs, la conception et la fabrication des emballages étant souvent effectués à l'étranger.
L'agrément unilatéral, en limitant l'intervention des autorités compétentes à une seule, celle du pays d'origine, vise implicitement le transport international. Dans le contexte généralisé dont nous avons parlé, elle nécessite davantage de précisions.

Est-il du ressort de l'AIEA elle-même d'apporter cette précision?
Il s'agit en effet de problèmes visant à la souveraineté des États, et qui sortent du domaine technique des recommandations.

Cependant, la solution de ces ambiguïtés est une des conditions de la survie effective de l'agrément unilatéral et il convient, à notre avis, que son application dans le domaine du transport international, dans le cadre des conventions existantes, soit aussi stricte que possible. Par ailleurs, il faut trouver des solutions dans les autres domaines d'application des règles de l'AIEA, qui permettent de sauvegarder l'unité de la sûreté et d'éviter des divergences paradoxales, par une vision claire des limites de l'agrément unilatéral. L'AIEA pourrait en tous cas, à une prochaine occasion, aborder ce problème, par exemple en enquêtant sur la manière dont il est considéré dans chacun des États Membres, et l'attitude de chacun de ces pays à propos des problèmes pratiques apparus à ce sujet. On pourrait ensuite confronter les différentes positions. La question est difficile mais son importance nous paraît la rendre digne d'être éclaircie.
CZECHOSLOVAK APPROACH TO LICENSING OF NUCLEAR MATERIALS TRANSPORT

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Abstract

CZECHOSLOVAK APPROACH TO LICENSING OF NUCLEAR MATERIALS TRANSPORT.

The paper discusses the Czechoslovak approach to the licensing and regulation of nuclear materials transport, especially the transport of spent nuclear fuel from Czechoslovakia to the USSR. Legal aspects of these transports are also discussed. The containers used for transport, as well as the operators' experience gained from the transportation of spent fuel, according to Czechoslovak Atomic Energy Commission provisions, are briefly described.

Czechoslovakia is an industrialized country without corresponding power resources of its own. The increasing demand for energy in Czechoslovakia during the 1970s led to an increase in the use of nuclear energy, which facilitated large scale construction of nuclear power plants, especially after the construction of fossil fuel power plants was terminated. Thus, all growth in power production is now being met by nuclear energy.

In accordance with the Agreement Concerning the Provisions of Assistance by the Union of Soviet Socialist Republics to the Czechoslovak Republic in Connection with the Development of Research into the Physics of the Atomic Nucleus and the Utilization of Atomic Energy for the Needs of the National Economy, Czechoslovakia started construction of nuclear power plants equipped with 440 MW PWR (light water cooled and moderated power reactors of WWER type) developed in the USSR. At the present time, there are six units equipped with WWER-440 reactors, each fully operational. The power generated by these nuclear power plants covers about 17% of total electricity production. Another six 440 MW reactors of the WWER type are under construction and the first nuclear power plant with four reactors, WWER-1000 MW, has been under construction in southern Bohemia since 1985. Suitable locations for other WWER-1000 MW power plants have already been chosen.

The large scale use of nuclear energy has raised the problem of the safe transportation of both fresh and spent fuel within Czechoslovak territory and the consequent need for its licensing and regulation. Czechoslovakia is a country with
an open nuclear fuel cycle and it owns neither reprocessing plants nor enrichment facilities. All fresh nuclear fuel assemblies are imported from the USSR and, according to the agreement between Czechoslovakia and the USSR, the nuclear spent fuel is, after a three- or five-year cooling period, transported back to the USSR. The safe transportation of nuclear materials has become an important part of the national programme for the use of nuclear materials and the generation of nuclear power.

There are two competent authorities approving and regulating this type of transport in Czechoslovakia. One is the Ministry of Transport, which monitors compliance with all international (e.g. Agreement on International Railroad Freight Traffic (SMGS) and Regulations Concerning the Carriage of Dangerous Goods by Rail (RID)) or national regulations related to transportation safety. The second is the Czechoslovak Atomic Energy Commission (CsAEC). The CsAEC is, by Act. No. 28/1984 concerning state supervision of the nuclear safety of nuclear facilities, authorized as the regulatory body for all aspects of nuclear safety. Thus it regulates the transport of nuclear materials from the point of view of nuclear safety and it also ensures that the provisions to assure safety in the course of transportation are met in practice.

The licensing procedure applied by the CsAEC to transportation has two components:

(1) Approval of the package design
(2) Approval of the shipment.

1) Approval of the package design. This first step complies fully with the IAEA Regulations for the Safe Transport of Radioactive Materials. An application for approval must be submitted to the CsAEC by the operator of the nuclear facility, the receiver or the shipper of nuclear materials. The content of the technical documentation supporting the applicant's request for approval differs according to package origin. If the container was manufactured abroad, then the following documents have to be presented to the competent authority (CsAEC):

(a) Package design approval certificate issued by the competent authority of the country of origin.
(b) Instructions for the use and maintenance of the container and of auxiliary devices.
(c) Description of actions carried out during the handling of the nuclear fuel.
(d) Assessment of accident possibilities during transport and handling.

If the container was produced in Czechoslovakia, the application for approval should comply with the requirements stated in paragraphs 705 and 711 of the 1985 Edition of the Regulations for the Safe Transport of Radioactive Material [1] and with the requirements listed in (b)–(d) above.

2) Approval of shipment. An application for shipment has to be submitted by the carrier and should include:
(i) Package design approval certificate  
(ii) Transport provisions and regulations  
(iii) Vehicle test assessments  
(iv) Transport route and timetable description  
(v) Accident emergency provisions  
(vi) Statement issued by the radiation protection authority.

All information included in the applications for approval is subject to careful checking. Also, the quality assurance programme for engineering of the packagings is analysed in detail. Furthermore, approval certificates issued by the CsAEC can stipulate relevant supplementary, temporary or permanent conditions related to operational requirements, radiation protection, etc.

All Czechoslovak transports of spent nuclear fuel are carried out by rail and thus must also meet all requirements stated in the Regulations for the Safe Transport of Spent Nuclear Fuel from Nuclear Power Plants in the Council for Mutual Economic Assistance (CMEA) Member Countries, Part 1 — Transport by Rail. These regulations, based on the regulations in IAEA Safety Series No. 6, were formulated under a programme for scientific and technical co-operation among CMEA member countries and are co-ordinated by Scientific and Technical Council No. 2 of the CMEA Permanent Commission. They were approved by the CMEA Permanent Commission on Co-operation in the Peaceful Uses of Atomic Energy. The CMEA Executive Committee certified these regulations in November 1977 and they were later adopted by the CMEA member countries as national regulatory standards.

Spent nuclear fuel transported under the requirements of these CMEA regulations must fully comply with the Technical Conditions for Spent Fuel Elements and Assemblies from Nuclear Power Plants of the Corresponding Types, which were also formulated by Scientific and Technical Council No. 2. The Technical Conditions contain provisions related to the condition of the spent fuel (e.g. burnup and residual thermal output), as well as the required accompanying documentation (e.g. tests assessment and radiation dose measurements).

At the present time, both the CMEA regulations and the Technical Conditions are being revised by Scientific and Technical Council No. 2 and new standards are being developed taking into consideration the 1985 Edition of IAEA Safety Series No. 6.

In the interim, CsAEC shipment approval certificates are being issued for a limited period of time regardless of the number of transports. Transports may be made according to specified conditions without separate application prior to each movement.

The CsAEC carries out inspections to ensure that all requirements specified in the approval certificates and the relevant provisions related to the transport have been satisfied. In particular, CsAEC inspectors examine:

(a) Compliance of the package design with the approval certificate;  
(b) Fulfillment of state authority approval provisions;
Based on these inspections, the CsAEC can levy a fine or withdraw the approval certificate.

The following describes experience gained during the transportation of spent fuel from Czechoslovak nuclear power plants to the USSR. There are two different types of nuclear spent fuel assemblies that are transported from Czechoslovakia to the USSR. One of them is spent fuel from the KS-150 reactor, which is now subject to decommissioning. A spent fuel transport container (T-15) was developed and manufactured at the Škoda works, Plzeň, Czechoslovakia. The container was tested according to the requirements in the IAEA Regulations and the CsAEC has issued the appropriate package design approval certificate, as well as a shipment approval certificate for a Type B(U) design. The certificate is valid until 1988. The container also complies fully with the requirements of the 1985 version of the IAEA Regulations. The certificate is of limited validity because periodic controls are requested by the national authority. The capacity of the container is 16 assemblies, with the assemblies being loaded in the vertical position, filled with nitrogen, checked and then moved to the horizontal position for transport.

The other container, of Soviet origin, was fabricated as a special container for the safe transport of spent fuel assemblies from the WWER-440 reactor. The container designation is TK-6 and its capacity is 30 spent fuel assemblies. The fuel elements are loaded and transported in a vertical position. Spent fuel assemblies during transportation can be cooled by water (wet method) or by nitrogen (dry method) according to the burnup and residual thermal output. The container is of Type B(U) for 'dry' transport and Type B(M) for 'wet' transport. The CsAEC shipment approval certificates for both types of shipments have already been issued.

Each consignment should be made up of eight TK-6 containers or two T-15 containers and the transport should be made under exclusive-use conditions. During loading, transport preparation and transportation, all provisions stated in the package design and shipment approval certificates issued by the CsAEC have to be observed. At the present time there are ten conditions included in the shipment approval certificate. The most important ones are those provisions that apply to nuclear fuel integrity, to the maximum value of burnup of each separate assembly, to the total residual thermal output in each container, to the full operability of the stowage and handling facilities and to the leaktightness of the closed containers, as well as others relating to radiation safety.

Compliance with the provisions concerning nitrogen purity and oxygen content in the container during transport preparation and transportation, the leaktightness test of the closed container using nitrogen overpressure and tests of the integrity of the fuel assembly cladding using activity measurements of $^{131}$I in dry residue are most important matters for both nuclear power plant operators and regulatory bodies. This compliance with all provisions is verified by CsAEC inspections carried out by
inspectors prior to each transportation. The inspections are also carried out in cooperation with the state radiation protection authority.

Experience gained so far, and the results of the spent fuel shipments already carried out, show that if all package design and shipment approval certificate provisions and all the regulations are observed, then safety of spent fuel transport is fully assured.

REFERENCE

TRENDS IN FEDERAL, STATE AND LOCAL ACTIVITIES RELATIVE TO TRANSPORTATION OF RADIOACTIVE MATERIALS IN THE UNITED STATES OF AMERICA

Old problems and new solutions

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Abstract

TRENDS IN FEDERAL, STATE AND LOCAL ACTIVITIES RELATIVE TO TRANSPORTATION OF RADIOACTIVE MATERIALS IN THE UNITED STATES OF AMERICA: OLD PROBLEMS AND NEW SOLUTIONS.

The paper describes the current legal and regulatory structure for transporting nuclear fuel cycle materials in the United States of America, particularly as this structure applies to irradiated reactor fuel. The respective responsibilities of the cognizant federal agencies, states and localities are discussed. Recent decisions affecting the division of authority among governmental bodies are examined to illustrate emerging trends in the resolution of institutional issues concerning indemnification, emergency response, routing, choice of transportation modes and imposition of fees. The crucial role of the United States Department of Energy in resolving these and other issues is discussed, specifically with respect to the design of new casks for transporting irradiated reactor fuel to a repository. Principles are suggested to help determine the appropriateness of suggested solutions to institutional issues.

1. INTRODUCTION

Those responsible for worldwide transportation of radioactive materials over the past forty years -- in private industry and the government -- have done an outstanding job in assuring that the public and transport workers have been protected against the risks inherent in this activity. Notwithstanding this excellent record and the many studies by reputable scientists showing that the risks of transporting radioactive materials are extremely low, particularly compared with the risks of transporting other essential commodities that are classified as hazardous, this activity continues to attract an extraordinary degree of attention by the news media, elected representatives, regulatory authorities and members of the public. Moreover, international shipments as well as those
made within the U.S.A. continue to be threatened with disruption as a result of activities by state and local officials.

In the U.S.A. there is intensified interest in the prospect of a much larger number of shipments of irradiated reactor fuel (spent fuel or fuel) associated with the efforts of the U.S. Department of Energy (DOE) to develop a geologic repository for storing such fuel. The DOE currently estimates that some 70,000 metric tons of spent fuel will be stored in the first repository. Of course, shipments to the repository would take place over a period of about 30 years and the number of such shipments is minuscule compared to the approximately 100 million shipments of all hazardous materials that take place every year in the U.S.A. Moreover, the number of shipments that are likely to be required in order to transport spent fuel to a repository may be much lower than has been forecast in the past due, for example, to the redesign of casks to accommodate fuel that has been cooled for a minimum of 5 years.

It was the recognition of the significance of 'institutional', e.g. regulatory, legal and public acceptance, issues to the safe, reliable and economic transportation of nuclear fuel cycle materials that led to the formation in the U.S.A. of the Electric Utility Companies' Nuclear Transportation Group (Group), consisting of 37 utilities that are constructing or operating 100 power reactors. The Group's mission is (i) to participate in regulatory, judicial and related activities where necessary to ensure that electric utilities' interests are adequately presented to administrative agencies and the courts and (ii) to promote the successful implementation of the transportation-related provisions of the Nuclear Waste Policy Act of 1982 (NWPA). The conclusions and perspective of the authors of this paper reflect their experience in directing and carrying out the Group's activities.

2. CURRENT LEGAL AND REGULATORY STRUCTURE FOR REGULATING TRANSPORTATION IN THE U.S.A.

Acting primarily pursuant to its authority under the Commerce Clause of the U.S. Constitution, the U.S. Congress has enacted several laws which create a comprehensive federal system for the regulation of transportation of radioactive materials in the U.S.A. Under the Supremacy Clause of the U.S. Constitution, these laws and their implementing regulations enacted by the cognizant federal agencies, particularly the U.S. Nuclear Regulatory Commission (NRC) and the U.S. Department of Transportation (DOT), preempt inconsistent or
conflicting state and local regulations. In addition to establishing these two regulatory agencies, Congress assigned responsibilities to the DOE and to the Federal Emergency Management Agency to carry out essential functions associated with transportation of radioactive materials. All these agencies' activities are thus part of the inclusive federal program for addressing the safe transportation of radioactive materials.

The authority of state and local governments to regulate the transportation of radioactive materials stems from their inherent police powers to protect the health and safety of their citizens and from the provisions of the U.S. Constitution which reserve to the states those powers not delegated to the federal government. However, state and local regulatory authority is defined largely by the scope of exercise of federal regulatory authority, because states and localities may not enforce laws or regulations that restrict the flow of international or interstate commerce or are preempted by federal laws or regulations.

The tension inherent in the concurrent exercise of federal, state and local authority over transportation of radioactive materials has been present since the development of nuclear energy for peaceful purposes. In a federal republic such as the U.S.A., with its relatively large land mass and strong tradition of decentralized government particularly as this applies to highway transportation, continuing problems exist in accommodating the need for centralized control over matters of national interest and demands for regional autonomy.

In recent years, however, the interaction of federal, state and local authority has become more complex. The byproducts of generating electricity with nuclear power plants must eventually be transported to permanent storage locations. There are a relatively few committed opponents of nuclear power who are prepared to use every tactic available to them to prevent spent fuel from moving through their communities. For such persons, the goal is an absolute ban on transportation through their jurisdictions. However, the majority of state and local governmental officials and members of the public are willing to accept a reasonable accommodation of the national interest and more parochial concerns. In the next section of this paper we describe recent developments in the continuing search for a proper balance among the competing interests.

3. EMERGING TRENDS IN FEDERAL, STATE AND LOCAL ACTIVITIES

The nature of intergovernmental relationships in the U.S.A. has led to the identification of several key problems relative
to transportation of radioactive materials. These include indemnification against risks, emergency response, routing and choice of modes of transportation and imposition of fees on movements of radioactive materials.

3.1 Indemnity Coverage for Transportation Accidents

Some have questioned the adequacy of insurance and indemnity coverage for an accident occurring during transportation of spent fuel. The coincidence of the implementation of the NWPA and the expiration of the Price-Anderson indemnity legislation in the U.S.A. in 1987 has led to renewed examination of this subject. The attention of commentators has been focused on the following issues:

- Are evacuation costs covered by the indemnity requirement if there were no release of radioactivity?
- Will indemnity coverage be available if spent fuel were diverted by terrorists who sabotage the cask and cause a release of radioactivity?
- Will states and localities be able to recover their costs in responding to an event occurring during transportation of spent fuel?
- Will rail carriers be protected against loss due to blockage of their tracks if an accident occurred while the railroads were carrying a cask?

It is too early to know how these questions will be resolved in the U.S.A. since Congress continues its consideration of proposals to amend the Price-Anderson legislation. In any event, it is important that state and local authorities have a clear understanding of available indemnity coverage.

3.2 Emergency Response

Responsibility for response to emergencies involving transportation of spent fuel is divided among carriers, shippers, state and local governments and the federal government. However, emergency preparedness is innately a governmental responsibility and the initial response to an emergency necessarily must come from state and local personnel, such as police and firefighters. Although the initial response must be by states and localities, the nature of this response is very similar to what would be called for if the accident involved other hazardous materials. Moreover, the federal government has provided guidance and training to state and
local officials and has also established a powerful network to assist states and localities, on request, in the event of a transportation emergency involving radioactive materials.

The possibility that shipments of spent fuel will impose substantial burdens for emergency response remains one of concern to many state and local officials. However, recent judicial decisions in the U.S.A. reinforce the principle that the interlocking nature of the federal regulatory scheme prevents state or local efforts to impose onerous requirements on shippers or carriers related to emergency response.

3.3. Routing and Choice of Modes

Questions concerning choice of routes and modes of transportation for spent fuel shipments are among the most enduring of the institutional issues since these choices directly affect the proximity to particular communities of movements of large amounts of radioactivity. There continues to be controversy over the extent to which detailed safety or environmental analyses must be performed before a particular route or transportation mode is chosen. One of the most important proceedings currently pending before the DOT involves New York City's assertion that it should be allowed to enforce a ban on highway transportation of spent fuel through the city in favor of a combination of barge and highway transportation through neighboring jurisdictions. The DOT has preliminarily ruled against the city, but the outcome will likely need to be resolved in the federal courts.

3.4 Imposition of Fees on Transportation by States and Localities

Another important proceeding that has just been decided by the U.S. Department of Transportation concerns the validity of a $1000 fee assessed by the State of Illinois per cask of spent fuel shipped. On June 4, 1986 the U.S. Department of Transportation ruled that the Illinois fee is not inconsistent with one of the governing Federal statutes in the U.S.A., the Hazardous Materials Transportation Act. According to the Department of Transportation, the fee does not effectively cause rerouting, restrictions, or delays in spent fuel shipments by highway or rail. The Department of Transportation distinguished its latest ruling from an earlier one by the Department, in which a fee imposed by the State of Vermont had been determined to be invalid, on the grounds that the Vermont fee was an integral part of an invalid state permit system. The Department of Transportation also concluded that since preparedness for transportation emergencies, including those involving spent fuel shipments, is a shared responsibility of
federal, state and local governments. Illinois was within its legal rights in requiring payment of a fee to cover its costs of providing for emergency preparedness.

No doubt this decision by the Department of Transportation needs to be carefully examined to determine whether it is likely that fees can legally be imposed on spent fuel shipments to cover a variety of costs allegedly incurred by other jurisdictions in the U.S.A. In this connection, it is important to note that the Department of Transportation stated that it was not prepared to accept the proposition that fees imposed by other governmental bodies would also be found to be valid. The Department of Transportation called attention to the fact that the present proceeding involved a fee which is part of a state program for emergency preparedness, the different role of municipalities in nuclear safety preparedness and the possibility that other fee programs might, in effect, be illegal bans. The Department of Transportation also noted that fee provisions may be illegal under other federal statutes or under the provisions of the Constitution of the U.S.A. Thus, although this decision will have an important bearing on intergovernmental authority to assess fees for transporting spent fuel, the long term effects of this decision by the Department of Transportation remain to be determined.

4. DOE's IMPLEMENTATION OF THE NWPA

DOE's plans to implement the NWPA will have the greatest long-term influence on the resolution of institutional issues affecting transportation. The NWPA established the responsibility of the DOE to dispose of spent fuel and associated high level radioactive waste from commercial nuclear power plants beginning in 1998. The success of DOE in accomplishing its statutory mandate is dependent upon the siting and construction of a suitable repository and the establishment of a transportation system that can safely, efficiently and economically transport these materials from power reactors to the repository. The Group is vitally interested in ensuring that a suitable transportation system is planned and built.

The DOE has issued a Transportation Business Plan that delineates DOE's program for designing, acquiring and operating the transportation system. The DOE's program for addressing institutional issues is addressed in its Transportation Institutional Plan. One of the principal issues addressed in the Transportation Business and Institutional Plans is the redesign of the casks that will be needed to transport the relatively large volume of spent fuel that is accumulating at power reactor sites. The new casks will be designed for a
variety of reasons, including development of dual purpose casks and in order to increase the amount of fuel transported per shipment. This does not mean that the use of existing casks is inappropriate. In many studies by the responsible regulatory agencies, it has been confirmed that the radiological risk of transportation in casks designed to the current international standards poses an extremely low level of risk.

Nevertheless, criticism continues to be voiced by some special interest groups that the current international safety standards are inadequate and that full-scale proof testing of casks is necessary to dispel public doubts. The NRC is currently conducting a research program to evaluate the international standards against real-world accident conditions. If there are any such accident conditions that could cause a release from the cask the probability of such an event occurring will be evaluated and a determination made of the consequences of the release for those events that are of concern.

It is highly desirable to utilize casks as part of a program to describe to the public how spent fuel can be safely transported. Nevertheless, the desirability of additional full-scale proof testing by the DOE is questionable. In principle it is possible to devise a test that will satisfy all but the most committed opponents of nuclear transportation that the casks are invulnerable. In practice, and considering the tendency by some special interest groups to question the results of the tests conducted by the Central Electricity Generating Board in the U.K. and by Sandia National Laboratories in the U.S.A., it is by no means clear that such a result is likely. In the absence of a clear definition of the objectives of the tests and evidence that such tests would be cost-effective in addressing the public's perception of the safety of the casks, a case has not been made for more full-scale proof testing of casks.

5. CONCLUSION

The amount of radioactive materials being shipped, particularly those associated with the 'back-end' of the nuclear fuel cycle, will increase substantially over ensuing decades. Public interest in this subject is growing, particularly along the likely routes of large-scale shipments of spent fuel. The institutional issues that must be resolved are reasonably well identified. What is called for is a sustained, coordinated effort to fashion widely acceptable solutions to these issues. From the perspective of the Group, these solutions should conform to the following basic principles:
• Transportation regulatory requirements must have a sound technical justification.

• Analyses of the risks of and need for transportation should normally be performed generically rather than on a shipment-by-shipment basis.

• There must be uniform, preemptive federal standards for transportation of nuclear fuel cycle materials, subject to appropriate state and local participation within the framework of applicable law.

• Shippers must be able to use any or all available modes of transportation in order to be able to choose the most advantageous and economical one as individual circumstances dictate.

• Carriers must not be allowed to impose their own requirements upon nuclear fuel cycle transportation that differ from federal standards.

• States should be integrated into efforts to enforce the federal regulations, particularly those relating to highway safety. Regional cooperative efforts by states should be encouraged. State inspection efforts should not be duplicative and should be reasonably related to the schedules and other operational features of shipments.

• The assessment by states and localities of fees for transporting spent fuel is objectionable for legal and practical reasons.

• The safety and environmental risks of transportation are a relatively minor factor in selecting a repository for storing spent fuel. However, the economic costs of transportation are a substantial part of the total cost of disposing of spent fuel and should be treated accordingly.
STANDARD PROBLEM EXERCISE TO VALIDATE CRITICALITY CODES FOR LARGE ARRAYS OF PACKAGES OF FISSILE MATERIALS

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Abstract

STANDARD PROBLEM EXERCISE TO VALIDATE CRITICALITY CODES FOR LARGE ARRAYS OF PACKAGES OF FISSILE MATERIALS.

A study has been conducted by a Working Group under the auspices of the Organisation for Economic Co-operation and Development, Committee on the Safety of Nuclear Installations that examined computational methods used to compute $k_{eff}$ for large $\geq 5^3$ arrays of fissile material (in which each unit is a substantial fraction of a critical mass). Five fissile materials that might typically be transported were used in the study. The 'packages' used for this exercise were simplified to allow studies unperturbed by the variety of structural materials which would exist in an actual package. The only material present other than the fissile material was a variation in the moderator (water) surrounding the fissile material. Consistent results were obtained from calculations using several computational methods. That is, when the bias demonstrated by each method for actual critical experiments was used to 'correct' the results obtained for systems for which there were no experimental data, there was good agreement between the methods. Two major areas of concern were raised by this exercise. First, the lack of experimental data for arrays with size greater than $5^3$ limits validation for large systems. Second, there is a distinct possibility that the commingling of two shipments of unlike units could result in a reduction in safety margins. Additional experiments and calculations will be required to satisfactorily resolve the remaining questions regarding the safe transport of large arrays of fissile materials.

1. INTRODUCTION

This report presents the results obtained by the working group for $k_{\text{eff}}$ calculations for large finite and infinite arrays of fissile material units. The current work consists of five different fissile material forms that have several thicknesses of water between the fissile units and a thick water reflector around the outside of the finite arrays. These were intended to be hypothetical transport packages with the importance of inter-spersed moderation being the parameter studied. This work was carried out in response to a request by an International Atomic Energy Agency (IAEA) working group that was reviewing the criticality safety aspects of the Agency's regulations for the safe transport of radioactive materials.

In an effort to study and validate only one parameter at a time, the effects of other packaging and structural materials are not considered in this report.

The initial effort of the current exercise was similar to that of a previous standard problem exercise reported in OECD-CSNI Report No. 71; that is, experimentally critical systems were chosen to be used in validating the computational procedures for systems with physical and material properties similar to that which one might find in Fissile Class II packages.

Owing to the lack of experiments with large (>5") arrays of fissile materials (in which each unit is a substantial fraction of a critical mass), it is not possible to validate the calculations as a function of array size. This leaves, without an easy solution, the problem of needing to extrapolate from the data that are available. There are two difficulties which cause concern about such extrapolations. First, for several experiments that have been performed in which the array size was varied, the bias in the calculations has been a reasonably strong function of the array size. Second, with a finite number of neutrons per generation for Monte Carlo calculations of very large arrays, a question must be raised regarding the adequacy of sampling. These two concerns were addressed by this study.

Another problem addressed by this study involved the effect of commingling Fissile Class II packages with different fissile material contents. The current IAEA regulations have as one of their criteria the condition that five times the number of packages allowed in a shipment must be subcritical. This provides a margin of safety in case two or more shipments are placed together by a transportation organization. The question that needed study was whether the safety margins thought to be in place would be reduced if the multiple shipments which were placed together consisted of different fissile materials. Hypothetical arrangements for commingling of different types of fissile materials were studied in an effort to understand whether commingling would reduce the safety margins.
Even though our study allowed us to make a recommendation regarding some of the above-mentioned problems, it became clear that additional critical experiments are needed to satisfactorily provide the complete guidance needed to assure the safe and economical transport of Fissile Class II packages.

2. OBJECTIVE OF THE EXERCISE

The objective of this exercise was to establish the validity of criticality safety computational methods for computing arrays of fissile transport packages which are defined as Fissile Class II under the IAEA Regulations for the Safe Transport of Radioactive Materials. This initial work has been directed toward examining the validity of criticality computer programs under conditions of varying amounts of hydrogenous moderators between the fissile units. Future work will need to address other issues such as the effects of structural material neutron poisons, other package components, along with additional work on the intermingling of unlike packages.

The problems chosen consist of experiments with materials that might typically be placed in a Fissile Class II package. Specifically, the problems involve:

1. Highly enriched uranium metal.
2. Highly enriched uranyl nitrate.
3. 5% enriched uranium oxide.
4. 5% enriched uranium oxide with H/U = 20.
5. Plutonium oxide.

Calculations were also made on experimentally critical arrays of units resembling the hypothetical packages of problems 1-4 with the objective of providing a basis for judging the validity of the computer program and the associated cross-section libraries. No appropriate experimental data were found to compare with problem 5.

The problems chosen for study consisted of $8 \times 8 \times 8$ ($8^3$) arrays of fissile material units with varying amounts of water as interspersed moderation and a water reflector. The $8^3$ array size was chosen because this number of units (512) was thought to be a reasonably "large" number of packages which would be used in Fissile Class II shipments.

In addition to the study of $8^3$ arrays for various fissile materials, another study was made to determine whether adequate neutron sampling was occurring for large arrays. Calculations were made by the participants for arrays of size $4^3$, $12^3$, and $16^3$. The results of these calculations were compared for consistency, and for evidence of inadequate sampling, with the results obtained for the $8^3$ calculations.
A study was also made to evaluate the effect on $k_{\text{eff}}$ which would result from commingling two different fissile material packages. The object was to determine if the $k_{\text{eff}}$ resulting from commingling two different fissile material packages would be greater than the $k_{\text{eff}}$ for an array of the same size which consisted of either of the two packages alone.

3. PROBLEM DESCRIPTIONS

Simulated Transport Package Problems 1-5 consisted of 8x8x8 ($8^3$) arrays of fissile material surrounded by a thick water reflector at the array boundary. The fissile material was spherical in shape and was surrounded by a spherical shell of water. The following is a short summary of the problem specifications.

<table>
<thead>
<tr>
<th>Problem</th>
<th>Material</th>
<th>Density</th>
<th>Radius of fissile material</th>
<th>Thicknesses of water shell</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>$^{235}\text{U metal}$</td>
<td>18.76 g/cm$^3$</td>
<td>6.262 cm</td>
<td>a) 0.0 cm b) 2.54 cm c) 10.16 cm</td>
</tr>
<tr>
<td>2</td>
<td>$^{235}\text{U nitrate solution}$</td>
<td>1.07892 g/cm$^3$</td>
<td>12.0 cm</td>
<td>a) 0.0 cm b) 2.0 cm c) 2.8 cm</td>
</tr>
<tr>
<td>3</td>
<td>$^{235}\text{U}_2\text{O}_2$ powder</td>
<td>5.0 g/cm$^3$</td>
<td>20.5 cm</td>
<td>a) 0.0 cm b) 2.2 cm c) 4.0 cm</td>
</tr>
<tr>
<td>4</td>
<td>$^{235}\text{U}_2\text{O}_2$ powder</td>
<td>2.2 g/cm$^3$</td>
<td>14 cm</td>
<td>a) 0.0 cm b) 4.0 cm</td>
</tr>
<tr>
<td>5</td>
<td>$^{239}\text{PuO}_2$ ($25%$ $^{240}\text{Pu}$, $12%$ $^{241}\text{Pu}$, $3%$ $^{242}\text{Pu}$)</td>
<td>5.0 g/cm$^3$</td>
<td>10.7 cm</td>
<td>a) 0.0 cm b) 2.2 cm c) 6.0 cm</td>
</tr>
</tbody>
</table>

Experimentally critical systems which were similar to the arrays of simulated transport packages are given below. Problems I-IV follow. [Note that Problem 3 had an experimental $k_{\text{eff}} = 1.014$ with no gap (i.e., criticality occurred with a small gap remaining between the two halves of the system).]
### I. Material

<table>
<thead>
<tr>
<th>Density</th>
<th>U(93.2% $^{235}$U) metal cylinders</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>$18.73 \text{ g/cm}^3$</td>
</tr>
<tr>
<td>Radius of fissile material</td>
<td>$5.753 \text{ cm}$</td>
</tr>
<tr>
<td>Height</td>
<td>$10.765 \text{ cm}$</td>
</tr>
<tr>
<td>Moderator</td>
<td>Plexiglas box surrounding each cylinder; outside dimensions $21.4\times21.4\times20.7 \text{ cm highwall}$</td>
</tr>
</tbody>
</table>

**a)** Array size: $2\times2\times2$
- Paraffin reflector thickness: $0.0 \text{ cm}$

**b)** Array size: $2\times2\times2$
- Paraffin reflector thickness: $15.2 \text{ cm}$

**c)** Array size: $3\times3\times3$
- Paraffin reflector thickness: $0.0 \text{ cm}$

### II. Material

<table>
<thead>
<tr>
<th>Density</th>
<th>$\text{H/}{^{235}\text{U}}$ nitrate solution cylinders</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>$1.083 \text{ g/cm}^3$</td>
</tr>
<tr>
<td>Radius of fissile material</td>
<td>$19.04 \text{ cm}$</td>
</tr>
<tr>
<td>Height</td>
<td>$17.77 \text{ cm}$</td>
</tr>
<tr>
<td>Moderator</td>
<td>Plexiglas cylinder with wall and end thicknesses of $0.64 \text{ cm}$</td>
</tr>
</tbody>
</table>

**Array size**
- $3\times3\times3$

### III. Material

<table>
<thead>
<tr>
<th>Density</th>
<th>$\text{U}(4.46%\ 235\text{U})_3\text{O}_8\cdot\text{H}_2\text{O}$</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>$4.47 \text{ g/cm}^3$</td>
</tr>
<tr>
<td>$\text{H/}{^{235}\text{U}}$</td>
<td>$0.77$</td>
</tr>
<tr>
<td>Cube fissile material</td>
<td>$15.3 \text{ cm on each side}$</td>
</tr>
</tbody>
</table>
| Moderator cube | $0.923 \text{ cm thick plexiglas surrounding fissile material}$

**Array size**
- $5\times4\times5$
- Reflector: Thick plexiglas reflector on all sides
- Experimental $k_{\text{eff}}$: $1.014$

### IV. Material

<table>
<thead>
<tr>
<th>Density</th>
<th>$\text{U}(4.89%\ 235\text{U})<em>3\text{O}<em>8\cdot(C</em>{17}{^{235}\text{U}}\text{H}</em>{35}\text{CO}<em>2)</em>{3}\text{C}_{3}\text{H}_5$</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>$9.86$</td>
</tr>
<tr>
<td>$\text{H/}{^{235}\text{U}}$</td>
<td>$40.64x40.64x58.17$</td>
</tr>
<tr>
<td>Cube fissile material</td>
<td>$40.64x40.64x58.17$</td>
</tr>
</tbody>
</table>

**Array size**
- $1\times1\times1$
- Reflector cube: Thick $\text{H}_2\text{O}$ or paraffin reflector on all sides
<table>
<thead>
<tr>
<th>Computer Program Cross Sections</th>
<th>SPHERE</th>
<th>MORET</th>
<th>MORET</th>
<th>KENO</th>
<th>KENO-IV</th>
<th>MORSE-X</th>
<th>KENO</th>
<th>GAM- THERMOS</th>
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</thead>
<tbody>
<tr>
<td>Case No.</td>
<td>EIR</td>
<td>CEA</td>
<td>CEA</td>
<td>ORNL</td>
<td>GRS</td>
<td>PTB</td>
<td>ENEA</td>
<td>ENEA</td>
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<tr>
<td></td>
<td>Switzerland</td>
<td>France</td>
<td>France</td>
<td>U.S.</td>
<td>FRG</td>
<td>Italy</td>
<td>Italy</td>
<td></td>
</tr>
<tr>
<td>La</td>
<td>0.9269</td>
<td>0.984 ± 0.005</td>
<td>0.985 ± 0.005</td>
<td>1.001 ± 0.004</td>
<td>0.995 ± 0.005</td>
<td>1.011 ± 0.007</td>
<td>0.997 ± 0.004</td>
<td>0.995 ± 0.004</td>
</tr>
<tr>
<td>Lb</td>
<td>1.0350</td>
<td>0.990 ± 0.009</td>
<td>0.998 ± 0.009</td>
<td>1.005 ± 0.005</td>
<td>1.000 ± 0.005</td>
<td>1.003 ± 0.013</td>
<td>0.999 ± 0.005</td>
<td>1.004 ± 0.004</td>
</tr>
<tr>
<td>Lc</td>
<td>0.9636</td>
<td>0.983 ± 0.005</td>
<td>0.975 ± 0.005</td>
<td>0.993 ± 0.005</td>
<td>0.993 ± 0.005</td>
<td>0.989 ± 0.004</td>
<td>1.007 ± 0.005</td>
<td></td>
</tr>
<tr>
<td>II</td>
<td>0.8799</td>
<td>1.005 ± 0.005</td>
<td>1.004 ± 0.005</td>
<td>1.002 ± 0.005</td>
<td>1.005 ± 0.006</td>
<td></td>
<td>1.019 ± 0.005</td>
<td></td>
</tr>
<tr>
<td>III</td>
<td>1.053</td>
<td>1.029 ± 0.006</td>
<td>1.002 ± 0.005</td>
<td>1.024 ± 0.005*</td>
<td>1.012 ± 0.005</td>
<td>1.009 ± 0.004</td>
<td></td>
<td></td>
</tr>
<tr>
<td>IV</td>
<td>1.0056</td>
<td>0.996 ± 0.005</td>
<td>0.986 ± 0.003</td>
<td>0.991 ± 0.005*</td>
<td>0.999 ± 0.009</td>
<td>0.987 ± 0.004</td>
<td>0.990 ± 0.006</td>
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<tr>
<td>Computer Program Cross Sections</td>
<td>KENO</td>
<td>MONK</td>
<td>KENO</td>
<td>KENO</td>
<td>KENO-IV</td>
<td>KENO Jr</td>
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<tr>
<td>Case No.</td>
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<td>ENEA</td>
<td>KENO</td>
<td>MONK</td>
<td>UKNDL</td>
<td>MGCL</td>
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<td></td>
<td>Italy</td>
<td>Italy</td>
<td>Sweden</td>
<td>Belgium</td>
<td>U.K.</td>
<td>Japan</td>
<td>Japan</td>
<td></td>
</tr>
<tr>
<td>La</td>
<td>0.983 ± 0.005</td>
<td>1.016 ± 0.007</td>
<td>1.000 ± 0.005</td>
<td>1.000 ± 0.005</td>
<td>0.992 ± 0.007</td>
<td>1.004 ± 0.003</td>
<td>0.979 ± 0.002</td>
<td></td>
</tr>
<tr>
<td>Lb</td>
<td>0.985 ± 0.005</td>
<td>1.019 ± 0.007</td>
<td>0.995 ± 0.006</td>
<td>1.006 ± 0.008</td>
<td>0.995 ± 0.008</td>
<td>1.011 ± 0.002</td>
<td>1.002 ± 0.002</td>
<td></td>
</tr>
<tr>
<td>Lc</td>
<td>0.987 ± 0.005</td>
<td>1.010 ± 0.007</td>
<td>0.995 ± 0.004</td>
<td>1.009 ± 0.004</td>
<td>0.989 ± 0.007</td>
<td>1.002 ± 0.003</td>
<td>0.990 ± 0.002</td>
<td></td>
</tr>
<tr>
<td>II</td>
<td>0.984 ± 0.0006</td>
<td>1.022 ± 0.008</td>
<td>0.997 ± 0.005</td>
<td>1.010 ± 0.003</td>
<td>1.002 ± 0.007</td>
<td>0.974 ± 0.003</td>
<td>1.010 ± 0.003</td>
<td></td>
</tr>
<tr>
<td>III</td>
<td>1.021 ± 0.005</td>
<td>0.999 ± 0.008</td>
<td>0.999 ± 0.004</td>
<td>1.033 ± 0.006</td>
<td>1.010 ± 0.006</td>
<td>1.033 ± 0.002**</td>
<td></td>
<td></td>
</tr>
<tr>
<td>IV</td>
<td>0.970 ± 0.005</td>
<td>1.015 ± 0.008</td>
<td>0.991 ± 0.006</td>
<td>0.972 ± 0.008</td>
<td>1.016 ± 0.007</td>
<td>0.977 ± 0.002</td>
<td>0.984 ± 0.003</td>
<td></td>
</tr>
</tbody>
</table>

*GAMTEC-II cross sections

**Multi-KENO computer program
4. RESULTS

4.1 Experimentally Critical Array Calculations

The experiments (Problems I-IV) chosen for this exercise were selected to be as similar as possible to the hypothetical class II fissile packages that were to be studied. Data for arrays as large as 8x8x8 (8") were not available; hence, the data chosen were not as complete as we would have desired. We were unable to obtain experimental data for arrays of plutonium oxide.

Table I presents the results obtained for the experimentally critical systems studied. The results can be observed to be in reasonably good agreement with expected values with the exception of the EIR (Switzerland) results. The EIR calculations were performed with a new computational procedure, which cannot yet completely treat the geometries present in these experimental systems. The rather good agreement between experimental and calculated results obtained with the majority of the methods provides considerable confidence that the methods will perform satisfactorily for calculations on the "hypothetical problems" chosen.

4.2 8\(^3\) Arrays of Simulated Fissile Class II Packages

The multiplication factor for 8x8x8 arrays of simulated Fissile Class II packages were each computed with three values of water moderation between the units (except for Problem 4 where only two values were computed). The water moderation was chosen to represent no moderation, optimal moderation and over-moderation. The simple moderation model chosen was recognized not to be a complete study of moderation effects. In actual analysis of such systems it is necessary to establish optimum moderation; therefore, these studies cover only a portion of the range of application which would be needed to establish the safety of a package.

The results in almost every case are in agreement with the results obtained for the calculations of the experimentally critical systems. That is, if a negative/positive bias (i.e., the computed \(k_{\text{eff}}\) of the experimentally measured system is lower/higher than the experimental \(k_{\text{eff}}\)) is observed, then the results for the simulated package are less/greater than the expected values based on a consensus of the results from all participants for the simulated package calculations.

The EIR results, which were not in good agreement with the experimentally measured systems, produced excellent results for the 8\(^3\) arrays. The method employed by EIR was demonstrated to adequately compute \(k_{\text{eff}}\) for systems which meet the geometrical qualifications of the method.
4.3 Larger Array Calculations

In studying the results received for the $8^3$ array problems, the problem coordinators became concerned about the convergence pattern of several of the Monte Carlo calculations. The question was raised as to whether there was a risk associated with computing an array with 500 to 5000 units while using only 100 to 500 neutrons per batch (or stage). Since in this situation it is quite possible that not every unit would be sampled during the processing of one batch, the concern was that an incorrect value of $k_{\text{eff}}$ might be calculated.

To study this effect it was agreed that we would compute arrays of $12^3$ and $16^3$ to observe if this produced evidence of inadequate sampling. Based on the results obtained, several participants decided to go the other direction in array size and computed a $4^3$ array.

Based on the collective results observed from these calculations, it was concluded that no evidence was observed which indicated that one would have problems with inadequate sampling in arrays of identical units. Based on some limited research by one of the participants, it appears that, if there is a problem, it would more likely occur in arrays of size $4^3$ to $8^3$. Since the use of 300-500 neutrons per batch would assure reasonable sampling in these systems, it is suggested that there is no cause for concern regarding sampling in uniform arrays.

4.4 Mixed Arrays

One of the questions raised by the IAEA's request for a study was whether the commingling of two unlike arrays would produce a $k_{\text{eff}}$ which would be higher than two like arrays.

This concern is raised because of the interpretation and use of current IAEA regulations concerning Fissile Class II units. The regulations were designed to provide for safety in case more than one shipment arrived at the same location during transportation. An interpretation also allows a single shipment to be made up of unlike units as long as a summation of their individual TI (transport index) does not exceed the allowable number.

The members of the working group considered which units of the five used in the $8^3$ array study, when commingled, would be most likely to produce a higher $k_{\text{eff}}$ than an array of like units.

It was determined that combinations of packages with dry plutonium oxide and packages with uranium nitrate solution produced $k_{\text{eff}}$'s significantly higher than was obtained for arrays of either dry plutonium or uranium nitrate solution alone.

The full consequences of our findings can only be determined by additional research on this topic. Many factors could affect the validity of our findings. We probably have not
uncovered the commingling situation which produces the greatest increase in $k_{\text{eff}}$. On the other hand, we have not considered the effect of the steel which is present in most typical packages and which could negate the increase in $k_{\text{eff}}$ due to commingling.

The working group considered whether the summation rule was valid when making up a single shipment. While there was at least one strongly dissenting opinion, the working group felt that in most practical situations the use of the summation rule in making up a single shipment should be allowed. The working group, however, recommended that the effect of commingling of multiple shipments should be studied further.

5. PRINCIPAL CONCLUSIONS AND RECOMMENDATIONS FROM THE STUDY

The main conclusion reached was that the criticality codes studied can satisfactorily handle the largest arrays likely to be of practical interest, providing precautions are taken to ensure that:

- the computed value of $k_{\text{eff}}$ has converged sufficiently, and
- there has been sufficient neutron sampling of all parts of the array. (This precaution may be particularly important for arrays of "loosely coupled" packages (i.e., between which neutron interaction is small) and for arrays of different package types (see next section). Undersampling may lead to an (nonconservative) underestimate of $k_{\text{eff}}$.)

It was recommended that besides the margins usually allowed on such calculated $k_{\text{eff}}$ values (e.g., 3-sigma in statistical approaches, plus an allowance for the difference between theoretical and experimental neutronic cross-section data), an additional allowance should be made when treating larger arrays to ensure subcriticality due to the uncertainties resulting from the lack of experimental data for large arrays.

5.1 Mixed Arrays of Fissile Material Packages

It appears impossible to give a general demonstration of the validity of the summation rule for criticality transport indices ($T_{\text{C}}$) when several types of packages are stacked together. (To ensure the criticality safety of a mixed array, the rule is that the number of packages stacked together must be limited so that $\Sigma(T_{\text{C}})_i < 50$, where $(T_{\text{C}})_i$ is the criticality transport index for package type $i$.) Given the practical importance of the question, members of the working group made calculations for a few mixed arrays composed of two package types. The package types used were selected from those defined for the large array study as being most likely to show a higher $k_{\text{eff}}$ for the mixed array than for an array of the same size of either
type alone. (This was intended to examine whether in some cases the summation rule was nonconservative.)

It was concluded that the summation rule is an acceptable tool for determining the allowable composition of a shipment of unlike packages (with $\text{T}_\text{C} > 0$) if the following two conditions are met:

i. The calculations for each array are performed with optimal moderation between the fissile units, and

ii. The conclusions and recommendations in the first two paragraphs of this section are adhered to.

The group did observe one case involving two simulated packages with $\text{T}_\text{C} > 0$, which when mixed produced a higher $k_{\text{eff}}$ than was observed for an array of the same size of either type alone. Even though this does not necessarily indicate that the summation rule can be violated for practical packages with $\text{T}_\text{C} > 0$, it does provide an incentive for greater vigilance and perhaps additional research regarding the possibility that the safety margin might be less than expected in the actual transport environment when several shipments are combined or come together.

6. General Conclusions From the Working Group's Studies; Future Work Needed

The exercises performed by the working group were based on theoretical package designs. In consequence, and because in general it is difficult to extrapolate experimental data to permit comparison to model cases, the working group is reticent about suggesting explicit general recommendations on technical grounds about "acceptable" computed criticality margins for transport loadings on the basis of the results of their joint study.

General recommendations would require evaluating how accurately the codes allow for the effects of all package structural materials, neutron poisons, shielding, etc., in the calculation. In their computational exercise, the working group deliberately avoided attempting to address in detail the effects of actual packaging materials. It was considered that the numerous additional calculations needed to study the effects of such materials would not add a great deal to the value of the exercise, which had been performed to demonstrate that the codes can make accurate calculations for large arrays. In any case, one of the main packaging parameters affecting array criticality is interspersed neutron moderation, and the working group had already studied this particular parameter by introducing water shells in the model packages used.

It was generally agreed that more work in individual countries is necessary before it will be useful to perform any further joint comparative study on materials effects.
THE QUALITY ASSURANCE TOPICAL REPORT – INTERNATIONAL CO-OPERATION FOR MANUFACTURING OF NUCLEAR PACKAGES

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Abstract

THE QUALITY ASSURANCE TOPICAL REPORT – INTERNATIONAL CO-OPERATION FOR MANUFACTURING OF NUCLEAR PACKAGES.

The paper addresses the manufacturing quality assurance (QA) concerns associated with fabrication of radioactive material packaging in one country for use in another. A case study is discussed which shows a method of implementation that has been successfully practised for casks fabricated in the Federal Republic of Germany (FRG) for use in the USA. The casks utilized are the Castor series, which are fabricated in the FRG from ductile cast iron. This material has several specialized fabrication and inspection requirements. Although international efforts, especially through the IAEA, have helped to standardize QA measures, detailed implementation can vary from country to country. A means of co-ordinating requirements between countries was required. The vehicle used to accomplish this was the Quality Assurance Topical Report (QATR), which is an administrative QA document that clearly defines the roles and responsibilities of all affected organizations. Research was performed to evaluate and compare the general QA requirements for both countries. The close correlation of requirements is documented in the QATR. In addition, a programme is described which permits the USA competent authority to depend on the FRG authority for inspection and enforcement procedures. Once a design has been approved, it is of great importance that the fabrication and inspection of any
component important to safety be properly controlled and documented. The USA requirements for these QA measures are enforced by the Nuclear Regulatory Commission (NRC). Enforcement of applicable regulations in the FRG is the responsibility of the competent authority, Bundesanstalt für Materialprüfung (BAM). As a part of the development of this QATR, NRC and BAM representatives met to establish a mutual understanding of QA philosophy and implementation. As a result of the co-operation, the QATR concept has been approved by both BAM and the NRC. A problem with manufacturing equipment in a foreign country is the need to periodically send company representatives to witness hold points and to review manufacturing records. This approach can be cumbersome in terms of scheduling activities and expensive in terms of time and travel. The methods described in the QATR provide a practical and workable alternative by permitting the BAM to serve as the purchaser (or utility) agent. The utility is only required to make one inspection trip. This approach has proved successful during the manufacture of five Castor casks for the USA. Under the conditions outlined by the QATR, it is agreed that the QA measures enforced by the BAM are sufficient to assure that the USA manufacturing requirements are met. Records and documentation developed during component fabrication are approved and controlled by BAM to provide complete evidence of conformance to specified requirements. A purchaser who takes advantage of this approach will be able to reduce and minimize schedule impacts associated with QA verification, while maintaining full confidence in the quality of the delivered product.

1. INTRODUCTION

It has become evident that nuclear B(U) packages and their construction, use, and unilateral approval have become an issue of both domestic and international interest. A portion of this concern focuses on quality assurance. Adequate quality assurance (QA) during fabrication is important for items manufactured in one country and used in another. Facilitating documents and approved programmes must be available that provide both nations with confidence that international and national requirements are being fully met.

In its totality, QA covers many areas including engineering design, testing, manufacturing, inspection, and finally operation. However, the area of particular interest to this paper is that of manufacturing. Included in this are the procedures for inspection and material testing, as well as the actual fabrication processes. To date, international efforts, particularly those of the IAEA, have been successful in standardizing QA requirements between nations. However, detailed implementation on a specific project must be carefully developed in order to fully match the practices which are known and prevail in each country. As a result, the actual QA programme used must be carefully formulated and adapted to the specific requirements of both the product, in this case the nuclear package, and the participating organizations. An approved planning and co-ordinating document is required which clearly defines the roles of the industrial participants, the user, and the competent authority in both nations. Correlation between national standards must be identified, as well as specific manufacturing inspections and qualifying procedures to be
used. Auditing schedules and responsibilities must be clearly defined. The facilitating administrative document that accomplishes this is called the 'Quality Assurance Topical Report' or QATR. It is a document, approved by the competent authorities in both countries, which specifically defines the roles of the manufacturer, management and auditing agencies.

The purpose of this paper is to present a case study which shows the history of the development of a QATR. Specific reference is made to the QA roles of the various participants and the areas of international co-operation. The QATR described has been approved by the competent authorities of both the Federal Republic of Germany and the United States of America. This QATR has been evaluated in actual use over the last one and a half years and proved successful.

2. GENERAL DISCUSSION

In this case study, the specific utilization of the QATR has been oriented towards the manufacturing of the Castor cask. This cask, which is manufactured in the Federal Republic of Germany, has been licensed for dry storage and has been purchased for use by a US utility.

As general information, the Castor cask employs a very cost-effective and efficient manufacturing technology. The cask body is a massive, poured and machined casting, fabricated from nodular ductile cast iron. This material has several specialized inspection requirements, specifically, ultrasonic testing of the cast body for flaws, and material testing by removal of core bar specimens in the cask wall. This ductile cast iron technology was successfully adapted during the last decade to nuclear packaging, by the Gesellschaft für Nuklear-Service mbH, or GNS, of Essen. At present, a US company, General Nuclear Systems, Inc., or GNSI, is developing this technology for US utility applications; GNSI is a joint venture of two companies — GNS (Federal Republic of Germany), and Chem-Nuclear Systems, Inc., Columbia, South Carolina (USA). Both companies are very much involved in nuclear waste handling, packaging, transport and disposal.

The Castor V is a dual purpose spent fuel storage and transport cask. It weighs 120 tons and has a capacity of nominally 10 to 11 tons of uranium as spent fuel. A topical safety analysis report (TSAR) was jointly prepared in 1984 by GNSI and GNS. Approval of the TSAR for spent fuel storage was received in 1985 from the US Nuclear Regulatory Commission. Approval was documented under Title 10, US Code of Federal Regulations, Part 72. The first Castor V was purchased by a US utility, the Virginia Power Company, in 1984 for a joint governmental (Department of Energy) testing programme. After successful testing in 1985 at the Idaho National Engineering Laboratory, Virginia Power purchased five additional Castor V's for use at the Surry Nuclear Power Station. This is the site of an independent spent fuel storage installation, or ISFSI. The Surry ISFSI is the pioneering facility for dry storage cask applications in the USA.
3. HISTORY OF THE QATR

It was recognized by both GNSI and GNS that a manufacturing QA programme with reduced manufacturing inspection requirements for the utility-purchaser would be needed. Typically it is not practical or cost effective for the US utility-purchaser to be involved on a daily basis with fabrication inspection in a distant foreign country. Detailed knowledge of Federal German standards and practices, and their relationship to US practices requires specific expertise which is difficult to acquire in the USA. Clearly a system of auditing by a trusted and knowledgeable Federal German authority was required. It was also recognized that the utility, as owner and licensee-operator, would be ultimately responsible for quality assurance. Both GNSI and GNS researched and compared the general QA requirements in the USA and the Federal Republic of Germany. The correlation of national codes and standards, such as DIN, IAEA, ASTM and ASME, were made. We found considerable comparability in both intent and practice between countries. To this end, a proposal for a joint international QA programme was forwarded to competent authorities in both countries, specifically the Bundesanstalt für Materialprüfung, or BAM, and the US NRC - Quality Assurance Branch.

In December 1984, an organizing meeting was held in Berlin (West), with GNS, GNSI, BAM, NRC and Virginia Power QA and licensing personnel participating. The purposes were:

1. To compare and adapt US and Federal German QA and compliance requirements for construction of nuclear safety equipment; and
2. To evaluate the possibility of a co-operative agreement between the NRC and BAM for ensuring the quality of NRC-licensed equipment manufactured in the Federal Republic of Germany.

The participants concluded that a workable programme of co-operation was feasible. The vehicle for this agreement was to be an approved QATR. The intent of the QATR was to establish an administrative programme which rigorously defined all organizational responsibilities. Together, GNS and GNSI prepared a QATR, which made extensive use of the existing GNS Quality Assurance Handbook. This document was specifically developed for the manufacture of safety class equipment destined for the USA. In it, reference was made to both US and Federal German codes, standards and procedures.

Submittals of the QATR were made to both the NRC and BAM. In February 1986, BAM officially concurred with the document and agreed that all processes and requirements attributed to them in the QATR would be met. In April 1986, GNSI received official NRC approval of the QATR. The approval specifically stated that “BAM and PTB (the Federal German competent authorities) activities described in the QATR will be accepted by the NRC as a substitute for inspections otherwise performed by the utility-purchaser”. The NRC QA programme review
and approval was made in accordance with 10CFR72. It was noted that utilities could reference the QATR in their cask procurement and ISFSI licensing documents. To date, GNS has completed five Castor casks in accordance with this QATR and referenced programmes. These casks will be shipped to the USA in the summer of 1986 for use at the Surry Station ISFSI.

4. RESPONSIBILITIES

Generally, the simpler the plan and the fewer the number of concerned parties involved, the greater the probability of success. However, the nature of international co-operation frequently moves in the opposite direction. In this case, at least six parties in two countries are involved. However, the responsibilities defined in the QATR are being implemented according to the plan. The underlying philosophy embodied in the QATR defines responsibilities based on these three principles:

1. A clear division of QA roles and an understanding and agreement on responsibilities by each organization involved was established.
2. A defined schedule of audits by each organization is in place, with a coordinating project manager assigned to ensure compliance.
3. Each organization has one controlling document which is referenced in the QATR, which rigidly defines their actions. This document is applicable to their special function or role. The organization is thoroughly familiar with its use.

The organizations, their controlling document, and nation are shown in Fig. 1. The specific administrative roles of each organization are as follows:

1. **The Purchaser (Utilities — USA)** is the ultimate licensee and operator. As such, they have final responsibility to ensure that the equipment is manufactured in accordance with the NRC licence. This responsibility includes assurance of conformance to all applicable technical codes and standards. Even though a competent Federal German authority audits manufacture, the purchaser-utility must schedule a comprehensive audit of the inspection process on at least an annual basis. This audit may be in combination with other purchasers. Their controlling document is the procurement specification which is in accordance with their ISFSI Part 72 licence.

2. **The Project Managers (GNSI — USA)** are responsible for executive management and overall co-ordination of the fabrication project and they routinely monitor that all quality actions are in accordance with the QATR (their controlling document). They are the focal point for facilitating and ensuring liaison between all US and Federal German organizations.

3. **Engineering and Manufacturing Management (GNS — Federal Republic of Germany)** ensures that all manufacturing is performed in accordance with
the licensed design. They prepare procedures for special material and testing requirements. For example, ultrasonic testing of the cask body, material chemistry and toughness properties, leak testing, etc. These are critical to cask function and safety. Their controlling document is the GNS QA handbook.

(4) *The Manufacturer (Federal Republic of Germany)*, as constructor of the cask, directly performs all inspections and testing processes. Standard factory proce-
dures are sometimes modified by special instructions, which are supplied by GNS. Their controlling documents are their QA manual and factory procedures as approved by GNS.

(5) *The Competent Authority (PTB, BAM – Federal Republic of Germany)* is the Physikalisch-Technische Bundesanstalt (PTB). Their work is supplemented by the BAM in the area of manufacturing and testing. The BAM is responsible for compliance assurance and serves as a third party auditor for fabrication and testing. Since they are an independent agency, and familiar with Federal German inspection practices and standards, they are an invaluable substitute auditor for the utility-purchaser. They are also thoroughly familiar with IAEA requirements for nuclear transportation packaging. Their controlling document is the FPP or fabrication and test plan. This is a standard checklist form, prepared by GNS and approved by BAM. The FPP is the BAM control for auditing. BAM approves and controls the test fabrication records developed during component fabrication to provide complete evidence of conformance.

(6) *The Competent Authority (NRC – USA)* is responsible for the QA of all nuclear equipment in the USA as it relates to the US public safety. As competent US authority, they audit, as necessary, the Quality System to ensure compliance to the approved QATR. Their controlling document is Subpart B of 10CFR72. They periodically audit the utility-purchaser to ensure compliance with their ISFSI licence per 10CFR72.

Obviously other organizations are involved, including the Federal German Ministry of Transport and the Department of Transportation (USA). Both have agreements and memoranda of understanding which utilize the expertise of their nation’s competent authority.

5. EXPERIENCE TO DATE

Quality assurance of the manufacture of nuclear material packaging is a key example of needed international co-operation in the nuclear field. The Castor QATR, as presented in this case study, is a pioneering agreement. The QATR presents both a logical and workable administrative approach that all organizations can adhere to. The programme fully meets all IAEA requirements. It could serve as a framework for agreement between other nations and organizations.

The BAM role in auditing manufacture within the Federal Republic of Germany is key to the success of this process. It provides the US purchaser–utility and the NRC confidence that the package will be manufactured to the acceptable standards in accordance with the NRC approved package licence. It is also an efficient system since casks could be supplied to several US purchasers—utilities if annual utility audit requirements are met. Extensive utility costs are precluded.
The Castor QATR programme has been working successfully for over one year and through the manufacture of five casks. Based on the results of a number of component authority audits and verifications, it is a practical and workable approach. Even though the number of involved organizations is large, the programme is successful and not complex for the following reasons:

(1) There is a defined and dedicated commitment by all parties to make the system work.
(2) The organizational duties are well known to all parties and similar to those normally performed in their respective countries.
(3) A clearly written administrative plan, as exemplified by the QATR, employing a project manager who is dedicated full time to this programme, ensures that co-ordination is achieved.

We trust that this case study shows that international co-operation in nuclear packaging can be successfully accomplished.
RESEARCH AND DEVELOPMENT
HEAT TRANSFER INVESTIGATIONS
FOR SPENT FUEL ASSEMBLIES
IN A DRY CASK

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Abstract

HEAT TRANSFER INVESTIGATIONS FOR SPENT FUEL ASSEMBLIES IN A DRY CASK.

For studying the heat transfer processes and predicting the maximum spent fuel element surface temperature in a spent fuel assembly (SFA) transported in a dry cask, model experiments have been performed with a gas filled model cask containing a simplified electrically heated model of a WWER-type SFA with 90 fuel elements. The temperature distribution of the SFA model is measured for different heat rates under vacuum in the model cask, and under normal pressure and overpressure (0.1-0.7 MPa) for several cooling gases (air, argon, helium) in order to separately investigate heat transfer processes by radiation and convection/conduction. The results were compared with the calculations. Computer programs as well as simplified calculation methods for temperature prediction were developed and checked. The results obtained are also useful for thermal analyses in the field of the dry storage of SFAs in a cask or a can.

1. INTRODUCTION

The heat transfer in a dry spent fuel cask has to be considered as a significant safety problem. According to the legal Transport Regulations in the German Democratic Republic [1] and the IAEA Regulations for the Safe Transport of Radioactive Material [2] it has to be demonstrated that the spent fuel element (SFE) surface temperatures of the spent fuel assembly (SFA) do not exceed permissible limits. For the exact prediction of these temperatures the heat transfer processes in the cask have to be studied and calculation models have to be developed and checked. Such investigations have already been made and published for a cask containing 30 SFAs of the PWR type, as far as the heat transfer from the SFAs to the cask inner wall and the prediction of the maximum SFA surface temperature in the cask is concerned [3]. In the present work the heat transfer processes inside an SFA are studied for the prediction of the maximum SFE surface temperature. For this purpose model experiments have been performed with
a gas filled model cask containing an electrically heated SFA model. The temperature distribution of the SFA model is measured as a function of various parameters and compared with calculations to verify computer programs as well as simplified methods.

2. OBJECTIVES AND RESEARCH PROGRAMME

It was the aim of the investigations:

- to study the heat transfer processes due to radiation, convection and conduction in the SFA
- to determine the SFE surface temperatures as a function of heat rate and internal cask pressure for different cooling gases, and
- to check calculation methods which can be applied to the original cask after verification by the model experiments.

For this purpose, the radial and axial temperature distribution of the SFA model was measured with the model in a vertical position in the model cask according to the following research programme:

- measurement under vacuum for the investigation of the radiant heat transfer and comparison with calculations
- measurement under normal pressure and overpressure using air, argon and helium as coolants for studying convective and conductive heat transfer and comparison with calculations.

3. EXPERIMENTAL EQUIPMENT

3.1. Experimental setup

The experimental setup is shown schematically in Fig. 1. It mainly consists of:

(a) The model cask with temperature measuring points (internal diameter 180 mm, height 600 mm, 2 mm wall of stainless steel), designed for an overpressure up to 0.7 MPa;
(b) The SFA model with temperature measuring points, described in Section 3.2, which is contained in the model cask;
(c) The vacuum system for measurements under vacuum in the model cask up to 7 mPa;
(d) The pressure system for overpressure measurements from 0.1 to 0.7 MPa for air, argon and helium;
(e) Devices for power supply, heat rate measurement and temperature measurement and registration.
3.2. Model of spent fuel assembly (SFA model)

The SFA model represents a simplified model of a WWER-type pressurized water reactor SFA with 90 fuel elements, which is reduced in length. The cross-section of the SFA is modelled nearly on the original scale (scaling factor 1.3). The fuel elements are simulated by electrically heated rods. They are hexagonally arranged as shown in Fig. 2, and they are positioned by three spacer grids. The temperature distribution of the SFA model is determined by means of thermocouples at selected measuring points, as can be seen from Fig. 2. The radial temperature profile is measured at middle height of the heating rods and the vertical temperature distribution at the innermost heating rod No. 5 and at the SFA wall. The coolant temperature is measured by two thermocouples centrally above and below the SFA model.
4. RESULTS

4.1. Measurements and calculations for radiant heat transfer

The radiant heat transfer in the SFA has to be considered as the dominant heat dissipation process for the resulting temperature distribution. For its separate investigation, measurements were made in the model cask in the pressure range from 700 mPa to 13 mPa. The measured radial temperature distributions of the SFA model are shown in Fig. 3. They are compared with calculations for radiant heat transfer made using the computer program [4], which is based on the calculation model by Watson [5]. Good agreement was found in the high temperature range. The deviation of maximally 8% at low temperatures is caused mainly by the contribution of residual gas thermal conduction in this temperature range, which is not yet negligible and which results in a small decrease of measured temperatures as compared to calculated ones for pure radiant heat transfer. For the development of a simplified thermal analysis method the following Eq. (1) has been derived for the
calculation of the radiant heat transfer between the innermost heating rod (No. 5) and the SFA wall:

\[
\dot{Q}_H = C_S \cdot \mathcal{F}(e_H, e_{SFA}) \cdot F_H \left( \frac{T_{\text{max}}}{100}^4 - \frac{T_{SFA}}{100}^4 \right)
\]

where

\( \dot{Q}_H \) is the heat rate of the heating rod
\( C_S \) is the black body radiation constant \( (5.78 \times 10^{-4} \, \text{W} \cdot \text{cm}^{-2} \cdot \text{K}^{-4}) \)
\( \mathcal{F}(e_H, e_{SFA}) \) is the radiant configuration factor of the SFA as a function of the emissivities of the heating rods \( e_H \) and the SFA wall \( e_{SFA} \)
\( F_H \) is the heat transfer area of the heating rod
\( T_{\text{max}} \) is the maximum heating rod surface temperature in the SFA in K (innermost heating rod No. 5)
\( T_{SFA} \) is the SFA wall temperature in K.
The radiant configuration factor was calculated by using the computer program [4] and is given in Fig. 4 as a function of the emissivities. It can be seen that the emissivity of the SFA wall has a greater influence on the radiant configuration factor than the emissivity of the heating rods. Compared with the model experiments ($\varepsilon_\text{H} = 0.39; \varepsilon_\text{SFA} = 0.35$) it was found that this simple calculation method of Eq. (1) for the radiant heat transfer in the SFA predicts the heat flow with an accuracy of about 10% or the maximum SFE surface temperature with an accuracy of about 5%.

4.2. Measurements and calculations for total heat transfer

In order to predict the maximum SFE surface temperature, the total heat transfer in the SFA has to be studied, which includes beside radiant heat transfer also the heat transfer by free convection and conduction. According to the research programme (see Section 2) the experiments were performed in such a way as to allow a separate investigation of theses processes and a separate fitting and checking of calculation methods.
4.2.1. Measurement results

The radial temperature distribution of the SFA model in the air filled cask under normal pressure and overpressure conditions is shown in Fig. 5 for two selected heat rates. Owing to the pressure increase, heat removal by convection is considerably improved, which causes a strong decrease in all temperatures and an equalization of the heating rod surface temperatures. Using the results for radiant heat transfer (see Section 4.1) it was found that under normal pressure, and depending on temperature, 30 to 45% of the heat rate is dissipated by thermal radiation. By increasing the pressure up to 0.7 MPa this fraction is reduced to only 7 to 10%.
On the basis of the measured vertical temperature distribution of the innermost heating rod (No. 5) it was found that the maximum heater surface temperature is shifted towards the upper heater end with increasing pressure, exceeding the surface temperature at middle heights of the heater by 3 to maximally 11%. The pressure dependence of this maximum heater surface temperature is shown in Fig. 6 along with the measured temperatures of the SFA wall, cask wall and air coolant. With increasing pressure the air coolant temperature approaches the maximum heater surface temperature very quickly. The pressure dependence of the temperatures in Fig. 6 shows that in particular pressure increases in the range from 0.1 to 0.5 MPa contribute to an effective improvement of heat removal.

The effect of coolant conductivity on the maximum heater surface temperature of the SFA model is evident from Fig. 7. As expected, the lowest temperatures
FIG. 7. Maximum heating rod surface temperature in the SFA, $T_H$, as a function of heat rate $\dot{Q}$ for argon, air and helium as coolants under normal pressure ($p = 0.1 \text{ MPa}$).

FIG. 8. Maximum temperature difference between innermost heating rod and SFA wall $\Delta T$ as a function of heat rate $\dot{Q}$ for argon, air and helium as coolants under normal pressure ($p = 0.1 \text{ MPa}$).
were measured for helium cooling. About 1.7 times more heat can be removed by helium as compared to air at the same maximum temperature in the model cask. At the actual maximum temperature difference between the innermost heating rod and the SFA wall, as shown in Fig. 8, a 2.5-fold greater heat rate can be removed by helium. Equally good conditions of heat removal as in the case of helium could be obtained with air as a coolant only by a pressure increase to about 0.5 MPa.

4.2.2. Calculation results and comparison with measurements

On the basis of the temperature distributions of the SFA model, measured for various pressure values and cooling gases, the calculation methods for total heat transfer were checked. For the calculations a two-dimensional computer program was used, developed on the basis of the computational model in [6]. This computer program is described in detail in [7]. It calculates the radial and axial temperature distribution of the SFA model, taking into account all heat transfer processes by radiation, convection and conduction. The temperatures, densities and velocities of the coolant are also determined. All parameters required for the calculations are used as functions of temperature and pressure. Calculations and measurements
are compared in Fig. 9 for the vertical temperature distribution of the innermost heating rod (No. 5) and the wall of the SFA model at selected heat rates and pressure values. Some deviations arise in the overall vertical temperature distribution at the various pressure values, caused mainly by the necessarily simplified theoretical and experimental modelling of real processes and structures. Good agreement is found for the maximum heater surface temperature as the decisive thermal safety parameter, and the maximum SFA wall temperature. Measured and calculated maximum temperatures agree within 5% at normal pressure. Sufficiently good agreement is also obtained under conditions of overpressure (0.3 and 0.7 MPa), where the calculated maximum temperatures are higher by maximally 10% than those measured.

A similarly good agreement between calculations and experiments was found for the coolants argon and helium. In particular, for helium under normal pressure, the calculated and measured temperatures agreed very well within measuring errors, so that the exact calculation of the thermal conduction processes in the SFA model by means of the computer program [7] could be confirmed.

On the basis of the model experiments and the basic equation of heat transfer we found the following calculation method for the prediction of the maximum heater surface temperature in the SFA model, useful for simplified thermal analysis:

\[ \dot{Q} = (\alpha_r + \alpha_c) F (T_{\text{max}} - T_{\text{SFA}}) \]  

(2)

where \( \dot{Q} \) is the heat rate of the SFA model
\( \alpha_r \) is the heat transfer coefficient for radiation
\( \alpha_c \) is the effective heat transfer coefficient for convection and conduction
\( F \) is the heat transfer area of heating rods
\( T_{\text{max}} \) is the maximum heater surface temperature in the SFA model
\( T_{\text{SFA}} \) is the wall temperature of the SFA model.

The parameter \( \alpha_r \) can be derived from studies of the investigations of radiant heat exchange according to Eq. (1) in the following form:

\[ \alpha_r = C_S \cdot \mathcal{F} \cdot \frac{[(T_{\text{max}}/100)^4 - (T_{\text{SFA}}/100)^4]}{(T_{\text{max}} - T_{\text{SFA}})} \]  

(3)

The parameter \( \alpha_c \) was determined by means of the model experiments for air cooling at different heat rates and internal cask pressures. It could be represented as a function of the dimensionless convective heat transfer Grashof (Gr) and Prandtl (Pr) numbers as follows:

\[ \alpha_c = 1.12 \times 10^{-4} \times (Gr \times Pr)^{1/2} \]  

(4)
FIG. 10. Measured and calculated maximum heating rod surface temperature $T_{\text{max}}$ in the SFA model as a function of heat rate $\dot{Q}$ for air cooling and different pressure values $p$.

By means of Eq. (4), the measured $\alpha_e$ values (in W·m$^{-2}$·K$^{-1}$) are reproduced with a deviation of maximally ±10% for the range of $10^8 \leq (Gr \times Pr) \leq 5 \times 10^9$ which could be realized in the model experiments. From this simplified calculation method good agreement is obtained with the measured maximum heater surface temperatures for air cooling at different heat rates and internal cask pressures, as can be seen in Fig. 10.

5. SUMMARY AND CONCLUSIONS

For the investigation of the heat transfer in a spent fuel assembly under dry transportation conditions, model experiments were performed, using an electrically heated SFA model of a pressurized water type reactor in a model cask. The radial and axial temperature distribution of the SFA model was measured in a vertical position under vacuum, normal pressure and overpressure, using air, argon and
helium as coolants. Calculation methods for temperature prediction were developed and checked. The following conclusions can be drawn:

- The heat removal from the SFA can be considerably improved by increasing the internal cask pressure or by using helium as coolant.
- The radiant heat exchange in the SFA model can be calculated with sufficient accuracy by means of the computer program [4] or the simplified analytical representation of Eq. (1). Both methods are directly applicable to the original SFA and useful in order to approximately calculate the maximum SFE surface temperature under normal pressure, if the fraction of heat transferred by radiation is allowed for.
- For the calculation of the total heat transfer the computer program [7] was developed and verified; this program permits the prediction in the SFA model of temperature as a function of heat rate, type of gaseous coolant and coolant pressure. This computer program can be directly applied to the original SFA for the calculation of the maximum SFE surface temperature.
- For the purposes of simplified thermal analysis an analytical representation for the total heat transfer could be derived; this permits the prediction of the maximum heater surface temperature in the SFA model. Its range of validity for practical applications on a larger scale has still to be confirmed by further investigations.
- The results obtained contribute to the study of the basic heat transfer processes to be considered for the thermal analysis of content/package behaviour in transportation, and also for thermal problems in the dry storage of spent fuel assemblies in a cask or can.

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REAL, AS OPPOSED TO REGULATORY, REQUIREMENTS

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Abstract

REAL, AS OPPOSED TO REGULATORY, REQUIREMENTS.

A major programme of flask testing and analysis was recently completed by the UK Central Electricity Generating Board (CEGB). This programme generated a great deal of data relating to the behaviour of Magnox irradiated fuel flasks under impact and fire accident conditions and it has enabled comparisons to be made between regulatory and ‘real accident’ conditions. This paper puts forward the concept of ‘true safety’ in which it is suggested that the flask designer should take account of potential real accident situations, in addition to purely regulatory requirements, during the early stages of the design process. Some examples are given which illustrate the application of this philosophy.

1. INTRODUCTION

The IAEA Regulatory tests for Type B packages are often criticised on the grounds that they do not represent severe enough conditions to encompass the range of "real accidents" which those packages might encounter during normal transport operations. The 9 m drop test and the half-hour fire test, in particular, have been criticised in this way.

This paper puts forward the proposition that, in the cases of impact and fire, there are many mitigating factors in the real world which, when accounted for, can result in a significantly enhanced flask capability in a "real accident" compared to that suggested by a Regulatory test. This proposition is supported by reference to recent experimental and analytical work on spent Magnox fuel flasks which has been carried out by the CEGB [1].

However, although it is argued that the Regulatory tests generally demonstrate a satisfactory basic level of flask performance, it is nevertheless unwise to design simply to meet a
Regulatory requirement, since this approach gives no indication of the margins of safety that exist in the design. Indeed, flask design based solely on meeting the Regulatory requirements may not always confer guaranteed safety in a real accident environment because areas of potential weakness may not be addressed by Regulatory testing. Examples are given in the paper where attention to "real accident" conditions might significantly affect detailed flask design.

It is proposed that a more rational approach is to design to the spirit of the Regulations but with a regard for the real accident environment to ensure a robust design with an acceptable margin of safety, which can then be demonstrated to meet the Regulatory requirements. In this way, "true safety" may be built into the design. The pursuit of "true safety" can be encouraged without prejudicing the Regulatory standard and an approach which is aimed at meeting this objective for CEGB flask movements in the U.K. is presented in the following sections.

2. THE SELECTION OF POSTULATED "REAL ACCIDENT" SCENARIOS

The discussion of flask behaviour in "real accidents" in the following sections is developed around a number of example scenarios. These accident scenarios have been developed as a result of an extensive transport and hazards survey of the routes over which the CEGB's spent fuel flasks are transported in the United Kingdom during normal operations [2]. This survey took account of the frequency of flask movements along each section of the route and it identified and assessed all those hazards which might be associated with a serious accident involving a flask. Examples of the hazards identified include other vehicles on the road, other vehicles on the railway; high bridges from which the flask might fall, tunnel abutments with which the flask might collide, and hazardous installations with significant stores of inflammable or explosive materials adjacent to a flask transport route.

From the hazards survey and the information on the frequency of flask movements, it was possible to construct a table of accident types ranked in terms of probability of occurrence. This table is reproduced in Fig. 1. The accident scenarios discussed below have been selected from this table either because of their severity, or because of their frequency of occurrence, or for a combination of these reasons.

3. IMPACT ACCIDENT SCENARIOS

It has been frequently stated that the IAEA Regulatory 9 m drop test for a Type B flask is inadequate since it involves an impact speed of only 30 mph (48 km/h) whereas flask transport by rail can take place at much higher speeds.
10^{-1}

1/10 years

Flatrol derailment (all classes of incident)
Fire on a flask train (all classes of incident)

10^{-2}

1/100 years

Train to flask train collision\(^{(1)}\) at speed above 30 mph
(all types of collision)

10^{-3}

1/1000 years

Flatrol struck directly in converging line collision
Derailed flatrol struck by oncoming train (all speeds)
Flatrol strikes face of bridge abutment above 30 mph
Flask train collides with train carrying flammable liquids
(no resulting fire)

10^{-4}

Derailed flatrol struck by Inter-City train at over 30 mph
Flask train collides with pipeline carrying flammable liquid
Flask train collides with train carrying flammable liquids
(giving rise to a consequential fire) \((2)\)
Flask train involved in fire following collision with a pipeline \((2)\)
Flask and/or flatrol falling more than 9 m onto a "rock target"\(\(^{(3})\)
Flask train collides with hazardous lineside installation
(no resulting fire)
Flask and/or flatrol falling more than 20 m vertically
onto a "rock target"
Flatrol struck by heavy goods vehicle at level crossing

10^{-6}

1/1,000,000 years

Flask train involved in fire following collision
with lineside hazard\((2)\)
Secondary impact of flask into a bridge or tunnel
abutment following a train collision

NOTES:

\((1)\) No allowance for impact attitudes has been made in any of the above scenarios.

\((2)\) Consequential fire does not necessarily fully engulf flask.

\((3)\) "Rock target" refers to that fact that rock formations are present at a site: These will often be overlain by surface deposits of softer material but the mitigation of these deposits has not been considered.

FIG. 1. Table of accident probabilities.
It is easy to imagine accident scenarios which potentially involve flask impacts at speeds in excess of 30 mph (48 km/h). Obvious examples are flasks involved in collision with massive bridge or tunnel abutments at high speed, flasks falling from a high bridge or viaduct, or flasks involved in a multiple accident in which an express train hits a flask lying in its path. In each case, the impact conditions are at first sight more onerous than those specified in the IAEA test. However, in each case, the real "target" is likely to be less damaging than the unyielding target specified in the Regulations.

With reference to the tunnel abutment impact, tests were mounted by the CEGB in which 1/4 scale model flasks were crashed into a series of targets which were designed to represent massive tunnel portals. The targets were built using engineering bricks and hard sandstone masonry. The tests were carried out in a horizontal rig driven by an electric winch. In some tests, the flasks were mounted in a trolley which released them at a specified impact attitude immediately in front of the target. In other tests, the flasks were mounted on a purpose built scale model railway wagon (flatrol) and the whole system was crashed into the target. The purpose of this latter series of tests was to examine the additional protection which might be provided by the flatrol in the event of a real accident. Most of the tests were carried out at 60 mph (96 km/h) although a small number of exploratory tests were carried out at a reduced speed. A detailed description of the work may be found in [3] and [4].

In all cases the damage to the flask was minimal, even though the nominal compressive strengths of the masonry and bricks used to construct the targets were high (typically 50 MPa and 90 MPa respectively). These tests largely confirmed the results of a series of drop tests previously carried out by the CEGB onto different target materials using shaped missiles and model flasks [5]. The present series of tests also underlined the fact that additional mitigation is afforded by the nature of masonry constructions over and above monolithic constructions of the same stiffness material. In an effort to make the masonry targets "stronger" steel braces were wrapped around the construction to prevent bursting, but even this extreme form of reinforced target was found to be relatively undamaging to the flask.

With reference to accident scenarios involving a fall from a high bridge, a series of calculations was performed in which the cushioning effect of a small covering of ground surface deposits was allowed for. Borehole samples were taken from sites where a high drop could occur on the United Kingdom flask transport routes and the material properties appropriate to each site were used in the calculations. It was generally concluded that the flask would have to fall on exposed hard rock before the impact forces become potentially as onerous as those generated during an
IAEA Regulatory 9 m drop test. In the case of the Magnox flask, the term "hard" rock may be taken to mean rock with a static unconfined compressive strength in excess of 70-100 MPa. The probability of a flask falling from a high bridge and hitting a substantial region of hard rock is very low indeed, as can be seen from Fig. 1.

As well as the relief afforded by the relative strength of most real targets compared with the unyielding Regulatory target, other structural components associated with the transport system have cushioning effects. The CEGB train crash test [1] showed how structurally weak the heavy locomotive was compared with the flask. This serves to limit the maximum impact forces which can be generated during such an event. Similarly, the model flatrol crash tests into tunnel abutments described above showed that the flatrol crushes during the impact causing considerable deceleration of the flask before it ever reaches the target abutment [4].

4. FURTHER IMPACT CONSIDERATIONS

From the foregoing, it may be concluded that the IAEA Regulatory impact tests do, indeed, provide sufficient margins to encompass the types of "real accident" which have been referred to. However, there are certain aspects of flask design which may not necessarily be examined by the IAEA tests and three examples are given below.

First, in the "real target" tests carried out by the CEGB, the flasks made significant indentations in the faces of the targets. Under these conditions, the target material can impinge upon quite large portions of the flask and this can impose high local loads on vulnerable features such as valves and exposed lid bolt heads. This type of load is never encountered in an IAEA test because the target is, by definition, flat and unyielding. It would not be prudent, therefore, to design a flask to withstand an IAEA impact test without giving some consideration to protecting vulnerable components from this type of load.

Second, it is possible, under some conditions, to generate a higher impact load on a "soft" target than on a "hard" one. This can occur where the profile of the impacting edge of the flask is irregular and the area of contact between flask and target suddenly increases as the flask bites further into the target. This situation is illustrated schematically in Fig. 2 where the contact forces generated by two different target materials are compared. By simple energy balance methods, it can be seen that the peak force generated by a soft target can be greater than that generated by a hard target. It is unwise, therefore, to assume that soft targets always do less damage than hard targets: each case must be dealt with on its own merits.
FIG. 2. Contact forces for hard and soft targets.
Third, once the target material is strong enough to withstand flask impact without serious indentation it is effectively an "unyielding target", i.e. it is no different from an IAEA target. The critical strength at which this occurs is related to the strength of the flask itself [5] but, for "soft" flasks (i.e. those with crushable shock absorbers), the critical target strength may be as low as 20-30 MPa. Under these circumstances, it is not reasonable to claim mitigation from many "real" target materials simply on the grounds that they are relatively weak compared to a steel faced target. This means that a "reserve of strength" which is frequently taken for granted when assessing real accidents does not, in fact, exist.

The three examples above have been put forward to underline the fact that a blind intention to comply with Regulatory test conditions does not necessarily produce a flask which is optimised from a "true safety" point-of-view. Whilst the designer may set out to meet the Regulatory standards as a baseline, he should nevertheless be aware of the loadings which might be applied to the flask in a real accident condition and improve his design accordingly if shortcomings are encountered.

5. FIRE ACCIDENT SCENARIOS

The IAEA Regulatory Fire accident defines a thermal environment which a flask must withstand while maintaining the required shielding and meeting the leakage standards. However, it should be recognised that the specified fire parameters, and in particular the average temperature environment, are not necessarily realistic in that they are difficult to create in practice even in a well engineered experiment. For example, the CEGB carried out full scale fire tests on Magnox flasks and delays of several weeks were experienced in trying to attain a wind speed of less than 2 m/s in order to meet the Regulatory prescription of a fully engulfing fire. Then, even with a wind shield surrounding the test area, flame temperatures varied temporally and spatially between 650° and 1100° with the overall average temperature being around 925°C.

Criticism could also be levelled at the specified flame emissivity of "at least 0.8". Again, taking the CEGB tests as an example, and accepting the difficulty of actually measuring emissivity within the fire, it is likely that a wide fluctuation occurred throughout the fire. However, the average value deduced was around 0.6.

At first sight, therefore, it may appear that valid objections could be made to the fire parameters in the Regulatory fire test. However, it is reassuring to find that the average radiative environment created by the Regulatory conditions gives an incident heat flux of between 67 and 75 kW/m² which encompasses the average value of around 70 kW/m² deduced from the
CEGB fire tests. It is also worth noting that the incident convective heat flux in the CEGB tests was found to be around 15 to 20% of the total and is also applicable to the Regulatory fire test.

It would therefore seem quite justifiable to claim that the fire environment defined by the IAEA Regulations represents closely the average environment which would be experienced if the flask were engulfed in a real fire of the same duration. It should be recognised, however, that locally more onerous conditions could exist and a flask design should not be vulnerable to these.

With regard to the 30 minute fire duration specified in the IAEA Regulations, it can readily be postulated that accident situations with potentially longer fires might occur and, indeed, several examples of fire accidents exist to support this view.

It is argued, however, that for the large flasks used for spent fuel transport it is very unlikely that a fire of sufficient extent to be fully engulfing could be sustained in a particular location for an extended period of time. This is reasonable since it would require the combination of several exceptional circumstances to support such an event, e.g. large supply of fuel, terrain and drainage conditions to allow the fuel to be contained, weather conditions to provide a steadily burning fire, lack of effective fire fighting resources, and a most essential element, the presence of a flask at the precise location. Furthermore, it is unlikely that a fire would fully engulf the flask since a significant area of the flask surface is likely to be resting on the ground or protected from the fire by the transporting vehicle or other debris.

An estimate of the probability of a serious fire accident involving a flask train was presented at the Sizewell "B" Inquiry [6]. Consideration was also given to the maximum length of fire that could arise based on a maximum credible amount of fuel that could be available. Taking a pessimistic approach it was considered that all the fuel from two adjacent 75 t rail tankers could feed a fire of the required size to fully engulf a PWR flask. From the Sandia fire test [7] a burn-rate of 75 t/h had been determined. Two 75 t rail tankers could therefore sustain a fire of 2 hours duration.

However, as already mentioned, the approach taken was pessimistic and a more realistic consideration of a number of factors would, in reality, result in a fire of considerably shorter duration. These are discussed in the following paragraphs and are then used to produce a revised estimate of fire duration which would be applicable to the U.K. rail transport system in general.
a) Drainage

The practice on the British Rail rail network is to rely on natural drainage where that is clearly adequate (e.g. on an embankment) but to install artificial drainage on ground level track, in cuttings and tunnels and where dips in the track could result in water collection points. On an embankment or level track it is therefore judged to be incredible that fuel could be contained only in the vicinity of the flask vehicle. Where artificial drainage is provided an estimate may be made as follows of fuel loss into the drains based on the design capacity of the system.

There is an equal chance of the fuel draining to either side of the track so it can be assumed that the drainage channels provided both sides of the track each receive half the available fuel. It is then calculated that the total drainage rate based on the capacity of the system is 2 t/min. Experiments carried out by the CEGB, have shown that only the top 80 mm or so of the fuel draining through the ballast on which the track is laid will support a significant fire on the surface. With an average ballast depth of 250 mm, a track width of 8.4 m and drainage catchpits at approximately 30 m centres, it is calculated that 20 t of fuel is effectively lost in the ballast.

If it is assumed that fuel not lost by drainage or within the ballast is available to be consumed by burning then a fire duration may be calculated as shown in the next paragraph.

b) Fire Extent and Duration

In this analysis it is pessimistically assumed that two tankers are breached at adjacent positions and that the combined breach size is an optimum to provide fuel to support a fire of a specific extent and burn-rate. Taking the basic assumptions of the availability of 150 t of fuel it can be postulated that for any track condition, the extent of the fire could vary from being a relatively narrow rivulet of burning fire to a much larger pool type arrangement. In the first case the fire could in no way be fully engulfing, and is therefore not a serious hazard. In the second case the pool has to be bounded in some way to prevent the fuel spreading unhindered over a very large area. A pessimistic assumption may be made that the fire extends to the width of the track (8.4 m) and over the length between the drainage catchpits which are provided at 30 m centres. The fire area in this case is therefore 252 m².

The burn-rate of 0.008 t/m²/min has been calculated based on the Sandia fire test [7] and when taken in conjunction with drainage gives a total fuel consumption rate of 4 t/min. The amount of fuel available when account is taken of the ballast reservoir is 130 t and the maximum length of a fully engulfing fire is therefore calculated to be around 33 minutes.
c) Other Factors

The requirement for the flask vehicle to be in the immediate vicinity of breached tankers is a very unlikely event considering the relative position in the train of the flask vehicle. However, even assuming it is near the tankers, the low centre of gravity of the flask vehicle makes it unlikely that it will overturn.

The protection therefore afforded to the flask by the substantial structure of the carrying vehicle extends under the flask and fore and aft of the flask by around 3 m. In the unlikely event of the flask being ejected from the vehicle, then one face would be resting on the ground or in contact with other substantial debris. It is therefore reasonable to assume that at least one face of a flask will be protected from the direct effects of the fire giving a significant reduction in heat input compared to a comparable fully engulfing fire condition.

There remains the circumstances of an accident involving, say, a tanker train and a flask train whilst both are in a constricted location, e.g. within a tunnel. The probability of such a coincidence occurring is very low indeed. However, even this can be avoided by appropriate cargo scheduling.

6. CONCLUSIONS

A number of impact and fire scenarios have been defined and the response of the Magnox flask to these potentially severe accidents has been discussed. Work carried out by the CEGB into real but unlikely transport accidents in the United Kingdom has shown that some transport accidents have an associated energy potential which could cause greater damage than that obtained in a Regulatory drop test. However, important mitigating factors associated with the strength of real constructions and the compliance of intervening structures serve to reduce very considerably the damage potential which really exists in these situations. Similar arguments can be used to assess the mitigating factors associated with real transport accidents involving fires. It is concluded that, in most cases, the accident conditions do not in general terms present more severe conditions than those imposed during the Regulatory Type B impact and fire tests.

Whilst the IAEA tests represent a good baseline for flask design standards, it is nevertheless prudent to consider potential "real accidents" as well. This is because the methods of applying the mechanical and thermal inputs to a flask in a real accident will inevitably be different from those which arise during a Regulatory test. It is not sensible, to design for a Regulatory test without making allowance for "real accident" effects where they can sensibly be anticipated.
The designer should therefore take account of the effect of real accidents in the design of flasks to ensure that they have an adequate margin of safety in the real environment, as well as meeting the Regulatory requirements. This attention to real accidents should be pursued by designers taking account of local factors which may vary from country to country. By referring to these more tangible parameters the public may be better able to perceive the "true safety" of irradiated fuel flask movements, and it is in the interest of all users to seek this objective.

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DOE/PNC JOINT PROGRAMME ON TRANSPORTATION TECHNOLOGY*

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Abstract

DOE/PNC JOINT PROGRAMME ON TRANSPORTATION TECHNOLOGY.

The paper summarizes the work performed in a co-operative programme on transportation technology between the US Department of Energy (DOE) and the Power Reactor and Nuclear Fuel Development Corporation (PNC) of Japan. This work was performed at Sandia National Laboratories (SNL) in Albuquerque, New Mexico. The joint programme emphasized the safety analysis for truck transportation of special nuclear materials (SNM) in Japan. Tasks included structural analyses and testing, thermal testing, leak rate studies and tests, and transportation risk assessments. The paper presents the results of full scale structural and thermal tests conducted on a PNC developed SNM transport system. Correlation of full scale impact test results with structural analysis and scale model testing are also reviewed.

INTRODUCTION

In Japan, nearly all overland transport of nuclear materials, except for spent fuel, is by truck. PNC has developed an SNM highway transport system consisting of a tractor and a trailer carrying up to three shipping casks to transport materials such as mixed-oxide fuel. In this joint program, we assessed the response of the SNM transportation system under impact and fire conditions and performed regulatory drop and puncture tests of a

* This work was supported by the US Department of Energy under Contract No. DE-AC04-76DP00789.
breeder reactor fresh-fuel cask loaded with simulated fuel. The full-scale accident tests included a head-on 90 km/h (56 mph) collision of the cask transport system against a massive concrete barrier, 9 m (30 ft) drop and 1 m (40 in.) puncture tests of the prototype breeder reactor fresh-fuel cask, and 1 h fire testing of the impacted cask transport system.

STRUCTURAL ANALYSIS

Before full-scale testing of the transport system, structural analysis and scale model testing were performed to evaluate the response of the system under the specified impact condition. In the computer evaluation phase of the study, we modeled the system analytically with a one-dimensional, lumped-parameter computer code. In this type of analysis, the system is modeled with a number of mass and spring coupling elements (Figure 1). The force displacement characteristics of the couplings were estimated by using the cross-sectional area and load-deflection characteristics for each load-carrying member of the transport system.

The purpose of the analytical model was to provide estimates of the response of the system to impacts against a rigid barrier. In this type of analysis the lumped masses are given an initial
velocity against the rigid barrier, and the computer program solves the associated equations of motion and calculates the response of the system as a function of time. The results included the amount of crushup in the system, the time the system takes to come to rest, and the forces in various system elements, including the tiedowns. It was of primary importance to determine if the cask would impact the target, an event that may cause damage to the cask.

During the lumped-parameter evaluation, we varied several parameters to determine a possible range of behavior. (The precise force-displacement characteristics of the various elements in the system are not known.) For example, the crushup of the cab could only be approximated. The ultimate breaking strength of the fifth-wheel connection is another item that could only be estimated; it was found to be a parameter that significantly affects results.

After completing the evaluation, we concluded that, although the proposed 90 km/h impact test would completely destroy the tractor, it would at worst deform the front end of the trailer only slightly. The results also indicated that the cask would remain attached to the trailer and would not impact the target. Figure 2 shows typical results for the calculated velocity-time history of the cask in an impact of 90 km/h. It was calculated that the cask would come to rest in slightly less than 0.3 s. Displacement results indicated that the cask would move forward with the trailer about 4.5 m (15 ft) after the system impacted the target. These results were later compared to results obtained with scale models and from the full-scale testing described below.
SCALE MODEL TESTS

Using full-scale drawings from PNC, we fabricated two quarter-scale models of the transport system for the head-on impact tests. The purpose of the scale-model tests was to verify calculational results and to study possible system response in greater detail. In designing the scale models, we simplified the full-scale system while retaining pertinent structural features. The models included considerable structural detail and were constructed of materials similar to those of the prototype. Two complete quarter-scale models of the PNC system were designed and fabricated, including a detailed cask, a trailer, and a tractor (Figure 3). Note in Figure 3 that the cask is visible through the sides of the trailer structure. Clear panels were used to allow the cask to be viewed during the impact test. Quarter-scale tiedowns, which had not been defined by PNC, were designed and fabricated. The system was designed to run at a sled-track facility where it would be accelerated by rocket motors.

The first scale-model test consisted of a 90 km/h impact environment, and the second test evaluated the response of a revised tiedown and transportation system. Figure 4 is a sequence of photographs taken during a scale-model impact test. Both model tests indicated that the tractor would be demolished.
and the trailer crushed slightly. The cask would remain attached to its tiedowns. The model impact lasted approximately 0.07 s, which translated into a full-scale impact duration of 0.28 s (events in the larger prototype take longer by the scale factor). The results from the scale-model tests were used to assist in the development of the analytical model and to provide empirical estimates of the response of the full-scale test. These model tests also served as an aid to confirm the design of tiedowns for the full-scale prototype hardware.

**FULL-SCALE IMPACT TEST**

The purpose of the full-scale impact test was to verify the adequacy of analytical and physical scale modeling. We designed the test transport system components to provide an adequate model of the prototype transport system. A full-scale transport system was constructed using both existing and new materials. A tractor
made available from PNC was modified to match the structural characteristics of the actual unit. A new cab structure was specially designed and constructed to fit on the Japanese chassis. A special enclosed trailer was fabricated from simplified drawings of the prototype. The trailer frame was specially fabricated to model the PNC prototype. Structural I-beams of the correct size and shape formed the main frame of the system. Many types of structural shapes and plating were used to form the trailer system so as to conform closely to the PNC prototype. The trailer superstructure was designed with openings on the sides and top to allow viewing the cask during the test.

As mentioned above, the PNC system was designed to carry up to three shipping casks. Before testing, PNC personnel decided that the full-scale impact test should simulate a fully loaded system. For the full-scale test, the trailer contained three casks: two mass models (to simulate the weight of fully loaded casks) and an actual prototype breeder reactor fresh-fuel cask supplied by PNC. A basket and simulated fuel assembly were also provided by PNC and used in the test.

Before testing, the system was instrumented with many accelerometers and strain gages. The data from these transducers would be sent by telemetry to a recording system at the test site. In the actual test, the system was accelerated to speed at a sled-track facility [1] and allowed to impact a very rigid target. The event was recorded by many high-speed cameras to determine the detailed response of the system.

The full-scale test was conducted at a velocity of 90 km/h. The system, propelled by rocket motors, impacted a 625 metric ton concrete mass backed by 1580 metric tons of earth. System response was evaluated by examining the posttest hardware, by analyzing the high-speed films, and by assessing the results obtained from onboard instrumentation. Figure 5 includes a series of photographs taken from one of the high-speed films of the test. As predicted from the structural analysis and scale-model tests, the tractor was demolished and the trailer was partially crushed during impact. The front end of the trailer sustained damage well within the predictions. The impact lasted approximately 0.3 s, as predicted. The test cask remained with the trailer; the tiedowns did not fail. Posttest examination of the cask revealed only minor damage to the exterior of the body near the impact end. Leak testing of the cask after the test indicated that the seals were intact. The full-scale test demonstrated that (1) an impact of this magnitude will not damage the cask or cause loss of contents, (2) the cask will remain attached to the trailer while decelerating at approximately 28g, and (3) the cask contents will not be damaged in this type of impact.
FIG. 5. Full scale impact test.
FULL-SCALE DROP AND PUNCTURE TESTS

After the full-scale system test, regulatory drop and puncture tests of the prototype cask were performed to verify its design integrity. Two 9 m drop tests, side and center-of-gravity over corner, onto an unyielding surface verified that the shock absorbing system was adequate. Four 1 m puncture tests (side, pedestal, center-of-gravity over damaged corner, and angled) onto a 15.2 cm (6 in.) mild steel probe evaluated the adequacy of the exterior shell. In the side and center-of-gravity over corner drop tests, peak decelerations measured were approximately 200 g and 140 g, respectively. The series of drop and puncture tests caused local deformation of the cask body but did not threaten its fuel-containment capability. The two drop tests proved more serious than the puncture tests, both tests resulting in deformations and deceleration values much higher than recorded in the puncture tests. The damage did not extend to the inner lid. The cask maintained its leak tightness in all of these tests.

FULL-SCALE FIRE TESTING

The crash-tested trailer with two mass models and the drop- and puncture-tested cask were reassembled on pedestals for fire testing. The tractor and front face of the trailer were not included since they were both severely damaged in the 90 km/h impact test. The trailer and cask were instrumented with approximately 100 thermocouples to monitor temperatures at different locations throughout the test. Four water-cooled instrumentation towers were placed by the system. These towers contained thermocouples and calorimeters to characterize the fire environment.

We exposed the transport system to a nominal 1 h jet fuel (JP-4) fire in a 9 by 18 m (30 by 60 ft) pool. After the fuel was ignited, flames quickly engulfed the system and lasted 72 min, with the system fully engulfed in flame 96% of the time. Figure 6 shows the system after the fire test. The trailer structure was considerably warped. The trailer deck sagged on its sides, allowing the side dummy cask to move down with respect to the center cask. Temperatures on the cask varied from about 815 C to 1093 C (1500 F to 2000 F). In general the top surface of the cask was hotter than the bottom, and the front was hotter than the rear. Thermocouples located along the top of the cask showed average peak temperatures of 925 C (1700 F) and bottom was about 825 C (1525 F). The cask surface came up to temperature quite rapidly, remaining at high temperature for about 1 hr. The cask performed satisfactorily by maintaining its leak tightness throughout the test. Data from the fire will be used to verify a thermal analysis performed separately by PNC.
CONCLUSIONS

Analyses and tests conducted as part of the joint PNC/DOE program provided information about the response of the system to very severe hypothetical accident environments. The structural analysis work relating to impact of the system onto a rigid barrier (based on analytical and physical scale models) indicated that the vehicular system would be seriously damaged but the cask would remain undamaged on the trailer structure. These results were confirmed by a full-scale impact test demonstrating the usefulness of these analysis techniques to a very satisfactory degree. The full-scale drop and puncture tests indicated that the PNC cask will pass regulatory requirements. The cask survived the 1 h long fire test by maintaining its containment integrity. In summary, the posttest examinations of the hardware also demonstrated that the cask maintained its leak tightness throughout multiple events in the test series.
Reference

DEMONSTRATION TEST OF 100 TON CLASS SPENT FUEL TRANSPORT CASK

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Abstract
DEMONSTRATION TEST OF 100 TON CLASS SPENT FUEL TRANSPORT CASK.
The paper reports the results of reliability verification tests on spent fuel transport casks, which the Central Research Institute of Electric Power Industry (CRIEPI) has conducted under contract with the Science and Technology Agency of the Japanese Government. The demonstration test was performed on the 50 ton and 100 ton class casks. This paper particularly reports the test of the latter cask under transport accident conditions. The casks tested were designed, manufactured, and analysed with the most advanced methods practised in Japan and were developed by incorporating various features of casks currently used in Japan. The tests were carried out by a facility specifically prepared for this test in the Yokosuka Laboratory of CRIEPI in accordance with Japanese legal requirements. In addition to the general cask test, a high water pressure loading test, with pressures of 20 kg/cm² and 300 kg/cm² were conducted on a separate cask. Shield and leakage tests were performed before and after the loading tests. The results indicate that the tests under accident conditions and the high water pressure test had little effect on the cask, not impairing its integrity. The leakage tests and shield tests indicated that the casks had adequate performance characteristics. Thus the test results verified that the casks conform to the legal requirements, and the current design, manufacture and safety analysis methods are appropriate.

1. INTRODUCTION
The Central Research Institute of Electric Power Industry (CRIEPI) has been conducting, under contract with the Science and Technology Agency of Japan, the spent fuel transport cask reliability demonstration test since 1977 to verify the safety and reliability of spent fuel transport casks. The first phase of this test was completed in 1985.

In this demonstration test, 50 ton and 100 ton class casks designed and manufactured by current methods were subjected to tests to verify the suitability of the design and manufacturing techniques used through observation of behaviour of the cask and confirmation of its soundness. The casks were subjected to tests under normal conditions and under the accident conditions stipulated in Japanese regulations and IAEA standards, and also to pressure tests, which were performed from the point of view of safety in shipping, although this is not stipulated in the Japanese regulations.
The test results on the 50 ton class casks were reported in PATRAM '80 and PATRAM '83. The results of the 100 ton class casks are reported in this paper.

2. OUTLINE OF 100 TON CLASS CASK

The 100 ton class cask can accommodate 27 BWR or 10 PWR spent fuel assemblies. The gamma ray shield is provided by lead and stainless steel and the neutron shield by the cavity water and silicon rubber. The outside cylinder has radiator fins, and both ends of the cask are provided with shock absorbers which are made mostly of wood (Fig. 1).

The specification of the cask is as follows:

Weight: about 104 t including 10 PWR assemblies
Outer dimension: 300 × 6200 mm
Materials: cask body (SUS 304, Pb), fin (Cu coated with Ni), absorber (wood covered with steel), neutron shield (silicon rubber).

For this 100 ton class cask, the safety analysis report was prepared in a manner similar to that for the currently used casks and subjected to the safety review by Japanese experts in accordance with Japanese standards. The cask was assessed as usable.
3. TEST METHODS

The tests were performed according to the sequence illustrated in Fig. 2. The tests were performed on a facility which was installed in the cask testing yard in the Yokosuka Laboratory of CRIEPI. The testing methods are presented in Table I.

4. TEST RESULTS

The impact of various tests on the cask under normal test conditions is so small that there was nothing worth reporting. Thus only the test results under accident conditions and the pressure test are presented.

4.1. Tests under accident conditions

4.1.1. Drop test (I)

In drop test (I), the cask was dropped vertically with its bottom down, which is the severest condition. As a result of this impact, the shock absorber absorbed
<table>
<thead>
<tr>
<th>Test item</th>
<th>Test method</th>
<th>Measurement items</th>
</tr>
</thead>
<tbody>
<tr>
<td>Drop</td>
<td>Free drop from 9 m</td>
<td>Acceleration and strain of the cask main body and fuel assembly, and cavity pressure cask</td>
</tr>
<tr>
<td>Fire test</td>
<td>The cask is placed in a pre-heated furnace, and heated for 30 min at 800°C</td>
<td>Temperature of each part: cavity pressure</td>
</tr>
<tr>
<td>Pressure test</td>
<td>Performed in a high pressure water tank having 500 kg/cm² capacity: test conditions are 20 kg/cm² for 1 h and 300 kg/cm²</td>
<td>Displacement, strain, cavity pressure and water-tightness at sealing boundary (the inside of cavity is filled with air)</td>
</tr>
<tr>
<td>Heat transfer test</td>
<td>The cask is left in a hood having inside temperature of 38°C: the heat is generated by a heater simulating the spent fuel residual heat</td>
<td>Temperature of each part and cavity pressure</td>
</tr>
<tr>
<td>Leakage test</td>
<td>Vacuum test and He leak test</td>
<td>Measurement of vacuum or He quantity</td>
</tr>
<tr>
<td>Shield test</td>
<td>Cylindrical shell sources of ⁶⁰Co and ²⁵²Cf are used: ( \phi 50 \times 50 \text{ cm} ).</td>
<td>Neutron — rem counter, etc.: gamma ray — ionization chamber, etc.</td>
</tr>
</tbody>
</table>

the impact energy and deformed by approximately 130 mm, and the tip of the radiator fins deformed by 5 to 15 mm downwards, but no deformation was observed on the cask itself. The acceleration and strain of the cask body and the basket and the cavity pressure are presented in Table II. All the measured values were less than those obtained in the safety analysis.

4.1.2. Drop test (II)

In drop test (II), the cask cylinder was impacted on a steel bar with the cask in a horizontal posture. This gives the cask the severest impact as the cover and bottom of the cask are equipped with shock absorbers. As a result of this test,
the steel bar crushed the fin, penetrated the neutron shield (silicon rubber) between the fins and reached the fin structural ring. However, only local residual deformation was produced on the cask body. The strains created by the impact of the steel bar on the cask main body are presented in Table II. The measured values are sufficiently smaller than the analytical values in the safety analysis.

4.1.3. Fire test

The cask tested has a layer of silicon rubber on the outer surface of the cylinder which functions as the neutron shield. The fire test was originally designed to maintain a furnace temperature of 800°C, but hydrogen and carbon in the silicon rubber burnt, making the average furnace internal temperature over 1000°C and creating a very severe test environment. The cask was taken out of the furnace after 30 minutes in accordance with the related standard, but the silicon rubber continued to burn for 60 minutes. It was observed after extinction of the fire that the silicon rubber had turned to white ash and the fin tips were partially melted and lost. The wooden material of the shock absorbers was also burnt as fire entered from cracks on the outer steel plate that were formed in drop test (I). The wood continued to smoulder for five days after the test.

The temperatures of the O-ring component forming the sealing boundary in the fire test and of the lead gamma shield and the cavity pressure are presented in Table II. The values obtained in the experiment were smaller than those in the safety analysis, the maximum allowable material temperature, and the design pressure.

4.1.4. Water immersion test

The cask was immersed in water of 1.5 kg/cm² for 8 hours in the high pressure water test facility and it was confirmed that the cask was not affected.

4.1.5. Heat transfer test

The cask was left in a chamber with an environmental temperature of 38°C for one week, after the series of tests under accident test conditions were completed. The temperature of each part is as illustrated in Table II, being below the allowable limit. It was judged that the cask has adequate heat transfer characteristics despite changes of cask status caused by the special condition tests.
### TABLE II. TEST RESULTS

<table>
<thead>
<tr>
<th>Test item</th>
<th>Tested part</th>
<th>Measurement item</th>
<th>Measurement</th>
<th>Analytical value</th>
<th>Design basis value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Drop test (I)</td>
<td>Main outer cylinder</td>
<td>Acceleration (g)</td>
<td>67</td>
<td>100</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Axial strain (x 10^{-6})</td>
<td>140</td>
<td>455</td>
<td>860</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Circumferential strain (x 10^{-6})</td>
<td>213</td>
<td>296</td>
<td>860</td>
</tr>
<tr>
<td>Basket</td>
<td></td>
<td>Acceleration (g)</td>
<td>72</td>
<td>100</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Axial strain (x 10^{-6})</td>
<td>135</td>
<td>528</td>
<td>796</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Cavity internal pressure (kg/cm²)</td>
<td>24</td>
<td>25</td>
<td>-</td>
</tr>
<tr>
<td>Drop test (II)</td>
<td>Main outer cylinder</td>
<td>Axial strain (x 10^{-6})</td>
<td>80</td>
<td>117</td>
<td>860</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Extent of penetration (cm)</td>
<td>0</td>
<td>6.3</td>
<td>12</td>
</tr>
<tr>
<td>Fire test</td>
<td></td>
<td>Lead (°C)</td>
<td>205</td>
<td>206</td>
<td>327</td>
</tr>
<tr>
<td></td>
<td></td>
<td>O-ring (°C)</td>
<td>170</td>
<td>203</td>
<td>230</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Cavity internal pressure (kg/cm²)</td>
<td>11.8</td>
<td>34.7</td>
<td>65</td>
</tr>
<tr>
<td>Test item</td>
<td>Tested part</td>
<td>Measurement item</td>
<td>Measurement</td>
<td>Analytical value</td>
<td>Design basis value</td>
</tr>
<tr>
<td>---------------------------</td>
<td>-------------</td>
<td>------------------</td>
<td>-------------</td>
<td>------------------</td>
<td>-------------------</td>
</tr>
<tr>
<td>Environment transfer test</td>
<td>Lead (°C)</td>
<td></td>
<td>143</td>
<td>134</td>
<td>327</td>
</tr>
<tr>
<td></td>
<td>O-ring (°C)</td>
<td></td>
<td>144</td>
<td>163</td>
<td>180</td>
</tr>
<tr>
<td>Pressure test (kg/mm²)</td>
<td>Cavity internal pressure (kg/cm²)</td>
<td>6.0</td>
<td>13.0</td>
<td>—</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Cylinder</td>
<td>Outer surface</td>
<td>12.2</td>
<td>17.3</td>
<td>33.6</td>
</tr>
<tr>
<td></td>
<td></td>
<td>of outer cylinder</td>
<td>7.6</td>
<td>9.5</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Lid</td>
<td>Outer surface</td>
<td>25.0</td>
<td>23.5</td>
<td>50</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>25.6</td>
<td>23.5</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Bottom plate</td>
<td>Outer surface</td>
<td>14.3</td>
<td>16.2</td>
<td>50</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>14.2</td>
<td>16.2</td>
<td></td>
</tr>
</tbody>
</table>
4.1.6. Performance tests

(1) Leakage

Before and after each test under the accident conditions, the cask sealing boundaries (cover seal, vent valve and drain valve) were vacuum tested. In addition, the leakage of helium sealed in the dummy fuel assembly was checked to verify the integrity of the assemblies against the impact of the drop test. The test results indicated that the leakage rate to the vacuum was \((2-8) \times 10^{-4\text{ atm-cm}^3/\text{s}}\) after the special condition test, providing sufficient seal characteristics. No leakage of helium was detected.

(2) Shield

As the actual spent fuel cannot be employed for testing the cask shield characteristics, the evaluation of these characteristics was performed by an experiment employing radioactive isotope sources (\(^{60}\text{Co}\) and \(^{252}\text{Cf}\)) in combination with analysis. In the test, scanning measurements were performed on all surfaces to confirm the uniformity of the cask shield, and an absolute dose rate measurement was made.

The scanning measurement indicated that the shielding performance of the cask was as uniform after the accident condition tests as before them.

The dose rate at a distance of 1 m from the cask surface after the test was increased by a factor of about 4 in terms of neutrons and about 2 in terms of gamma rays owing to combustion of hydrogen and carbon in the silicon rubber. These results were consistent with the analytical values.

4.1.7. Conclusion

From the test results in 4.1.1-4.1.6, it was confirmed that the 100 ton class cask maintained its integrity and characteristics in conformance with regulations even after the accident condition tests. Thus it was judged that the design and manufacturing methods employed for this cask were appropriate.

4.2. Pressure test

The pressure test with 20 kg/cm\(^2\) for 1 hour and then 300 kg/cm\(^2\) hypothesizes sinking of the cask in ocean transportation. These tests were performed by a pressure test facility having a capability of 500 kg/cm\(^2\). The pressure test was performed in the absence of any components inside the cask which did not affect the structural strength of the body (such as fuel assembly, basket and cavity water). Strain gauges, displacement gauges and water leak detection devices were attached to the cask to measure the strain, displacement and sealing characteristics. In order to measure the strain and displacement
of the outside cylinder of the cask body, parts of the fins and neutron shield material were removed.

The test results were as follows:

1. No leakage was detected at any sealing boundary during the pressure test and no reduction in the sealing characteristics was observed by comparison with tests performed before and after the pressure tests;

2. The stress intensity of the cask main body was very small in the 20 kg/cm² - 1 hour test. Most of the cask body was within the elastic limit for the water pressure of 300 kg/cm², and the allowable stress intensity of the material was not exceeded at any part (Table II);

3. Deformation by the water pressure is 1 mm or less for the inside cylinder and 2 mm or less for the cover at 300 kg/cm²;

4. The shield tests performed before and after the pressure test indicated that there was no reduction in the shielding performance for gamma rays or neutrons.

It was judged for these test results that the cask has adequate pressure resistance characteristics.

5. CONCLUDING REMARK

On the basis of these results, it was demonstrated that the cask satisfies the conditions required by the rules and regulations. It was also confirmed that the design, fabrication and safety analysis methods which are generally used are appropriate.

The wet type cask was tested in this phase of the verification programme. This will be succeeded by a fire test of the dry type cask in 1986 and 1987.
SAFETY RESEARCH ON THE TRANSPORT OF RADIOACTIVE MATERIALS IN JAPAN

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Abstract

SAFETY RESEARCH ON THE TRANSPORT OF RADIOACTIVE MATERIALS IN JAPAN. The generation of electricity by commercial nuclear power plants has been steadily growing in Japan. With the increase in nuclear power plants and facilities related to the nuclear fuel cycle, many kinds of packages are expected to be required and the transport of nuclear fuel and radioactive waste materials will increase. It becomes more and more important to secure the safety of these transports. The paper explains the nuclear fuel cycles in Japan and the schedules of the new five year safety research programme formulated by the Japanese Nuclear Safety Commission.

1. INTRODUCTION

The generation of electricity by commercial nuclear power plants in Japan is six times as much as that of ten years ago and is increasing. Moreover, new types of reactor, such as ATRs and FBRs, are under development. The proportion of nuclear power generation is expected to increase in the future.

Figure 1 shows the nuclear fuel cycles in Japan. These include the effective use of LWRs, the reprocessing of the spent fuel, the utilization of plutonium, and the enrichment of uranium. In the figure, solid arrows denote the cycles in operation (high quantity), dash-dot arrows the cycles at research stages (or lower quantity), and the dashed arrows the cycles expected to be available in the future. Uranium hexafluoride or uranium oxide from outside Japan is transported to uranium conversion and fuel fabrication facilities to be made into fuel assemblies for LWR and new type reactors. These fuel assemblies are transported to each reactor. Part of the spent fuel is sent to a reprocessing facility at Tokai, and the rest to France and the United Kingdom. Uranium and plutonium recovered from the spent fuels in these countries is to be returned to Japan, together with radioactive waste materials.

New nuclear fuel cycle facilities, involving a reprocessing plant, uranium enrichment plant and low level radioactive waste storage facility, are expected
FIG. 1. Nuclear fuel cycle in Japan.
to be constructed at Shimokita in northern Japan in the early 1990s. The waste materials returned after reprocessing in France and the UK will be temporarily stored there. At that time, it will be necessary to transport new types of packages involving radioactive waste materials or natural uranium hexafluoride. With the improvement of LWRs, high burnup spent fuel packages will be needed.

Vessels and trucks using expressways are mainly used to transport the packages. Air transport is used for almost all radioisotopes for medical use.

In order to secure the safety of these types of packages and transport methods, the IAEA Regulations have been incorporated actively into Japanese regulations. Also, various research and development work in this field is being carried out by both the Government and private companies.

The Nuclear Safety Commission in Japan has formulated a new five year safety programme (1986-1990) to promote R&D coherently and comprehensively. The Government budgets in this field were $2.5 million for the 1983 Japanese fiscal year, $1.8 million for 1984, $2.1 million for 1985 and $2.9 million for 1986. Some of the main studies carried out in the past have been:

1. Experiments on lateral collision of vehicles carrying nuclear fuel packages
2. Experiments on secondary impact from falling with the use of scale models
3. Evaluation of cases of fire accidents involving tank lorry trucks
4. Demonstration tests to confirm the reliability of spent fuel packages
5. Experiments on pressure resistance of cylinders used for shipments of UF₆

Work in private companies in recent years has been concerned mainly with analyses and experiments that are necessary for evaluation of package designs for fresh and spent fuels. Also, studies related to packages for radioactive waste materials and spent fuel packages in dry condition and packagings for storage have been carried out.

Table I shows proposed research work on packages which is expected to be implemented from now on, including advanced research. The work involves computation methods on structure, heat, shielding and criticality to verify safety and experiments on the fire resistance of neutron shielding materials and on the impact characteristics of wood as a shock absorbing material. With regard to packaging design, some significant work has been done in the development of new types of packaging for the more highly enriched fuel or plutonium.

2. SCHEDULES OF SAFETY RESEARCH ON TRANSPORT OF NUCLEAR FUEL MATERIALS IN JAPAN

The safety research schedules of the new five year programme on the transport of nuclear fuel materials in Japan under the budget of the Japanese Government are shown in Table II.
<table>
<thead>
<tr>
<th>Research area</th>
<th>Package</th>
<th>Spent fuel package</th>
<th>UF₆ package</th>
<th>Plutonium package</th>
<th>Cast iron package</th>
<th>Radioactive waste package</th>
</tr>
</thead>
<tbody>
<tr>
<td>Packaging structures and materials</td>
<td></td>
<td>Drop tests of 48Y cylinder with and without overpack</td>
<td>Pressure test for cylinder</td>
<td></td>
<td>Characteristics of packaging materials at low, normal and high temperatures</td>
<td>Calculation for drop impact Tests for large TRU radioactive waste, low level solid waste package Tests of industrial package</td>
</tr>
<tr>
<td>Heat and fire resistance</td>
<td>Open fire test for package using resin as shielding material Overheat resistance for a short period</td>
<td>Fire test of 48Y cylinder with and without overpack</td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>Containment</td>
<td></td>
<td></td>
<td></td>
<td>Leakage test for O-ring materials</td>
<td></td>
<td>Long period sealing performance Lightweight package with double containment</td>
</tr>
<tr>
<td>Shielding and criticality</td>
<td>Shielding of γ-rays and neutrons emitted from high burnup spent fuel Criticality safety of high enriched fuel package</td>
<td></td>
<td></td>
<td>Establishment of shielding analysis system</td>
<td></td>
<td>Method for evaluation of radiation source Method for measuring neutron dose rate of re-processing radioactive waste</td>
</tr>
<tr>
<td>Characteristics of contents</td>
<td>Response behaviour of assembly against drop impact</td>
<td></td>
<td></td>
<td>Chemical reaction of UF₆–water Evaluation for leakage condition of UF₆</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Safety analysis codes</td>
<td>Structure analysis</td>
<td>Containment analysis</td>
<td>Criticality analysis</td>
<td></td>
<td></td>
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<td></td>
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<tr>
<td>Thermal analysis</td>
<td></td>
<td></td>
<td>Safety analysis code system, databank</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Shielding analysis</td>
<td></td>
<td></td>
<td>Development of Pu package resisting airplane accident</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

<p>| Development of Pu air transport packaging |                      |                                    | Demonstration test on safety of radioactive waste package |
| Demonstration tests |                      |                                    |                                    |
|                    | Demonstration test on safety of spent fuel package |                                    |                                    |
|-------------|------|------|------|------|------|-----------------------|
| 1. Research on packaging structure and materials | | | drop tests of scale models/similarity rule | | | MEL |
| Applicability of scale model | crush test/evaluation | | | | | MEL |
| Establishment of method for crush test | | | | | | |
| Evaluation of brittleness of metal materials at low temperature | method of measurement | evaluation based on fracture toughness at low temperature/high speed deformation characteristics | | | NRIM |
| Safety evaluation of cast iron packaging | comprehension of mechanical characteristics/non-destructive analysis/quality assurance criteria for | | | | NRIM |
| Structural integrity of package for radioactive waste | dynamic response | | | static response | SRI |
| 2. Research on heat and fire resistance | | | | | | |
| Thermal behaviour of UF₆ package | safety of UF₆ package with and without overpack | | | | PNC |
| Thermal behaviour under fire accident in tunnel | fabrication of tunnel model | fire test using scale model | | | FRI |
| 3. Research on containment | | | | | | |
| Leakage behaviour of radioactive material from packaging | basic study | | experimental study method/evaluation | | MEL |
| | on leakage | | | | PNC |</p>
<table>
<thead>
<tr>
<th>4. Research on shielding and criticality</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Shielding of package for high burnup spent fuel</td>
<td>measurement method for radioisotope ratio, intensity of source /analysis code</td>
</tr>
<tr>
<td>Evaluation of radiation in a spent fuel transport vessel</td>
<td>measurement of radiation dose distribution/calculation</td>
</tr>
<tr>
<td>Shielding study on transport of radioactive waste in a vessel</td>
<td>source data</td>
</tr>
<tr>
<td>Criticality safety study on package</td>
<td>calculation/experiment</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>5. Research on characteristics of content</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Effect of impact on spent fuel</td>
<td>irradiation test, material testing/drop test</td>
</tr>
<tr>
<td>Behaviour of spent fuel during transport in dry condition</td>
<td>oxidation test of cladding, UO₂ pellet/mechanical characteristics</td>
</tr>
<tr>
<td>Behaviour of leakage from damaged fuel pin</td>
<td></td>
</tr>
<tr>
<td>Study of behaviour of UF₆ package under accident</td>
<td></td>
</tr>
<tr>
<td>6. Research on safety analysis codes</td>
<td></td>
</tr>
<tr>
<td>Development of safety analysis codes for packages</td>
<td></td>
</tr>
<tr>
<td>Production of database, handbook</td>
<td></td>
</tr>
<tr>
<td>7. Research on development of Pu air transport package</td>
<td>test evaluation</td>
</tr>
<tr>
<td>8. Demonstration tests</td>
<td></td>
</tr>
<tr>
<td>Radioactive waste package</td>
<td>preliminary test fabrication of package normal test/accident test</td>
</tr>
<tr>
<td>Dry type spent fuel package</td>
<td>thermal conductivity fire test</td>
</tr>
</tbody>
</table>

| JAERI | SRI | PNC | CRIEPI |
2.1. Research on packaging structures and materials

2.1.1. Research on evaluation method for the demonstration test by the use of scale model

At the Mechanical Engineering Laboratory (MEL), the similarity rule for packaging has been studied by the use of scale models without shock absorbers and with a three-layer structure (carbon steel for the outer shell, lead for shielding and stainless steel for the inner shell). The rule will also be studied for other types of scale models of packaging with shock absorbers, taking into account the strain rate of the structural materials.

2.1.2. Research on structural strength of lightweight packagings

At MEL, the preliminary tests for the drop test III, stipulated in the 1985 Edition of the IAEA Regulations for the Safe Transport of Radioactive Material, have been carried out by the use of simulated models. Experiments and analyses will be carried out on the effect of the crush posture of the packaging to accumulate basic data and establish the evaluation method.

2.1.3. Evaluation of brittleness of packaging materials at low temperatures

At the National Research Institute for Metals (NRIM), a method will be established for evaluating the brittleness of packaging materials at the low temperature of $-40^\circ$C required for B(U) packages. Usually, the mechanical characteristics of materials at low temperatures are judged by their transition temperature in the Charpy impact test. Instead of this, testing and evaluating methods based on fracture mechanics will be developed for clarifying the dynamic fracture toughness of materials at low temperatures. The materials used are 0.1%C carbon steel, and nodular cast iron, which is now a focus of some attention. The behaviour of materials under high speed deformation will also be studied. From the results, the relationship between two sets of test data will be investigated.

2.1.4. Evaluation of the safety of cast iron packagings

At NRIM, the ultrasonic characteristics of microstructures of nodular cast iron will be studied to assure the quality of large cast iron packagings with thick shells.

At the Central Research Institute of Electric Power Industry (CRIEPI), non-destructive inspection methods for defects in cast iron will be formulated for quality control criteria.
2.1.5. Research on the structural strength of packagings for returnable waste materials

At the Ship Research Institute (SRI), the dynamic response of packages to impact and vibration in vessel transport will be evaluated by the use of computer codes in relation to shipments of waste materials returned from France and the UK.

2.2. Research on heat and fire resistance

2.2.1. Evaluation of thermal behaviour of UF₆ packages

At the Power Reactor and Nuclear Fuel Development Corporation (PNC), experiments will be carried out to find the time which a UF₆ package under fire accident conditions takes to reach the design temperature with and without an overpack in order to evaluate the fire resistance characteristics of the package.

2.2.2. Evaluation of the behaviour of packages under fire accident in a tunnel

At the Fire Research Institute (FRI), the fire temperature, the thermal convection and the fire duration time in a tunnel will be studied by the use of a tunnel model to clarify the thermal behaviour during a hypothetical tunnel fire accident of various types of package on a vehicle.

2.3. Research on containment

2.3.1. Research on leakage behaviour of radioactive materials from packagings

At MEL, the evaluation method for leakage behaviour of a gas from packagings will be studied experimentally by the use of a gap model which simulates sealed boundaries. A measuring method for the leakage behaviour of powders will also be studied.

At PNC, basic studies will be done on the leakage of radioactive materials from various types of package. Experiments will be carried out to find the sealing characteristics of the O-ring structures which are generally used.

2.4. Research on shielding and criticality

2.4.1. Research on the safety of package shielding for high burnup spent fuel

At the Japan Atomic Energy Research Institute (JAERI), a measuring and estimation method will be studied for the neutron source intensity of high burnup
spent fuels resulting from LWR improvement. Computer codes will also be
developed to evaluate the shielding.

2.4.2. Research on radiation shielding of spent fuel packages transported by vessel

At SRI, a method for evaluating neutrons and secondary gamma rays from
high burnup spent fuel and for determining the spatial distribution of radiation
dose rate will be studied in work aimed at protecting ships crews from radiation
exposure from spent fuel packages transported by vessel.

2.4.3. Research on criticality safety of packages

At JAERI, experiments and evaluations will be carried out on the sub-
criticality of packages in relation to the use of more highly enriched fuel for LWRs.

2.5. Research on the characteristics of contents

2.5.1. Research on impact in drop tests of spent fuel

At JAERI, the threshold values and the propagation behaviour of a pinhole
or crack in the fuel cladding will be studied to examine the integrity of sealing
characteristics in spent fuel package drop accidents. The behaviour of the fuel
assembly due to the damaging effect of water will also be studied.

2.5.2. Research on the behaviour of spent fuel transported in a dry condition

At JAERI, the impact strength and creep strength under internal pressure
of LWR fuel cladding during transport in a dry condition or for storage over a
long period will be studied experimentally. Experiments will also be carried out
to evaluate the effect of the oxidation of spent fuels in damaged fuel cladding
in air.

2.5.3. Leakage behaviour of radioactive materials from damaged fuel cladding

At JAERI, studies will be carried out on the leakage behaviour of spent
fuel materials and their fission products from fuel claddings with defects such as
pinholes, so as to measure the amount of leaching and the leaching rate into
water. A leaching model will be developed.

2.5.4. Research on the behaviour of UF₆ in an accident

At PNC, studies are being carried out to evaluate the impact on UF₆ packages
when they fall to the bottom of the sea in a shipwreck. The studies also include
experimental clarification of the mechanism of the chemical reaction between UF₆ and seawater, and the characteristics and the diffusion of its products.

2.6. Development of safety analysis codes

2.6.1. Development of safety analysis codes for packages

At JAERI, various types of safety analysis codes will be developed in relation to the structure, containment, heat, sealing, shielding and criticality of various types of packages.

2.6.2. Compilation of safety analysis database for packages

At JAERI, data related to packages will be collected to facilitate efficient safety examination of packages.

2.7. Research on the development of plutonium air transport packaging

At PNC, experiments will be carried out on the development of packagings for plutonium in air transport.

2.8. Demonstration tests

2.8.1. Demonstration test for the safety of radioactive waste packages

At CRIEPI, demonstration tests will be carried out on packages for high level vitrified wastes, and for low or medium level wastes, under normal and accident conditions, to verify the safety of packages of returnable waste materials generated by reprocessing in France and the UK.

2.8.2. Demonstration test for the reliability of spent fuel packagings

At CRIEPI, tests will be carried out on the thermal conductivity and fire resistance characteristics of dry packagings for spent fuels.

In the field of international co-operation, Japan has been participating in co-ordinated research programmes (CRP) on the application of the INTERTRAN code. We are planning to participate in the following research tasks in a new CRP:

(a) Studies on the behaviour of UF₆ in the environment and on the heat and fire resistance of UF₆ cylinders to ensure a safety margin through assessment in a test which is more severe than that of the IAEA Regulations
(b) Studies on improvement of the thermal analysis method for spent fuel packages
(c) Studies on establishment of shielding analysis for damaged industrial packages or Type A packages.

The PNC has been implementing a research programme on the development of air transport packages with the co-operation of the nations concerned.

3. CONCLUSION

Research in Japan on the safe transport of nuclear fuel is being carried out over a wide range and the results have been used effectively in the following fields:

(1) To improve safety guidelines and to collect technical data for the evaluation of safety, and

(2) To improve the safety measures of current transport systems and their components.

In the near future it will be necessary to systematize this research, and to allocate it to private companies. Thus, research projects will be promoted rationally and efficiently and will contribute further to securing the safety of package transport. International co-operation in this field is very important. We would like to co-operate with many countries and the IAEA in order to secure transport safety on a worldwide level, to reduce costs, and to increase the efficiency of research by exchanging results.
EVALUATION OF THE SAFETY FACTORS FOR RADIOACTIVE MATERIAL TRANSPORT PACKAGINGS UNDER SEVERE FIRE CONDITIONS.

The international Regulations for the Safe Transport of Radioactive Materials guarantee a high level of safety, achieved mainly by a packaging design which is suited to the potential risk inherent in the materials being transported. In fact, many of the packages used at present are designed so that they behave satisfactorily in environmental conditions which are much more severe than the regulatory conditions. Any risk evaluation should be made with data which are as realistic as possible, including the behaviour of the packaging in relation to the nature of the accident (fire, collision, immersion and so on) and the associated source term. This is a basic objective for the validation of codes such as INTERTRAN developed by the IAEA. For all these reasons, a databank is needed on the behaviour of the main packagings used. Fire is an example of an accident environment which should be considered for two reasons: the probability of fires occurring with durations and temperatures greater than the conditions applied to Type B packages ($800^\circ$C for 30 min) is not insignificant, particularly in the case of air and sea transport, and the associated risk may be considerable (transfer by inhalation). Studies have been conducted to identify safety factors for the packagings. These studies have been carried out in some cases by calculations and in others by tests. The paper presents the results obtained for different types of packagings; they show a high safety factor by comparison with regulatory fire conditions ($800^\circ$C for 30 min).

EVALUATION DES MARGES DE SECURITE DES EMBALLAGES DE TRANSPORT DE MATIERES RADIOACTIVES DANS DES CONDITIONS D'ENVIRONNEMENT DE FEU SEVERES.

Le Règlement international de transport des matières radioactives garantit un degré de sûreté élevé obtenu principalement par une conception de l'emballage adaptée au risque potentiel présenté par la matière transportée. En réalité, beaucoup de colis couramment utilisés sont conçus de telle sorte qu'ils présentent un état satisfaisant dans des conditions d'environnement beaucoup plus sévères que les conditions réglementaires. Toute évaluation de risque doit être faite avec des données aussi réalistes que possibles, concernant notamment le comportement des emballages en fonction de la nature de l'accident (feu, choc, immersion, etc.) et le terme-source associé. Ceci est un objectif fondamental pour valider des codes tels que le code INTERTRAN développé par l'AIEA. Pour l'ensemble de ces raisons, il est nécessaire de disposer d'une banque
de données sur le comportement des principaux emballages utilisés. L'incendie est un cas d'environnement accidentel important à considérer pour deux raisons: la probabilité d'occurrence d'incendies ayant des durées et des températures supérieures aux conditions appliquées aux colis de type B (800°C, 30 min) n'est pas négligeable, en particulier pour le transport par voie aérienne et maritime, et le danger associé peut être important (voie de transfert par inhalation). Des études ont été entreprises dans le but de connaître les marges de sécurité des emballages. Ces études ont été réalisées, dans certains cas par des calculs, dans d'autres cas par des essais. Ce mémoire présente les résultats obtenus pour différents types d'emballage, résultats qui font apparaître une grande marge de sécurité par rapport aux conditions de feu réglementaires (800°C, 30 min).

1. INTRODUCTION

Les réglementations internationales concernant le transport des matières radioactives garantissent un haut niveau de sûreté. Ceci est principalement obtenu en adaptant la conception de l'emballage au risque potentiel. Néanmoins, il y a lieu de considérer que certains accidents sont plus sévères que ceux représentatifs des critères de référence; la probabilité annuelle de rencontrer de telles conditions, compte tenu du trafic concerné, peut être de l'ordre de grandeur des probabilités d'accidents qui sont prises en compte dans la conception des installations nucléaires.

Les emballages de type B disposent généralement de marges de sécurité par rapport aux accidents représentatifs des tests B, marges qui, malheureusement, ne peuvent être appréciées à partir des seuls tests réglementaires. La connaissance de ces marges de sécurité, du mode de ruine et du terme-source associés est un élément indispensable à l'évaluation du risque. L'utilisation de codes d'évaluation de risques pour le transport, tel que le code INTERTRAN développé sous l'égide de l'AIEA, n'a de sens que si elle est basée sur des valeurs réalistes du comportement des colis. Il en est de même pour l'utilisation de méthodes telles que les méthodes coût-bénéfice, ou l'analyse multicritère couplée avec les méthodes d'évaluation de risque, en vue de sélectionner les options les plus avantageuses pour diminuer le risque.

Toutes ces raisons conduisent à disposer d'une banque de données du comportement des colis allant jusqu'à la ruine. Un des scénarios important à considérer est le comportement à l'incendie des colis pour les deux raisons principales suivantes: la probabilité d'occurrence d'incendie de sévérité plus grande que celle prise en compte dans la réglementation (800°C, 30 min) n'est pas négligeable, notamment pour les transports par voie maritime et aérienne, et le terme-source en cas d'ouverture (formation d'aérosols) constitue une des sources d'atteinte à l'individu les plus dangereuses.
2. COMPORTEMENT A L'INCENDIE D'UN EMBALLAGE DE TRANSPORT D'ELEMENTS COMBUSTIBLES IRRADIES

2.1. Introduction

Un certain nombre de colis de combustibles irradiés sont transportés par voie maritime. La question ayant été posée du comportement des emballages dans l'hypothèse d'un incendie non maîtrisé en mer, on a évalué la marge de sécurité dont on dispose sur un emballage de type TN-12 (fig. 1) couramment utilisé en France pour le transport des éléments combustibles de REP.

L'évaluation a été faite par calcul. On a établi la carte des températures dans les différentes parties du colis et notamment au niveau du joint en fonction du temps. Les joints utilisés sont en Viton dont l'intégrité est garantie par le constructeur jusqu'à 316°C.
2.2. Calculs

Le calcul a été fait pour une puissance transportée de 120 kW, puissance prise en compte dans l’agrément, alors que la puissance transportée ne dépasse pas actuellement 80 kW pour des raisons d’exploitation de l’usine de retraitement.

La mise au point de la modélisation du colis a été faite à l’aide des mesures de référence obtenues au cours de l’épreuve thermique. Le code bidimensionnel DELFINE permettant de représenter l’emballage par un modèle axisymétrique a été à cette fin utilisé.

Seule la moitié supérieure du colis a été étudiée en imposant un flux nul sur le plan médian.

Le colis a été schématisé par un maillage comportant 3030 nœuds et 2720 éléments. Les conditions limites retenues sont conformes à la réglementation. En ce qui concerne les frontières externes en plus du rayonnement, les phénomènes de convection sont pris en compte dans le milieu à la température de l’incendie.

Les hypothèses suivantes ont été retenues:
— température initiale et après refroidissement: 38°C;
— température de palier: 840°C pour des durées de 30, 60 et 90 min;
— durée de montée et de descente en température: 3 min.

Pour chaque palier, le calcul a été poursuivi jusqu’à ce que la température au niveau des joints décroisse.

2.3. Résultats

2.3.1. Phase de l’incendie

Le balsa ne commence à être traversé par le flux de chaleur qu’à la fin d’un incendie de 90 min et seulement dans ses parties les moins épaisses. La chaleur pénètre par les parties métalliques exposées à l’incendie (ailettes) et se propage à travers le corps en acier à la fois radialement vers le centre et axialement vers les joints (les isothermes montrent le contournement du balsa).

2.3.2. Phase de refroidissement

a) Comportement général:

Les parties internes du balsa, le bouchon et les zones du corps protégées par le balsa voient des variations de température infimes par rapport aux variations externes. L’acier a une bonne diffusivité et les phénomènes externes se propagent très rapidement dans la direction radiale.

b) Comportement du joint supérieur (J₀ sur la figure 1):

Le joint est situé dans la région protégée par le balsa. Dans les cas étudiés, le joint n’atteint pas la limite de 316°C. La durée de l’incendie a peu d’influence
sur le temps à partir duquel la température du joint diminue. Par contre, une prolongation de feu de 30 min produit une augmentation de température au niveau du joint de 15°C environ.

c) Comportement du joint inférieur (J₁ sur la figure 1):
Les temps pour atteindre les températures maximales sont plus courts que pour le joint supérieur (environ 6 h après l'extinction de l'incendie).

2.4. Durée maximale de l'incendie pour atteindre la température limite admissible

Il est difficile de donner une idée précise de la durée d'incendie pour laquelle le joint atteint une température de 316°C lors du refroidissement de la structure.

En effet, aux phénomènes étudiés, s'ajoute en cas d'incendie prolongé le flux radial à travers le balsa. Ce flux correspond au transfert dans l'acier de la quantité de chaleur jusque là accumulée dans le balsa.

Les résultats montrent que la température du joint croît au cours de l'incendie d'environ 15°C toutes les 30 min. C'est à cette valeur que nous estimons l'effet du flux dans le balsa.

Considérant que le flux dans l'acier n'intervient pas dans cette approche, son effet propre ne change pas: il est de 15°C par 30 min. L'effet global est donc de 30°C par 30 min.

Cette évaluation pessimiste conduit à une durée d'incendie admissible de 3 h. Par contre, si on néglige l'apport de chaleur dans le balsa, l'augmentation de température reste de 15°C par 30 min. La durée maximale est alors de 5 h (ce qui donne une température de 319°C). La véritable limite se situe entre 3 et 5 h.

2.5. Conclusion

Les joints principaux de l'enveloppe de confinement ne sont pas très sensibles à la durée d'un incendie à 800°C: la température n'atteint pas la valeur limite de 316°C avant au moins 3 h d'incendie.

Au cours de l'incendie prévu par la réglementation pour les études de feu (800°C pendant 30 min), la température atteint seulement 183°C. Les températures maximales au niveau du joint sont atteintes environ 10 h après l'extinction de l'incendie.

3. COMPORTEMENT DE L'EMBALLAGE FS-47 DE TRANSPORT DE PuO₂ A UN INCENDIE DE LONGUE DUREE

3.1. Introduction

L'emballage FS-47 (type B fissile) a été développé en France pour le transport d'oxyde de plutonium sous forme de poudre. On a évalué, à l'aide du
code Delfine, le comportement thermique du colis soumis à un incendie de durée (30, 60, 90 min) et de température (800 et 1000°C) variables.

3.2. Description du colis

D'une hauteur de 2 m environ et d'un diamètre de 0,75 m, sa masse est, à pleine charge, de l'ordre de 1500 kg (fig. 2).

3.3. Calculs

La mise au point de la modélisation du colis a été faite à l'aide des mesures de référence obtenues au cours de l'épreuve thermique, un lest de barytine simulant le PuO₂.

Des simplifications ont été faites. Elles sont liées à l'utilisation du code Delfine, à la nécessité d'utiliser un modèle axisymétrique et à la difficulté d'associer des matériaux de diffusivité très différentes dans un transitoire rapide.

Le maillage comprend 2626 nœuds, et 2349 éléments rectangles. La taille des mailles permet un pas de temps de 300 s lors des transitoires rapides de début et de fin d'incendie.
La principale difficulté de la modélisation est la détermination des différents jeux entre les différentes pièces constituant la partie courante. Les jeux ont été maintenus constants.

Les autres points délicats sont:
— l’incertitude sur les propriétés thermiques équivalentes de la région ailetée (compound + cuivre);
— la méconnaissance de la barytine: ce produit a été choisi pour remplacer la poudre de PuO₂ pour l’essai de chute; il a la même masse volumique que la poudre de PuO₂, mais ses propriétés thermiques sont mal connues.

Les conditions prises en compte sont:
1) Les échanges externes: toute la surface du colis échange de la chaleur avec le milieu extérieur par convection et rayonnement.
2) Les échanges internes: dans tous les jeux, les surfaces en regard échangent par rayonnement face à face et par conduction dans la lame d’air.
3) La puissance totale dégagée, égale à 77 W, répartie uniformément dans la poudre de PuO₂.
4) La température initiale ambiante: 8°C.
5) Le palier de température d’incendie: 800°C puis 1000°C.

3.4. Résultats du comportement à l’incendie du colis FS-47

3.4.1. Incendie à 800°C (durée variable: 30 min, 60 min, 90 min)

Pendant l’incendie, le capot de balsa constitue une protection thermique très efficace. La région des joints est moyennement protégée par ce capot. En effet, si l’on considère l’incendie de 90 min, on peut retenir les températures suivantes:
— région la mieux protégée (entre capot et bouchon): moins de 60°C;
— région des joints: 100 à 150°C;
— virole interne: 150°C et plus.

Pendant le refroidissement, certains points continuent de s’échauffer pendant quelques heures après l’extinction du feu. La température maximale du joint est de l’ordre de 150°C pour un incendie de 90 min, cette température étant atteinte au bout de 2 h après arrêt de l’incendie.

3.4.2. Incendie à 1000°C (durée: 90 min)

Au niveau du joint, la température maximale atteinte au bout du même temps n’est supérieure que de 20°C, soit 170°C.

3.5. Conclusion

La température des joints, dans le cas le plus défavorable (1000°C, 90 min), n’atteint pas 170°C et reste donc très inférieure à la limite acceptable de 316°C. On dispose d’une marge importante sur l’étanchéité du colis.
4. **COMPORTEMENT D'UN EMBALLAGE DE TYPE A AU COURS D'UN INCENDIE: GENERATEUR DE TECHNETIUM**

4.1. Introduction

Le générateur Elumatic III à usage médical est un système automatique hautement protégé qui permet d'obtenir aisément une solution stérile et apyrogène de technétium 99m sous forme de pertechnétate de sodium. Cette solution est éluée à partir d'une colonne chromatographique d'alumine sur laquelle est fixé le 99Mo de fission (T = 66 h), parent de 99Tc (T = 6,02 h).

L'Elumatic III est livré en fût métallique étanche: c'est un colis de type A (fig. 3).

Dans le cadre de l'exportation vers l'Italie, un trafic important de ces générateurs a lieu en empruntant le tunnel du Mont-Blanc d'une longueur de 12 km. La prise en considération de l'éventualité d'accidents provoquant un incendie a conduit à évaluer le risque résultant de ce trafic. Dans ce but, des essais ont été entrepris pour connaître le comportement de ces colis à un incendie. On a considéré l'incendie de référence de la réglementation AIEA pour l'emballage de type B (800°C, 30 min).

4.2. Programme d'étude

Une préétude a été faite et a permis de montrer que le produit ne se décompose pas au cours d'un incendie. L'analyse thermogravimétrique et l'analyse thermique différentielle ont donné les résultats suivants: changement de phase à...
l'état solide: $T_f = 690^\circ C$; pas de décomposition du produit jusqu'à $800^\circ C$; pas de perte de poids du produit à cette température pendant toute la durée du palier (2 h).

Un essai de tenue à l'incendie a été fait sur le colis Elumatic III instrumenté de 11 thermocouples.

4.3. Résultats et observations

Après une minute d'exposition au feu, on observe l'ouverture du couvercle du fût due à la surpression interne: les pièces en polystyrène composant le calage du générateur s'embrasent au contact de la flamme du foyer.

Au bout de 16 min débute la fusion du plomb, avec écoulement de ce dernier vers l'extérieur au bout de 23 min.

Après 24 min d'incendie, la température au niveau de la colonne en verre (pyrex recuit) contenant l'alumine sur laquelle est fixé le $^{99}\text{Mo}$ de fission a atteint $675^\circ C$; cette température est à peu près stable jusqu'à la fin de l'essai.

Après refroidissement naturel, à l'issue des 30 min d'incendie, les observations suivantes ont été faites sur le colis. Le conteneur en nylon et le calage en mousse de polyéthylène se sont entièrement consumés, 13,3 kg de plomb se sont complètement écoulés; on note par ailleurs une légère déformation du tube en pyrex et la disparition des bouchons de caoutchouc et des capsules métalliques de fermeture de la colonne chromatographique, mais le filtre en verre fritté et le bouchon en laine de verre sont restés en place et n'ont pas été détériorés, ce qui a permis à l'alumine de rester à l'intérieur du tube, assurant le confinement du molybdate de sodium.

4.4. Conclusion

Cet essai a permis de mettre en évidence que la protection biologique de l'emballage Elumatic III n'est plus assurée au bout de 16 à 17 min pour un feu de $800^\circ C$. Par contre, il n'y a pas de risque de dispersion sous forme d'aérosols de la matière radioactive transportée.

5. COMPORTEMENT À L'INCENDIE D'UN CONTENEUR 30B, REMPLI D'UF$_6$, AVEC COQUE PF1 (fig. 4)

5.1. Introduction

Les essais avaient pour objectif de vérifier le comportement de l'ensemble conteneur-coque-bouteille échantillon aux épreuves réglementaires de chute et de feu prévues pour l'agrément de type B, prouvant la capacité de résister aux accidents en cours de transport, ceci dans les conditions d'un remplissage avec de l'hexafluorure d'uranium.
5.2. Programme et résultats des essais

Le même ensemble conteneur-coque-bouteille échantillon a subi les trois épreuves successives ci-après:
- chute libre de 9 m;
- chute de 1 m, sur poinçon;
- incendie de 30 min, le plan de joint des deux demi-coques étant placé verticalement de façon que la zone affectée par les deux épreuves précédentes soit en regard des flammes.

Le but final était de réaliser les essais sur un conteneur rempli d’UF₆ mais, pour des raisons de sécurité de l’essai, une approche méthodique par étapes a été retenue.
a) Essais avec lest inerte simulant l'UF₆
Le lest simulant l'UF₆ était constitué d'un mélange de billes d'acier de 3 mm de diamètre et de paraffine.
Les essais de chute et de poinçonnement ont permis de vérifier la bonne tenue de l'ensemble.
Pour l'essai d'incendie, l'ensemble conteneur-coque-bouteille échantillon était instrumenté de 28 thermocouples. Le feu a été prolongé jusqu'à 90 min pour estimer les marges de sécurité.

b) Transposition des résultats à l'essai avec l'UF₆
A partir des résultats précédents, un calcul a été effectué et a montré que les températures atteintes par le conteneur dans l'essai avec l'UF₆ resteraient très inférieures, au bout de 30 min, à la température maintenue au moment du remplissage ou de la vidange (115°C) et que, dans le cas le plus pessimiste, la masse d'UF₆ fondue serait très limitée.

c) Essais avec l'UF₆
Un nouvel ensemble conteneur-coque-bouteille échantillon a été utilisé. Le conteneur 30B était rempli de 2150 kg d'UF₆ appauvri. Les essais de chute et de poinçonnement ont confirmé les résultats obtenus aux essais effectués avec lest. Pour l'essai d'incendie, une aire spéciale a été construite et équipée de différents moyens pour faire face à toute fuite d'UF₆. Le conteneur a été aménagé, afin de déceler rapidement toute fuite d'UF₆, en particulier au niveau de la vanne et des bouchons, et canaliser la fuite éventuelle à l'extérieur du bac de kérosène. L'ensemble à tester était instrumenté de 32 thermocouples (fig. 4). Au terme des 30 minutes d'incendie:
- aucune fuite d'UF₆ n'a été constatée;
- la température maximale sur le conteneur est de 73°C, avec une moyenne de 43°C sur les 9 points de mesure (elle passe par un maximum de 75°C six minutes après l'arrêt du feu);
- les températures sur les collecteurs, à proximité de la vanne et du bouchon de vidange, ainsi que sur la bouteille échantillon sont respectivement de 34°C, 72°C et 16°C. Cette dernière passe par un maximum de 82°C dans la phase ultérieure, le bois contenu dans les extrémités de la coque continuant de se consumer.

Après l'essai, les parties des tôles extérieures de la coque prenant appui sur le bâti-support étaient légèrement enfoncées. Ceci est dû à un affaissement du colis sur le bâti par diminution, pendant l'incendie, des caractéristiques mécaniques de la mousse phénolique contenue dans la coque.
Après ouverture de la coque, il a été constaté que:
- le conteneur 30B n'a subi aucun dégât pendant l'essai d'incendie;
- le plan de joint de la coque, côté feu, est presque entièrement carbonisé, ainsi qu'une bonne partie du bois contenu aux extrémités des demi-coques;
- la boîte de protection en bois et la bouteille échantillon sont intactes.
5.3. Conclusion

Les résultats obtenus sont très satisfaisants. Les dégâts occasionnés par les essais de chute et de poinçonnement sont très limités et ne compromettent en rien la tenue mécanique du conteneur. Par ailleurs, au cours de l'incendie, les températures maximales relevées sur l'extérieur du conteneur 30B restent très inférieures à la température normale de remplissage et l'étanchéité de celui-ci est maintenue.

Les résultats ont permis de valider le code de calcul utilisé et d'extrapoler l'évaluation pour des durées d'incendie plus longues. Avec une durée d'incendie de 90 min, on disposerait encore d'une marge de sécurité appréciable.
RESEARCH AND DEVELOPMENT WORK ON RADIOACTIVE MATERIAL TRANSPORT IN THE FEDERAL REPUBLIC OF GERMANY

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Abstract

RESEARCH AND DEVELOPMENT WORK ON RADIOACTIVE MATERIAL TRANSPORT IN THE FEDERAL REPUBLIC OF GERMANY.

The paper presents a survey of research and development programmes on the transport of radioactive materials in the Federal Republic of Germany. The so-called 'Projekt Sicherheit der Entsorgung' sponsored by the Ministry of Research and Technology was completed in 1985. The aim of this project was to develop a methodology for safety analyses of the transport of radioactive materials under the special conditions in the Federal Republic. Initiated by the IAEA, a programme was started to analyse with the aid of the INTERTRAN computer code incident free transport and transport with hypothetical accidents. Research and development activities related to nodular cast iron Type B(U) casks for the transport and storage of radioactive materials have been continued. These types of casks will be used not only for transport and storage of spent fuel elements from light water reactors but also for other kinds of radioactive material, for instance high level radioactive waste or spent pebble bed reactor fuel. The Ministry of Research and Technology has sponsored a project on a new package design for plutonium nitrate transportation; the results of the first design study will be reported at the end of 1986. A programme has been started on the transportation of low level radioactive waste to the Konrad geological repository sited in Lower Saxony. A classification of different low level waste materials has been given and a comparison made between the different categories and the IAEA Regulations for the Safe Transport of Radioactive Material. The operation of the Konrad repository will start at the end of the 1980s.

Transportation of radioactive materials provides a necessary connection between the various steps of the nuclear fuel cycle. Since the last PATRAM Symposium in 1983, nuclear fuel cycle activities in the Federal Republic of Germany have grown. Transport of all kinds of radioactive material has increased and become more important.

By June 1986, 17 nuclear power stations with a power of 17.2 GW(e) were in operation in the Federal Republic. At the end of that year another light water reactor is planned to start operating and the total power will be 18.5 GW(e). Nuclear power stations produce nearly 40% of electric power in the country. Other nuclear fuel cycle facilities are under construction or in operation. At the front
end of the fuel cycle an enrichment facility started up in 1985. This type of facility works on the basis of the centrifuge principle. It is located at Gronau near the Federal German/Dutch frontier and the project is sponsored by the UK, the Netherlands and the Federal Republic. Over the three years 1983–1986 an important number of facilities at the back end of the nuclear fuel cycle were under construction:

- intermediate repositories for nodular cast iron transport and storage casks situated at Gorleben and Ahaus;
- a reprocessing plant at Wackersdorf, Bavaria, with a capacity of 350 tons/year;
- planned disposal facility for low radioactivity, non-heat-generating waste in the Konrad iron mine;
- other facilities such as a disposal plant for high level radioactive waste and a fuel element conditioning facility in Lower Saxony.

The expansion of the nuclear industry in the Federal Republic of Germany implies an increase in the transport of radioactive materials. This fact is shown by two examples. Transportation of spent fuel casks by rail increased strongly over the last 10 years. In 1975, only 4 rail shipments of spent fuel casks were carried out; in 1983 the number was 89. A similar increase exists in the case of uranium hexafluoride (UF₆) and fresh fuel elements. For instance, in 1985 the transport capacity of UF₆ in the Federal Republic was 2100 tons with 110 road transports and 45 by rail. These facts document the importance of the transport of radioactive materials.

Various research and development programmes in this area are being sponsored by the Federal Ministry of Research and Technology, the Ministry for Nuclear Safety or the industry itself.

First, a retrospective view of the work over the period 1983–1986 will be given. An important point was the 'Projekt Sicherheit der Entsorgung PSE,' sponsored by the Ministry of Research and Technology. Detailed information on this programme was presented at PATRAM '83 in New Orleans [1, 2]. The final paper of the programme was published in 1985 [3].

The aim of the PSE was to develop a methodology for safety analyses of the transport of radioactive materials under the special conditions in the Federal Republic of Germany.

Both incident free transportation and possible accidents had to be taken into account. In the first case, results for the contribution due to direct exposure to gamma and neutron radiation during incident free transportation were presented. For the second case a risk analysis was prepared and an estimation of the radiation exposure due to a transport accident was published. The following are some of the important results of the PSE project:

- A review of the accident statistics for road and rail transportation of dangerous goods in the Federal Republic of Germany;
- The calculation of forces acting on different packages during a hypothetical accident;
— Derivation of radiation exposure in the case of a hypothetical accident;
— Derivation of radiation exposure during incident free transportation;
— A comparison with other projects, i.e. the so called 'Projekt andere Entsorgungstechniken РАЕ' (project on a geological repository for spent fuel elements).

The radiological impact on different population subgroups resulting from the transport of radioactive materials has been assessed with the aid of the INTERTRAN computer code [4]. Incident free transport and transport involving accidents have been analysed for an important subset of radioactive materials within the Federal Republic of Germany. The study concerned well known types of packaging, and common physical and chemical forms of radioactive material, with standard radionuclide content, transport routes and annual number of transports.

A major advantage of the INTERTRAN computer code is that the analyst has a program at hand which models the radiation exposure resulting from external radiation for incident free transport and from possible dispersion of radioactive material in the event of accidents. The analyst can therefore fully concentrate on acquiring appropriate parameter values for the INTERTRAN model. The quality of the input parameters strongly determines the quality of the results obtained. In addition, the importance of different parameters in relation to the radiological risk evaluated by the program can be readily determined by parameter variations.

Since INTERTRAN supplies default values for almost all parameters used in the program, a gradual replacement of these parameters is possible as the analyst proceeds in determining parameters most appropriate to the special conditions of transport in his country or improved values from measurements and experiments. Such parameters include, for instance:

— Population densities and fractions of the route in different population zones;
— Population zone dependent parameters like transport velocity, traffic count, shielding factors or accident rates;
— Severity classification of impacts including probabilities of occurrence;
— Quantity and character of released materials;
— Atmospheric dispersion and dosimetric parameters.

The procedures used to derive appropriate parameters for the subset of transported radioactive materials have been considered. The parameter values derived reflect conditions specific to the Federal Republic of Germany as well as more general properties of the transported materials.

Research and development activities related to nodular cast iron Type B(U) casks for the transport and storage of radioactive materials have been continued. Type B(U) casks of nodular cast iron are needed in the Federal Republic for the dry storage of spent fuel elements in an intermediate repository away from a nuclear power station. Over the last 10 years there has been progress in the work
on package design to meet the general requirements of the IAEA Transport Regulations. For both types of nodular cast iron casks Castor and TN-1300, a large number of tests have been carried out and many papers have been published since the 1980 PATRAM Symposium in Berlin (West) [5-7]. In the near future this type of nodular cast iron cask will be used not only for spent fuel elements from LWRs but also for some other kinds of radioactive material. Examples of cask usage are:

- Type B(U) casks for the transport and storage of spent fuel elements from the Super Phénix reactor in France;
- Transport and storage casks for vitrified high level wastes Castor HAW 21, and transport casks HAW and Castor HAW C for liquids;
- Type B(U) transport and storage cask TN-900 for spent pebble bed reactor fuel.

In the Federal Republic of Germany eight different types of nodular cast iron casks have a Type B(U) licence issued by the Physikalisch-Technische Bundesanstalt in Braunschweig. The competent authority for all investigations of specific design requirements is the Bundesanstalt für Materialprüfung in Berlin (West). Two or more new Type B(U) licences are expected for the year 1986. For the nodular cast iron casks Castor and TN-1300, nearly 30 different types are under construction or development. Further improvement of the leaktightness of lids and gaskets for dry transport and storage casks are necessary. For intermediate storage for some 10 years, metal gaskets are preferred. Another aim of the programme is the determination of the Type B(U) leakage rate after severe transport accidents.

Reactor plutonium from the reprocessing plant Cap de La Hague in France is transported with a special security truck in the chemical form of plutonium oxide. In a few cases, small amounts of plutonium nitrate are transported from the Federal German reprocessing plant WAK, Karlsruhe, to the fuel element fabrication plant in Hanau.

For the fabrication of mixed oxide fuel elements the use of plutonium nitrate will have some advantages. The Ministry of Research and Technology has sponsored a project planning a new package design for plutonium nitrate transportation. A study on the evaluation of existing investigations on the transport of plutonium [8] has been carried out. The authors of the report summarize fundamental conditions for a new package design for the transport of plutonium nitrate.

As a result of the mechanical and thermal behaviour of a Type B(U) accident resistant package design for plutonium there is no difference in the risk assessment for the transportation of plutonium in solid or in liquid form. An example of a new package design for plutonium nitrate will be given at the end of 1986.

At the beginning of 1985 a programme was started on the transportation of low level radioactive waste from different nuclear fuel cycle facilities to the Konrad geological repository. As a first step, a classification of different low level radioactive waste materials was made. Packaging materials were selected according to the type of waste, the waste producing facility, the activity content and the representa-
tive mixture of radionuclides. In the second step, a comparison was made between
the different waste categories and the rules for transportation laid down in the
the completion of this programme it should be possible to reduce the radiation
exposure during incident free transportation within the nuclear fuel cycle. For the
transportation of low level radioactive waste to the Konrad geological repository,
standardized packages will be used. The majority will be strong industrial packages
for low specific activity material, Type LSA-II and LSA-III. Only 10\% of all
packages will be Type B. The operation of the Konrad repository will start at the
end of the 1980s.

The aim of all research and development programmes has been to guarantee
the safety and security of the transport of radioactive materials. Up to 1986 there
were no deaths or injuries attributed to the release of radioactive materials during
the transportation of nuclear fuel cycle material in the Federal Republic of
Germany.

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(1985).


these Proceedings, Vol.1.
RADIATION PROTECTION
AND RISK ASSESSMENT
RISKS OF PLUTONIUM TRANSPORTATION

A comparative literature survey*

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Motor Columbus Consulting Engineers, Inc.,
Baden, Switzerland

Abstract

RISKS OF PLUTONIUM TRANSPORTATION: A COMPARATIVE LITERATURE SURVEY.

Several risk analyses have been carried out in different countries to answer the questions: what are the risks of plutonium transportations and are they acceptable? Many of them are evaluated thoroughly in the paper and compared with each other, the different boundary conditions of each being taken into account. The main goal of this comparative survey was to estimate the risk of transporting plutonium within the total risk of nuclear energy generation and in comparison with other technical and natural risks. An additional goal was to work out the difference — if any — between the transportation risks of the two chemical forms (Pu oxide powder and Pu nitrate solution). The most important results of this study are: (1) the potential risk of Pu transportation is less than 0.1% of the total risk associated with nuclear energy production with Pu recycling; (2) the chemical form of Pu has no significant influence on transportation risks; (3) the use of large transport containers for Pu nitrate solutions does not increase the transport risks; there are even several advantages; (4) the transport mode (rail or road) does not influence the transportation risks significantly; (5) risks of Pu transportation are several orders of magnitude lower than many other natural or technical risks and therefore can be considered to be acceptable.

1. INTRODUCTION AND BOUNDARY CONDITIONS

In the following study only the potential radiological risks of a possible accident during a plutonium (Pu) transport are examined. The potential gamma exposure of personnel and population during normal transportation (without accidents) is not considered. Also, the radiation exposure during remedial actions at the location of an accident has not been evaluated in the present study.

In several countries (such as the USA, the Federal Republic of Germany (FRG), the United Kingdom (UK), France (F)) and also in the European Community (EC) the risks of Pu transportation have been investigated according to the boundary conditions prevailing in each country concerned. The risks have been evaluated, together with their dependence on the chemical forms of Pu (solid

* This study was sponsored by the Federal Ministry for Research and Technology (BMFT), Bonn, Federal Republic of Germany and carried out in collaboration with MC Energy Consult, Stuttgart, Federal Republic of Germany.
as Pu oxide powder or liquid as Pu nitrate solutions) and the transport mode (road or rail). Air transport has not been evaluated in the present study.

Many of the studies reviewed date back as far as 1974, so that only transport containers of older types (satisfying the 1973 Revised Edition of the IAEA Transport Regulations [1]) have been taken into account.

2. METHODOLOGY FOR THE RISK ANALYSES

The probabilistic risk analysis method is necessary to evaluate the safety status of the components of a transport system, consisting of the transport containers, the transport mode, the traffic network involved, and the meteorological and demographic parameters along the traffic network. They are based on definitions of risk as proposed by Farmer [2] for the safety evaluation of nuclear systems. The Farmer methodology has been developed and was applied to NPPs by Rasmussen [3], who considered all possible event combinations that potentially could lead to an accident with radiological consequences on the environment and population.

In 1974, the Battelle Pacific Northwest Laboratories (BNWL) in the USA [4] further refined the Rasmussen methodology and adapted it to waste treatment and management systems. This method could also be applied to the analysis of the transport of radioactive material [5–7]. The risk analysis method of BNWL starts by identifying non-desirable accidental events and then identifies all possible event combinations that may lead to these accidents by using the 'fault-tree' method (deductive) instead of starting from the use of initiating events in 'event-tree' methods (inductive). Also, the probabilities of single events that lead to accidents with radiological consequences have been derived by using the 'fault-tree' methodology.

The most significant advantage of this method consists in its completeness. All later risk analyses concerning radioactive material transportations have used this BNWL method [5]. Therefore, all risk analyses which have been evaluated in the present study have been compared systematically on the basis of the BNWL method.

3. EVALUATION AND COMPARISON OF RISK ANALYSES FOR PLUTONIUM TRANSPORTATION

After a complete literature survey of risk analyses concerning plutonium transportation [8, 9], the most significant and representative studies up to mid-1984 (deadline for submission of this study to BMFT) for several countries with nuclear energy generation and Pu recycling programmes have been selected for a more detailed evaluation and comparison. These studies concerning the
risk of Pu transportation in both chemical forms (oxide and nitrate) are summarized
with their main characteristics in Tables I and II. In all these risk analyses only
the risks of Pu transportation on the road have been investigated. The risks of rail
transportation of Pu have been studied by BNWL only [7]. A comparison of the
results of both BNWL studies for Pu transportation by road [5] and by train [7]
showed that there are no significant differences between the two [18]. Worldwide
the majority of studies have been carried out for the transport mode 'road'.
Therefore, attention has been concentrated in the present study on the risk analyses
of Pu transportation by road.

Not all of these risk analyses have been carried out to the same depth and the
results were not easily comparable. Also, some intermediate data required to
evaluate important aspects such as radiological consequences were missing.
Therefore it was necessary as a first step to evaluate all available data and to
bring them to a common base from which it was possible to make comparisons.
The main aspects of risk analyses for Pu transportation by road are evaluated,
discussed and briefly summarized below.

3.1. Release of plutonium

3.1.1. Basic data for the analyses

Transport distances: In the French study CEPN-49 [12] and in the CEC
report [11] the distances are not defined. For further evaluation the transport
distance has been assumed to be 600 km, taking into account European
conditions. This assumption has an uncertainty factor of 2, which is not
significant for risk calculations.

Transport containers: With the exception of French [12] and British
studies [13, 14] the majority of risk analyses considered the older generation of
transport containers (Type 6M for Pu oxide powder and L-10 for Pu
nitrate solutions). For these containers experimental tests have been made [19]
to define the failure limits for each barrier (inner and outer containers,
sample cans, heat insulation materials, etc.). In the case of French containers
FS-47, FS-51/52 (for Pu oxide), British containers UK-250 (for Pu nitrate),
Federal German containers GWK-Pu and 188 (for Pu nitrate), which represent
the new generation of transport containers, there have been some experimental
investigations. These show that the resistance of these containers to mechanical
and/or thermal impact has been improved considerably in comparison with the
older generation [20]. All these new containers are licensed in the relevant
countries (F, UK, FRG and EC) according to IAEA Transport Regulations [1].
Some of them meet even the improved requirements of the revised edition of the
IAEA Transport Regulations (1985) such as FS-51 for Pu oxide and 18B for
Pu nitrate.
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<td>400</td>
<td>600</td>
<td>(600) (a)</td>
<td>2400</td>
<td>(600) (a)</td>
<td>50</td>
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<td>Pu amount per transport (kg)</td>
<td>100</td>
<td>100</td>
<td>240</td>
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<td>100</td>
<td>54</td>
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<td>6 M</td>
<td>6 M-2</td>
<td>6 M-5</td>
<td>FS-51</td>
<td>L-10</td>
<td>GWK-Pu</td>
<td>UK-250</td>
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<td>puncture</td>
<td>puncture</td>
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<td>1.0</td>
<td>1.0</td>
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<td>2200</td>
<td>2650</td>
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<td>Airborne fraction of released portion</td>
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<td>0.1/200 g</td>
<td>(0.1/220 g)</td>
<td>(0.1/265 g)</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0/4/10 g</td>
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<tr>
<td></td>
<td>1600 Ci</td>
<td>2600 Ci</td>
<td>3300 Ci</td>
<td>23000 Ci</td>
<td>23000 Ci</td>
<td>30000 Ci</td>
<td>50 Ci</td>
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<td>3.52.E-6</td>
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<td></td>
<td>(0.92.E-6) (f)</td>
<td>(0.92.E-6)</td>
<td>(0.92.E-6)</td>
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<td>(5.E-8) (g)</td>
<td>3.E-6</td>
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<tr>
<td>Release frequency (km⁻¹)</td>
<td>8.1E-7 (5.8E-7) (g)</td>
<td>3.8E-8 (4.8E-8) (g)</td>
<td></td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>Release frequency (GW(e)·a)</td>
<td>5.6E-7 (5.8E-7) (g)</td>
<td>3.8E-8 (4.8E-8) (g)</td>
<td></td>
<td></td>
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</tr>
</tbody>
</table>

1 Ci = 37 GBq

(): Values in brackets are either analogously derived figures for data which are lacking from other studies or values calculated by the present authors
(a): The transport distance is not given, therefore the same distance is used as in Ref. [12]
(b): Loss of heat insulation by mechanical forces and fire load on the inner container
(c): The mechanical forces are not described in detail, very likely it is crush
(d): Recalculated value (see Section 4.2)
(e): EPRI (USA) Study [15]
(f): According to PSE study 1985 [16]
(g): Under the assumption that the heat insulation does not fail (through better design and better heat insulation of the container)
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<td>(BNWL-1846)</td>
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<td>- F -</td>
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<td>(SRD-R-187)</td>
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<td></td>
<td>[13, 14]</td>
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<td>Release and exposure time (h)</td>
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<td>&lt;24</td>
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<td>Release height and distance from the place of the release (m)</td>
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<td>0-1, &gt;100 m, 0-100 km</td>
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<td>Atmospheric stability classes and wind speed (m/s)</td>
<td>PASQUIL; B, D, E, F 1, 3, 5, 7, 10, 18</td>
<td>KLUG; I-V, KLUG; I (PASQUIL: F) (c) DOURY PASQUIL; B, D, E, F PASQUIL; F</td>
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<tr>
<td>Type of consequences (early or late fatalities)</td>
<td>late, early, late</td>
<td>late, late, late</td>
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<tr>
<td>Dose limits (road) for somatic fatalities (ref. organ: lung)</td>
<td>&gt;6000: 100%, 3000, &gt;6000; 50%</td>
<td>&gt;6000</td>
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<td>No. of exposed persons (km²)</td>
<td>35 approx., 100-7000 (a)</td>
<td>500</td>
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<td>No. of early fatalities</td>
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<td>not given, &gt;10</td>
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<td>2.1.5 (2.1.4) (g)</td>
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<tr>
<td>(man-rem)</td>
<td></td>
<td></td>
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<tr>
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<td>5.1.E-8</td>
</tr>
<tr>
<td></td>
<td></td>
<td>6.1.E-8</td>
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<td></td>
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</tr>
<tr>
<td>Frequency of late fatalities/(GW(e)-a)</td>
<td>7.1.8</td>
<td>5.1.E-8 (c)</td>
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<tr>
<td></td>
<td></td>
<td>4.1.E-8</td>
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<td></td>
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<tr>
<td>Risk man-rem/GW(e)-a</td>
<td>0.01 (0.001) (g)</td>
<td>&lt;0.01 (c)</td>
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<tr>
<td></td>
<td></td>
<td>0.05</td>
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</table>

1 rem = 10^-2 Sv

(a) Real population densities along the transport routes have been investigated in this study.
(b) The extent and frequency of early fatalities are defined according to the corresponding transport routes.
(c) Calculated using Ref. [17].
(d) Recalculated value (see Section 3.2).
(e) An approximate value (not given in original ref.).
(f) Under assumption that the heat insulation does not fail.
(g) For an airborne fraction of 0.01 of Pu oxide powder [15] instead of conservative value 0.1 used in this table.
Release scenarios: Containment puncture is defined as the dominant scenario for Pu oxide release in the US [5] and in FRG studies [10]. In the case of the French study [12] the mechanical failure scenario is not defined very clearly, but is very likely crush. Also, formation of a critical assembly of Pu containers has been assumed as a potential consequence of an accident under certain circumstances (such as for a van speed of >124 km/h). For Pu nitrate transport a combination of mechanical forces and fire load has been considered as the dominant scenario for release in the US and in CEC studies [12], whereas in the British studies [13, 14] the fire load has been assessed to be two orders of magnitude less likely than mechanical destruction (crush) of the big transport containers UK-250.

Release amounts and airborne fractions: Only the airborne fraction of the released amount of the total Pu inventory in a container is of importance for assessment of radiological consequences. In nearly all risk analyses represented in Table I the total Pu inventory has been assumed to be released after the accident (an exception is the BNWL study [5] where the amount is 1/2 of the inventory). Only 1/10 of the released portion of the Pu has been considered to be airborne within 24 h after a transport accident. These are of course conservative worst case assumptions. In reality (according to experimental experience) the airborne fraction is less [15].

3.1.2. Accident frequencies and release frequencies

Accident frequencies per kilometre can easily be determined in each country from police statistics of traffic accidents. There is a surprising similarity between the data in Table I. The uncertainty of about a factor of 3 for European conditions seems to be quite low.

Release frequencies can be calculated from the accident frequencies considered during accident scenarios. Through comparison of mechanical forces prevailing during an accident and failure limits of different barriers of a container during tests (such as IAEA recommended tests), it is possible to work out the failure frequencies of single barriers (outer container, inner container, sample can and thermal insulator) per accident. There is a good conformity between the failure probabilities of single barriers in several risk studies [5, 10, 11]. This can be attributed to the fact that later studies have taken over the frequencies of basic events (failure of single barriers) in the 'fault-tree' from the initiating BNWL study [5]. In the case of the French study with criticality as the release scenario [12] and the British risk assessment with crushing as the significant release mechanism [13, 14] the calculations of failure frequencies are independent of the BNWL study. Using the failure frequencies of single barriers the release probabilities per accident and finally per transport can be calculated (see Table I).
3.2. Consequences of Pu release and risks of Pu transportation

Boundary conditions for the estimation of radiological consequences and corresponding risks are summarized in Table II.

The average height of the potential release after an accident is assumed to be 0—1 m above the ground, which is very conservative, especially for the release of Pu nitrate under fire load.

Only the Federal German study [10] considers the real meteorological conditions along the chosen transport routes. All other risk analyses used averaged values, which are weighted over a year with corresponding frequencies for atmospheric stability characteristics and wind speeds.

Also, the demographic characteristics along the transport routes are averaged and weighted values for all studies with the exception of the Federal German risk assessment [10], for which real population densities along exactly defined transport routes have been used.

3.2.1. Radiological consequences and their corresponding frequencies

For both early (somatic) and late (latent cancer) fatalities inhalation has been assumed to be the dominant exposure path with <24 h exposure time. For the calculation of radiological doses an integration time of 1 year for early and 50 years for late (stochastic) fatalities has been used.

Only in Federal German [10] and French [12] risk assessments are early fatalities estimated using individual personal doses (rem). In all studies with the exception of the Federal German study the number of late fatalities has been determined using collective doses (man-rem) and cancer risk factors (5.E-5 rem⁻¹ for USA, 1.E-4 rem⁻¹ for ICRP and 2.E-4 rem⁻¹ for France).

As can be seen from Table II, there is a remarkable conformity between the frequencies of the late radiological consequences per gigawatt (electrical)-year for Pu oxide transportation. This could be explained partially by the mutual compensation of the contrary effects of some parameters such as transport distance, gigawatt-year and population density in the USA and Europe.

The corresponding frequencies for Pu nitrate transportation are not very consistent with each other. Therefore, a recalculation of these frequencies has been performed by using the original frequencies in each risk study for all basic events that lead together to an accident with Pu release and taking into account the probability of radiological consequences. The results of this calculation are given in brackets in Table II with corresponding comments.

The new elaborated frequencies for late radiological consequences of potential accidents during Pu nitrate transportation are about one order of magnitude lower than the analogous values for Pu oxide. This can be explained by a 10 times higher airborne fraction and by a 100 times lower probability.
for the release of Pu nitrate in the fire scenario with a loss of thermal isolation. The release frequency and consequently the probability of late fatalities would be reduced by one or more orders of magnitude by using improved transport containers for Pu nitrate with better thermal insulation and mechanical stability (such as the 18B container).

3.2.2. Risks of Pu transportation

To have a common base for comparison of the results of several risk studies it was decided to use the reference unit (man-rem/GW(e)-a) for collective radiological risks, for a quantity of Pu equal to that produced in an NPP of 1 GW(e). It is assumed that this amount will be recycled completely and therefore transported between reprocessing and fuel fabrication (MOX fuel) plants.

As can be seen in Table II (last line) the collective risk for Pu transportation amounts to 0.01–0.3 man·rem/GW(e)-a, whereas for Pu nitrate transportation 0.3 man·rem/GW(e)-a represents the conservative upper limit. After correction of the probabilities for the release and its radiological consequences the collective risk of Pu nitrate transportation has been reduced to 0.01 man·rem/GW(e)-a. Even this figure could be reduced to lower values using improved transport containers such as the 18B instead of the old version L-10 that has been considered. A similar reduction of risks for Pu oxide transportation could also be expected by using an improved container design such as FS-51. Both transport containers (FS-51 and 18B) fulfil the new IAEA design requirements and are licensed.

If the cumulative uncertainty of the collective risks is now set conservatively to be ±2 orders of magnitude, a maximum value of around 1 man·rem/GW(e)-a can be expected for the collective risks of Pu transportation, independent of the chemical form of plutonium.

A comparison of this risk for Pu transportation with the total collective radiological risk from nuclear energy production (with a total planned capacity of 25 GW(e)) in the Federal Republic of Germany [21] shows clearly¹ that the collective risk of Pu transportation contributes less than 0.1% to the total risk of nuclear energy production with Pu recycling. This result is confirmed by a US-EPRI risk assessment [15], in which the contribution of Pu transportation is even less (~0.01% of the total collective risk of nuclear energy production).

Parallel to the present study the risks of transportation of mixed oxide powder ² (MOX powder) by train or by road have been evaluated thoroughly using real transport routes in the Federal Republic of Germany [16]. The work was completed at the end of 1984 and therefore could not be considered in this comparative survey of work up to mid-1984. However, a cross-check of the results from the two studies now shows a very good agreement between them.

¹ The total collective risk for a nuclear programme with 25 nuclear power plants (~25 GW(e) installed capacity) is: 6.6 × 10⁶ man·rem/a.

² Includes up to 40–45 wt% Pu oxide.
According to PSE [16, Vol. 8], the collective radiological risk for MOX transport by road is 0.09 man·rem/14 GW(e)·a (or 0.0064 ≈ 0.01 man·rem/GW(e)·a). For rail transport the same risk is about half that from road transport.

4. CONCLUSIONS

Several risk analyses performed worldwide since 1975 show that the potential risks of Pu transportation in both chemical forms (oxide or nitrate) or as MOX powder are less than 0.1% of the total collective risk of nuclear energy generation with Pu recycling and can therefore be considered as negligible.

The chemical form of Pu has practically no significant influence on transportation risks. Despite this fact, the transport of Pu as nitrate solution may have some advantages. It allows easier handling both for transportation and MOX fuel fabrication, e.g. easier $^{241}$Am separation after long storage periods.

The use of large transport containers for Pu nitrate solutions does not decrease transport risks; there are even several advantages such as:

- reduction of the transport frequency by increasing the net transport weight for Pu per container
- reduction of fire load as a result of much higher heat capacity of the armoured outer container
- reduction of the potential risk for diversion (better physical protection)
- easier handling due to a smaller number of operational steps.

The transport mode (rail or road) does not influence transportation risks significantly. Rail transportation has a slightly lower risk than road by a factor of about 2–3 for the same amount of Pu [7, 16, 17]. But this difference cannot be distinguished within the uncertainty limits of such risk analyses. Road transport has the advantage of greater flexibility in selecting more suitable routes. Also, the diversion risks for Pu can be kept lower by road transportation.

REFERENCES


RISK ASSESSMENT WITH INTERTRAN FOR A SUBSET OF TRANSPORTS OF RADIOACTIVE MATERIALS IN THE FEDERAL REPUBLIC OF GERMANY

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Abstract

RISK ASSESSMENT WITH INTERTRAN FOR A SUBSET OF TRANSPORTS OF RADIOACTIVE MATERIALS IN THE FEDERAL REPUBLIC OF GERMANY.

An important subset of all transports of radioactive materials are those by rail. All transports in one year (1983) have been analysed with the aid of the computer code INTERTRAN, supplied by the IAEA to Member States in 1983. The resulting collective radiation exposure to different subgroups of the public and of railway workers has been determined. Work was concentrated on the derivation of well founded input values in order to obtain meaningful results. The values adopted are discussed. Finally the calculated radiation exposure to railway workers is compared with independent estimates of the Federal German railway authorities on the basis of parameters such as sojourn times, exposure distances, etc. The comparison indicates that the results presented should form a reasonable basis for a comparison of health impacts from incident-free transports of radioactive materials and from transports involving accidents.

1. INTRODUCTION

The computer code INTERTRAN [1] has been made available by the IAEA to Member States since 1983. It is a system for assessing the radiological impact from transporting radioactive materials. The computer code is to a large extent based on the RADTRAN II code developed at Sandia National Laboratories [2], but represents a more internationally oriented version suitable for application by many users. The INTERTRAN program can separately handle incident-free, i.e. normal case, transports and transports involving accidents. The results of the code are expressed in quantities related to the risk associated with an annual set of transports. For incident-free transport the resulting collective dose for different subgroups of the population is calculated. For transports involving accidents individual and population doses are calculated for every accident analysed and the annually expected number of health effects (early fatalities, early morbidities, latent cancer fatalities and genetic effects) resulting from a set of transports determined.
It is appropriate to emphasize that a clear distinction between the incident-free part and the accident part of the computer code INTERTRAN has to be made. For the incident-free case exposure of persons results from external radiation originating from radioactive material within the package and therefore the modelling of the resulting radiation exposure to different population subgroups is fairly straightforward. In evaluating the radiological risk associated with transport accidents, modelling is far more complex since parameters such as accident rates, associated mechanical and/or thermal impact on packages, package response and release fractions dependent on material behaviour have to be determined. In addition, atmospheric dispersion of the material released has to be modelled and the resulting radiation exposure from different pathways to be evaluated. Therefore, in order to obtain meaningful results a large number of parameters have to be carefully determined.

The INTERTRAN code has a number of advantages:

1. It is used in many countries and there is therefore the possibility of comparing the results of different users on the same basis, i.e. with the same modelling;
2. The user is not deflected by code development work but can devote his efforts to determining parameters and data, which is essential in order to obtain meaningful results;
3. Since INTERTRAN supplies default values for almost all parameters used in the program a gradual replacement of the parameters is possible as the analyst proceeds in determining parameters more appropriate to the special conditions of transports analysed or improved values from measurements and experiments.

2. SUBSET OF TRANSPORTS ANALYSED

The results of an assessment with the aid of INTERTRAN of radiation exposure associated with incident-free transports of radioactive materials by rail in the Federal Republic of Germany is presented below. All transports by rail for the reference year 1983 as summarized in Table I are analysed. The information relating to mean transport distances and mean dose rates at a distance of 1 m are also based on statistics from the railway authorities. Emphasis will be placed on the derivation of parameters used by the code.

3. INPUT PARAMETERS – INCIDENT-FREE CASE

A number of input parameters important for resultant radiation exposure have to be supplied by the user; alternatively, default values can be used. The derivation of values adopted for the analysis is discussed in the following subsections.
### TABLE I. TRANSPORT OF RADIOACTIVE MATERIALS BY RAIL IN THE FEDERAL REPUBLIC OF GERMANY IN 1983
(1 Ci = 37 GBq)

<table>
<thead>
<tr>
<th>Type of shipment</th>
<th>Number of wagons</th>
<th>Mean distance (km)</th>
<th>Tl (mrem/h at 1 m)</th>
</tr>
</thead>
<tbody>
<tr>
<td>UF₆ (natural or enriched, &lt;5 Ci)</td>
<td>89</td>
<td>375</td>
<td>0.37</td>
</tr>
<tr>
<td>UO₂ or U₃O₈ powder (natural or enriched)</td>
<td>8</td>
<td>375</td>
<td>0.86</td>
</tr>
<tr>
<td>Fresh fuel elements</td>
<td>25</td>
<td>375</td>
<td>1.95</td>
</tr>
<tr>
<td>Spent fuel elements (≤13 MCi)</td>
<td>89</td>
<td>375</td>
<td>6.7</td>
</tr>
<tr>
<td>Radiation sources (≤2 × 10⁵ Ci)</td>
<td>37</td>
<td>375</td>
<td>2.4</td>
</tr>
<tr>
<td>Uranium ore or concentrates (0.1–8.5 Ci)</td>
<td>436</td>
<td>375</td>
<td>1.1</td>
</tr>
<tr>
<td>UF₆ (depleted)</td>
<td>36</td>
<td>375</td>
<td>0.86</td>
</tr>
<tr>
<td>Low level waste (e.g. 200 L drums)</td>
<td>56</td>
<td>375</td>
<td>3.6</td>
</tr>
<tr>
<td>Piece goods (mean weight 90 kg)</td>
<td>Packages 1900</td>
<td>450</td>
<td>0.8</td>
</tr>
<tr>
<td>Goods shipped by express (mean weight 9 kg)</td>
<td>Packages 16000</td>
<td>400</td>
<td>0.6</td>
</tr>
</tbody>
</table>

### 3.1. Population zones

A 795 km long railway route within the Federal Republic of Germany has been analysed in detail with respect to population density in a 1 km wide corridor on either side of the transport route. The data are summarized in Table II and indicate how the input parameters — mean population density in the three population zones and fractional part of transport route assigned to population zones — have been derived.

It is interesting to note that for the railway route analysed (from the Stade nuclear power plant near Hamburg via the Ruhr area to Forbach near Saarbrücken) an average population density of 1104 inhabitants per square

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1 The data on population densities and the fractional distribution in population density intervals were supplied by GUW — Gesellschaft für Umweltforschung mbH, Aldenhoven, Federal Republic of Germany.
TABLE II. DISTRIBUTION OF POPULATION DENSITIES ALONG SELECTED RAILWAY ROUTE (Stade to Forbach, 795 km) AND DERIVED INPUT VALUES FOR INTERTRAN

<table>
<thead>
<tr>
<th>Range of population density (km$^{-2}$)</th>
<th>Fraction of route (%)</th>
<th>Average population density (km$^{-2}$)</th>
<th>Assignment to population zone</th>
<th>Fraction of route (%)</th>
<th>Population density (km$^{-2}$)</th>
</tr>
</thead>
<tbody>
<tr>
<td>0−150</td>
<td>53.0</td>
<td>13.6</td>
<td>Rural</td>
<td>55</td>
<td>20</td>
</tr>
<tr>
<td>151−250</td>
<td>2.0</td>
<td>189.2</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>251−500</td>
<td>5.6</td>
<td>358.0</td>
<td>Suburban</td>
<td>18</td>
<td>670</td>
</tr>
<tr>
<td>501−1200</td>
<td>12.6</td>
<td>812.3</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1201−3000</td>
<td>13.4</td>
<td>2030.1</td>
<td>Urban</td>
<td>27</td>
<td>3600</td>
</tr>
<tr>
<td>Greater than 3000</td>
<td>13.4</td>
<td>5240.7</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Average: 1104 (km$^{-2}$)

kilometre in a 1 km broad corridor on either side of the railway route results, compared with the average population density of about 248 km$^{-2}$ in the Federal Republic of Germany.

3.2. Transport velocity

Including stops at signals, the average velocity of goods trains while under way was estimated as 50 km/h, independent of population zone.

3.3. Radiation exposure at switchyards

From information supplied by the Federal German railway authorities a stop time of 11 h per 24 hour trip was determined. Measurements of sojourn times and distances of switchmen from shipments of spent fuel casks have led to an upper estimate of 0.5 mrem collective dose$^2$ per switchyard and shipment.

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$^2$ 1 rem = $10^{-2}$ Sv.
This value is reproduced by INTERTRAN by adjusting the input values – persons exposed while stopped – to 1.5 (switchmen) and the average stop exposure distance to 20 m in conjunction with the above stop time per 24 hour trip. One switchyard per 200 km travel distance for goods trains is assumed. It turns out that the shape factor introduced in the computer code to reflect the influence of the dimensions of the package, cask or wagon on the radiation field as a function of distance is adequate to give reasonable agreement with estimates of radiation exposures of switchyard personnel for other radioactive materials.

3.4. Exposure of people sharing the transport link

The average number of persons in a passenger train was determined as 213 from railway transport statistics. From timetables averaged over 24 h a traffic count of 2.3 passenger trains per hour passing a specific point resulted, independent of population zone.

3.5. Exposure of people surrounding the transport link while the shipment is moving

The collective dose to persons within a corridor on either side of the transport link ranging from 30 to 800 m depends among other things on population density and shielding from buildings. The default values of INTERTRAN for the shielding factor are 1 (rural), i.e. no shielding by buildings, 0.5 (suburban) and 0.05 (urban). Since the dose to people surrounding the transport link is proportional to the product of population density and shielding factor, the combined effect is a lower effective population density if the shielding factor is smaller than 1. Our analysis of shielding by the type of buildings typical for the three population zones within the Federal Republic of Germany leads to shielding factors of 0.3 (rural), 0.05 (suburban) and 0.004 (urban). But in order to take account of people outside of buildings and not to underestimate the resulting collective dose, the default values have been adopted for the analysis.

3.6. Handling of transports of radioactive materials

The only significant exposure to railway personnel is assumed to be associated with the handling of express and piece goods (receipt of the delivered packages, temporary storage, transport to platform, loading into the train, etc.). A collective dose of 0.25 mrem per handling per unit transport index is assumed by INTERTRAN for small packages. The number of handlings was set to 4 per shipment of a small package. The parameters are assumed to include the exposure of warehouse personnel.
3.7. Number of crew and mean distance to radioactive material

The computer code does not foresee an effective shielding factor between crew (engine driver, escort crew, package car personnel) and the radioactive material. Therefore, the parameter — distance from source to crew — has to be chosen appropriately to take account of the radiation field in the axial direction of large packages or wagons and of shielding by material and may not necessarily coincide with the physical distance between source and crew. A value of 2 crewmen at a distance of 15 m for railway wagons and of 7 m for baggage cars has been adopted.

4. RESULTS OBTAINED WITH INTERTRAN

The results obtained from running the computer code for all transports of radioactive materials by rail during one year (1983) with parameters as discussed above are summarized in Table III. Resulting collective doses for personnel (crewmen, handlers, switchmen) and the public (people sharing the transport link, i.e. in passenger trains and people living in a corridor on either side of the railway route) are given for each type of shipment. For different types of railway personnel the total collective radiation doses are compared with estimates of the Federal German railway authorities based on measurements of sojourn times and distances from radioactive materials [3].

As can be seen from Table III, the largest calculated dose to crewmen — in this case baggage car personnel — results from the comparatively large number of transports of piece and express goods. The agreement between the total exposure calculated for this subgroup (6.7 man-rem) and the estimate of the Federal German railway authorities (9.1 man-rem) is reasonable. This is also the case for the collective dose from handling of piece and express goods (11.1 man-rem calculated, 9.2 man-rem estimated by Federal German railway authorities). Also for switchmen the agreement within about a factor of 2 is reasonable. In total, the railway authorities estimated about 19 man-rem for their personnel from transports of radioactive materials of one year. It is interesting to have the additional information about the approximate numbers of personnel involved with transports of radioactive materials: switchmen (360), baggage car personnel (540), engine drivers (680), handling of express (970) and of piece goods (460). The average dose to personnel directly involved with the transport of radioactive materials is accordingly of the order of $6 \times 10^{-3}$ man-rem.

The total collective doses to the public — surrounding population while shipments are under way and people (in passenger trains) travelling on the transport link — are 0.73 man-rem and 3.0 man-rem, respectively. The larger exposure to people in passenger trains passing the shipment in spite of the
TABLE III. RESULTS OF COLLECTIVE DOSE (MAN·REM) FROM RAILWAY TRANSPORTS IN ONE YEAR (1983) OF RADIOACTIVE MATERIALS AS CALCULATED BY INTERTRAN: COMPARISON WITH ESTIMATES AND MEASUREMENTS OF FEDERAL GERMAN RAILWAY AUTHORITIES

<table>
<thead>
<tr>
<th>Type of shipment</th>
<th>Crewmen</th>
<th>Handlers</th>
<th>While moving</th>
<th>Switchmen</th>
<th>Totals</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>Off link</td>
<td>On link</td>
<td></td>
</tr>
<tr>
<td>UF₆</td>
<td>3.1E-2</td>
<td>8.6E-3</td>
<td>3.5E-2</td>
<td>6.0E-3</td>
<td>8.1E-2</td>
</tr>
<tr>
<td>UO₂ or U₃O₈ powder</td>
<td>6.5E-3</td>
<td>1.8E-3</td>
<td>7.4E-3</td>
<td>1.2E-3</td>
<td>1.7E-2</td>
</tr>
<tr>
<td>Fresh fuel elements</td>
<td>4.6E-2</td>
<td>1.3E-2</td>
<td>5.2E-2</td>
<td>8.8E-3</td>
<td>1.2E-1</td>
</tr>
<tr>
<td>Spent fuel elements</td>
<td>5.6E-1</td>
<td>1.6E-1</td>
<td>6.4E-1</td>
<td>1.1E-1</td>
<td>1.5E+0</td>
</tr>
<tr>
<td>Radiation sources</td>
<td>1.3E-2</td>
<td>3.7E-3</td>
<td>9.5E-2</td>
<td>2.6E-3</td>
<td>3.4E-2</td>
</tr>
<tr>
<td>Uranium ore/concentrates</td>
<td>4.5E-1</td>
<td>1.3E-1</td>
<td>5.1E-1</td>
<td>8.7E-2</td>
<td>1.2E+0</td>
</tr>
<tr>
<td>UF₆ (depleted)</td>
<td>2.9E-2</td>
<td>8.1E-3</td>
<td>3.3E-2</td>
<td>5.6E-3</td>
<td>7.6E-2</td>
</tr>
<tr>
<td>Low level waste</td>
<td>2.2E-2</td>
<td>6.1E-3</td>
<td>2.2E-1</td>
<td>4.2E-3</td>
<td>5.7E-2</td>
</tr>
<tr>
<td>Piece goods</td>
<td>8.4E-1</td>
<td>1.5E+0</td>
<td>5.1E-2</td>
<td>2.1E-1</td>
<td>1.1E+0</td>
</tr>
<tr>
<td>Goods shipped by express</td>
<td>4.7E+0</td>
<td>9.6E+0</td>
<td>2.8E-1</td>
<td>1.2E+0</td>
<td>2.0E-1</td>
</tr>
<tr>
<td>Totals:</td>
<td>6.7E+0</td>
<td>1.1E+1</td>
<td>7.3E-1</td>
<td>3.0E+0</td>
<td>4.5E-1</td>
</tr>
<tr>
<td>Estimates + measurements of railway authorities</td>
<td>9.1E+0⁸</td>
<td>9.2E+0⁹</td>
<td>1.1E+0</td>
<td>1.9E+1⁸</td>
<td></td>
</tr>
</tbody>
</table>

⁸ This number combines 3.7 man·rem (engine drivers) and 5.4 man·rem (baggage car personnel).
⁹ This number includes only exposures from handling of piece and express goods.
⁸ This number includes only exposure of railway personnel.

smaller number results from the much shorter exposure distance. It is evident that individual doses are so small that it is doubtful from the radiological point of view whether they have any significance with respect to the associated health risk. In any case the calculated collective dose to the public is about a factor of 5 smaller than the dose to railway personnel.
5. CONCLUSIONS

For incident-free transports of radioactive materials, the models for calculating resulting collective doses for different population subgroups are rather straightforward. The modelling as used in INTERTRAN is able to produce reasonable results. It is the quality of the parameters supplied by the user which essentially determines the quality of the results. It is felt that results presented for all transports of radioactive materials by rail in one year are a reasonable estimate of associated collective doses. In the case of different groups of railway personnel this is supported by the agreement with estimates of collective doses derived by the Federal German railway authorities and based on sample measurements of dose rates, sojourn times and distances of personnel close to radioactive materials. Efforts are presently under way to collect data relevant to the estimation of risks associated with transports involving accidents with the aid of INTERTRAN. The results presented here for incident-free transports should form a reasonable basis for the comparison of the relative risks from normal case transports and transport accidents with radioactive materials.

ACKNOWLEDGEMENT

The author would like to thank Mr. Kwiedor from Bundesbahn-Zentralamt, Minden, Westfalia, for a detailed discussion of procedures of the Federal German railway authorities to derive the radiation exposure of railway personnel.

REFERENCES

Abstract—Résumé

EVALUATION OF ACCIDENT PROBABILITIES DURING THE ROAD TRANSPORT OF RADIOACTIVE MATERIALS.

The number of packages containing radioactive materials transported by road in Belgium during the reference year (1977) was 65 181, of which 14 686 were Type B and 50 495 were industrial Type A packages. These packages were conveyed in 9380 journeys representing 0.35 million vehicle kilometres. The ratio of the transport of radioactive materials to that of freight conveyors (lorries) and general traffic is of the order of $1.2 \times 10^{-4}$ and $1.2 \times 10^{-5}$, respectively. These figures show that the transport of radioactive materials accounts for a minimally small proportion of the general traffic, even though it makes a significant contribution in the field of medicine, research and industry. Surveys of accidents which have occurred during the transport of hazardous materials, especially radioactive materials, have been carried out in different European countries and the United States of America and it has emerged that the number of accidents involving radioactive materials is extremely small. Conclusions have been drawn from these statistics regarding the configuration of accidents, the obstacle encountered, the energy absorption capacity of the vehicle and the speed of impact. One of these conclusions is that the average probability of an accident occurring in Europe is of the order of 50–63% for accidents with head-on impact and 9–41% for accidents with lateral impact. An average severe accident rate per kilometre for Europe has been determined for light vehicles (total permissible laden weight ≤ 3.5 t) and heavy vehicles (total permissible laden weight > 3.5 t), which is $8 \times 10^{-7}$ and $4 \times 10^{-8}$, respectively.

EVALUATION DES PROBABILITES D'ACCIDENT LORS DU TRANSPORT ROUTIER DE MATIERES RADIOACTIVES.

Le nombre de colis contenant des matières radioactives transportés par la route en Belgique pendant l’année de référence (1977), a été de 65 181 dont 14 686 de type B et 50 495 de types industriel et A. Ces colis ont été acheminés en 9380 transports représentant 0,35 million de véhicules-kilomètres. Le rapport du transport de matières radioactives avec celui du trafic «fret» (camions) et du trafic général est de l’ordre, respectivement, de $1,2 \times 10^{-4}$ et $1,2 \times 10^{-5}$. Ces valeurs montrent que le transport de matières radioactives participe pour une part minime au trafic général tout en faisant partie d’une activité médicale, de recherche et industrielle non négligeable. D’autre part, des relevés d’accidents survenus lors du transport de marchandises dangereuses et, notamment, de matières radioactives ont été effectués pour divers pays.
européens et les États-Unis. Il est apparu que le nombre d'accidents pour lesquels des matières radioactives sont concernées est excessivement faible. De ces statistiques, des considérations ont été déduites en ce qui concerne la configuration des accidents, l'obstacle rencontré, la capacité d'absorption d'énergie du véhicule et de la vitesse d'impact. Il en résulte notamment que la probabilité d'occurrence moyenne européenne est de l'ordre de 50 à 63% pour des accidents avec choc frontal et de 9 à 41% pour des accidents avec choc latéral. Un taux moyen européen d'accidents graves par kilomètre a été déterminé pour les véhicules légers (PTAC ≤3,5 t) et les poids lourds (PTAC ≥3,5 t) et qui est respectivement de $8 \times 10^{-7}$ et $4 \times 10^{-8}$.

1. LE TRANSPORT ROUTIER DE MATIÈRES RADIOACTIVES PAR RAPPORT AU TRAFIC GENERAL

Une évaluation de l'importance du transport de matières radioactives par rapport au trafic général a été réalisée en 1980 [1]. Un extrait de cette étude est repris ci-après dans le cadre du transport routier.

Rappelons qu'en Belgique l'exécution de transports de substances radioactives, quel que soit le moyen de transport utilisé, y compris les véhicules personnels, ne peut être effectuée que moyennant l'autorisation préalable du ministère de la santé publique. Aussi les détenteurs d'autorisations de transport ont-ils l'obligation d'adresser à l'autorité compétente un relevé mensuel des transports effectués.

Sur la base des relevés mensuels reçus par le ministère de la santé publique au cours de l'année 1977 prise comme référence, il résulte que:
- 78 252 colis de matières radioactives ont été officiellement transportés suivant les divers modes de transport (ferroviaire, aérien et routier); toutefois, certains des colis, ayant été acheminés vers leurs destinataires par deux ou trois modes de transport différents, ont été comptabilisés deux ou trois fois;
- l'activité totale des expéditions a été de 60 millions de curies; cette activité est due principalement aux transports de combustibles irradiés.

Le tableau I donne, à titre d'information, la répartition du nombre de colis acheminés par les trois modes principaux de transport. Il permet de situer le transport routier de ces matières par rapport aux autres modes de transport.

Pour chaque mode de transport, l'activité totale du colis est reprise par type d'emballage, d'une part en types industriel et A groupés, et d'autre part en type B.

Le nombre de transports routiers ou le nombre de tonnes-kilomètres ont été calculés également par type d'emballage.

L'évaluation du trafic routier des matières radioactives par rapport à celui de toutes les marchandises est exposée au tableau II. Celui-ci met en évidence que le transport des matières radioactives, calculé en véhicules-kilomètres, prend une part minime du trafic général et, de ce fait, ne représente pas un élément perturbateur de ce dernier tout en faisant partie d'une activité médicale, de recherche ou industrielle importante en Belgique.
**TABLEAU I. TRANSPORTS DE MATIERES RADIOACTIVES (ANNEE 1977)**

<table>
<thead>
<tr>
<th>Type de transport</th>
<th>Ferroviaire</th>
<th>Aérien</th>
<th>Routier</th>
<th>Totaux</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>A + I</td>
<td>B</td>
<td>A + I</td>
<td>B</td>
</tr>
<tr>
<td>Nombre de colis</td>
<td>9 526</td>
<td>158</td>
<td>3 197</td>
<td>190</td>
</tr>
<tr>
<td>Activité (Ci)(^1)</td>
<td>1 643</td>
<td>7 294</td>
<td>390</td>
<td>54 761</td>
</tr>
<tr>
<td>Nombre de transports routiers</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Nombre de tonnes-km</td>
<td>14 300</td>
<td>1 900</td>
<td>6 000</td>
<td>3 525</td>
</tr>
<tr>
<td>Nombre de véhicules-km</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

\(^1\) 1 Ci = 3.70 \(\times\) 10\(^10\) Bq.
### TABLEAU II. EVALUATION DE L'IMPORTANCE DU TRANSPORT DE MATIÈRES RADIOACTIVES PAR RAPPORT AUX TRAFICS FRET ET ROUTIER

<table>
<thead>
<tr>
<th></th>
<th>Trafic routier (véhicules-kilomètres)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Transport de matières radioactives</td>
<td>$3,5 \times 10^5$</td>
</tr>
<tr>
<td>Trafic routier «fret» (1973)</td>
<td>$2,8 \times 10^9$</td>
</tr>
<tr>
<td>Trafic routier «général» (1973)</td>
<td>$2,8 \times 10^{10}$</td>
</tr>
</tbody>
</table>

Rapport du transport de matières radioactives par rapport:
- au trafic «fret» $1,2 \times 10^{-4}$
- au trafic «général» $1,2 \times 10^{-5}$

### TABLEAU III. DISTRIBUTION DES ACCIDENTS EN FONCTION DE LA DIRECTION DE L'IMPACT (%)

<table>
<thead>
<tr>
<th>Pays</th>
<th>Direction</th>
<th>Etats-Unis</th>
<th>Suède</th>
<th>Grande-Bretagne</th>
<th>France</th>
<th>Moyenne</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Frontale</td>
<td>46</td>
<td>38,54</td>
<td>50</td>
<td>53</td>
<td>63</td>
</tr>
<tr>
<td></td>
<td>Latérale</td>
<td>5</td>
<td>17,07</td>
<td>41</td>
<td>15</td>
<td>9</td>
</tr>
<tr>
<td></td>
<td>Arrière</td>
<td>12</td>
<td>34,77</td>
<td>9</td>
<td>24</td>
<td>28</td>
</tr>
<tr>
<td></td>
<td>Retournement</td>
<td>34</td>
<td>7,68</td>
<td>-</td>
<td>-</td>
<td>8</td>
</tr>
<tr>
<td></td>
<td>Autres</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
</tbody>
</table>

Référence [4] [5] [6] [7,8] [9]

2. STATISTIQUES D'ACCIDENTS

2.1. Critères d'accidents

Sur la base des statistiques relevées pour divers pays européens ainsi qu'aux Etats-Unis, les conditions accidentelles ont été groupées en fonction des quatre critères précisés ci-après.
TABLEAU IV. DISTRIBUTION DES ACCIDENTS EN FONCTION DE LEUR NATURE

<table>
<thead>
<tr>
<th>Nature</th>
<th>CITMD (%)</th>
<th>CEPN (%)</th>
<th>Italie (%)</th>
<th>Etats-Unis (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>1</td>
<td>2</td>
<td>3</td>
<td></td>
</tr>
<tr>
<td>Obstacles fixes</td>
<td>11,5</td>
<td>8,75</td>
<td>8,50</td>
<td>6</td>
</tr>
<tr>
<td>Obstacles mobiles</td>
<td>42,0</td>
<td>40,00</td>
<td>33,25</td>
<td>72</td>
</tr>
<tr>
<td>Renversement</td>
<td>39,0</td>
<td>42,50</td>
<td>51,00</td>
<td>22</td>
</tr>
<tr>
<td>Divers (+feu)</td>
<td>7,5</td>
<td>8,75</td>
<td>7,25</td>
<td>22</td>
</tr>
</tbody>
</table>


TABLEAU V. DISTRIBUTION DES ACCIDENTS EN FONCTION DE LA VITESSE (%)

<table>
<thead>
<tr>
<th>Vitesse (km/h)</th>
<th>Pays</th>
<th>Belgique</th>
<th>France</th>
<th>République fédérale d'Allemagne</th>
<th>Suède</th>
<th>Etats-Unis</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>50</td>
<td>54,4</td>
<td>45,5</td>
<td>61,2</td>
<td>40,0</td>
<td>39,0</td>
<td></td>
</tr>
<tr>
<td>50– 80</td>
<td>45,4</td>
<td>52,0</td>
<td>37,4</td>
<td>53,0</td>
<td>30,0</td>
<td></td>
</tr>
<tr>
<td>80–100 (110)</td>
<td>–</td>
<td>2,5</td>
<td>1,4</td>
<td>7,0</td>
<td>28,0</td>
<td></td>
</tr>
<tr>
<td>110</td>
<td>–</td>
<td>–</td>
<td>–</td>
<td>–</td>
<td>3,0</td>
<td></td>
</tr>
</tbody>
</table>


a) La configuration d’accident: la distribution des accidents en fonction de la direction de l’impact est donnée au tableau III. Comme le montre également ce tableau, il apparaît que le choc frontal représente la configuration d’accidents la plus fréquente et la plus sévère.

b) L’obstacle rencontré: la distribution des accidents en fonction de leur nature est consignée au tableau IV. Si les statistiques française et américaine sont cohérentes en ce qui concerne les accidents contre un obstacle fixe, celles relatives
aux accidents contre un obstacle mobile ou résultant d'un renversement sont divergentes. Cette incohérence est difficile à expliquer. Elle pourrait résulter de modes de conduite différents entre les deux continents.
c) Capacité d'absorption d'énergie du véhicule: la sévérité d'un choc dépend également de la capacité d'absorption d'énergie de l'obstacle. Contre un obstacle rigide, l'énergie cinétique du véhicule au moment de l'impact doit être absorbée en totalité par le véhicule; l'effet du choc sur le véhicule sera donc le plus grand.
d) Distribution des accidents en fonction des types de véhicules: Les études du Centre d'étude pour la protection nucléaire permettent de dégager les statistiques suivantes sur le plan du transport français de matières dangereuses [2, 3]:

<table>
<thead>
<tr>
<th>Type de véhicule</th>
<th>% d'accidents</th>
</tr>
</thead>
<tbody>
<tr>
<td>Camions à 2 ou 3 essieux</td>
<td>28,1%</td>
</tr>
<tr>
<td>Semi-remorques à 3, 4 ou 5 essieux</td>
<td>68,8%</td>
</tr>
<tr>
<td>Camions + remorques</td>
<td>3,0%</td>
</tr>
</tbody>
</table>

2.2. Evaluation des vitesses de véhicules avant et pendant les accidents

2.2.1. Vitesse du ou des véhicules avant l'accident

La distribution des accidents (en %) en fonction de la vitesse est donnée au tableau V pour la Belgique, la France, la République fédérale d'Allemagne, la Suède et les Etats-Unis. En ce qui concerne ce dernier pays, la conversion des miles/h en km/h a été adaptée aux vitesses européennes. Par exemple 30 miles/h correspondent à environ 50 km/h.

Deux remarques générales sont à formuler au sujet de la distribution des vitesses. En premier lieu, la vitesse au moment du choc est généralement inférieure à la vitesse à laquelle roulait le véhicule avant l'accident. Ceci est en particulier dû au freinage plus ou moins important qui précède le choc dans la plupart des cas. Deuxièmement, la plupart des véhicules dont il est question ont leur vitesse limitée à 60 km/h en toutes circonstances.

Le tableau V met en évidence qu'une zone de 93 à 99,8% des accidents, dans les pays européens, est couverte en prenant 80 km/h comme limite supérieure pour la vitesse du ou des véhicules avant l'accident.

2.2.2. La vitesse au moment de l'impact

Il est admis que la vitesse au moment de l'impact sera inférieure à celle juste avant le moment de l'impact par suite de l'action de freinage de la part du chauffeur. Suivant les informations obtenues auprès de l'UTAC (France), les vitesses à considérer au moment de l'impact et juste avant ce dernier sont respectivement de 50 km/h et de l'ordre de 80 km/h. L'action de freinage contribue donc à une réduction de la vitesse de croisière d'environ 31%.
TABLEAU VI. TAUX MOYEN D'ACCIDENTS PAR KILOMETRE

<table>
<thead>
<tr>
<th>Pays</th>
<th>Année</th>
<th>Taux d'accidents par km</th>
<th>Référence</th>
</tr>
</thead>
<tbody>
<tr>
<td>Etats-Unis</td>
<td>1966</td>
<td>$1,5 \times 10^{-6}$</td>
<td></td>
</tr>
<tr>
<td></td>
<td>1967</td>
<td>$1,5 \times 10^{-6}$</td>
<td></td>
</tr>
<tr>
<td></td>
<td>1968</td>
<td>$1,56 \times 10^{-6}$</td>
<td>[4]</td>
</tr>
<tr>
<td></td>
<td>1969</td>
<td>$1,5 \times 10^{-6}$</td>
<td></td>
</tr>
<tr>
<td></td>
<td>1970</td>
<td>$1,69 \times 10^{-6}$</td>
<td></td>
</tr>
<tr>
<td>France</td>
<td>1976</td>
<td>1,2 à $1,8 \times 10^{-6}$</td>
<td>[3]</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0,78 à $1,7 \times 10^{-6}$</td>
<td></td>
</tr>
<tr>
<td>Rép. féd. d'Allemagne</td>
<td></td>
<td>$1,65 \times 10^{-6}$</td>
<td>[13]</td>
</tr>
</tbody>
</table>

TABLEAU VII. TAUX D'ACCIDENTS GRAVES/KM DANS DIFFERENTS PAYS EUROPEENS

<table>
<thead>
<tr>
<th>Type de camion</th>
<th>Véhicules légers</th>
<th>Poids lourds</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sur la base de statistiques belges</td>
<td>$8,19 \times 10^{-7}$</td>
<td>$8,45 \times 10^{-8}$</td>
</tr>
<tr>
<td>Sur la base de statistiques françaises</td>
<td>$7,6 \times 10^{-7}$</td>
<td>$3,4 \times 10^{-8}$</td>
</tr>
<tr>
<td>Sur la base de statistiques de la République fédérale d'Allemagne</td>
<td>-</td>
<td>$4,1 \times 10^{-8}$</td>
</tr>
</tbody>
</table>

2.2.3. Probabilités d'accident

Des probabilités d'accident ont été définies aux Etats-Unis, en France, et en Allemagne:

- un taux moyen d'accidents par kilomètre de $1,52 \times 10^{-6}$ peut être considéré comme conservatif (voir tableau VI);
- un taux moyen d'accidents graves par kilomètre a été également défini pour divers pays européens (voir tableau VII). Il est de $8 \times 10^{-7}$ pour les véhicules légers et de $4 \times 10^{-8}$ pour les poids lourds.
REFERENCES


ASSESSMENTS OF RADIATION EXPOSURES FROM TWO TRANSPORT OPERATIONS IN THE UNITED KINGDOM

Technetium generators
and intermediate level radioactive wastes

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National Radiological Protection Board,
Chilton, Didcot, Oxfordshire,
United Kingdom

Abstract

The paper presents summaries of work undertaken by the United Kingdom National Radiological Protection Board (NRPB) assessing the radiation exposures arising from two separate transport operations. Both studies used the results of radiation surveys and work studies and these were supported by theoretical assessment procedures, where appropriate. An NRPB study of the normal transport of radioactive material within the UK showed that nearly half of the annual occupational collective dose arose from the carriage of $^{99}$Mo in the form of technetium generators. As a result, the Health and Safety Executive commissioned a study to reduce the radiological impact of transporting technetium generators. Distant handling can reduce whole-body exposures by a factor of 25 for a typical generator, although the extremity dose would increase marginally. Since manual handling accounts for about 70% of the dose, distant handling would yield a 60% overall reduction in dose and thus reduce the annual exposures of the few transport personnel who receive annual doses of up to 20 mSv to below 10 mSv. The NRPB was commissioned by the Department of the Environment to determine the radiological impact resulting from the movement of intermediate level wastes between conditioning plants and storage or disposal facilities. The assessment, based on the premise that all movements occur at the maximum radiation levels permitted in the IAEA Transport Regulations, has shown that the radiation exposure of the public is low, regardless of the mode of transport adopted. Collective doses to transport personnel are also assessed to be low. However, using the assumptions adopted in this study, it is shown that certain individuals, in particular the lorry drivers and the loaders at railheads, could theoretically receive significant exposures. The practical significance of these findings will be determined by actual work patterns and the actual level of radiation penetrating the transport containers.
Radiological Impact arising from the Transport of Technetium Generators in the UK

1.1 Introduction

An earlier study[1] in 1981/82 of the normal transport of radioactive materials within the UK had shown that the collective dose (an abbreviation of collective effective dose equivalent) received by transport workers was distributed broadly as: 13% nuclear fuel cycle, 20% industrial isotopes, 18% general medical isotopes and 49% technetium generators, amounting to about 1 man.Sv in total.

Approximately 20 000 personnel are directly involved in all these transport movements; the majority being concerned with the nuclear fuel cycle, about 1,000 each with the industrial isotopes and general medical sources but less than 100 transport workers with the technetium generators. A few individuals in this latter group have recorded annual doses (an abbreviation of effective dose equivalent) of 10-20 mSv; some 1,000 other workers are estimated to receive doses of up to 1 mSv and the remainder are estimated to receive considerably less than 1 mSv.

As a result of these findings, the Health and Safety Executive commissioned a study on reducing the radiological impact of transporting technetium generators.

A generator contains a molybdenum-99 parent radioisotope, decaying to technetium-99m. This latter isotope accounts for over 70% of all radioisotope usage by 120 nuclear medicine departments in UK hospitals[2]. In 1983, it was estimated that in the UK nearly 16 000 generators were transported, half for domestic use, half for export. Of nearly 8,000 generators used in the UK, slightly more than half were imported from three suppliers outside the UK.

Five different models of generator were in use and available for this study. One model was available with either a lead shield or a depleted uranium shield. The latter shield was not available for study but provided substantially greater protection than the lead-shielded generator during transport.

1.2 Methodology

Each supplier was approached for details of generator construction and packaging, annual numbers transported, transport routes and handling techniques. Visits were made to transport depots to study these techniques and the transporting vehicles.
Samples of the 5 different models of generators were also obtained for laboratory study and evaluation. The evaluation was conducted in two parts (i) shielding measurements and (ii) a comparison of the two handling techniques of different generators observed during visits to depots.

Subsequently, field measurements were carried out on generator packages to supplement the laboratory results.

1.3 Generator systems

Two generator models incorporate low specific activity molybdenum-99 (as sodium molybdate), obtained from reactor irradiation of molybdenum-99, and three models incorporate high specific activity molybdenum-99 obtained from fission of uranium-235 and fission product processing. The molybdate is absorbed onto alumina granules in a chromatographic column and the technetium-99m is extracted by a saline elution as sodium pertechnetate. The parent isotope emits 0.8 MeV photons (air kerma rate $33 \mu$Gy.m$^{-2}$.h$^{-1}$.GBq$^{-1}$) and the daughter isotope emits 0.14 MeV photons (air kerma rate $14 \mu$Gy.m$^{-2}$.h$^{-1}$.GBq$^{-1}$) with corresponding shielding requirements of 30 to 50 mm thick lead and 2 to 6 mm thick lead, respectively, for a typical generator column and its eluant.

The columns are mounted vertically in these substantial lead shields (cylindrical for one model, cylindro-conical for the other models) with stepped slots or grooves cut to allow the saline/eluant pipework to penetrate through the shields into and from the columns. Eluant collection vials are mounted above or alongside these shielded columns and are provided with the thinner lead shields.

1.4 Measurements

1.4.1 Shielding

Shielding effects were measured utilizing a 4.6 GBq caesium-137 to simulate a 10.2 GBq molybdenum-99 source, with appropriate energy compensation on attenuation measurements. While optimum shielding was closely achieved on only two models out of five, two of the non-optimum models were being phased out and shielding improvements would have proven difficult (complete redesign), inefficient (additional weight considerations rendered handling difficult), and would have involved redesign of the column to locate the activity central to the shield (prohibitively expensive).
A comparison was made of measured dose rate, at a simulated package surface, with field measurements on a labelled 11 GBq generator package. The field measurements were a factor of 5 greater. This led to an investigation of labelled activity. The generator activity was marked on each package for the date of delivery at a hospital - not for the date of despatch which could be one, two or even three (trans-Atlantic shipments) half-lives earlier. Thus package dose rates could be a factor of 8 greater than the activity given on the label would imply.

1.4.2 Handling techniques

Handling techniques were investigated using a tissue equivalent phantom and a 390 MBq caesium-137 source, yielding a dose rate of 30 μGy/h at 1 m, equivalent to a transport index of 2.5, typical of many higher activity generators.

Generator packages, generally, are either 0.4 m cubic packages (80%) held against the chest when carried, or are 0.3 m diameter drums with a handle (10%) carried at knee-level. Carrying the drum at knee height reduces whole-body exposure by a factor of 25, reduces gonad exposure by 8 and increases extremity exposure by 28.

Consideration was given to increasing the dimensions of the cubic packages in order to reduce the surface dose rate. An increase from 0.38 m to 0.46 m reduces the surface dose rate by 30% but increases the volume of the package by nearly 80% and would make handling more difficult.

The principal UK manufacturer had previously determined that their transport drivers receive 30% of their dose during driving and 70% during handling of packages. Distant handling would considerably reduce the whole-body handling exposure from 70% to 3% of the total exposure, an overall reduction (for the most exposed workers) from 20 mSv.a⁻¹ to 8 mSv.a⁻¹.

1.5 Optimisation procedures

If distant handling of generator packages can be obtained, then savings in dose to workers can be achieved.

The provision of a handle, strap or net would enable this to be accomplished. At the time of this study, nets commonly used to carry bulky 13 kg loads cost £80 per 1000 or £0.08 per net. The cost of a generator was approximately £250, thus the extra cost of a net would be <0.1%.
The overall cost for 14,000 cubic packages in a year would be £1,100 for a saving of \(120 \times 10^{-3}\) man.Sv (£9,000 per man.Sv). In its latest publication[3] on determining whether radiation doses are as low as reasonably achievable, the Board recommends a value of about £20,000 per man.Sv for costing of optimisation of protection for workers with annual individual doses in the range \(10^{-2}\) to \(2 \times 10^{-2}\) Sv: this corresponds with the most exposed of the transport workers considered in this study.

On a practical level, the manufacturers of the generators studied are not recommended to change their current packages. Instead the package need only be placed in the net; broad mesh is suggested - 0.1 m net pitch would avoid obscuring the labels. Some simple strengthening of the hand-held area could be achieved and these measures would ensure minimal interference with the current generator packages in use in the UK and probably much of the world.

Other improvements are possible - rearrangement of shield structures, relocation of sources central to shields, redistribution of shielding to minimize the effects of sources which are non-central to the package. It is believed that these would entail redesign of generators at a very high cost, prohibitively expensive, for only marginal improvements in transmission dose rates.

1.6 Conclusion

Distant handling of technetium generators should be examined for justifiable cost-effective reductions in individual doses and in the collective dose received by transport workers.

2. Transport of low and intermediate level wastes

2.1 Introduction

The need for a national strategy for radioactive waste management was recognised by the Royal Commission on Environmental Pollution in 1976[4]. A programme of research to compare the merits of a variety of waste management systems has been co-ordinated by the Department of the Environment (DoE). Waste storage, treatment, transport and disposal have been considered for actual and projected waste arisings. Transport issues will influence decision making because of the costs involved, the impact on occupational dose, the resulting exposure of the public, and other social and political factors. The Board was commissioned by DoE to determine the radiological impact resulting from the movement of low and
intermediate level wastes (ILW) between conditioning plants and storage or disposal facilities. Much of this work required the use of theoretical assessment procedures. However, where possible, the results of radiation surveys and work studies were applied. This paper summarises the results of this work which have been published[5-7] for use in the formulation of Government policy.

2.2 Assessment of the radiation exposure associated with the overland transport of ILW

2.2.1 Modelling assumptions

A theoretical assessment of the radiation exposure of members of the public and workers associated with the options available for the normal transport of ILW generated between 1988 and 2019 was undertaken[2]. The amount of waste arising was consistent with a 'minimum power scenario' in which it was assumed that existing reactors are not replaced and that no new reactors come on-line. For this study DoE specified that the radiation source terms should be based on the maximum dose rates permitted by the International Atomic Energy Agency's Transport Regulations[8], and that the penetrating gamma radiation should have a mean energy of 1 MeV. As no decision has been made regarding the location of any future disposal facility in the UK, notional sites were selected in the north-east of England and East Anglia for deep cavity disposal and engineered trench disposal, respectively. Information on transport logistics and routing requirements for the system were provided by DoE. Both road and rail transport were considered.

2.2.2 Results of the assessment

The assessment has shown that the radiation exposure of the public is low regardless of the mode of transport adopted, a total of about 10 man.Sv from all movements. The highest individual effective dose equivalents are received by people living near to the busiest railway marshalling yard where waste containers may dwell for a few hours. The maximum theoretical annual individual effective dose equivalent is most unlikely to exceed 90 μSv. The collective dose for transport personnel is also assessed to be low, a total of about 10 man.Sv from all movements. However, using the cautious assumptions adopted in this study, certain individuals, in particular the lorry drivers and the loaders at railheads could receive significant exposures. A lorry driver could acquire an effective dose equivalent of 5 mSv during 250 hours in the cab and a loader involved in 140 operations might receive a similar dose. The practical
significance of these findings will be determined by actual work patterns and the actual levels of radiation penetrating the transport containers. In order to make more realistic estimates of the radiation levels penetrating transport containers loaded with radioactive waste, it was necessary to consider the assignment of waste streams to designs of transport container[6] as specified in the DoE research program. Good design and operating procedures result in radiation levels below regulatory limits and it was assumed that an operational maximum of 0.02 mSv.h\(^{-1}\) will prevail at a distance of 2 m from the transport containers. The radionuclide inventory of the waste arisings for each waste stream were supplied by DoE and for the purposes of this work a 1:1 ratio of waste to concrete was assumed for the waste conditioning process. For many ILW streams, cobalt-60 is the dominant radionuclide contributing to external dose rate. Isotopes of caesium are also important for some waste streams. Certain wastes will not give rise to radiation penetrating the shielding of the container while others may require storage prior to transportation to comply with dose criteria. For those wastes requiring storage, it is usually the contribution from cobalt-60 which dictates the duration of storage. Cobalt-60 has a half-life of 5.27 years and storage times of several decades are predicted to be necessary for some situations. Furthermore a short period of storage may enable a less heavily shielded container to be used. These predictions are tentative since actual dose rates will be dependent on the changes in activity concentrations during waste conditioning. Also, waste from a number of different waste streams may well be placed in a single transport container.

2.3 A review of the radiation exposure of transport personnel during radioactive waste sea disposal operations

The report of the Independent Review of Disposal of Radioactive Waste in the North-east Atlantic[9], chaired by Professor Holliday, was published in November 1984 and accepted by the UK Government. As a response to the recommendation that an assessment should be made of the Best Practicable Environmental Option (BPEO), DoE coordinated a study giving particular attention to the comparison of sea dumping and land disposal options. The Board was commissioned to review the radiation exposure of transport personnel during radioactive waste sea dumping operations.

Since 1977 sea dump operations have been managed by the UK Atomic Energy Authority. Information on the exposure of
the dockers and ship's crew was contained in the reports of the Escorting Officers for the years 1977 to 1982. Collective doses to both dockers and ship's crew have reduced by about eight-fold since 1977. In 1982 the collective effective dose equivalent received by dockers was about $\frac{1}{3} \times 10^{-2}$ man.Sv and that to the ship's crew was about $3 \times 10^{-3}$ man.Sv. Individual doses have also been reduced and in 1982 these were less than 1.5 mSv for both groups of workers. These dose savings were achieved despite a general increase in the quantity of radioactive waste dumped. This was probably due to the improved shielding of the packages and radiologically improved work procedures. If ocean disposal of solid or solidified radioactive waste was to be resumed, the exposure of transport personnel might be expected to be comparable to the low doses received in the early 1980s. However changes in packaging, handling procedures and the frequency of movements could have major effects on radiation exposures.

REFERENCES

METHODOLOGIES FOR ASSESSING THE RADIOLOGICAL IMPACT ARISING FROM THE TRANSPORT OF RADIOACTIVE MATERIALS

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Abstract

METHODOLOGIES FOR ASSESSING THE RADIOLOGICAL IMPACT ARISING FROM THE TRANSPORT OF RADIOACTIVE MATERIALS.

The paper seeks to provide advice on the suitability of existing radiological consequence models for specific transport situations. The review was commissioned by the Health and Safety Executive of the United Kingdom as part of its programme of research to provide a firm basis for its requirements in relation to transport operations. Many mathematical models were identified as having been applied to transport issues. Four important models were selected for closer scrutiny: CRAC 2, RADTRAN II, INTERTRAN and MARC. Large deterministic codes, for example CRAC 2 and MARC, are useful when applied to situations that warrant detailed probabilistic assessments of radiological consequences. This may be the case for large notional releases from Type B packages, which may be shown to be of particular significance from the results of a less detailed consequence model, for example, in a risk analysis code, such as RADTRAN II or INTERTRAN. It is noted that all attempts to quantify the risk from the transport of radioactive materials suffer from a lack of good input data concerning the probabilities of occurrence for defined accident severity categories and the associated values for release and aerosolization factors.

1. Introduction

Methodologies for assessing the radiological impact arising from the transport of radioactive materials (RAM) can be used in retrospective studies. To date the need for such studies has been limited because transport accidents involving RAM have been rare and very few have resulted in a loss of material to the environment, hence the impact has been small. Assessments of radiological impacts also have roles to play in the development of statutory requirements, optimisation exercises, design criteria, emergency planning procedures, risk analyses, reviews of radiation exposure and in contributing to the public debate concerning transport issues.
This review of methodologies[1] was commissioned by the Health and Safety Executive of the United Kingdom as part of its programme of research to provide a firm basis for its requirements and recommendations in relation to transport operations. The objectives of this paper are to provide advice on the suitability of existing models for specific transport situations with respect to radiation protection.

At least two types of situations have to be addressed by transport accident models. Firstly, loss of shielding incidents which give rise to enhanced external radiation levels but which do not necessarily involve a release of RAM. Secondly, accidents which cause a loss of containment leading to a release of a respirable aerosol. A recent review of accidents and incidents during the transport of RAM in the UK [2,3] has shown that loss of shielding events have been the main contributors to radiation exposure.

Mathematical models are the most practicable method of analysing postulated incidents involving radioactive materials. Application of these models to transport operations is particularly difficult because the location of an incident can be almost anywhere and there is a very wide diversity of materials, package types and modes of transport to consider. In the absence of an ideal model, the use of mathematical models alone will not provide complete solutions to the problems to which they are applied. The choice of model, either from the range available or to be developed, will be determined by the project objectives and the effort and systems available.

A comprehensive assessment of radiological consequences will cover the following aspects: atmospheric dispersion, meteorology, population distribution, dosimetry, health effects and in some cases countermeasures, agricultural production and economic impacts. The model should address all potentially significant pathways of exposure:

(i) Direct exposure from penetrating radiation from a consignment with impaired shielding;
(ii) Penetrating gamma radiation emerging directly from a plume of released activity;
(iii) Exposure of the skin to beta radiation emerging directly from the plume;
(iv) Direct inhalation of material that has become respirable;
(v) Inhalation of material which has been deposited and later resuspended;
(vi) External irradiation from skin contaminated by radioactive material;
(vii) External irradiation from ground contaminated by the deposition of airborne activity;

and for some releases:

(viii) Exposure from the consumption of contaminated foodstuffs.

For many circumstances relating to transport operations it will be unwarranted to model all pathways of exposure. Some models cover a larger number of aspects and exposure pathways than others but all share some common ground.

2. A summary of existing radiological consequence models

A literature search was conducted to identify published radiological consequence models which have been applied to the transport of RAM. Several of the models identified are related to one another. Many are adapted from reactor accident codes.

CRAC (Calculation of Reactor Accident Consequences)\[4\] was developed at Sandia National Laboratories in 1975 and revised in 1981 to become CRAC 2[5]. It is a consequence model that has been applied to postulated releases during transport operations.

RADTRAN\[6\] is a risk assessment methodology written in 1977 and updated in 1980 to become RADTRAN II[7]. Work on RADTRAN III is currently being undertaken at Sandia.

INTERTRAN[8] is an international code for risk assessment coordinated by the IAEA in 1979 and assisted by the Swedish Nuclear Power Inspectorate. RADTRAN II is the basis for the INTERTRAN code.

MARC (Methodology for Assessing Radiological Consequences)\[9\] was developed by the NRPB in 1982. Like CRAC it is a tool for use in risk assessments that has been applied to transport operations.

NECTAR[10] was written by and for the CEGB of the UK in 1982 and is used in the design and safety studies for future plant. It is essentially a deterministic code that has been applied to transport studies.

TREC II[11] was initiated at Pacific Northwest Laboratory in 1979 and was transferred to Sandia in 1980. It was developed for the US Department of Energy as a risk analysis tool for evaluating high level waste management systems.
TIRION[12] was developed by the Safety and Reliability Directorate (SRD) of the UK Atomic Energy Authority (UKAEA). Version 4 was completed in 1978.

TRIP[13] estimates the risks to the public arising from the overland transport of hazardous cargoes. It was written by SRD in 1980.

3. A comparison of selected models

For detailed comparison CRAC 2, RADTRAN II, INTERTRAN and MARC were selected as being the models most often applied to transport studies.

3.1 Atmospheric dispersion

CRAC 2 and MARC employ a similar level of complexity based on implementations of the Gaussian plume model. Plume rise and both wet and dry deposition are modelled. Both codes were developed primarily for accident consequence assessments of a nuclear plant. Source terms for transport accidents are generally smaller and it is important for the interval lengths at which concentrations are calculated to be correspondingly finer to compensate for large potential variations in concentrations over short downwind distances. RADTRAN II and INTERTRAN share a simplified approach to atmospheric dispersion using a tabulation of time-integrated dilution factors. Plume rise and wet deposition mechanisms are not modelled.

3.2 Meteorology and meteorological sampling

Accident consequence models can be used to predict the impact of a release of radioactive material in a particular set of meteorological conditions. However their most frequent prospective application is to estimate the statistical distributions of consequences which arise because a notional release will be subject to a range of meteorological conditions with associated probabilities of occurrence. CRAC 2 and MARC have similar meteorological sampling routines which ensure that the entire range of consequence for a given release are adequately covered to probabilities of occurrence as low as $10^{-3}$. These models are able to take account of changes in atmospheric conditions during the time that the plume is travelling.

In INTERTRAN and RADTRAN II, meteorological conditions are characterised by the Pasquill stability categories A to F. The effects of rainfall are ignored and wind direction is unimportant because uniform population distributions are
TABLE I. SUMMARY OF PATHWAYS OF EXPOSURE CONSIDERED BY THE MODELS FOR ATMOSPHERIC RELEASES OF RADIOACTIVITY

<table>
<thead>
<tr>
<th>Pathway of exposure</th>
<th>CRAC 2</th>
<th>RADTRAN II</th>
<th>INTERTRAN</th>
<th>MARC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cloud-(\gamma) irradiation</td>
<td>Yes</td>
<td>No</td>
<td>No</td>
<td>Yes</td>
</tr>
<tr>
<td>Cloud-(\beta) irradiation</td>
<td>No</td>
<td>No</td>
<td>No</td>
<td>Yes</td>
</tr>
<tr>
<td>Inhalation of the plume</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
</tr>
<tr>
<td>Resuspension</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
<td>Optional</td>
</tr>
<tr>
<td>External (\gamma)-irradiation from deposit</td>
<td>Yes</td>
<td>No</td>
<td>No</td>
<td>Yes</td>
</tr>
<tr>
<td>Ingestion</td>
<td>Yes</td>
<td>No</td>
<td>No</td>
<td>Yes</td>
</tr>
</tbody>
</table>

employed. Although a range of windspeed is possible in each Pasquill category a single representative speed is assigned to each category. Constant meteorological conditions are assumed to persist for the duration of the plumes travel. This approach is very sparing of computational resources, but will not give the full range of consequences reflecting the spectrum of meteorological conditions.

3.3 Population distribution

The population distribution is defined in terms of the number of people living with each of a number of angular sectors and distance bands in CRAC 2 and MARC. Users may choose to use uniform population densities: concentric distributions or actual population densities using census data. RADTRAN II and INTERTRAN use up to three evenly distributed population zones corresponding to urban, suburban and rural districts.

3.4 Pathways of exposure for atmospheric release of radioactivity

A summary of the pathways of exposure considered by each of the models is presented in Table I. CRAC 2 and MARC model all major pathways of exposure while RADTRAN and INTERTRAN include only the most significant pathways for most releases.
TABLE II. SUMMARY OF HEALTH EFFECTS CONSIDERED FOR ATMOSPHERIC RELEASES OF RADIOACTIVITY

<table>
<thead>
<tr>
<th></th>
<th>CRAC 2</th>
<th>RADTRAN II</th>
<th>INTERTRAN</th>
<th>MARC</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Early effects</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Mortality</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>bone marrow irradiation</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
</tr>
<tr>
<td>lung irradiation</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
</tr>
<tr>
<td>G.I. Tract irradiation</td>
<td>✓</td>
<td></td>
<td></td>
<td>✓</td>
</tr>
<tr>
<td><strong>Morbidity</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>lung fibrosis</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
</tr>
<tr>
<td>prodromal vomiting</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
</tr>
<tr>
<td>stem cell loss</td>
<td>✓</td>
<td></td>
<td></td>
<td>✓</td>
</tr>
<tr>
<td>temporary sterility</td>
<td>✓</td>
<td></td>
<td></td>
<td>✓</td>
</tr>
<tr>
<td>(males)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>early thyroid effects</td>
<td>✓</td>
<td></td>
<td></td>
<td>✓</td>
</tr>
<tr>
<td><strong>Latent effects</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Fatal cancer</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>bone</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
</tr>
<tr>
<td>leukaemia</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
</tr>
<tr>
<td>lung</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
</tr>
<tr>
<td>breast</td>
<td>✓</td>
<td></td>
<td></td>
<td>✓</td>
</tr>
<tr>
<td>liver</td>
<td></td>
<td></td>
<td></td>
<td>✓</td>
</tr>
<tr>
<td>thyroid</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
</tr>
<tr>
<td>lower large intestine</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
</tr>
<tr>
<td>skin</td>
<td></td>
<td></td>
<td></td>
<td>✓</td>
</tr>
<tr>
<td>stomach</td>
<td>✓</td>
<td></td>
<td></td>
<td>✓</td>
</tr>
<tr>
<td>pancreas</td>
<td>✓</td>
<td></td>
<td></td>
<td>✓</td>
</tr>
<tr>
<td>others</td>
<td>✓</td>
<td></td>
<td></td>
<td>✓</td>
</tr>
<tr>
<td><strong>Non-fatal cancer</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>thyroid</td>
<td>✓</td>
<td></td>
<td></td>
<td>✓</td>
</tr>
<tr>
<td>skin</td>
<td>✓</td>
<td></td>
<td></td>
<td>✓</td>
</tr>
<tr>
<td>breast</td>
<td>✓</td>
<td></td>
<td></td>
<td>✓</td>
</tr>
<tr>
<td><strong>Hereditary effects</strong></td>
<td></td>
<td></td>
<td></td>
<td>✓</td>
</tr>
</tbody>
</table>

3.5 Health effects modelling for atmospheric releases of radioactivity

The health effects modelled by each of the codes are identified in Table II. There are significant differences in the methods used by the codes to estimate the incidence of the same health effect. Constraints of space do not allow these differences to be covered in this paper. Some of the most important aspects are listed in Section 5.
4. The application of radiological consequence models to the near field

The near field is taken to correspond to an area extending to a radius of about 100 m from the incident. It is an area which is not covered in detail by the models reviewed in this paper. Clearly it is an area of prime importance and the exposure of persons in the vicinity, including the crew of the conveyance and the emergency teams who may be expected to undertake duties in this area, should be taken into account.

Generic assessments of near field radiological consequences are difficult. Exposures incurred will be highly dependent on incident specific details including: the precise location of the release, local topographical features, involvement of injury to personnel, time of day, the geometrical configuration of the RAM with respect to the packaging and conveyance and the numbers of persons present. It is acknowledged that similar uncertainties prevail in all consequence modelling but exposures in the near field are particularly sensitive to these factors.

In the event of an accident occurring in the UK, contingency plans and emergency arrangements are established to make it likely that operations in the near field will be conducted under health physics supervision within about 2 hours of the rescue services attending the scene. Subsequent radiation exposures in the near field would be planned as far as possible. Emergency services include dealing with radioactive materials as part of their training.

5. Recommendations

The choice of model to apply to a given problem will be determined by the objectives of the project and the effort available. The quality and sophistication of the model output will be related to the resources required to provide commensurate input data.

Codes such as CRAC 2, MARC and NECTAR address a comprehensive range of pathways of exposure, health effects and meteorological conditions and use relatively complex methods to model atmospheric dispersion. Considerable effort will be required by a user to become familiar with the large amount of input data. These large demands, particularly on computer resources, make it unlikely that such models will be applied to comprehensive risk analyses of transport operations. Such work requires radiological consequence models to be run for ranges of release fractions, package
types, package contents and release locations. Large
deterministic codes are useful when applied to transport
events that warrant detailed probabilistic assessments of
radiological consequences. This may be the case of large
notional releases from some Type B packages in urban areas,
identified perhaps, as being of particular significance using
a less detailed consequence model in a risk analysis.

RADTRAN II and INTERTRAN are designed to be simple methods
for the assessment of risk from transport operations involving
RAM. Probabilities of occurrence are assigned to each of
eight severity categories as functions of mode of transport,
degree of degradation of package integrity and release
fractions of package contents. To achieve the objective of
risk assessment, the radiological consequence model must be
run for each release considered. Therefore the model has to
be relatively sparing of computer resources to execute in a
reasonable timescale. Comments on the default input data of
these codes regarding the eight severity categories lies
outside the scope of this paper. With respect to the
radiological consequence models, the following points are
important when considering the output:

(i) The atmospheric dispersion model is simplistic and
    considers only constant meteorology;
(ii) Only dry weather conditions are modelled;
(iii) Uniform population distributions are assumed;
(iv) Direct exposure from radiation emerging directly
    from the plume is not modelled;
(v) The incidence of early mortality is based solely on
    irradiation of the lung;
(vi) The incidence of early morbidity is computed using
    a 50-year committed dose equivalent;
(vii) The incidence of fatal cancer is calculated using
    risk factors which are applicable to a population
    which will live long enough for the total risk to
    be expressed;
(viii) It is assumed that the population will be evacuated
    after an exposure time of 24 hours for a period of
ten days and only returned if doses are less than a
specified clean-up level.

In 1985 the IAEA convened a Technical Committee[14] which
was charged to assess the radiological impact from the
transport of RAM. One of the working groups specifically
examined the problems encountered by users of INTERTRAN in
order to identify these problems in an organised manner, to
develop potential solutions where possible, and to recommend
ways in which the usefulness of INTERTRAN could be improved.
The report of that working group considered the accident
section of INTERTRAN, although suffering from a number of important deficiencies, to be a useful framework for accident assessment, in particular to facilitate the exchange of information between countries. The report also stated that users should supplement their risk analysis with a consequence analysis or safety assessment.

Loss of shielding events may give rise to enhanced external radiation levels while the conveyance is moving or stationary. These events can be assessed using models designed to estimate the collective and individual doses from external gamma radiation during normal transport operations. The degree of sophistication required to model such exposures is far less than that needed to model releases of RAM to the environment. Such models are included in RADTRAN II, INTERTRAN and an NRPB model called TRANSDOS[15] which is able to use the population data based on the 1 km² grid for Great Britain.

All attempts to quantify the risk from the transport of RAM suffer from a lack of good input data concerning the assignment of probabilities of occurrence to defined accident severity categories, values of release fractions and aerosolisation factors. It is in these areas that the most important advances can be made.

REFERENCES

RADIOLOGICAL IMPACT OF TRANSPORT ACCIDENTS AND INCIDENTS IN THE UNITED KINGDOM OVER A TWENTY YEAR PERIOD

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Abstract

RADIOLOGICAL IMPACT OF TRANSPORT ACCIDENTS AND INCIDENTS IN THE UNITED KINGDOM OVER A TWENTY YEAR PERIOD.

A review has been performed in the United Kingdom (UK) of the radiological impact resulting from accidents and incidents occurring during transport over the period 1964 to 1983. This work was jointly commissioned by the UK Health and Safety Executive and the Department of Transport. The materials transported consisted of nuclear fuel cycle materials, and radioisotopes for medicine and industrial use. The modes of transport studied were road, rail, sea and air. Information in this paper should be used with caution because the database is likely to be incomplete and selective owing to the difficulties of data collation for the review. These difficulties arise because information is sparse as a result of the low frequency of occurrence of such events and to the lack of published information on their impact. In the 20 year period a total of about 330 events were recorded for an estimated 720,000 shipments. Of these events only 42 had the potential to exceed or did exceed the radiological impact associated with normal transport conditions. Over 98% of the total collective dose of about 5 man • Sv may be attributed to 15 events. It is in part due to regulatory control that there has never been a serious accident involving the dispersal of radioactivity during transport in the UK. The regulations are directed to ensuring that safeguards appropriate to the nature and quantity are built into the design of the package in which the material is to be transported. A review extending over a twenty year period cannot be used to derive the probability of occurrence of severe accidents which have very low probabilities. The review has demonstrated that the majority of significant events in the UK are related to procedural and quality assurance failures. These failures are aspects which should be addressed in a comprehensive assessment of the radiological impact of transport operations.

Introduction

The UK Health and Safety Executive, jointly with the Department of Transport, commissioned the National Radiological Protection Board to carry out a study of accidents and incidents arising from the transport of radioactive materials, in the UK, covering the period 1964 to 1983.
It was important from the start of the study to specify the events to be covered. Events were classified as occurring either during the moving phase of transport or during handling prior to, during, or subsequent to the movement. Further, the events were categorised either as accidents or as incidents. The former involved significant damage to the transport or package whilst the latter dealt with other events, for example, theft, incorrect procedures, delays and incorrect packaging prior to shipping. The term 'significant damage' was limited to the situation where the load was subject to potential disruption; it excluded situations of trivial damage with negligible risk to packages.

Formal reporting of accidents was not required until 1970, when formal requirements were introduced for reporting to the appropriate authority [1,2]. The formal reports were brief, giving few details and, where the radiological impact was low, commenting little or only that dose limits were not exceeded. Few data were available on the exposure of members of the public.

Methodology

Investigation of the formal reports yielded limited data on some 200 events during the 20 year period. In addition, the Board held files on transport incidents and accidents involving radioactive materials under the NAIR scheme (National Arrangements for Incidents involving Radioactivity). Many of these reports referred to transport accidents or incidents but, again, held only limited data because the radiological consequences were always low.

Board assistance had been requested from time to time in evaluating substantial doses to radiographers travelling from remote locations to their headquarters with a gamma radiography source out of its shielded container. This group of events - transport incidents involving incorrect procedures prior to commencing a journey - was included in the report.

A questionnaire was prepared and distributed to all major transporters of radioactive material, explaining the classification and requesting data on relevant events. Once data had been collated from all readily available sources, site visits were made in order to extend the database and to expand the information on known events.

As the study progressed it became clear that few companies had retained data for more than 5 to 10 years previously, since the majority of all events were radiologically trivial and company archives are often cleared at regular intervals in
order to retain only significant information for long-term record purposes. Thus for the early years of this study, data were sparse, but sufficient for the overall purposes of this report.

It is believed that no significant events have been overlooked or not recorded for the period of the study.

Materials and their modes of transport

From within the UK, radioactive materials are transported by road, rail, sea and air - about 37,000 shipments (containing various numbers of packages) in 1981/82 [2] and, because of limited data from earlier years, this same value is applied without modification to the earlier years of this study.

Materials transported were considered under three broad headings (with sub-divisions where appropriate), namely non-irradiated nuclear fuel cycle material, irradiated spent nuclear fuel plus waste products arising and radioisotopes. Consideration was given to their physical and chemical form - whether solid or liquid, sealed and encapsulated or in the form of drums of chemical powders, in order that allowances could be made for whether the radiological hazards were external or included the potential for internal exposure. Criticality has been considered where relevant.

During the period under study, road transport accounted for 68% of movements, air transport 21%, rail transport 8% and sea transport 3%.

As can be seen from Table I, radioisotopes used for radiography constitute the largest single number of annual shipments with other general radioisotopes the next largest group.

Reports received

Some 330 events have been reported out of nearly three quarters of a million transport shipments in the twenty year period of this study. Events were randomly distributed around the UK with a slight emphasis on nuclear materials in Lancashire and Cumbria in the later 1960s and early 1970s because of a combination of the presence of ports, chemical and reprocessing plants and particular operations in those areas. The nuclear material principally concerned was uranium ore concentrate (UOC) carried in industrial drums. Events during unloading of vessels at ports (10 reports) and road accidents (4 reports) led to spills of UOC, with consequential
Table I
MATERIALS AND TRANSPORT MODE (1981/82)

Numbers of consignments carried annually

<table>
<thead>
<tr>
<th></th>
<th>Non-irradiated</th>
<th>Irradiated</th>
<th>Isotope use</th>
<th>Radiography</th>
</tr>
</thead>
<tbody>
<tr>
<td>Road</td>
<td>3 000</td>
<td>1 000 (flasks)</td>
<td>9 000</td>
<td>12 000</td>
</tr>
<tr>
<td></td>
<td></td>
<td>1 000 (wastes)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Rail</td>
<td>Nil</td>
<td>1 100</td>
<td>1 600</td>
<td>260</td>
</tr>
<tr>
<td>Air</td>
<td>Negligible</td>
<td>Nil</td>
<td>8 000</td>
<td>Negligible</td>
</tr>
<tr>
<td>Sea</td>
<td>1 000</td>
<td>&lt;50</td>
<td>200</td>
<td>Negligible</td>
</tr>
</tbody>
</table>

The table shows numbers of consignments with the majority representing single packages except for radioisotopes and low-level wastes. For radioisotopes, a typical road consignment could contain 300 packages. For low-level wastes, typically up to 100 drums could be carried as a single load by road.

Light contamination, low internal and external exposure of drivers and members of the public resulting in low radiological consequences, and with no exposure exceeding 2 mSv for dock workers or drivers and 0.01 mSv for a member of the public. Containerisation of these drums since 1978 has resulted in no further releases being reported.

Radioisotopes are transported throughout the UK with radiography isotope movements concentrating around industrial centres and the edges of cities. Some of these latter movements have taken place with unshielded sources, either by accident or by improper procedure prior to the transport movement. Thus, in addition to radiation doses to the radiographers (who also transport these sources), members of the general public have also been exposed but again it is estimated that none of this latter group have exceeded a radiation dose of 2 mSv arising from any single event.
Nearly half the reports (140) related to events occurring in the cargo area of the principal UK airport. With more than one daily road delivery of medical isotopes intended for world-wide export by air, several hundred individual packages are handled daily (70 000 per annum) [2]. Typical of many bulk-cargo handling depots, packages were moved within the cargo area stacked loose on pallets mounted on lift trucks. This has resulted in packages falling from the front of the pallet, not being noticed by the driver and being run over by the lift truck or other vehicles making collections or deliveries to the airport cargo centre. In only a single event was the inner capsule, containing a radioactive liquid, broken, releasing its contents and resulting in minor contamination of the aircraft and two handlers loading the plane. One other event with a broken capsule occurred after the incorporation of absorbers in packages and so no release resulted.

In none of the other 138 events was there any damage to the inner capsules. Loose stacking has been reduced in recent years and is reflected in a reduction in airport incidents reported.

Rail transport of spent nuclear fuel and discharged fuel flasks has resulted in reports of nearly 40 incidents, mostly involving low speed collisions or derailments, principally at marshalling yards. In none of these events has there been any radiological consequences, either from external or internal exposure, nor has any irradiated nuclear fuel flask suffered any damage.

Some 20 reports of contamination on irradiated nuclear fuel flasks or their associated flatrol transporters occurred evenly over the study period. Most of these reports arise from reactor cooling pond water being contaminated, and this contamination being preferentially absorbed in painted flask surfaces. Despite subsequent decontamination, the contaminating isotopes - principally caesium-137 and caesium-134 with lesser amounts of strontium-90 - have not proved entirely removable. Adverse conditions lead to later release of these materials from the surfaces and four reports refer to the transfer of contamination from flasks to protective gloves worn by British Rail staff handling the flasks to and from their flatrols. No contamination of persons has been reported.

The contamination reports led to a further investigation of the consignor's reports. Only Magnox fuel flask data were available and these showed a frequent low level of flask contamination, largely below the derived working limit. On
theoretical considerations, with all flasks assumed to be uniformly contaminated at the working level and this contamination released completely throughout a transport movement, calculations showed that doses arising from inhalation of this material resulted in wholly trivial annual doses to individual members of the public or individual transport workers.

About a third of reports associated with irradiated nuclear fuel flasks refer to incorrect procedures. None of these events led to any reported additional exposure of transport workers or members of the public.

Package and transport failures

Package failures resulting in breaches of the containment have occurred and been noted during this study. However, almost without exception, these have been subject to stresses greatly in excess of test limits.

Large industrial packages, subject to a drop test, containing 300 kg UOC have failed after 10 m drops or 50 km/h speeds, when thrown from lorries. Small industrial packages used for radiopharmaceuticals are subject to a compression test. Such packages have failed twice in 140 events, mainly as a result of compressions estimated to be one hundred to one thousand times the package weight.

Steel pressure cylinders used for the transport of uranium hexafluoride were twice involved in severe accidents, once in a road crash, once at sea. In both instances the cylinders were dislodged or torn from their mountings, and were thrown onto the road surface or vessel decking and suffered no other damage. No leakage and no contamination occurred.

An early despatch of new fuel involved in a road accident was spilt onto the road when the steel chest burst on impact. No radiation doses were recorded but these were subsequently estimated to be trivial - below 0.1 mSv.

There have been few transport failures reported in the period of the study - twenty two colliding or crashing road vehicles, two vehicle fires, some twenty derailments of railway rolling stock and two aircraft crashes but no sea-vessel failures.

The road vehicle events resulted in external damage to packages but with negligible radiological consequences. The rail reports had no radiological consequences and one of the air crashes resulted in the outer packaging only of a
radiography container being destroyed but without breaching of
the inner containment. The other air crash with resulting
fire resulted in severe damage to about 30 packages with loss
of containment, shielding and - for volatile sources - loss of
contents. As a result of rapid radiological supervision, the
radiological consequences were negligible.

Incorrect procedures

Incorrect procedures have led to the largest radiological
consequences encountered in this study. There are 48 reports
for radioisotopes and 12 for nuclear fuel cycle materials.

The nuclear fuel cycle reports related mainly to spent
fuel flasks (11) with negligible radiological impact.

The 48 reports relate to radioisotopes with 15 of
radiological significance, referring to radiography sources.
This latter group provided the major radiological impact in
this study. Radiography sources, ranging in activity from
0.03 to 1 TBq of iridium-192 (13) caesium-137 (1) and
cobalt-60 (1) have been transported out of containers for
periods ranging from 1 hour up to 3 days, with overnight
parking in public areas in the latter case. The events have
generally arisen from operators failing to monitor radiography
exposure containers, which also serve as transport containers,
at the end of a working day. The unshielded sources have been
transported in road vehicles, at distances of 1 to 3 m from
the radiographer. In three instances, overnight parking has
been with the vehicle either 10 m or 50 m from adjacent
housing for 12 hour exposure periods, with an assumed minimum
of 200 mm thick brick walls intervening. Individual worker
radiation doses ranged from a few millisievert up to 2.4 Sv
(measured) individual public radiation doses ranged from 0.1
to 2 mSv (estimated).

The occupational collective dose to approximately 80
workers (all connected with radiography) amounted to 5 man.Sv,
whilst the collective dose to the public amounted to 0.02
man.Sv for the same 15 events. The remaining collective
occupational and public doses were less than 2% of these
values (Table II) for all other events combined.

Contingency planning

For the first ten years of the study, contingency planning
for the consignors was broadly limited to the large
organisations - the three major groups in the nuclear power
industry and the major UK radioisotope producer. These
companies had adequate resources and appropriately trained
Table II

OCCUPATIONAL AND PUBLIC COLLECTIVE DOSE

<table>
<thead>
<tr>
<th>Material transported</th>
<th>Estimated dose (20 year period)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Occupational (man.Sv)</td>
</tr>
<tr>
<td>Radiography sources [Ir-192, Cs-137, Co-60]</td>
<td>5</td>
</tr>
<tr>
<td>Nuclear fuel cycle material</td>
<td>0.06</td>
</tr>
<tr>
<td>Other radionuclides for medical and industrial use</td>
<td>0.003</td>
</tr>
</tbody>
</table>

Footnote. The annual collective dose to the UK population from natural sources is about 105 000 man.Sv.

staff and were able to respond when called to the scenes of accidents and incidents.

Prior to the arrival of these consignors, control was then, and still is, essentially with the police force, who were invariably the first to be called to the scenes of accidents and who secured control of the areas surrounding the events, excluding other persons from entering until radiological supervision could be provided, usually from the NAIR scheme.

The NAIR scheme provides the police with a rapid technical level of support from local professional staff, e.g. hospital or power station health physicists, with supplementary support of a full team if required. This latter facility has seldom been required because few events have yielded any significant radiological impact.

Unfortunately the most significant events - those concerning radiographic sources - have rarely been recognised as events until the equipment has been returned to the consignee. The group of workers most closely involved, namely the site radiographers, are subject to close inspection.
Conclusion

Based on an estimated three quarters of a million transport movements in the 20 year study period, there have been some 330 known events reported, with 15 providing over 98% of the radiological impact.

The major component of both occupational and public doses are directly attributable to the movements of radiography sources. Only a very minor contribution arises from the transport of nuclear fuel cycle materials.

Site radiography is frequently undertaken in conditions which are difficult and hostile, such as construction sites, pipelines, and motorway bridges. These conditions are less conducive to the normal standards of safety associated with nuclear plants, factories and laboratories.

Although early data were sparse, details obscure and, for the earliest years, missing entirely from some archives, it is believed unlikely that any major event has been omitted from this study.

The majority of events are related to procedural and quality assurance failures and improvements in this area would reduce the radiological impact. This is an aspect which should be addressed in a comprehensive assessment of the radiological impact from the transport of radioactive materials. The requirements for packaging and transport have been shown to be generally adequate. Improved reporting is recommended particularly with respect to data relating to emergency personnel.

The occupational collective dose amounted to 5 man.Sv in a 20 year period and the public collective dose in the same period amounted to 0.025 man.Sv. These collective doses are low compared to exposures from the normal transport of radioactive materials and are extremely low compared to exposures from natural sources of radiation.

It is partly due to regulatory control that there has never been a serious incident involving the dispersal of radioactive material during transport in the UK. The regulations are directed to ensuring that safeguards appropriate to the nature and quantity are built into the design of the package in which the material is to be transported.
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URANIUM HEXAFLUORIDE IN TRANSPORT ACCIDENTS

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Abstract

URANIUM HEXAFLUORIDE IN TRANSPORT ACCIDENTS.

After a brief description of the physical and chemical properties of UF and of the products of its hydrolysis, UO₂F₂ and HF, the problem of the radiological and chemical risks in transport accidents is analysed. 'Acceptable levels of exposure' for making rough safety decisions based on chemical hazards are suggested. The present practice in the transport of UF₆ is described and some apparent inconsistencies and the different levels of safety used to protect against chemical or radiological hazards in transport are noted. Finally, a mass limit for UF₆, based on chemical hazards, is derived and the main questions to be answered in relation to safety requirements in the transport of UF₆ are presented.

1. INTRODUCTION

In the nuclear power industry, large quantities of uranium are converted from one chemical form to another and, when the nuclear fuel cycle is based on enriched uranium, one of the most widely handled and transported chemical forms is uranium hexafluoride (UF₆). As a gaseous compound of uranium at relatively low temperature, the UF₆ is used in the enrichment process at diffusion or centrifugation plants. Therefore, massive amounts of natural, depleted or enriched uranium are transported as UF₆ to and from enrichment facilities.

At room temperature and atmospheric pressure, uranium hexafluoride is a white solid of relatively high density (about 5 g·cm⁻³) that sublimes slowly in dry air. Under higher pressures and temperatures (e.g. 0.4 MPa and 70°C), the solid UF₆ melts to form a colourless liquid of high density (about 3.6 g·cm⁻³) that, when the temperature increases, converts into vapour. At high temperatures and at atmospheric pressure or below, the solid can be converted directly into vapour (the atmospheric sublimation point is about 56.4°C) [1]. While the gaseous form is used in the enrichment process, the liquid form is usually employed in the operations of filling or emptying storage and transport containers and the product is transported as a solid at pressures slightly below atmospheric pressure [2].

Uranium hexafluoride is highly reactive with hydrogenous compounds, such as water and oils. When UF₆ reacts with water, a substance always expected to be present in a transport environment, the reaction products are uranyl fluoride.
(UO$_2$F$_2$) and hydrogen fluoride (HF). Therefore, when UF$_6$ is released into the atmosphere, it rapidly reacts with ambient moisture to form an aerosol of UO$_2$F$_2$ and HF. Anhydrous UO$_2$F$_2$ is hygroscopic, as is HF, and both substances tend to become hydrated. The particulate UO$_2$F$_2$ is easily visible as a white cloud or 'smoke'. The combination of possible releases of UF$_6$ under different conditions relating to the presence of water, the further reaction or combination of the UF$_6$ or of its initial hydrolysis products with the moisture in the air and the possible interactions among them do not allow for a reasonable prediction of the actual products to be found and of their aerosol characteristics at different distances from the release point [3, 4].

2. RADIOLOGICAL AND CHEMICAL RISKS

Both UF$_6$ and UO$_2$F$_2$ present chemical and radiological risks, while HF is a highly corrosive substance and only involves chemical risks. Although the radiological hazard of UF$_6$ increases with enrichment, due to the increase of $^{234}$U, the consequences of an accidental release of UF$_6$ during transport are largely associated with the chemical hazards caused by UF$_6$, UO$_2$F$_2$ and HF.

The criticality risk, only possible with significantly enriched UF$_6$, does not appear to be a problem because the criticality control of UF$_6$ systems is not difficult under normal conditions by controlling the presence of HF in the UF$_6$ system, and because the accidental incorporation of a common moderator, such as water, produces immediately the exothermic reaction indicated above. Therefore, a critical configuration during or after the chemical reaction of UF$_6$ with water [5, 6] is quite improbable and, in some cases, physically impossible in a transport accident.

2.1. Radiological risk

The radiological risk of uranium increases with enrichment because the enrichment process produces a relative increment of the $^{234}$U content. The final relative content of $^{234}$U, for a defined $^{235}$U/$^{238}$U ratio, is a function of the way in which the enrichment plant is operated and of the previous history of the uranium processed [7, 8]. Conservative estimates of the maximum expected concentration of $^{234}$U as a function of the degree of enrichment can be made [8, 9].

Even for very high enrichments (e.g. 90%), the mass of $^{234}$U is negligible if compared with the sum of the masses of $^{235}$U and $^{238}$U, but its semidisintegration period is significantly shorter than those of the others and, therefore, its contribution to the total alpha activity ranges between 50% in natural uranium and practically 100% in very highly enriched uranium. Consequently, the specific activity of enriched uranium increases significantly with the degree of enrichment, while the mass intake — equivalent to the Annual Limit of Intake (ALI) — is smaller for highly enriched uranium than for low enriched or natural uranium.
The ALI value recommended by the International Commission on Radiological Protection for uranium is $5 \times 10^4$ Bq [10], equivalent to an annual intake of a mass of about $2 \times 10^3$ mg of natural uranium, 135 mg of 20% enriched uranium, or 23 mg of 90% enriched uranium. A quantity, named Radiological Daily Derived Mass of Intake (RDDMI), can be obtained for each degree of enrichment by dividing the mass value, equivalent to the ALI, by the number of working days in a year (usually 250). Moreover, since uranium is toxic, there is a Toxic Daily Limit of Intake (TDLI), based on potential kidney injuries in routine exposures (the TDLI value for very soluble uranium compounds, such as $\text{UO}_2\text{F}_2$, is about 2 mg per day). From enrichments higher than about 10%, the RDDMI is lower than the TDLI; in these cases and in routine exposures, efforts to control the intake for radiological reasons are more stringent than the ones required for toxicological reasons. It is said that ‘the radiological risk dominates the toxicological risk’. Obviously, this statement is not applicable to accidental situations because the RDDMI can be exceeded by a factor $10^3$ without any significant radiological consequences, while, if the TDLI is exceeded by an order of magnitude or more, the person exposed could suffer severe health effects or die. Therefore, the radiological risk is not further considered in the context of this paper.

2.2. Chemical risks

In a transport accident, it is better to consider the toxicity of UF₆ as the sum of the toxicities of $\text{UO}_2\text{F}_2$ and HF, in view of the readiness with which UF₆ reacts with atmospheric water. Uranyl fluoride is one of the most soluble compounds of uranium and, as such, it presents a high toxicity hazard when inhaled. The health effect of $\text{UO}_2\text{F}_2$ is kidney damage, which could imply the death of the exposed person if the intake is high (e.g. 200 mg), disregarding likely remedial actions. Hydrogen fluoride is a highly corrosive substance and, under acute exposure conditions, the health hazard is the induction of pneumonitis and pulmonary oedema [11, 12].

For making safety decisions in UF₆ transport accidents, one of the main problems is defining an ‘Acceptable Level of Exposure’ (ALE). The ALE is a condition of exposure such that, if not exceeded, no person will be significantly affected in accidental cases. From the information available on recommended values for acute exposures to UF₆, $\text{UO}_2\text{F}_2$ and HF [11, 12], as well as the comparison among them and among the values recommended for routine occupational exposures [13], it is the authors’ opinion that — for short exposure times (e.g. 30 minutes) — the ALE value for $\text{UO}_2\text{F}_2$ could be 150 mg-UF₆-m⁻³-min and the ALE value for HF could be 300 mg-HF-m⁻³-min, assuming a breathing rate of about 1.2 m³-h⁻¹ for both cases.

Although it is recognized that the health effects will vary with the age and individual susceptibility of the exposed persons, these values seem to be adequate for making rough safety decisions, such as those related to the safety level required for the packages, and it is in this sense that they are used in this paper.
3. THE PRESENT PRACTICE IN THE TRANSPORT OF UF₆

The present practice in the transport of UF₆ seems to be established on a historical and pragmatic basis. As stated in Section 2, the chemical risk dominates in most transport accidents and, therefore, a level of safety equivalent to that used for the transport of substances of similar chemical risks, such as HF and HCl, was initially used as a reference. The ANSI N14.1 standard [8] is widely used with some adaptations to domestic conditions. In this standard, the level of safety of the primary vessel is mainly based on the conjunction of an artificially high internal design pressure (associated more with the load/unload process of the cylinders than with the transport itself) and of detailed material specifications. Although attempts were made by the nuclear industry to evaluate the performance of these vessels in cases of impact or fire (for instance Ref. [14]), it should be recognized that performance requirements directly related to potential transport accidents are not yet available. Therefore, the development of an international standard by the ISO will not occur until those requirements are developed [15].

As the radiological risk is not relevant, the IAEA Transport Regulations [9] establish low package requirements for the transport of depleted, natural or low enriched uranium. A Type B package is required for high enriched uranium when the A₂ value is exceeded. In addition, the IAEA Regulations require criticality controls when the enrichment is higher than 1%, but this requirement does not necessarily imply the use of packages with a high level of safety, such as Type B, because criticality controls can be obtained by other means, such as limiting the total mass per shipment. On the other hand, some recommendations in the ANSI standard seem to be far from the IAEA requirements and not related to the chemical risk problem. For instance, if on the basis of the reasons stated above, the following assumptions are made: (a) chemical risk dominates radiological risk, (b) criticality control of low enriched UF₆ is not difficult and in accidental cases can be based on requirements which do not imply the retention of the content of the package, and (c) chemical risk is proportional to the UF₆ mass content in a package or shipment; it cannot be technically explained why, as in the present practice, a Type B package should be used for the transport of about 2 kg of UF₆ (1.1% enrichment), with zero criticality risk, and packages of a lower level of safety can be used for the transport of some tons of UF₆ if enrichment is lower than 1%.

As a final observation, it is noted that the level of safety used to control the chemical risk of UF₆, although similar to that used for other dangerous chemical substances [16], is lower than the level of safety used to control the radiological risk of radioactive substances which, in the event of a transport accident, could produce similar or lower consequences in terms of health effects or number of deaths.
4. ESTIMATED UF₆ MASS LIMIT

Under the framework defined in Sections 1–3 of this paper, one of the problems to be addressed seems to be a determination of the maximum UF₆ mass which could be packaged without special transport package requirements. This is a concept equal to the one used for defining the transition from a Type A package to a Type B package on the basis of the radiological risk, but now the approach is based on the chemical risk. Two outdoor scenarios were considered: (a) rupture of the cylinder by mechanical forces with the presence of water as a liquid on the ground and as humidity in the air, and (b) rupture of the cylinder during a fire. The latter situation appears as non-conservative, owing to the high dispersion associated with a fire. The former situation was developed under the following assumptions: (i) the ALE levels defined in Section 2.2 will not be exceeded, (ii) there will be a total rupture of the cylinder, (iii) there will be enough water on the ground to react with the total content but without a significant reduction of the atmospheric release by dilution of the hydrolysis products of UF₆ (it is assumed that 10% of the total mass content will be released into the atmosphere), (iv) a person will be exposed at a relatively short distance from the release point (15–30 m) and (v) the exposure time will be about 30 minutes.

As stated above, for short distances from the release point, it is impossible to model the atmospheric dispersion or to estimate the composition of the release products with accuracy. Furthermore, for short dispersion distances and low level releases, it seems impossible to develop a general dispersion model, particularly considering the effects of buildings, trees or the ground shape on the phenomenon. Assuming no dispersion along the vertical axis (heavy cloud), no deposition on the ground (short distances) and a relatively low wind velocity (about 1 m·s⁻¹), a comparison of the results obtained from different dispersion models [17, 18] suggests that, for short distances from the release point (15–20 m) and at a low altitude from the ground level (2–3 m), a dilution factor of about 10⁻² s·m⁻³ could be adopted. This value corresponds to a steady-state condition and a further assumption of constant concentration during the exposure time is implicitly made.

The reaction of one unit of mass of uranium as UF₆ with the formation of the anhydrous components UO₂F₂ and HF implies the reaction of about 1.48 units of mass of UF₆ with 0.15 unit of mass of H₂O and the formation of 1.29 and 0.34 units of mass of UO₂F₂ and HF, respectively; with the ALE value for uranium (150 mg·U·m⁻³·min) and a dilution factor of 10⁻² s·m⁻³ (1.7 X 10⁻⁴ mg·U·m⁻³·min), the mass of uranium which can be released into the atmosphere without exceeding the ALE value is 0.88 kg (I). With the ALE value for HF (300 mg·HF·m⁻³·min) and the same dilution factor, the mass of HF which can be released without exceeding the respective ALE value is 1.8 kg and this corresponds to a uranium mass of 5.3 kg (II). From a comparison between (I) and (II), it is clear that the uranium toxicity dominates under the assumptions made in this paper, disregarding potential remedial actions on the exposed person (it should be noted that at long distan-
ces, where deposition of $\text{UO}_2\text{F}_2$ could be significant, the situation could be reversed). Taking the lower mass of uranium (0.88 kg) and assuming that only 10% of the total mass content of the package will be released into the atmosphere, the mass limit for uranium as $\text{UF}_6$ in a package which is not designed to withstand accidents is 8.8 kg of U or 13 kg of $\text{UF}_6$. Therefore, in the light of present information, a rounded value of 10 kg of $\text{UF}_6$ is suggested as the maximum allowed mass in an $\text{UF}_6$ package not designed to withstand accidental conditions.

5. GENERAL COMMENTS

It is recognized that the main factors used to derive the value of the mass limit, namely the ALE values, the assumed fraction of mass released and the dilution factor could perhaps be modified after a detailed revision or additional studies. However, it is the authors' opinion that a combination of these modifications leading to a significant change in the suggested mass limit is quite improbable.

6. CONCLUSIONS

The present practice in the transport of $\text{UF}_6$ [8] ensures a level of safety equal or higher than the level of safety recommended for the transport of similar dangerous goods [16], but some basic questions arise requiring the attention of the nuclear industry and competent authorities: (a) Is it reasonable to establish a lower level of safety for chemical risk than for radiological risk, particularly in the case of $\text{UF}_6$, a material closely related with the nuclear fuel cycle? (b) Is it reasonable to design transport packages on the basis of pressure vessel requirements and of material specifications, or should transport performance requirements be developed? Both questions will require an answer in the near future and, if a decision is made to apply a level of safety equivalent to the one provided by Type B packages, this level should only be applied when the content exceeds a given mass limit. Besides, if transport performance tests are developed, such as those concerning impact and fire, the acceptable leakage after tests should be expressed as a function of the ALE values for chemical risks.

This paper has attempted to describe the $\text{UF}_6$ transport problems and to present preliminary estimates, based on the chemical hazards of $\text{UF}_6$ and its hydrolysis products. These problems refer both to the mass limit and to ALE values for chemical risks, as a first step in improving the requirements for the safe transport of $\text{UF}_6$. Finally, it is noted that the best way to improve safety seems to be the use of an outer packaging aimed at providing an additional capacity for the primary vessel to withstand impact and fire performance tests [19].

Note: this paper was written in the framework of IAEA Research Agreement No. 4369/CF on the evaluation of $\text{UF}_6$ transport requirements.
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RADIOLOGICAL LIMITS FOR TYPE B PACKAGES WITHIN THE TERMS OF THE IAEA TRANSPORT REGULATIONS

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Abstract

RADIOLOGICAL LIMITS FOR TYPE B PACKAGES WITHIN THE TERMS OF THE IAEA TRANSPORT REGULATIONS.

The paper discusses the radiation levels and activity release limits applicable to Type B packages with particular reference to consignments of irradiated nuclear fuel. Within the IAEA Regulations for the Safe Transport of Radioactive Material, external irradiation and maximum permissible non-fixed surface contamination levels are defined in terms of conventional radiological protection criteria, whereas permitted activity releases are expressed in terms of the non-special form Type A package contents limit A2 values and are not directly comparable. However, by considering a range of possible exposure scenarios relevant to the transport of Type B packages and noting that an intake of $A_2 \times 10^{-6}$ corresponds to an effective dose equivalent of 50 mSv, it is possible to derive the dosimetric limits implied by the permitted activity releases contained in the Regulations. This facilitates comparisons with external radiation and surface contamination limits under both normal transport and accident conditions. In the latter case, the safety standards for Type B packages implied within the Regulations may be compared with those widely used in the establishment of emergency guidelines at power reactor sites and other fixed nuclear installations.

1. INTRODUCTION

Within the IAEA Transport Regulations [1] external radiation limits and maximum permissible levels of non-fixed surface contamination for Type B packages are defined in terms of conventional radiological protection criteria, viz. mSv·h⁻¹ and Bq·cm⁻² respectively. In contrast, permitted activity releases form Type B packages under conditions of normal transport or following tests to simulate a severe accident are not so directly defined. Instead, they are expressed in terms of the non-special form Type A package contents limit A2 values, which are not directly comparable with standard dosimetric criteria. However, noting that an intake of $A_2 \times 10^{-6}$ corresponds to an effective dose equivalent of 50 mSv, the permitted activity releases from Type B packages may be expressed as equivalent dose or dose rate limits.
Type A packages are intended for the transport of low activity consignments and the dosimetric criteria appropriate to them are defined in the assumptions of the Q system used in the derivation of their A₁ and A₂ contents limit values [2]. Type B packages are designed for the transport of quantities of radioactivity in excess of the Type A limits and a principle embodied in the Regulations is that intentional leakage should be avoided. However, as absolute containment cannot be guaranteed, and in order to define appropriate and practical test procedures, maximum allowable leakage rates for Type B packages are specified as follows:

(a) \( A₂ \times 10^{-6} \) per hour following tests to simulate the conditions of normal transport, and

(b) an accumulated loss of \( A₂ \) (or 100 TBq for \( ^{85}\text{Kr} \)) in a period of up to one week following tests to simulate the conditions of a severe accident.

The dosimetric limits implied by these activity release limits may be evaluated by reference to a range of possible exposure scenarios relevant to the transport of Type B packages [3]. Examples of the derivation of such dosimetric limits are presented in this paper and are compared with external radiation limits specified in the IAEA Transport Regulations, as well as with dosimetric criteria applied elsewhere in the nuclear industry.

2. NORMAL TRANSPORT

The external radiation dose rate limits for Type B packages specified in the IAEA Transport Regulations [1] are 0.1 mSv·h⁻¹ at 1 m from the surface of a package or at 2 m from a vehicle under exclusive use. In addition, the radiation level at any point on the exposed surface of the package or transport vehicle should not exceed 2 mSv·h⁻¹, while in the special case of road transport that at any normally occupied position in the vehicle is limited to 0.02 mSv·h⁻¹. Also, maximum permissible levels of non-fixed surface contamination are specified as 4 Bq·cm⁻² for \( \beta/\gamma \) and low toxicity \( \alpha \)-emitters, and 0.4 Bq·cm⁻² for all other \( \alpha \)-emitters, both being averaged over an area of 300 cm² of any part of the package surface.

The original derivation of the normal transport activity release limit of \( A₂ \times 10^{-6} \) per hour considered exposure of a worker in an enclosed vehicle of 50 m³ volume, with ten air changes per hour [4]. This and other scenarios appropriate to exposure of transport workers and the public during the normal transport of Type B packages are used below to express the above activity release as an equivalent dose rate limit.

2.1. Exposure of transport workers

At equilibrium, and assuming an average adult breathing rate of 1.2 m³·h⁻¹, the above scenario leads to an intake rate of activity of \( A₂ \times 2.4 \times 10^{-9} \) h⁻¹. Based
on the dosimetric equivalence of an intake of $A_2 \times 10^{-6}$ noted earlier, this represents an effective dose equivalent rate of 0.12 mSv·h$^{-1}$, which is comparable with the external radiation limit at 1 m from the surface of a Type B package of 0.1 mSv·h$^{-1}$ cited above. For a person working 2000 hours per year and spending 20% of this time in an enclosed vehicle the above intake results in an annual dose of 48 mSv. This is just within the ICRP recommended maximum annual dose for radiation workers of 50 mSv, although the scenario outlined above was judged to represent the most adverse expected condition [4].

An alternative exposure situation relevant to transport workers is exposure in a store room or cargo handling bay of 3000 m$^3$ volume, with 4 air changes per hour [3]. Assuming the adult breathing rate of 1.2 m$^3$·h$^{-1}$ and uniform mixing of the release activity, under steady state conditions the Type B package release limit of $A_2 \times 10^{-6}$ per hour corresponds to an intake rate of $A_2 \times 9.9 \times 10^{-11}$ h$^{-1}$. This represents an implied dose rate limit of 5 μSv·h$^{-1}$, which compares with an external radiation limit of order 1 μSv·h$^{-1}$ at distances in the range 10–20 m when extrapolated from the 0.1 mSv·h$^{-1}$ level specified at 1 m from the surface of a Type B package.

2.2. Exposure of the public

Members of the public are unlikely to be in close proximity to Type B packages in transit for extended periods of time, hence in this situation it is appropriate to consider exposure out-of-doors to an airborne release of radioactivity. The effective dose equivalent rate corresponding to the IAEA limit of $A_2 \times 10^{-6}$ per hour for a release occurring under average category C/D weather conditions evaluated on this basis is illustrated in Fig. 1. Also shown is the external dose rate as a function of distance corresponding to a level of 0.1 mSv·h$^{-1}$ at 2 m from the centre of the package. Since local entrainment and turbulence effects close to the source are likely to minimise the spatial variations in airborne activity concentrations in this region, the results shown in Fig. 1 indicate that the Type B package activity release and external radiation limits represent a limiting effective dose equivalent rate of $\sim 10^{-2}$ μSv·h$^{-1}$ at downwind distances in the range 50–200 m. This is less than 10% of typical natural background radiation dose rates.

3. ACCIDENT CONDITIONS

Under accident conditions the prime exposure scenario of interest is that of a release occurring out-of-doors. For Type B packages accidents of the severity simulated in the tests specified in the IAEA Regulations are unlikely to occur indoors, or if they did the resulting conditions would be such as to require immediate evacuation of all persons in the vicinity [5]. The IAEA accident activity release limit of $A_2$ in a period of up to one week may be expressed as an individual dose limit in a
manner similar to that employed in Section 2.2. Since it is unlikely that a significant accidental release will persist for the full period of one week, the release is assumed to occur over a period of 12 hours. This is selected as a conservative estimate of the time which might be required for emergency services to reach the scene of an accident and take effective remedial actions to limit the release of activity from the damaged package.

On the above basis the doses as a function of distance from the source for airborne releases occurring at or near ground level under average category C/D and stable category F weather conditions are reproduced in Fig. 2. As noted earlier, possible entrainment and turbulence effects, and also plume rise if a fire were involved, will tend to reduce the spatial variation in doses within the first few tens of metres of the source. Also shown in Fig. 2 is the external radiation dose, integrated over the one week period specified in the Regulations, corresponding to the accident dose rate limit of 10 mSv·h\(^{-1}\) at 1 m from the surface of the damaged package for a
source of 1 MeV photons. This includes the 100 times enhancement of external radiation dose rate permitted under accident conditions compared with that during normal transport [1]. At distances beyond a few tens of metres downwind the external radiation dose over a week is more than an order of magnitude less than the implied limits within the Regulations. In absolute terms these represent an effective dose equivalent limit via inhalation at distances in the range 50–200 m of the order 1–50 mSv, depending upon the prevailing meteorological conditions.

The above approach may also be utilized to evaluate the limiting levels of ground contamination implicit in the IAEA activity release limits for Type B
packages under accident conditions. Assuming an average dry deposition velocity of $3 \times 10^{-3} \text{ m.s}^{-1}$, a release of $A_2$ over a period of 12 hours under average category C/D conditions would result in a contamination level of $A_2 \times 7.5 \times 10^{-2} \text{ m}^{-2}$ at about 100 m downwind from the source. For the more radiotoxic radionuclides or for mixtures typical of irradiated thermal reactor fuels the $A_2$ values for fission product and actinide isotopes are $\sim 0.2 \text{ TBq}$ and $\sim 0.004 \text{ TBq}$ respectively. These $A_2$ values yield limiting ground contamination levels of 15 and 0.3 Bq.cm$^{-2}$ for $\beta/\gamma$ and $\alpha$-emitters respectively. Also, for 1 MeV photons the $\beta/\gamma$ level corresponds to an external radiation dose rate at 1 m above a uniformly contaminated plane surface of about 0.3 $\mu$Sv.h$^{-1}$, or approximately 2-3 times the natural background radiation dose rate. Under stable category F weather conditions the above levels would be increased by a factor of about five.

4. DISCUSSION AND SUMMARY

Under normal transport conditions the limiting dose rates implied by the activity release limits specified for Type B packages within the IAEA Transport Regulations have been shown to be comparable with the corresponding external radiation dose rate limits. At the distances likely to be of interest in routine transport, namely within a few metres for packages being handled indoors or $\sim 100$ m for packages in transit out-of-doors, the limiting dose rates are a few $\mu$Sv.h$^{-1}$ and $\sim 10^{-2}$ $\mu$Sv.h$^{-1}$ respectively. Bearing in mind likely residence times for workers or members of the public, radiation doses received by both groups are highly unlikely to exceed the maximum permitted annual limits recommended by the ICRP [6].

Under severe accident conditions the doses implied by the activity release limit of $A_2$ in a period of up to one week are at least an order of magnitude greater than the permitted external radiation dose over this period at distances in the range 50-200 m downwind from a damaged Type B package. Typically a release of $A_2$ over a few hours has been shown to result in an effective dose equivalent limit of $\sim 10$ mSv at these distances. The corresponding levels of ground contamination are up to about an order of magnitude higher than the permissible levels of non-fixed surface contamination for Type B packages specified in the Regulations for normal transport and, as noted above, for $\beta/\gamma$ emitters would result under average weather conditions in external radiation dose rates about 2-3 times that due to natural background.

Finally, it is of interest to compare the implied accident dose limit of 1-50 mSv effective dose equivalent corresponding to the Type B package activity release limit of $A_2$ over a period of up to one week derived in Section 3 with emergency dose limits applied at fixed nuclear installations. In the UK emergency action levels at CEGB nuclear power station sites are based upon a whole body dose of 100 mSv, or 300 mSv to the thyroid or other single organs [7], while the whole body dose range for evacuation countermeasures in the event of an accident recently recommended by
the ICRP and IAEA is 50–500 mSv [8, 9]. Thus these intervention levels are about an order of magnitude above the implied Type B package accident limit, a situation which may be justified by noting that transport accidents may occur in relatively densely populated urban areas, whereas power stations and other fixed nuclear installations tend to be sited in more remote rural areas. Indeed, the collective doses associated with a given release at typical power station sites and in urban areas within the UK are approximately in the ratio 1:10 (see Fig. 2 of Ref. [10]), thus the collective dose resulting from a reactor accident at the ICRP and IAEA intervention levels would be comparable with that for a severe transport accident occurring in an urban area with an activity release corresponding to the IAEA limit for Type B packages.

REFERENCES

PRELIMINARY RISK ASSESSMENT FOR THE TRANSPORT OF REACTOR DECOMMISSIONING WASTE IN THE UNITED KINGDOM

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Abstract

PRELIMINARY RISK ASSESSMENT FOR THE TRANSPORT OF REACTOR DECOMMISSIONING WASTE IN THE UK.

It has been decided to decommission the UKAEA Windscale Advanced Gas-cooled Reactor (WAGR) to a 'green field' site to provide information on procedures and costs. This paper describes a risk assessment for the transport to a suitable repository of 200 large concrete packages containing about 700 t of resulting waste material. The packages are cubic with external dimensions a little over 2 m. Prototypes have been drop tested from heights of 0.65 m and 5 m and they sustained insufficient damage to impair the containment and shielding significantly. An impact equivalent to a 15 m drop was calculated to be necessary to expose the package contents. An event tree approach was adopted to determine the frequency of severe mechanical and thermal loading during road and rail transport. Collisions with a second vehicle or fixed object, falls from bridges, serious fires and crane failure during unloading were considered. The frequency of rail accidents sufficiently severe to expose the package contents was calculated to be about $5 \times 10^{-7}$ per year. The frequency of these severe impacts in conjunction with a serious fire was found to be two orders of magnitude smaller. The package weight would limit the vehicle speed on UK roads and no accident which could expose the package contents was considered plausible. Preliminary calculations indicated that the consequences of a transport accident (increased gamma radiation and dispersion of flammable waste by fire) would be relatively minor. It was concluded that transport risks should not present obstacles to the decommissioning of the WAGR.

1. INTRODUCTION

The Windscale Advanced Gas-cooled Reactor (WAGR) was a graphite-moderated, carbon dioxide cooled, prototype reactor built by the UKAEA at Sellafield on the north-west coast of England. It had an electrical output of about 30 MW and, after 18 years of successful operation, was finally shut down in 1981. Prior to that, it had been decided to decommission the reactor itself (but not the ancillary buildings) to a 'green field' site to provide valuable information on procedures and costs.
The reactor core (graphite with steel supporting structure and restraints) is located inside a steel pressure vessel (protected by an internal thermal shield of steel plate) which, in turn, is located within a thick reinforced concrete biological shield. This, together with four large heat exchangers (also contained within reinforced concrete biological shields) are located within a steel containment.

Nearly 90% of the total mass to be demolished is inactive. About 1800 t has some activity, of which about 700 t will require special disposal, either at sea or by shallow land burial. In either case the waste will require packaging and transporting from the site. The materials to be disposed of consist mainly of mild and stainless steels, reinforced concrete and graphite of which, in terms of mass, mild steel forms the greatest part. The packaging operation is scheduled to commence in about three years' time.

2. WASTE PACKAGES

2.1 Description

The Disposal Box design which was assessed is almost cubic with sides a little over 2 m in length [1]. Five sides of the Box are constructed from reinforced concrete 230 mm thick, clad on the outside with 12 mm mild steel plate. The waste material is packed inside and concrete grout is poured in to fill all the remaining space. The top of the Box is then completed with reinforced concrete and the top steel plate is welded in position. Equipment to cut up remotely the waste material and to carry out the packaging behind shield walls has been designed.

The sides of Boxes containing the more active waste will be constructed from "Super-Shot" (high density) concrete and various combinations of waste materials and concretes give expected total Box weights ranging from 28 to 50 t. The contents of each Box will vary considerably as shown in Table I.

The Box is designed to meet the IAEA requirements for industrial packages carrying LSA III material [2]. The most onerous test it must satisfy is a free drop from 0.3 m onto an unyielding surface with no loss of radioactive contents and no more than a 20% increase in radiation levels. The structure's response to more severe impact loading is of interest to the transport risk analyst, and the full scale tests performed so far are described in the next section.
TABLE I. CONTENTS OF DISPOSAL BOXES

<table>
<thead>
<tr>
<th>Waste material</th>
<th>Type of shielding</th>
<th>Payload (t)</th>
<th>Activity (TBq)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Stainless steel</td>
<td>S</td>
<td>0.7</td>
<td>70-220</td>
</tr>
<tr>
<td>Stainless steel and mild steel</td>
<td>S</td>
<td>2.4-11</td>
<td>60-120</td>
</tr>
<tr>
<td>Stainless steel and mild steel</td>
<td>0</td>
<td>15</td>
<td>12</td>
</tr>
<tr>
<td>Mild steel</td>
<td>S</td>
<td>0.6-15</td>
<td>40-60</td>
</tr>
<tr>
<td>Mild steel</td>
<td>0</td>
<td>5-18</td>
<td>0.6-30</td>
</tr>
<tr>
<td>Graphite</td>
<td>S</td>
<td>5</td>
<td>3</td>
</tr>
<tr>
<td>Graphite</td>
<td>0</td>
<td>5</td>
<td>1</td>
</tr>
<tr>
<td>Reinforced concrete</td>
<td>0</td>
<td>8</td>
<td>0.7-1</td>
</tr>
</tbody>
</table>

2.2 Testing

Full scale drop testing of prototype Boxes was carried out at the UKAEA site at Winfrith, Dorset. The outdoor facility there includes a 150 t crane which is capable of raising loads of over 90 t to a hook height of 30 m [3]. The target consists of a 700 t concrete block faced with 150 mm steel plate.

Four prototype Boxes were constructed weighing 40 t and containing steel plates to simulate sections of the WAGR thermal shield [4]. Drops in four orientations were conducted, i.e. onto a base, edge, corner and lifting lugs. The IAEA regulatory drop height of 0.3 m was increased to 0.65 m for the first series of tests to compensate for the mass of the prototype being less than the proposed maximum and for possible variations in material properties. This greater height incorporates a large measure of conservatism. Information was recorded using high speed cameras and accelerometers, and in addition a gamma
source was inserted into specially prepared channels before and after the drops to check for any changes in radiation shielding capability.

The results of the tests were very much as might be expected. The flat base drop gave the highest deceleration (150g) and the greatest deformation occurred in the corner drop. Damage in all cases was slight with no splitting of the steel envelope. Cores taken from impact regions later showed localised concrete cracking but no reduction in shielding properties was detected using the gamma source.

In view of the minor nature of the damage sustained in these tests it was decided to conduct four further drops from a height of 5 m (the limit imposed by the Winfrith drop test facility at that time), to provide further data about the impact behaviour of the structure.

A deceleration of 600g was recorded for the base drop with about 0.5 m vertical plate weld splitting at each corner. A small amount of crumbled concrete spilled out from these gaps, but insufficient to give a detectable reduction in shielding using the gamma source test. For the edge drop there was again a degree of weld splitting at two corners and a small amount of concrete wall was lost. The greatest crushing deformation occurred in the corner drop but the steel plate containment remained intact. The lifting lugs appeared to provide effective shock absorption in the remaining drop as little damage other than lug distortion was observed.

2.3 Failure

The prototype Disposal Boxes withstood severe impacts, much in excess of the regulatory requirements, with no significant damage, i.e. no loss of contents and no detectable loss of shielding capability.

In order to determine transport risks it is necessary to assess the ability of the package to withstand extremely severe (although most improbable) loading. It was decided to evaluate the impact severity necessary to remove, in effect, the outer 230 mm wall shielding by concrete crushing, thus exposing the core of waste and concrete grout. This was assessed using SRD computer modelling techniques [5] and the drop test data, to require an impact equivalent to a 15 m drop onto an unyielding surface.

3. TRANSPORT ROUTES

The Boxes were originally designed with sea disposal in mind. In view of the current moratorium on this option, land
burial was also considered. Three potential transport routes were examined:

(a) by rail to the nearest convenient sea port (60 km)
(b) by rail to a potential land repository site (450 km)
(c) by road to the same land repository site (510 km).

The most direct major transport routes were chosen in accordance with current UK practice, with no attempt to, for example, avoid major centres of population,

It was assumed that 67 Boxes would be transported per year.

4. FREQUENCIES OF ACCIDENTS

4.1 Approach

Rail and road accidents considered were:

(a) collision with a second rail or road vehicle
(b) collision with a fixed object near the line or carriageway
(c) fall from a high bridge
(d) accidents involving a second vehicle carrying flammable cargo and resulting in fire
(e) crane failure during the unloading operation.

An event tree approach was adopted to assess the frequencies of significant Box damage.

4.2 Rail transport

It was assumed that the Box-carrying rail vehicles will travel at up to 100 km.h⁻¹ and that they may encounter other rail traffic travelling at speeds up to 200 km.h⁻¹. Historical rail data were used to assess derailment and collision frequencies (1.5 x 10⁻⁷ per vehicle.km for derailments) [6]. These were multiplied by probabilities determined for the important impact parameters such as speed, angle, Box orientation, impact energy absorption by vehicles, the number and nature of line-side hazards, and Box behaviour after impact. Where historical, route or vehicle data were inadequate, assumptions were made which were clearly pessimistic.

The Boxes will be transported either in dedicated trains or in mixed freight trains but without vehicles carrying flammable cargo. In addition, dangerous cargoes are segregated in
TABLE II. ASSESSED ACCIDENT FREQUENCIES

<table>
<thead>
<tr>
<th>Route</th>
<th>Frequency of minor accidents (no radiological consequences) (a⁻¹)</th>
<th>Frequency of superficial Box damage accidents (no radiological consequences) (a⁻¹)</th>
<th>Frequency of Box contents exposure (a⁻¹)</th>
<th>Frequency of impact to expose Box contents and subsequent serious fire (a⁻¹)</th>
</tr>
</thead>
<tbody>
<tr>
<td>By rail to notional land repository site</td>
<td>2 x 10⁻³</td>
<td>3 x 10⁻⁶</td>
<td>5 x 10⁻⁷</td>
<td>4 x 10⁻⁹</td>
</tr>
<tr>
<td>By road to notional land repository site</td>
<td>4 x 10⁻²</td>
<td>7 x 10⁻⁶</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>By rail to possible sea port</td>
<td>5 x 10⁻⁴</td>
<td>3 x 10⁻⁵</td>
<td>2 x 10⁻⁸</td>
<td>5 x 10⁻¹⁰</td>
</tr>
</tbody>
</table>
marshalling yards. Thus the chance of a Box being involved in a serious fire will be very small.

The most likely serious accident was found to be a derailment and subsequent high speed collision with a second train (about $5 \times 10^{-7}$ a$^{-1}$). The only other event predicted to cause significant impact damage to a Box is a collision with a tunnel abutment (but at a frequency about two orders of magnitude lower). Damage to Boxes in collisions with bridge piers, after falls from bridges along the routes, and after falls during unloading operations was assessed to be less than the failure criterion given in Section 2.3, i.e. the effective exposure of the grouted waste. The results are summarised in Table II.

In addition, the frequencies of superficial damage accidents (equivalent to a 5 m drop onto an unyielding surface) and minor accidents involving no radiological consequences (such as derailment with no direct Box impact) were assessed (Table II).

4.3 Road transport

Current UK legislation would limit the road speed to 20 km.h$^{-1}$ (because of the package weight). Higher speeds (on motorways for example) could be permitted at the discretion of the police and the assessed maximum speed was increased by a pessimistic factor of three to allow for this and possible future speed limit increases. Other Heavy Goods Vehicles are limited to 100 km.h$^{-1}$ in the UK but again allowance for exceeding this speed was made. Injury road accident statistics [?] with a pessimistic factor for damage only events gave an overall accident rate (0.9-2.4 per $10^6$ vehicle.km, depending on road class) although the majority of these incidents would be completely inconsequential in the context of significant Box damage. Unpublished Department of Transport figures, route data, and assumptions where data were not available, were used to establish probabilities for bridge pier collisions, vehicle collision orientations and speed distributions.

Only fires involving tankers carrying flammable fluids were considered to have the potential to threaten the Disposal Boxes.

It was concluded that impacts severe enough to cause the Box wall failure described in Section 2.3 could not occur, principally because of the relatively low speed of the vehicle. For this degree of Box damage a direct Heavy Goods Vehicle/Box impact at over 120 km.h$^{-1}$ was estimated to be necessary. Whilst relative velocities this great could occur occasionally, it was considered that actual impact speeds would be less.
Frequencies for superficial damage accidents and minor accidents (such as collisions with a car) are also shown in Table II. The most likely accident giving rise to superficial damage was assessed to be a fall from an elevated section of roadway.

5. CONSEQUENCES

Following the analysis of accident frequencies, summarised in Table II, the consequences were assessed in three damage categories, namely, minor, superficial and significant.

Minor accidents would have no associated radiological consequences, although people, car occupants for example, could be killed or injured.

Superficial damage accidents also would have no radiological consequences. There would be no release of radioactivity and no significant increase in external radiation levels, but the Box might suffer noticeable damage.

Significant damage is specified in Section 2.3 as localised exposure of the grouted waste. This would lead to increased radiation levels from that portion of the Box but the inner grout would still provide shielding, and the components of the waste would furnish a degree of self shielding. The radiation level would vary a good deal depending on the nature of the waste being transported, but would necessarily be less than the regulatory requirement of 10 mSv.h\(^{-1}\) at a distance of 3 m [2]. Such accidents would cause some disruption but it should not prove difficult to control exposure and make good any damage.

The most active waste material comprises activated steel components which are clearly not dispersible. Mechanical damage could release small quantities of powdered graphite, but this would not present a big problem as the specific activity is not high (at most about 3 TBq in a 5 t payload). The only feasible dispersion mechanism is the exposure of graphite by impact followed by a serious fire. However, the graphite would not ignite easily and the payload would only be partially exposed to the fire, so it is likely that only a small fraction of the total activity would be dispersed. Preliminary calculations considering the principal active components (\(^{14}\)C and \(^{60}\)Co) indicate that individual and collective doses would not be great even if complete dispersion occurred.

6. FURTHER WORK

Changes to the Disposal Box design assessed here, which are currently being considered, include the removal of the outer
steel plates on cost and superfluity grounds, alternative grouting mixtures to help ensure no air gaps remain, and modified lifting lugs to improve repository stacking arrangements.

SRD propose to develop their concrete package impact assessment techniques through programmes of scale model and full size drop testing and theoretical analysis.

7. CONCLUSIONS

Prototype Disposal Boxes for the transport and storage of radioactive decommissioning waste material have been designed, built and tested. A preliminary risk assessment for the proposed transport of these Boxes suggests that the frequencies of radiologically significant accidents would be extremely low \((<10^{-6}\text{ a}^{-1})\) and the radiological consequences of such accidents would not be severe. Transport risks should not, therefore, present obstacles to the decommissioning of the Windscale AGR.

REFERENCES


PROJECTED ENVIRONMENTAL IMPACTS OF RADIOACTIVE MATERIAL TRANSPORTATION TO THE FIRST US REPOSITORY SITE*

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Abstract

PROJECTED ENVIRONMENTAL IMPACTS OF RADIOACTIVE MATERIAL TRANSPORTATION TO THE FIRST US REPOSITORY SITE.

The relative national environmental impacts of transporting spent fuel and other nuclear wastes to each of nine candidate repository sites in the United States of America were analysed. Two scenarios were examined for each repository: (1) shipment of 5 year old spent fuel and defence high level waste (DHLW) directly from their points of origin to a repository (Reference Case); (2) shipment of 5 year old spent fuel to a Monitored Retrievable Storage (MRS) facility and shipment (by dedicated rail) of 10 year old consolidated spent fuel from the MRS to a repository. Transport by either all truck or all rail from the points of origin were analysed as bounding cases. The computational system used to analyse these impacts included the WASTES II logistics code and the RADTRAN III risks analysis code. The radiological risks for the Reference Case increased as the total shipment miles to a repository increased for truck; the risks also increased with mileage for rail but at a lower rate. For the MRS scenario the differences between repository sites were less pronounced for both modal options because of the reduction in total shipment miles possible with the large dedicated rail casks. All the risks reported are small in comparison to 'natural background'.

Introduction

Spent fuel from commercial nuclear power reactors in the United States will be permanently disposed of in mined geologic repositories. The Nuclear Waste Policy Act (NWPA) of 1982 outlined the implementation of this approach by the US Department of Energy under Contract No. DE-AC04-76DP00789.

* Work supported by the US Department of Energy under Contract No. DE-AC04-76DP00789.
Energy (DOE). The DOE has begun selection of a site for a first repository from among nine sites in three geologic media - salt, tuff, and basalt. A Monitored Retrievable Storage (MRS) facility may be included in the system; spent fuel would be stored for up to 5 years at an MRS, which would also consolidate the fuel before shipping it to the repository. This paper discusses the relative national environmental impacts of transporting nuclear wastes to each of the nine candidate repository sites in the United States (Reference 1). Several of the potential sites are closely clustered and, for the purpose of distance and routing calculations, are treated as a single location. These are: Cypress Creek Dome and Richton Dome in Mississippi (Gulf Interior Region), Deaf Smith County and Swisher County sites in Texas (Permian Basin), and Davis Canyon and Lavender Canyon sites in Utah (Paradox Basin). The remaining sites are: Vacherie Dome, Louisiana; Yucca Mountain, Nevada; and Hanford Reservation, Washington.

For compatibility with both the repository system authorized by the NWPA and with the MRS option, two separate scenarios were analyzed. In brief, they are (1) shipment of spent fuel and high level waste (HLW) directly from waste generators to a repository (Reference Case) and (2) shipment of spent fuel to a Monitored Retrievable Storage (MRS) facility and then to a repository.

Problem Definition

In order to perform cost and risk analyses of the impacts of transportation for a future US nuclear waste management system, a large array of data is required. These data include information on the transport links and surrounding populations, routing information (e.g. distances traveled), the packaging (e.g. cask capacity), transport mode characteristics (e.g. train speeds), radionuclide inventory, and pertinent operational characteristics of the system such as accident rates. These data are used as inputs for two major computational tools, the WASTES II logistics code and the RADTRAN III risk analysis code. This complex computational system is described more fully in Reference 2.

For the Reference Case, the primary waste stream is spent nuclear fuel (SF) from reactors. Secondary waste streams considered for this case include defense high level wastes (DHLW) from the Savannah River Plant in South Carolina, the Hanford Reservation in Washington, and the Idaho National Engineering Laboratory in Idaho, and commercially-generated high level waste from West Valley, New York (WVHLW). Acceptance of DHLW in a commercial repository was endorsed by the President of the United States in 1985 (Reference 3). In this case, all reactors will ship 5-year-old or older unconsolidated spent fuel directly to a candidate repository site. High level commercial and defense wastes will also be shipped directly to the repository. Two primary modal options are examined for the Reference Case: all truck and all rail from reactors and HLW generators. The resultant costs and risks will bound the transportation impacts. No
attempt has been made to forecast the actual fractions of truck and rail transport that might be used. The shipping system ultimately used for transportation of spent fuel and HLW will be a combination of modes determined by considerations such as the capabilities of handling facilities at the origins, freight rates, and operational constraints of the system.

MRS input data and scenarios are compatible with those being used by the MRS program. Final MRS documentation to be presented to Congress will, however, include additional alternatives not discussed here.

For the MRS cases, as in the reference case, reactors will ship 5-year-old or older unconsolidated spent fuel, but to an MRS rather than a repository. All spent fuel leaving the MRS will be consolidated and at least 10 years old. Additional secondary wastes would be generated at an MRS by the proposed spent fuel consolidation and possible overpacking operations. These MRS-related secondary wastes would consist of assembly hardware, high activity waste (HAW), and transuranic waste (TRU), which would also be shipped to the repository. Transport from an MRS would be by one of two possible shipping options: (1) 100-ton (100T) dedicated rail shipments of overpacked consolidated spent fuel and waste byproducts generated in the consolidation process and (2) 150-ton (150T) dedicated rail shipments of nonoverpacked consolidated spent fuel and byproducts. As in the Reference Case, high level commercial and defense wastes are shipped directly to the repository. For shipments from the MRS bounding values for total cask weight and payload characteristics were used either to minimize or to maximize cask capacity and, hence, to put upper and lower limits on the number of shipments from the MRS to the repository.

Results

Results of the analysis performed for the Reference Case are summarized in Tables 1-3, below. The differences in cost and impacts among the various repository sites are related primarily to the total shipping distances (Table 1). As can be noted from the table, spent fuel shipments account for the largest fraction of the total shipping distance for both modal options, comprising from 70-80 percent of the total truck travel and from 62-75 percent of the total rail travel. In either case the largest percentages are associated with travel to the westernmost site (Hanford, Washington). The fraction of total travel attributable to spent fuel transport increases as the potential repository site is shifted to the west because most of the spent fuel inventory projected to require shipment to the first repository is from reactors in the eastern United States. The relative contribution of high level wastes requiring shipment to the repository is between 19 and 29 percent for truck and 25 and 37 percent for rail. Although the projected mileage increases as the more western repository options are analyzed, the relative influence of high
### TABLE 1. TOTAL SHIPMENT-MILES (millions of miles*)

**REFERENCE CASE - Direct to Repository**

<table>
<thead>
<tr>
<th>Mode/Waste Type</th>
<th>GIR</th>
<th>Vacherie</th>
<th>Permian</th>
<th>Paradox</th>
<th>Yucca Mt</th>
<th>Hanford</th>
</tr>
</thead>
<tbody>
<tr>
<td>100% Truck SF</td>
<td>67.4</td>
<td>71.7</td>
<td>94.4</td>
<td>115.1</td>
<td>141.8</td>
<td>149.7</td>
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<tr>
<td>DHLW</td>
<td>28.0</td>
<td>28.0</td>
<td>26.0</td>
<td>28.0</td>
<td>33.0</td>
<td>35.0</td>
</tr>
<tr>
<td>WWHLW</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0</td>
<td>2.0</td>
<td>2.0</td>
<td>2.0</td>
</tr>
<tr>
<td>TOTAL</td>
<td>96.4</td>
<td>100.7</td>
<td>121.4</td>
<td>145.1</td>
<td>176.8</td>
<td>186.7</td>
</tr>
<tr>
<td>100% Rail SF</td>
<td>11.0</td>
<td>11.7</td>
<td>15.4</td>
<td>18.8</td>
<td>23.2</td>
<td>24.6</td>
</tr>
<tr>
<td>DHLW</td>
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<td>6.5</td>
<td>6.1</td>
<td>6.5</td>
<td>7.6</td>
<td>8.4</td>
</tr>
<tr>
<td>WWHLW</td>
<td>0.2</td>
<td>0.2</td>
<td>0.2</td>
<td>0.2</td>
<td>0.3</td>
<td>0.3</td>
</tr>
<tr>
<td>TOTAL</td>
<td>17.7</td>
<td>21.2</td>
<td>21.7</td>
<td>25.5</td>
<td>31.1</td>
<td>33.3</td>
</tr>
</tbody>
</table>

* 1 mile = 1.609 km

### TABLE 2. TOTAL TRANSPORTATION COSTS ($M)

**REFERENCE CASE - Direct to Repository**

<table>
<thead>
<tr>
<th>Mode/Waste Type</th>
<th>GIR</th>
<th>Vacherie</th>
<th>Permian</th>
<th>Paradox</th>
<th>Yucca Mt</th>
<th>Hanford</th>
</tr>
</thead>
<tbody>
<tr>
<td>100% Truck CAPITAL</td>
<td>227.2</td>
<td>234.2</td>
<td>261.2</td>
<td>290.1</td>
<td>325.1</td>
<td>337.2</td>
</tr>
<tr>
<td>OPERATING</td>
<td>708.9</td>
<td>730.0</td>
<td>866.0</td>
<td>1015.1</td>
<td>1213.6</td>
<td>1277.8</td>
</tr>
<tr>
<td>TOTAL</td>
<td>936.1</td>
<td>964.2</td>
<td>1127.2</td>
<td>1305.2</td>
<td>1538.7</td>
<td>1615.0</td>
</tr>
<tr>
<td>100% Rail CAPITAL</td>
<td>267.3</td>
<td>277.7</td>
<td>300.9</td>
<td>322.5</td>
<td>354.2</td>
<td>362.8</td>
</tr>
<tr>
<td>OPERATING</td>
<td>714.7</td>
<td>734.9</td>
<td>821.6</td>
<td>885.3</td>
<td>991.0</td>
<td>1013.8</td>
</tr>
<tr>
<td>TOTAL</td>
<td>982.0</td>
<td>1012.6</td>
<td>1122.5</td>
<td>1207.8</td>
<td>1345.2</td>
<td>1376.6</td>
</tr>
</tbody>
</table>

level wastes on the results decreases. Data in Table 1 indicate that miles traveled to the westernmost sites (Yucca Mt, Nevada, and Hanford, Washington) are almost double the total shipment miles required for transport to the easternmost sites in the Gulf Interior Region.

Transportation costs for the repository location options are summarized in Table 2. These costs increase with the total number of shipment-miles; however, because of the tariff structures of the transport modes, they do not increase in a linear manner. Truck costs increase by approximately 75 percent between the most eastern site in the Gulf Interior Region and the Hanford site in the West. Consistent with the rail rate structure, total rail costs for these sites vary by only about 40 percent. Truck costs are lower than rail for the easternmost sites and higher than rail for the western sites. The contribution of spent fuel cost to the total is consistent with the fraction of shipment mileage at-
### TABLE 3. SUMMARY OF THE TOTAL RISKS OF TRANSPORTATION REFERENCE CASE - Direct to Repository

<table>
<thead>
<tr>
<th>MODE</th>
<th>Repository</th>
<th>GIR</th>
<th>VACHERIE</th>
<th>PERMIAN</th>
<th>PARADOX</th>
<th>YUCCA MT</th>
<th>HANFORD</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>100% Truck from origin</strong></td>
<td></td>
<td></td>
<td></td>
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<td></td>
</tr>
<tr>
<td>SF</td>
<td>Radiological</td>
<td>4.6</td>
<td>5.0</td>
<td>6.2</td>
<td>7.7</td>
<td>9.2</td>
<td>10.0</td>
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<tr>
<td></td>
<td>Nonradiological</td>
<td>13</td>
<td>14</td>
<td>18</td>
<td>24</td>
<td>29</td>
<td>31</td>
</tr>
<tr>
<td>HLW</td>
<td>Radiological</td>
<td>1.8</td>
<td>1.7</td>
<td>1.7</td>
<td>1.8</td>
<td>2.1</td>
<td>2.1</td>
</tr>
<tr>
<td></td>
<td>Nonradiological</td>
<td>6.2</td>
<td>5.8</td>
<td>6.2</td>
<td>6.1</td>
<td>7.4</td>
<td>7.4</td>
</tr>
<tr>
<td><strong>100% Rail from origin</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>SF</td>
<td>Radiological</td>
<td>.16</td>
<td>.17</td>
<td>.18</td>
<td>.21</td>
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<td>.25</td>
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<td>Nonradiological</td>
<td>.81</td>
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<td>1.6</td>
<td>1.6</td>
</tr>
<tr>
<td>HLW</td>
<td>Radiological</td>
<td>.062</td>
<td>.067</td>
<td>.063</td>
<td>.066</td>
<td>.079</td>
<td>.074</td>
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<tr>
<td></td>
<td>Nonradiological</td>
<td>.63</td>
<td>.69</td>
<td>.64</td>
<td>.66</td>
<td>.84</td>
<td>.79</td>
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<td><strong>TOTALS</strong></td>
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<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Truck from origin:</td>
<td>Radiological</td>
<td>6.4</td>
<td>6.7</td>
<td>7.9</td>
<td>9.5</td>
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<td>12</td>
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<td>Nonradiological</td>
<td>19</td>
<td>20</td>
<td>24</td>
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<td>36</td>
<td>38</td>
</tr>
<tr>
<td>Rail from origin:</td>
<td>Radiological</td>
<td>.22</td>
<td>.24</td>
<td>.24</td>
<td>.28</td>
<td>.32</td>
<td>.32</td>
</tr>
<tr>
<td></td>
<td>Nonradiological</td>
<td>1.4</td>
<td>1.5</td>
<td>1.6</td>
<td>2.0</td>
<td>2.4</td>
<td>2.4</td>
</tr>
</tbody>
</table>

1 Radiological health effects include latent cancer fatalities and genetic effects in all generations.
2 Nonradiological fatalities

...tributable to spent fuel transport for truck; it is somewhat less than the fraction of total mileage for rail.

Because the points of origin of most shipments (i.e. reactors) are primarily in the eastern United States, the average fractions of total travel in rural, suburban, and urban population-density zones are about the same for spent fuel transport to each candidate repository site. Consequently, total travel distance becomes the major discriminator of risk between sites for a given shipment scenario. Table 3 shows that the Gulf Interior Region (GIR) and Vacherie, Louisiana, sites, which are closest to the origin points, have the lowest overall risks associated with them; while those sites farthest from the majority of the country's reactors have the highest associated risks. However, the total risks associated with the closest repository sites only differ from those for the most distant site by about a factor of 1.9 to 2.1 for truck and by about a factor of 1.5 to 1.8 for rail. These factors generally parallel increases in shipment-miles except for the radiological risk of rail transport, which increases at a significantly lower rate than the mileage. A component of radiological risk for rail transport, but not for truck transport, is associated with required endpoint classification and
### TABLE 4. TOTAL SHIPMENT-MILES (millions of miles)
MRS CASE - MRS at Oak Ridge

<table>
<thead>
<tr>
<th>Mode/Waste Type</th>
<th>GIR</th>
<th>Vacherie</th>
<th>Permian</th>
<th>Paradox</th>
<th>Yucca Mt</th>
<th>Hanford</th>
</tr>
</thead>
<tbody>
<tr>
<td>Truck from Origin</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>SF to MRS</td>
<td>48.8</td>
<td>48.8</td>
<td>48.8</td>
<td>48.8</td>
<td>48.8</td>
<td>48.8</td>
</tr>
<tr>
<td>DHLW to Repos.</td>
<td>28.0</td>
<td>28.0</td>
<td>26.0</td>
<td>28.0</td>
<td>33.0</td>
<td>35.0</td>
</tr>
<tr>
<td>WVHLW to Repos.</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0</td>
<td>2.0</td>
<td>2.0</td>
<td>2.0</td>
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<td>Rail from Origin</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>SF to MRS</td>
<td>8.0</td>
<td>8.0</td>
<td>8.0</td>
<td>8.0</td>
<td>8.0</td>
<td>8.0</td>
</tr>
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<td>DHLW to Repos.</td>
<td>6.5</td>
<td>6.5</td>
<td>6.1</td>
<td>6.5</td>
<td>7.6</td>
<td>8.4</td>
</tr>
<tr>
<td>WVHLW to Repos.</td>
<td>0.2</td>
<td>0.2</td>
<td>0.2</td>
<td>0.2</td>
<td>0.3</td>
<td>0.3</td>
</tr>
<tr>
<td>Rail from MRS to Repository (150T, nonoverpacked SF)</td>
<td>0.2</td>
<td>0.3</td>
<td>0.6</td>
<td>0.8</td>
<td>1.5</td>
<td>1.0</td>
</tr>
</tbody>
</table>

**TOTALS**

<table>
<thead>
<tr>
<th>Mode/Waste Type</th>
<th>GIR</th>
<th>Vacherie</th>
<th>Permian</th>
<th>Paradox</th>
<th>Yucca Mt</th>
<th>Hanford</th>
</tr>
</thead>
<tbody>
<tr>
<td>Truck from Origin</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>150T from MRS</td>
<td>78.0</td>
<td>78.1</td>
<td>76.4</td>
<td>78.6</td>
<td>85.3</td>
<td>86.8</td>
</tr>
<tr>
<td>Rail from Origin</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>150T from MRS</td>
<td>14.9</td>
<td>15.0</td>
<td>14.9</td>
<td>15.5</td>
<td>17.4</td>
<td>17.7</td>
</tr>
</tbody>
</table>

### TABLE 5. TOTAL TRANSPORTATION COSTS ($M)^1
MRS CASE - MRS at Oak Ridge

<table>
<thead>
<tr>
<th>Mode/Waste Type</th>
<th>GIR</th>
<th>Vacherie</th>
<th>Permian</th>
<th>Paradox</th>
<th>Yucca Mt</th>
<th>Hanford</th>
</tr>
</thead>
<tbody>
<tr>
<td>Truck from Reactors, HLW Sites</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>CAPITAL</td>
<td>201.0</td>
<td>202.1</td>
<td>204.3</td>
<td>209.8</td>
<td>214.2</td>
<td>217.5</td>
</tr>
<tr>
<td>OPERATING</td>
<td>613.7</td>
<td>608.1</td>
<td>601.1</td>
<td>615.8</td>
<td>639.0</td>
<td>652.9</td>
</tr>
<tr>
<td>Rail from Reactors, HLW Sites</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>CAPITAL</td>
<td>232.3</td>
<td>237.7</td>
<td>235.9</td>
<td>239.5</td>
<td>246.7</td>
<td>250.3</td>
</tr>
<tr>
<td>OPERATING</td>
<td>643.7</td>
<td>646.1</td>
<td>647.5</td>
<td>644.2</td>
<td>667.9</td>
<td>664.4</td>
</tr>
<tr>
<td>Rail from MRS to Repository (150T, nonoverpacked)</td>
<td></td>
<td></td>
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<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>CAPITAL</td>
<td>78.6</td>
<td>78.6</td>
<td>78.6</td>
<td>78.6</td>
<td>100.6</td>
<td>84.1</td>
</tr>
<tr>
<td>OPERATING</td>
<td>172.7</td>
<td>199.0</td>
<td>265.3</td>
<td>306.8</td>
<td>468.7</td>
<td>346.8</td>
</tr>
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</table>

**TOTALS**

<table>
<thead>
<tr>
<th>Mode/Waste Type</th>
<th>GIR</th>
<th>Vacherie</th>
<th>Permian</th>
<th>Paradox</th>
<th>Yucca Mt</th>
<th>Hanford</th>
</tr>
</thead>
<tbody>
<tr>
<td>Truck from Origin</td>
<td></td>
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</tr>
<tr>
<td>150T from MRS</td>
<td>1066.0</td>
<td>1087.8</td>
<td>1149.3</td>
<td>1211.0</td>
<td>1422.5</td>
<td>1301.3</td>
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<td>Rail from Origin</td>
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<tr>
<td>150T from MRS</td>
<td>1127.3</td>
<td>1161.4</td>
<td>1227.3</td>
<td>1269.1</td>
<td>1483.9</td>
<td>1345.6</td>
</tr>
</tbody>
</table>

^1The totals presented in this table are for the case in which all spent fuel and HLW wastes are shipped by the mode indicated; dedicated rail shipments from the MRS to the repository are added.
### TABLE 6. SUMMARY OF THE RISKS OF TRANSPORTATION OF SPENT FUEL AND HIGH LEVEL WASTES:
MRS CASE - (All SF to MRS, 150T cask)

<table>
<thead>
<tr>
<th>Mode</th>
<th>GIR</th>
<th>VACHERIE</th>
<th>PERMIAN</th>
<th>PARADOX</th>
<th>YUGCA MT</th>
<th>HANFORD</th>
</tr>
</thead>
<tbody>
<tr>
<td>100% Truck from Origin</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>SF</td>
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<td>3.6</td>
</tr>
<tr>
<td>Radiological</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Nonradiological</td>
<td>9.1</td>
<td>9.1</td>
<td>9.1</td>
<td>9.1</td>
<td>9.1</td>
<td>9.1</td>
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<tr>
<td>HLW</td>
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<tr>
<td>Radiological</td>
<td>1.8</td>
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<td>1.7</td>
<td>1.8</td>
<td>2.1</td>
<td>2.1</td>
</tr>
<tr>
<td>Nonradiological</td>
<td>6.2</td>
<td>5.8</td>
<td>6.2</td>
<td>6.1</td>
<td>7.4</td>
<td>7.4</td>
</tr>
<tr>
<td>100% Rail from Origin</td>
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<tr>
<td>Radiological</td>
<td>.92</td>
<td>.92</td>
<td>.92</td>
<td>.92</td>
<td>.92</td>
<td>.92</td>
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<tr>
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<td>.062</td>
<td>.067</td>
<td>.063</td>
<td>.066</td>
<td>.079</td>
<td>.074</td>
</tr>
<tr>
<td>HLW</td>
<td>.63</td>
<td>.69</td>
<td>.64</td>
<td>.66</td>
<td>.84</td>
<td>.79</td>
</tr>
<tr>
<td>Radiological</td>
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<td></td>
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<td></td>
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</tr>
<tr>
<td>Nonradiological</td>
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<td>2.6</td>
<td>3.8</td>
<td>5.3</td>
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<td>150T Rail from MRS</td>
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<tr>
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<td>TOTALS</td>
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<td></td>
</tr>
<tr>
<td>Truck from Origin</td>
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<td></td>
<td></td>
<td></td>
<td></td>
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<td>5.3</td>
<td>5.3</td>
<td>5.4</td>
<td>5.8</td>
<td>5.7</td>
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<td>17</td>
<td>18</td>
<td>19</td>
<td>20</td>
<td>26</td>
<td>22</td>
</tr>
<tr>
<td>Rail from Origin</td>
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<td></td>
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<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Radiological</td>
<td>.22</td>
<td>.25</td>
<td>.24</td>
<td>.25</td>
<td>.27</td>
<td>.26</td>
</tr>
<tr>
<td>Nonradiological</td>
<td>2.9</td>
<td>4.2</td>
<td>5.3</td>
<td>6.9</td>
<td>12</td>
<td>7.7</td>
</tr>
</tbody>
</table>

1Radiological health effects include latent cancer fatalities and genetic effects in all generations.
2Nonradiological fatalities

Inspection stops. Because this component is distance-independent (i.e. the same for all trips, short or long), the influence of distance traveled on total radiological risk for rail is less pronounced than for truck.

Insertion of an MRS into the system tends to reduce the variation in cost and risk between the potential repository sites because of the reduction in shipment-miles possible with the large dedicated rail casks. The 100T cask can carry between 18 and 45 consolidated, canistered spent fuel assemblies; the 150T cask capacity is between 48 and 171 assemblies. The actual payload depends on the fuel type (BWR or PWR) and the geologic medium of the repository because the consolidated fuel is packaged differently according to whether the repository is developed in salt, tuff, or basalt. Further, the MRS also reduces the difference in
costs and risks between modal options from the reactors and high-level-waste sites, which dominate the total impacts. The 150T rail cask in particular reduces the impacts of transportation from the MRS to the repository because of its large payload per shipment.

Use of repository-specific canisters and overpacks for the MRS cases influences the relative ranking of the Yucca Mountain (tuff) and the Hanford (basalt) repository sites because the canister and overpack for tuff are lower in capacity than the canister and overpack for basalt (all of the other sites use the canister and overpack for salt). In addition, the projected rail routings between the MRS locations and Yucca Mountain are more circuitous than the rail routings between the MRS locations and Hanford. The combination of increased shipment-miles and reduced canister and overpack capacities causes Yucca Mountain to rank higher in cost and risk than the Hanford repository site. Tables 4-6 summarize the shipment-miles, costs, and risks for the MRS case for an MRS located in Oak Ridge, Tennessee, with 150T dedicated rail casks between the MRS and the repository.

Summary

To summarize, between 17 and 38 truck accident fatalities, between 1.4 and 7.7 rail accident fatalities, and between 0.22 and 12 radiological health effects can be expected to occur as a result of radioactive material transportation during the 26-year operating period of the first repository. During the same period in the United States, about 65,000 total deaths from truck accidents and about 32,000 total deaths from rail accidents would occur; also an estimated 58,300 cancer fatalities are predicted to occur in the United States during a 26-year period from exposure to background radiation alone (not including medical and other manmade sources) (Reference 4). The risks reported here are upper limits and are small by comparison with the "natural background" of risks of the same type.

REFERENCES


ANALYTICAL METHODOLOGY FOR ESTIMATING THE ENVIRONMENTAL IMPACTS OF THE TRANSPORTATION OF NUCLEAR WASTE*

R.E. LUNA, J.W. CASHWELL, K.S. NEUHAUSER
Sandia National Laboratories,
Albuquerque, New Mexico,
United States of America

The U.S. Department of Energy Offices of Defense Programs and Civilian Radioactive Waste Management are in the process of locating fixed facilities for the processing and/or storage of radioactive wastes. To support the Environmental Assessments (EA) for these programs, Sandia National Laboratories, with the assistance of Battelle Pacific Northwest Laboratory and Oak Ridge National Laboratory has developed and utilized a system of computerized models and databases for the estimation of transportation-related costs and risks. This presentation outlines the major components of the system and the interactions between the components.

The interactions of these models (as applied to the user-defined input assumptions for the system to be analyzed) allow national transportation costs and risks to be compared for the scenarios of interest. A recent analysis performed for the first U.S. repository, in which this modeling structure was used, is described. Summary output tables for occupational and nonoccupational risks are presented for normal and accident cases.

In order to perform cost and risk analyses of the impacts of transportation for a future nuclear-waste-management system, a number of assumptions must be made regarding the physical, operational, and geographical characteristics of the system to be analyzed. For this reason, many of the analyses performed to provide the systems simulations required by the National Environmental Policy Act or the Nuclear Waste Policy Act are comparative in nature during the EA stage. An increasing level of specificity will be required for the final Environmental Impact Statement as well as for actual budgeting and operational forecasting.

These analyses have been performed to obtain comparative costs and risks as opposed to absolute or anticipated costs and risks. Future development and refinement of the models and their interactions will focus on the calculation of the regional and route-specific costs and risks of transportation. Calculation of the impacts of transportation on these subareas is required for the preparation of Environmental Impact Statement(s) for facility siting.

* This work supported by the United States Department of Energy under Contract DE-AC04-76DP00789. Only a summary is published here.
ON-BOARD EXPERIMENT AND ANALYSIS OF DOSE RATE DISTRIBUTIONS IN A SPENT FUEL SHIPPING VESSEL

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Abstract

ON-BOARD EXPERIMENT AND ANALYSIS OF DOSE RATE DISTRIBUTIONS IN A SPENT FUEL SHIPPING VESSEL.

An on-board experiment was carried out in a spent fuel shipping vessel, the Pacific Swan, in which 13 casks of the TN-12A and Excellox 3 type were loaded in five holds, and neutron and gamma ray dose rates were measured on the hatch covers of the holds. Before shipping, dose rates were also measured on the cask surfaces one by one for the purpose of eliminating radiation from other casks. The Monte Carlo coupling technique was successfully employed to analyse the measured neutron dose rate distributions in the vessel. As a result of this study, the Monte Carlo coupling code system, MORSE-CG/CASK-VESSEL, on which the MORSE-CG was based, was established. The agreement between the measured and the calculated neutron dose rates on the TN-12A cask surface was satisfactory. The calculated neutron dose rates agreed with the measured values within a factor of 1.5 on the Hold 3 hatch cover but to within a factor of 2 on the Hold 5 cover in which a concrete shield was fixed.

1. INTRODUCTION

An on-board experiment was carried out in a spent fuel shipping vessel, the Pacific Swan, in which 13 casks were loaded in five holds; eight casks were of the TN-12A type and five were Excellox 3. Neutron and gamma ray dose rates were measured in detail on the hatch covers of the vessel. The dose rates on and at 1 m from the cask surfaces were also measured for each cask before shipping.

The first purpose of the on-board experiment was to provide good quality measured dose rates on the cask surfaces as well as in the spent fuel shipping vessel. These measured dose rates could serve as reference data for the calculational method proposed in this paper. In addition, the measured dose rates are useful in cask design and for providing information on the dose rate profile in the vessel.

In this study, we propose a Monte Carlo coupling code system — the MORSE-CG/CASK-VESSEL on which the MORSE-CG code [1] was based [2, 3] — as a reliable code system for the shielding analysis of a spent fuel shipping vessel. In the first
step of the calculation in the MORSE-CG/CASK-VESSEL code system, the complex cask configurations were modelled in detail. Next, up to 10 casks and the large complex shielding system of the vessel were taken into account in the second step.

The calculated results from MORSE-CG/CASK-VESSEL were compared with the dose rates obtained from experiments. The calculated neutron dose rates on the TN-12A cask surface were in satisfactory agreement with the measured values. The calculated neutron dose rate was 1.6 mrem/h, which corresponds to 1.5 mrem/h for the averaged measured value on the middle surface of the TN-12A. The calculated value of 9.7 mrem/h on the lid side-surface of the TN-12A corresponds to a measured neutron dose rate of 10.0 mrem/h. The significant increase in the neutron dose rate around the lid side-surface is due to lack of any resin shield. The calculated neutron dose rates on the Hold 3 and Hold 5 hatch covers agreed with the measured values to within factors of 1.5 and 2, respectively. The calculational procedures needed to obtain the final results of interest, and the efficiency of the Monte Carlo coupling technique have already been discussed in previous works [2—4].

Gamma ray shielding calculations are rather easily performed by a point kernel code with build-up factor, such as the QAD code [5], even if the shielding system is complex or large. Accordingly, we applied the MORSE-CG/CASK-VESSEL code system to the neutron analysis. The application of the coupling code system to the gamma ray analysis is easily carried out with a minor modification of the code system. The point kernel RANKERN code [6] with neutron and gamma ray removal cross-sections has been applied successfully to analysing the measured neutron and gamma ray dose rates in the Pacific Crane [7]. The RANKERN code employed removal cross-sections, and the source intensity of neutrons and gamma rays was not obtained from an isotope generation and depletion code like ORIGEN-2 [8] but estimated from the dose rate on the cask surface using removal cross-sections. Accordingly, the RANKERN code could not be used to make a detailed general shielding analysis of casks and spent fuel shipping vessels.

2. DESCRIPTION OF VESSEL, CASK AND NEUTRON SOURCE

2.1. Vessel layout

The measurement of dose rates was carried out in the Pacific Swan, a spent fuel shipping vessel. It is 104 m long and operates at around 2950 tons dead weight. The vessel has five holds, each with an air cooling system for the loaded casks. Two types of shields are provided in the vessel. One is the 75 cm thick water tank sandwiched between iron slabs of 6.4 cm and located between Hold 5

\[1 \text{ rem} = 10^{-2} \text{ Sv}.\]
and the engine room; the other is concrete shields of 4 to 13.5 cm thick, fitted in the Hold 4 and 5 hatch covers.

The general layout of the Pacific Swan loaded with 13 casks is indicated in Fig. 1. The arrangement of the vessel structures and the loaded casks was taken into account in the second step calculation of the MORSE-CG/CASK-VESSEL code system, except for some miscellaneous structures with smaller shielding effect.

2.2. Casks

Two types of spent fuel shipping casks were loaded in the Pacific Swan; one was the dry type TN-12A and the other was the wet type Excellox 3. A TN-12A cask can contain 12 assemblies of PWR spent fuel or 30 BWR assemblies; on the other hand, an Excellox 3 cask carries 5 PWR assemblies or 12 BWR. For each cask, the maximum allowable total dose rate (neutrons plus gamma rays) is 200 mrem/h on the cask surface; meanwhile, in Japan the dose rate is limited to 10 mrem/h at any point 1 m from the cask surface.

In this study, the measurement of dose rates was carried out on the Hold 3 and 5 hatch covers. Some neutron and gamma ray contribution from the Excellox 3 casks loaded in Holds 1 and 2 to the dose rates observed on the Hold 3 and 5 hatch covers must be considered in a shielding calculation. However, as indicated in Fig. 1, the main part of the dose rates measured on the Hold 3 and 5 hatch covers could be due to the TN-12A casks loaded in Holds 3 and 5. Four TN-12A casks were in Hold 3 and four in Hold 5. There was no cask in Hold 4. We assumed the Excellox casks loaded in Holds 1 and 2 to be TN-12A casks.

Total, energy, and angular neutron fluxes were obtained on the TN-12A cask surface, and these fluxes were employed as the boundary source on the surface of all casks loaded in the vessel in the second step calculation of the MORSE-CG/CASK-VESSEL code system. The complex cask configurations were taken into account in detail in the first step calculation of the coupling code system to obtain precise flux distributions on the cask surface. However, miscellaneous structures such as shock absorbers, trunnion, and other items with smaller shielding effects were excluded from the calculation.

2.3. Neutron source

Neutrons are produced by the spontaneous fission of transplutonium nuclides and by (α,n) reactions of alpha particles emitted from decaying transplutonium nuclides primarily with 18O nuclei in the spent fuel. The neutron source intensity was calculated by the ORIGEN-2/82 code and the effective multiplication factor was obtained from the Monte Carlo code KENO-IV [9] for the TN-12A cask. The
FIG. 1. Model for calculation of neutron dose rate distribution in the Pacific Swan loaded with 13 casks.
The total neutron source intensity in a cask system $S_n$ is estimated as follows

$$\hat{S}_n = \frac{S_n}{1 - k_{\text{eff}}}$$

where $S_n$ is the neutron source intensity produced by spontaneous fission and by ($\alpha$,n) reactions.

3. MEASUREMENT

3.1. Dose rates on the cask surface and at 1 m from the surface

The neutron and gamma ray dose rates were measured on each TN-12A cask surface and at 1 m from the cask surface for the purpose of eliminating radiation from other casks before the shipping. Typical measured neutron and gamma ray dose rate distributions on the TN-12A cask surface are summarized in Fig. 2.
The measured gamma ray dose rate distribution on the cask surface had the following characteristics:

1. The dose rates were distributed rather uniformly on the cask surface and decreased toward the lid and bottom of the cask.
2. The averaged gamma ray dose rate on the middle cask surface was approximately 5.0 mrem/h.

The measured neutron dose rate distribution on the cask surface was characterized as follows:

a. Two significant peaks were observed in the distribution; one was on the side-surface of the lid, and the other was on the bottom. The reason why these peaks occurred around the lid and the bottom could be that the resin shield for neutrons did not cover them. The 10 cm thick resin shield covers the active zone (i.e. spent fuel neutron source), but there is no neutron shield around the lid and the bottom. The resin shield can be expected to reduce the neutron dose rate by a factor of 10 or less in the cask shielding system.

b. The maximum dose rate at the peak in Fig. 2 was about 10 mrem/h.

c. The dose rate distribution showed little variation on the cask surface except for the two peaks, and the averaged neutron dose rate was approximately 1.5 mrem/h.

3.2. Dose rate distributions on hatch covers

The neutron and gamma ray dose rates were measured in detail on the Hold 3 and 5 hatch covers.

The gamma ray dose rates on the Hold 3 hatch cover varied from 0.10 to 1.25 mR/h and from 0.06 to 0.37 mR/h on the Hold 5 hatch cover. On the other hand, the neutron dose rates were from 0.25 to 0.79 mrem/h on the Hold 3 hatch cover and from 0.05 to 0.30 mrem/h on the Hold 5 hatch cover. The neutron dose rates were about 0.07 mrem/h for all the measuring points in the rooms indicated in Fig. 1 (as A, B, C and D).

On the centre line of the Hold 3 and Hold 5 hatch covers, the gamma ray dose rates were from 0.45 to 0.95 mR/h and from 0.13 to 0.36 mR/h, respectively. The neutron dose rates were observed to be from 0.41 to 0.79 mrem/h and from 0.10 to 0.19 mrem/h, respectively.
4. COMPARISON WITH EXPERIMENTS

4.1. On the Hold 3 hatch cover

The comparison of neutron dose rates between the measured values and those calculated by the MORSE-CG/CASK-VESSEL code system is summarized in Fig. 3. The distributions shown in Fig. 3 are all on the centre line of the Hold 3 hatch cover. In calculating the neutron dose rates on the Hold 3 hatch cover, we assumed 4 casks loaded in Hold 3 and also 4 in Hold 2 in the second step calculation of the coupling code system. The concrete shields were fixed in the Hold 4 and 5 hatch covers, but there was no concrete in the Hold 3 hatch cover. The neutron dose rate on the cask surfaces for all the casks included in the second step calculation was assumed to be 1.6 mrem/h. The angular fluxes on the cask surface were stored in the modified subroutine SOURCE.

As indicated in Fig. 3, the calculated dose rates on the Hold 3 hatch cover agreed with the measured values to within a factor of 1.5, and the FSDs (fractional standard deviation) of the calculated dose rates were within 0.08 in the second step calculation. The second step calculations were carried out for 15 000 source particles. The slight overestimate of the calculated dose rates compared with the measured values could be due to the fact that the calculation could not take into account many complex reinforcing structures such as ribs in the second step. However, the calculated results and FSDs agree quite satisfactorily from the shielding analysis point of view.
TABLE I. MEASURED AND MONTE CARLO COUPLING CALCULATED NEUTRON DOSE RATES AND RATIOS OF CALCULATED AND EXPERIMENTAL DOSE RATES (C/E)

<table>
<thead>
<tr>
<th>Dose point</th>
<th>Centre line dose rates</th>
<th>Measured (mrem/h)</th>
<th>Calculated (mrem/h)</th>
<th>C/E</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Neutron</td>
<td>Neutron</td>
<td>Neutron</td>
<td></td>
</tr>
<tr>
<td>A</td>
<td>0.007 (±0.003)</td>
<td>0.0063 (±0.0013)</td>
<td>0.90</td>
<td></td>
</tr>
<tr>
<td>B</td>
<td>0.007 (±0.003)</td>
<td>0.0057 (±0.0006)</td>
<td>0.81</td>
<td></td>
</tr>
<tr>
<td>C</td>
<td>0.007 (±0.003)</td>
<td>0.0070 (±0.0014)</td>
<td>1.00</td>
<td></td>
</tr>
<tr>
<td>D</td>
<td>0.007 (±0.003)</td>
<td>0.0087 (±0.0017)</td>
<td>1.24</td>
<td></td>
</tr>
</tbody>
</table>

4.2. On the Hold 5 hatch cover

In calculating the neutron dose rates on the Hold 5 hatch cover, we assumed 4 casks in Hold 5 and 4 in Hold 3. There was no cask in Hold 4, as shown in Fig. 1. The concrete shields were included in detail in the second step of the coupling code system. The total, energy, and angular fluxes on the cask surfaces had the same values as those in the calculation for the Hold 3 hatch cover.

Owing to the shielding effect of the concrete shields in the Hold 4 and 5 hatch covers, the neutron dose rates on the Hold 5 hatch cover were lower than those of the Hold 3 cover by a factor of 4. The shielding effect of the concrete was confirmed by a Monte Carlo coupling calculation.

The calculated dose rates agreed with the measured values to within a factor of 2 and the FSDs of the second step calculations were within 0.15. The measured values as well as the FSDs were a little lower than the values for the Hold 3 hatch cover. However, the results could be meaningful. The overestimate of the calculated dose rates may also be due to the fact that the calculation could not take into account miscellaneous structures such as the hatch cover ribs.

4.3. In the rooms

The neutron dose rates in the rooms shown in Fig. 1 at the dose points A, B, C, and D were also calculated by the MORSE-CG/CASK-VESSEL code system. Comparison between the measured and calculated neutron dose rates is summarized in Table I. The measured dose rates were very low (0.07 mrem/h) and there was no distinguishable difference between the points A, B, C and D. The maximum allowable dose rate in a room is 0.18 mrem/h in Japan.
The values of C/E (calculation/experiment) were between 0.81 and 1.24. The FSDs of the second step calculation were not so good (within 0.21). On the other hand, the measured count rates at the dose point were so few that the measured values have a large statistical error of about 40%. Accordingly, no significant comparison between the measured neutron dose rates and the calculated values could be made in Table I.

5. CONCLUSIONS

The Monte Carlo coupling technique has been used to analyse the measured neutron dose rate distributions in a spent fuel shipping vessel, the Pacific Swan, loaded with 13 casks, and the Monte Carlo code system, MORSE-CG/CASK-VESSEL, was produced through this study. In the first calculation step of the coupling code system, complex configurations of the cask were modelled in detail and the total, energy, and angular fluxes were calculated on the cask surface to serve as the boundary source conditions for the second calculation step. Up to 10 casks in the vessel as well as the large complex shielding system of the vessel could be taken into account in the second step calculation, to obtain realistic three dimensional neutron dose rate distributions in the vessel.

The agreement between the measured and calculated neutron dose rates on the TN-12A cask surface was quite satisfactory both on the middle side-surface and on the lid and the bottom side-surfaces.

The calculated neutron dose rates on the Hold 3 hatch cover in which there was no concrete shield agreed with the measured values to within a factor of 1.5; on the other hand, agreement was within a factor of 2 on the Hold 5 hatch cover in which concrete shields were fixed.

In its current state, the MORSE-CG/CASK-VESSEL code system can be used not only as an effective tool but also as an accurate method for analysing radiation shielding problems in a spent fuel shipping vessel. Furthermore, the code system proposed in this study can be applied immediately to a ship such as a radioactive waste shipping vessel.

The present measured data on the neutron and the gamma ray dose rates on the casks and on the hatch covers can serve directly as a reference for the shielding design of a cask as well as for planning radiation protection in a spent fuel shipping vessel.

REFERENCES


AN ENGINEERING ASSESSMENT
OF THE PROBABILITY FOR DAMAGE
TO RADIOACTIVE MATERIAL TRANSPORT
CASKS DUE TO BARGE COLLISIONS

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Abstract

AN ENGINEERING ASSESSMENT OF THE PROBABILITY FOR DAMAGE TO RADIOACTIVE MATERIAL TRANSPORT CASKS DUE TO BARGE COLLISIONS.

The conditional probability for damage to radioactive material (RAM) transportation casks due to barge collisions was examined using level III methods from risk and reliability theory. For each of 12,500 collision cases examined, a time domain simulator for marine collisions was used to generate data concerning penetration, residual energy and relative velocity at each time step. The simulator developed for this purpose represents an extension of Minorsky's one dimensional collision model to three dimensions (six degrees of freedom). The model also includes the contributions of Jones and Van Mater which account for hull membrane structural resistance up to the point of hull rupture. Twenty-five hundred realistic randomly generated barge collision cases were simulated for each five classifications of US navigable waters (total of 12,500 simulated accidents). Monte Carlo methods were used to develop, from the set of predicted collision state variables, the joint demand processes for the struck barge cargoes. These demand processes were then compared with RAM cask capabilities in order to estimate the conditional probabilities of RAM cask damage given a collision in a specified classification of navigable waters.

1.0.0 INTRODUCTION

Several recent studies have determined that 80 percent of the operational reactor sites in the United States of America could utilize barge transportation for some portion of the spent fuel shipment. This paper presents, in synoptic form, a portion of a larger study [1] prepared for Sandia National Laboratories concerning the risk factors associated with barge transportation of radioactive material (RAM) as a consequence of barge collisions, rammings and groundings. The portion of that larger study here presented concerns only barge collision accidents. This topic, barge collisions, has also been presented at somewhat greater length than the present paper before the Society of Naval Architects and Marine Engineers [2].

The objective of this study was to estimate the conditional probabilities for specified levels of RAM transport cask damage, given...
that a barge collision has occurred on a specified classification of U.S. domestic navigable waters. No attempt was made in these studies to correlate the specified levels of RAM cask damage with probabilities for release of cask contents. The estimation of these conditional probabilities for RAM cask damage was developed using well known concepts from risk and reliability theory wherein the probability that system demand exceeds system capacity is evaluated. The usual representation of this relationship is:

\[
F(D>C) = \int_{-\infty}^{\infty} f_D(x) \ast F_C(x) \, dx = \int_{-\infty}^{\infty} [1 - F_D(x)] \ast f_C(x) \, dx
\]

where:
- \( F(D>C) \) is the probability that demand exceeds capacity
- \( f(\cdot) \) is a probability density function
- \( F(\cdot) \) is a cumulative probability function

and the subscript 'D' and 'C' denote process demand and capacity (also known as capability) respectively.

The demand and capability distributions are seen to be of central importance to the determination of the probability for RAM cask damage. The remainder of this paper will describe the development of these demand and capability distributions, and the details of the logic used to compose the final estimates of the conditional probabilities for cask damage.

### 2.0.0 DEMAND DISTRIBUTIONS

Demand distributions were developed which correspond to RAM cask capability with respect to identifiable cask damage processes. The selected measure of the demand process was the joint distribution of residual collision energy and relative velocity when the bow of the striking vessel crosses the cargo loading inset plane. Demand processes across two cargo loading inset planes were studied, one inset one-fifth of the barge beam which corresponds to the ANSI standard for barge transportation of high level waste [3], and the other inset 3.8 meters or one-fifth of the barge beam, whichever is greater. This latter loading restriction is designated the 'alternate loading restriction' within this study.

U.S. domestic navigable waters were grouped into five classifications for the purposes of this study, those being:

1) The so-called "Western" rivers
2) The Mississippi River system
3) The Gulf Intracoastal Waterway
4) Other inland rivers and waterways
5) Coastwise voyages offshore

Joint distributions of penetration distance, residual collision energy and relative velocity were determined using Monte Carlo techniques operating on collision case data generated using a time domain simulator. This time domain simulator was developed as a generalization and extension of Minorsky's [4] original one-dimensional ship collision model. The features of the simulator used for this study are:
1) The simulator used for this study was extended to three-dimensions and a total of six degrees-of-freedom (three horizontal plane DOF per vessel).

2) The tensor properties of hydrodynamic added mass were included. (See references [1, 2] for further details).

3) The hull shell membrane resistance of Jones and Van Mater [5] was incorporated to replace the empirical constant of Minorsky's original analysis.

4) Friction was included.

5) As a time domain simulator the joint demand process can easily be determined at any intermediate instant of the collision process evolution.

The basic structural interaction mechanism of Minorsky wherein the energy absorbed is proportional to the volume of in-plane structure deformed (a constant pressure process) was retained.

For each navigable domain 2500 accident cases were simulated. The engineering parameter vectors required as input to the time domain collision simulator were developed in two stages. First, a striking vessel type and length are chosen according to their joint frequency distributions in the U.S. Coast Guard Commercial Vessel Casualty database over the period from 1963 to 1980. The physical dimensions, displacement and design speed of the striking vessel are determined by random generation from conditional distributions with constraints to ensure realistic design. Then the physical dimensions, relevant structural parameters, loading conditions and speed are randomly determined for the struck barge using joint distribution data for the general cargo barge populations operating in the various navigable domains. The relative heading and the velocities of the two vessels at the instant of collision are also determined from random distributions conditioned on the operating domain and vessel types.

The final state of each simulated collision was classified as one of the following five conditions:

1) A glancing blow (minimal damage is presumed)
2) Collisions where the struck barge shell is dented but not ruptured
3) Collisions where the total penetration distance is less than the one-fifth beam ANSI inset
4) Collisions where the total penetration was greater than the ANSI inset and less than the alternate inset
5) Collisions where the total penetration was greater than both the ANSI inset and the alternate inset.

The distributions of collision penetration distances obtained for two important classifications of navigable waterway are shown in Table I. Residual collision energy and relative collision velocity when crossing the ANSI and alternate cargo inset planes respectively are shown in Table II. Similar data for the other waterways studied are given in reference [1].

The distributions of the various components of the demand processes were fit to analytical forms using least squares methods. The coefficients for the important relative velocity processes when crossing the cargo
TABLE I. DISTRIBUTION OF COLLISION PENETRATIONS (BASED ON 2500 SIMULATED COLLISIONS PER WATERWAY)

<table>
<thead>
<tr>
<th></th>
<th>Mississippi</th>
<th>Offshore</th>
</tr>
</thead>
<tbody>
<tr>
<td>Glancing Blows</td>
<td>2.96%</td>
<td>26.68%</td>
</tr>
<tr>
<td>Non Hull Ruptures</td>
<td>8.64%</td>
<td>9.72%</td>
</tr>
<tr>
<td>Penetrations Less than 0.2*Beam</td>
<td>62.76%</td>
<td>47.12%</td>
</tr>
<tr>
<td>Between ANSI and Alternate Insets</td>
<td>23.84%</td>
<td>2.16%</td>
</tr>
<tr>
<td>Greater than Alternate Inset</td>
<td>1.80%</td>
<td>14.32%</td>
</tr>
<tr>
<td></td>
<td>100.00%</td>
<td>100.00%</td>
</tr>
</tbody>
</table>

TABLE II. RESIDUAL ENERGY (GJ) AND RELATIVE VELOCITY (m/sec)

<table>
<thead>
<tr>
<th></th>
<th>Mississippi</th>
<th>Offshore</th>
</tr>
</thead>
<tbody>
<tr>
<td>Min. Percentile</td>
<td>50%</td>
<td>50%</td>
</tr>
<tr>
<td>Energy at ANSI Plane</td>
<td>6.1</td>
<td>11.6</td>
</tr>
<tr>
<td>Velocity at ANSI Plane</td>
<td>2.0</td>
<td>3.1</td>
</tr>
<tr>
<td>Energy at Alt. Plane</td>
<td>3.8</td>
<td>11.9</td>
</tr>
<tr>
<td>Velocity at Alt. Plane</td>
<td>1.7</td>
<td>3.1</td>
</tr>
</tbody>
</table>

TABLE III. LEAST SQUARES FIT DISTRIBUTIONS FOR RELATIVE VELOCITY

\[ \log_{10} \left[ 1/(1-F(x)) \right] = A + B x^2 \], where \( x \) is velocity (m/sec)

<table>
<thead>
<tr>
<th></th>
<th>MIN. VELOCITY</th>
<th>A</th>
<th>B</th>
</tr>
</thead>
<tbody>
<tr>
<td>Western Rivers</td>
<td>0.55</td>
<td>-7.62 \times 10^{-2}</td>
<td>1.58 \times 10^{1}</td>
</tr>
<tr>
<td>Mississippi</td>
<td>0.79</td>
<td>-7.81 \times 10^{-2}</td>
<td>9.95 \times 10^{-2}</td>
</tr>
<tr>
<td>Gulf Intracoastal</td>
<td>0.25</td>
<td>-3.14 \times 10^{-2}</td>
<td>1.26 \times 10^{1}</td>
</tr>
<tr>
<td>Inland</td>
<td>0.00</td>
<td>1.56 \times 10^{-2}</td>
<td>8.47 \times 10^{-2}</td>
</tr>
<tr>
<td>Offshore</td>
<td>1.05</td>
<td>-3.80 \times 10^{-2}</td>
<td>3.61 \times 10^{2}</td>
</tr>
</tbody>
</table>

loading inset planes are given in Table III. Similar data for other processes are given in reference [1].

3.0.0 CAPABILITY DISTRIBUTIONS

The structural capability of eight type B spent fuel casks of current design were modeled for impact loading processes imposed by a rigid ships bow. The casks were structurally modeled as concentric circular cylindrical shells which in most designs consisted of an outer structural shell, a gamma shield and an inner shell or liner. This structural system
was conservatively modeled by considering that each shell element acted in parallel with the others without any shear stresses acting at the interfaces between adjacent shells.

The impact loading process was considered to be that of a rigid ships bow striking the cask side. The striking bow was modeled as a vertical bar stem with a 3.8 cm face width and infinite vertical extent. The basic structural response was modeled as plastic denting of the shell taking place along a mechanism consisting of five plastic hinge lines. For each shell the reactive force required to continue the indentation process at any level of indentation was determined using published shell denting curves [6, 7]. Under the presumption that the shells act in parallel the total resistance to denting was then the sum of the resistances from each shell.

The dynamics of the impact process were modeled in the time domain to determine the impact velocity required to produce specified levels of damage for an impact centered at any particular location along the cask. The time domain model assigned properties associated with infinite mass to both the transport barge and the striking vessel. Thus the base to which the RAM cask sea-fastenings attached was treated as stationary and the velocity of the striking bow was unaltered by the resistance offered by the cask. The cask was assumed to be held to the barge with elasto-plastic sea-fastenings at each end of the cask. The sea-fastenings were modeled with structural area sufficient to resist a one 'g' lateral load at 90 percent yield. During the time domain simulations the individual sea-fastenings failed when a 10 percent strain was achieved.

Capability curves for each cask design were determined from the response surfaces developed in time domain analysis of cask dynamics. Attention was focused on two levels of cask damage. The first damage level corresponded to the indentation and energy absorption associated
with the hypothetical accident "puncture test" for cask certification wherein the cask is dropped one-meter onto a 15 cm diameter vertical steel bar. The second level of damage corresponds to an assumed limit state for the local shell denting mode of response.

Figure 1 shows a typical pair of capability curves for one of the casks examined in the study. Similar curves for all eight casks examined in the study are presented in reference [1]. The velocities at which damage occurs are noticeably less than for truck and rail transportation. This is principally the effect of the sea-fastenings which restrain the cask to participate with the mass of the barge which is orders of magnitude greater than that of truck or rail vehicle bodies.

4.0.0 CONDITIONAL PROBABILITY OF CASK DAMAGE

The elements necessary to estimate the conditional probability of RAM cask damage, given that a barge collision has occurred, have now been determined. This conditional probability is given the following decomposition:

\[
\begin{align*}
\text{Damage Level} & \quad \text{Across ANSI Boundary} & \quad \text{Across Alternate Boundary} \\
\text{Cert. Test} & \quad P_7 = P_1 \times P_2 \times P_3 \times P_5 & \quad P_7' = P_1' \times P_2' \times P_3' \times P_5' \\
\text{Limit State} & \quad P_8 = P_1 \times P_2 \times P_4 \times P_6 & \quad P_8' = P_1' \times P_2' \times P_4' \times P_6'
\end{align*}
\]

where: \( P_7 \) and \( P_7' \) are the conditional probabilities for the lower level of cask damage corresponding to the certification "puncture test", given a barge collision.

\( P_8 \) and \( P_8' \) are the conditional probabilities for the higher level of cask damage corresponding to the assumed limit state for the local shell denting process, given a barge collision.

\( P_1 \) is the probability for crossing the ANSI boundary

\( P_1' \) is the probability for crossing the alternate boundary

\( P_2 = P_2' = 0.7 \) is a stowage factor

\( P_3 \) and \( P_3' \) are the probabilities that the relative velocity exceeds the cask capability for the lower level of damage, given respectively that the ANSI or alternate plane has been crossed

\( P_4 \) and \( P_4' \) are the probabilities that the relative velocity exceeds the cask capability for the higher level of damage, given respectively that the ANSI or alternate plane has been crossed

\( P_5, P_5', P_6 \) and \( P_6' \) are all conditional probabilities that the residual collision energy exceeds the minimum energy absorption of the cask to the specified level of damage, given that the relative collision velocity at the cargo plane exceeds the minimum velocity required to produce the specified level of damage.
TABLE IV. PROBABILITIES FOR CROSSING CARGO LOADING BOUNDARIES

<table>
<thead>
<tr>
<th></th>
<th>ANSI Boundary</th>
<th>Alternate Boundary</th>
</tr>
</thead>
<tbody>
<tr>
<td>Western Rivers</td>
<td>0.1524</td>
<td>0.0028</td>
</tr>
<tr>
<td>Mississippi</td>
<td>0.2564</td>
<td>0.0180</td>
</tr>
<tr>
<td>Intracoastal</td>
<td>0.1692</td>
<td>0.0088</td>
</tr>
<tr>
<td>Inland Waterways</td>
<td>0.1836</td>
<td>0.0300</td>
</tr>
<tr>
<td>Offshore</td>
<td>0.1648</td>
<td>0.1432</td>
</tr>
</tbody>
</table>

TABLE V. CONDITIONAL PROBABILITY FOR LOWER LEVEL OF DAMAGE

<table>
<thead>
<tr>
<th></th>
<th>ANSI Boundary</th>
<th>Alternate Boundary</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Minimum</td>
<td>Maximum</td>
</tr>
<tr>
<td>Western Rivers</td>
<td>7.1 x 10^{-5}</td>
<td>1.6 x 10^{-2}</td>
</tr>
<tr>
<td>Mississippi</td>
<td>1.6 x 10^{-3}</td>
<td>6.2 x 10^{-2}</td>
</tr>
<tr>
<td>Intracoastal</td>
<td>1.9 x 10^{-4}</td>
<td>2.1 x 10^{-2}</td>
</tr>
<tr>
<td>Inland</td>
<td>8.6 x 10^{-4}</td>
<td>3.2 x 10^{-2}</td>
</tr>
<tr>
<td>Offshore</td>
<td>1.6 x 10^{-2}</td>
<td>8.3 x 10^{-2}</td>
</tr>
</tbody>
</table>

Table IV gives the probabilities \( P_1 \) and \( P_1' \) that the cargo inset planes have been crossed.

Given that the cargo loading boundary has been crossed it is still possible that the striking bow does not encounter a RAM cask. This possibility is represented by a stowage factor, \( P_2 \) or \( P_2' \) which is given the value 0.7 as a realistic upper bound.

The probabilities \( P_3, P_3', P_4, \) and \( P_4' \) are all determined by evaluating the level III risk integral:

\[
F(D>C) = \int_0^\infty f_D(x) * F_C(x) \, dx = \int_0^\infty [1 - F_D(x)] * f_C(x) \, dx
\]

where the demand distributions are for relative collision velocity at the cargo inset plane and the capability distributions are for cask capability to the specified level of damage.

The conditional probabilities \( P_5, P_5', P_6, \) and \( P_6' \) were studied and approach unity with a high level of confidence. Any striking vessel which possessed adequate collision velocity at the cargo plane but did not possess sufficient kinetic energy would have to be of very small mass. The structural capability of the striking vessel under such circumstances would not lead to the expectation that penetration to the cargo plane could have occurred or that damage to the cask could result. Therefore \( P_5 = P_5' = P_6 = P_6' \rightarrow 1.0.\)
A range of final conditional probabilities, reflecting the range in cask capabilities, are developed in each navigable domain. Details of the damage probabilities by cask are to be found in reference [1]. The range of final conditional probabilities for the lower level of damage corresponding to the "puncture test" for certification are given in Table V respectively for collisions across the ANSI and alternate cargo loading planes.

The range of final conditional probabilities for the higher level of damage corresponding to the assumed limit state for the local shell denting process are given in Table VI for collisions across the ANSI and alternate cargo loading planes respectively.

The conditional probability for the higher level of RAM cask damage can be seen to be considerably greater in the case of offshore transport than that obtained in any other navigable domain. This reflects the hazard of relatively high speed collisions with large ships, a hazard that does not exist on the other classifications of navigable waters.

**REFERENCES**


EVALUATION OF THE DOSES RECEIVED BY WORKERS AND THE GENERAL PUBLIC DURING RADIOACTIVE MATERIAL TRANSPORT.

This work is the continuation of a study begun in 1982 to find out more about the doses received in France by workers involved in the transport of radioactive materials and doses received by the general public. It summarizes the doses received between 1983 and 1985 during the transport of radiopharmaceutical products, spent fuel, various radioactive materials and wastes. The doses received during the transport of radiopharmaceutical products are the best documented. Progress has been made in the investigation of the doses received during the transport of spent fuel and wastes, but the results are still incomplete. Little is known about the doses received during the transport of gammagraphic equipment. The increase in the doses resulting from the transport of radioactive radiopharmaceutical products is linked to the increase in the number of packages sent. Doses resulting from the transport of spent fuel, wastes and other materials remain relatively low. Transport accounts for less than 1% of the exposure arising from the nuclear fuel cycle.

INTRODUCTION

L'application des règles contenues dans les normes fondamentales de radiation protection de l'AIEA relatives à la protection des travailleurs et, en particulier, à l'optimisation des expositions nécessite une bonne connaissance des doses reçues.

La réglementation des transports de l'AIEA ne fait pas obligation d'effectuer une dosimétrie individuelle pour le personnel affecté au transport des matières radioactives. La décision d'effectuer une dosimétrie individuelle ou une dosimétrie de zone est donc laissée à l'appréciation des employingots en fonction du degré de risque plus ou moins évident attaché au transport et à la manipulation des matières transportées.

Le transport des matières radioactives utilise un grand nombre d'itinéraires et des moyens de transport variés. Les transports font souvent intervenir un grand nombre de participants appartenant à des entreprises diverses et qu'il est parfois difficile d'identifier. Enfin, il est rare qu'une dosimétrie individuelle soit effectuée tout au long du parcours des matières transportées.

Une estimation globale de l'impact radiologique des transports des matières radioactives en France est donc difficile à obtenir et on doit se contenter d'une évaluation partielle relative à un certain nombre de transports types qui sont les transports des produits radiopharmaceutiques, les combustibles irradiés, les déchets, le transport des matières diverses, l'uranium enrichi et le plutonium.

L'étude présentée ici concerne uniquement les transports routiers.

1. LE TRANSPORT DES PRODUITS RADIOPHARMACEUTIQUES

Une étude globale du transport des produits radiopharmaceutiques en France a été effectuée par Brenot et al. [1]. Les données précises concernant la commercialisation et le transport des produits radiopharmaceutiques en France ont été fournies par l'Office des rayonnements ionisants (ORIS), situé à Saclay, qui fabrique et expédie des produits radiopharmaceutiques en France et à l'étranger. Le nombre des colis expédiés par l'entreprise est passé de 56 000 en
TABLEAU I. DOSE ANNUELLE CORRESPONDANT À LA PRÉPARATION DES EMBALLAGES DE PRODUITS RADIOPHARMACEUTIQUES À L’ORIS

<table>
<thead>
<tr>
<th>Année</th>
<th>Nombre d’agents</th>
<th>Equivalent de dose moyen (mSv)</th>
<th>Equivalent de dose maximum (mSv)</th>
<th>Equivalent de dose collectif (homme·mSv)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1980</td>
<td>8</td>
<td>15</td>
<td>22</td>
<td>123</td>
</tr>
<tr>
<td>1983</td>
<td>9</td>
<td>11</td>
<td>19</td>
<td>100</td>
</tr>
<tr>
<td>1984</td>
<td>9</td>
<td>9</td>
<td>15</td>
<td>81</td>
</tr>
</tbody>
</table>

1980 à 106 000 en 1985. L’activité commercialisée est constituée à 80% par des générateurs $^{99}$Tc-$^{99}$Mo (près de 40 000 colis annuels).

Une partie des colis est enlevée sur place par les utilisateurs. Le reste est transporté par route, par le personnel de l'installation:

- directement chez les usagers (hôpitaux) à Paris,
- aux gares ferroviaires de Paris,
- aux aéroports (Orly, Roissy).

1.1. Doses reçues lors de la préparation du colis pour expédition (tableau I)

On peut considérer que le travail de la préparation du colis pour expédition et les expositions résultantes font partie du transport. Ces opérations sont effectuées par une équipe de 8 à 9 personnes dont l’exposition moyenne est de l’ordre de 10 mSv par an. On constate que les équivalents de dose moyens maxima et collectifs ont diminué entre 1980 et 1986, et ceci malgré un doublement du nombre de colis expédiés.

1.2. Doses reçues au cours du transport par route des colis (tableau II)

Le transport routier est assuré par une équipe de 9 personnes depuis l'installation jusqu’aux utilisateurs, gares ou aéroports. Les chauffeurs assurent en même temps la manutention des colis. Les doses reçues par les chauffeurs se répartissent de façon égale entre l’opération de transport proprement dite et les manutentions.

1.3. Doses reçues dans des manipulations en transit dans les gares et aéroports et pour le transport des colis jusqu’à l’utilisateur

Si on tient compte que 70% des livraisons sont effectuées hors de la région parisienne, il y a lieu d’ajouter les doses reçues dans les manipulations en transit dans les gares et aéroports et pour le transport des colis à l’arrivée jusqu’à l’utilisateur.
TABLEAU II. DOSE ANNUELLE CORRESPONDANT AU TRANSPORT ROUTIER DES PRODUITS RADIOPHARMACEUTIQUES DE L’ORIS

<table>
<thead>
<tr>
<th>Année</th>
<th>1980</th>
<th>1983</th>
<th>1984</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nombre d’agents</td>
<td>9</td>
<td>9</td>
<td>9</td>
</tr>
<tr>
<td>Equivalent de dose moyen (mSv)</td>
<td>11</td>
<td>14,6</td>
<td>12,9</td>
</tr>
<tr>
<td>Equivalent de dose maximum (mSv)</td>
<td>17</td>
<td>20,7</td>
<td>18,8</td>
</tr>
<tr>
<td>Equivalent de dose collectif (homme-mSv)</td>
<td>98</td>
<td>131</td>
<td>116</td>
</tr>
</tbody>
</table>

N’ayant pas d’information sur ces opérations, une évaluation a été effectuée à partir des données obtenues pour ces opérations menées par les agents de l’ORIS. Cette évaluation conduit à une dose collective égale à 81 hommes-millisieverts.

1.4. Dose totale

La dose totale due au transport de produits radiopharmaceutiques en France pour l’année de référence 1984 s’établit à 300 hommes-millisieverts. Cette dose inclut celle qui résulte de la préparation du colis qu’il nous paraît logique de prendre en considération dans l’opération de transport. On remarque que la dose collective est pratiquement restée constante depuis 10 ans alors que le nombre de colis a été multiplié par 6 et que le temps prévu pour certains parcours a été considérablement allongé (transfert de Orly à Roissy pour les aéroports).

Ces résultats ont pu être obtenus grâce à l’adoption d’un certain nombre de dispositions, notamment l’automatisation du conditionnement des colis en deux chaînes (chaîne des générateurs de technétium, chaîne des autres colis A) et l’automatisation du stockage des colis.

L’exposition a pu être ainsi réduite au minimum, ce qui s’est traduit par un gain substantiel sur la dose individuelle maximale qui a diminué d’un facteur 2 en 10 ans (passant de 30 mSv à 19 mSv).

On doit également signaler, parmi les dispositions favorables, l’adoption, pour les générateurs de technétium, de mesures permettant de diminuer le niveau de dose: fût métallique avec poignée pour la manutention, évitant ainsi le contact avec la poitrine, et minimisation de toute fuite de rayonnement vers le haut. Les générateurs de technétium constituent, en effet, une des sources les plus importantes de dose. La nécessité d’avoir un encombrement et un poids permettant une manipulation manuelle aisée conduit à limiter la protection de plomb avec, pour conséquence, des niveaux de rayonnement superficiel et des indices de transport élevés.
TABLEAU III. DOSE ANNUELLE POUR LE TRANSPORT ROUTIER DES ELEMENTS COMBUSTIBLES IRRADIES

<table>
<thead>
<tr>
<th>Année</th>
<th>1983</th>
<th>1984</th>
<th>1985</th>
</tr>
</thead>
<tbody>
<tr>
<td>Equivalent de dose moyen (mSv)</td>
<td>1,1</td>
<td>0,6</td>
<td>1</td>
</tr>
<tr>
<td>Equivalent de dose maximum (mSv)</td>
<td>4,2</td>
<td>5,8</td>
<td>7,2</td>
</tr>
<tr>
<td>Equivalent de dose collectif (homme·mSv)</td>
<td>26</td>
<td>15</td>
<td>23</td>
</tr>
</tbody>
</table>

Enfin, on a recherché une protection plus poussée des cabines des chauffeurs ainsi qu'une meilleure organisation du travail pour le remplissage des véhicules et les attentes avant transport.

2. LE TRANSPORT DES COMBUSTIBLES IRRADIES

Le transport des combustibles irradiés se fait en France essentiellement par chemin de fer. Dans la plupart des cas, la route n'est utilisée qu'entre les centrales dites «non embranchées» et la gare la plus proche et entre la gare terminale et le site de retraitement.

Six entreprises emploient 23 chauffeurs dont quelques-uns seulement participent à la manutention; tous sont classés comme directement affectés à des travaux sous rayonnement (catégorie A) et sont donc soumis à une dosimétrie individuelle mensuelle. Les quantités annuelles transportées sont de l'ordre de 1500 tonnes.

Les résultats dosimétriques relatifs aux années 1983, 1984 et 1985 sont résumés dans le tableau III.

Le détail des résultats fait ressortir que les personnes qui reçoivent les doses les plus élevées sont celles affectées à la manutention ou les chauffeurs qui parfois participent à la manutention.

Le transport par chemin de fer des combustibles irradiés sur la majorité du parcours explique que les doses reçues par les transporteurs routiers soient faibles. Par contre, les doses reçues lors des manipulations effectuées au départ et à l'arrivée dans les usines ne sont pas comptabilisées ici.

3. LE TRANSPORT DES DECHETS RADIOACTIFS

Le transport des déchets radioactifs entre les entreprises productrices et le centre de stockage superficiel de l'Agence nationale pour la gestion des déchets radioactifs (Andra) situé sur le site de La Hague utilise des circuits qui peuvent
TABLEAU IV. DOSE ANNUELLE POUR LE TRANSPORT ROUTIER DES DECHETS

<table>
<thead>
<tr>
<th>Année</th>
<th>1983</th>
<th>1984</th>
<th>1985</th>
</tr>
</thead>
<tbody>
<tr>
<td>Equivalent de dose moyen (mSv)</td>
<td>1,79</td>
<td>1,74</td>
<td>2,6</td>
</tr>
<tr>
<td>Equivalent de dose maximum (mSv)</td>
<td>5,55</td>
<td>6,8</td>
<td>7,55</td>
</tr>
<tr>
<td>Equivalent de dose collectif (homme·mSv)</td>
<td>41,1</td>
<td>40,1</td>
<td>47,4</td>
</tr>
</tbody>
</table>

 être soit entièrement routiers, soit partiellement routiers et partiellement ferroviaires, comme c’est le cas pour les combustibles irradiés. Ces circuits ont été définis par l’Andra en concertation avec les producteurs de déchets.

Les transports sont effectués par plusieurs entreprises «agrées» d’importance inégale.

Une société emploie environ 18 chauffeurs dont la majorité est employée fréquemment et un petit nombre occasionnellement au transport des déchets radioactifs. Tous sont considérés comme directement affectés à des travaux sous rayonnement (catégorie A) et font l’objet d’une dosimétrie individuelle mensuelle.

Une seconde entreprise emploie cinq personnes pour le transport des déchets. Ces personnes ne sont pas directement affectées à des travaux sous rayonnement (catégorie B) mais subissent cependant une surveillance individuelle par film dosimètre.

Le bilan dosimétrique pour les années 1983, 1984 et 1985 est résumé dans le tableau IV.

Comme pour les combustibles irradiés, le transport des déchets se fait en grande partie par chemin de fer, ce qui peut expliquer les doses relativement faibles reçues par les transporteurs routiers.

La comparaison avec les résultats donnés en 1982 montre une réduction sensible de l’équivalent de dose moyen et une relative stabilité de l’équivalent de dose collectif.

Cependant, nous estimons que le transport des déchets nécessite une attention particulière, étant donné notamment la diversité des déchets et de leurs modes de conditionnement et l’accroissement prévu au cours des prochaines années du transport de ces matières.

4. LES MATIERES RADIOACTIVES DIVERSES

Les matières radioactives diverses désignent dans ce mémoire les matières autres que les produits radiopharmaceutiques, les combustibles irradiés, les déchets et les sources de gammagraphie. Elles englobent donc l’UF₄, l’UF₆ naturel,
### TABLEAU V. DOSE ANNUELLE POUR LE TRANSPORT ROUTIER DES AUTRES MATIÈRES RADIOACTIVES

<table>
<thead>
<tr>
<th>Année</th>
<th>1983</th>
<th>1984</th>
<th>1985</th>
</tr>
</thead>
<tbody>
<tr>
<td>Équivalent de dose moyen (mSv)</td>
<td>0,6</td>
<td>0,4</td>
<td></td>
</tr>
<tr>
<td>Équivalent de dose maximum (mSv)</td>
<td>2,6</td>
<td>2,7</td>
<td>Pas de résultat</td>
</tr>
<tr>
<td>Équivalent de dose collectif (homme·mSv)</td>
<td>7,4</td>
<td>9,3</td>
<td></td>
</tr>
</tbody>
</table>

### TABLEAU VI. DOSE ANNUELLE POUR LE TRANSPORT ROUTIER DE L'URANIUM ENRICHÉ ET DU PLUTONIUM

<table>
<thead>
<tr>
<th>Année</th>
<th>1983</th>
<th>1984</th>
<th>1985</th>
</tr>
</thead>
<tbody>
<tr>
<td>Équivalent de dose moyen (mSv)</td>
<td>0,16</td>
<td>0,07</td>
<td>0,18</td>
</tr>
<tr>
<td>Équivalent de dose maximum (mSv)</td>
<td>1,3</td>
<td>0,3</td>
<td>1,5</td>
</tr>
<tr>
<td>Équivalent de dose collectif (homme·mSv)</td>
<td>1,6</td>
<td>0,7</td>
<td>1,8</td>
</tr>
</tbody>
</table>

l'UO$_2$, des combustibles neufs, quelques assemblages combustibles provenant des réacteurs de recherche, des effluents en citerne et des sources radioactives diverses.


5. **PLUTONIUM ET URANIUM ENRICHED**

Une entreprise emploie des chauffeurs affectés au transport du plutonium et de l'uranium enrichi. Ces personnes, bien que non directement employées à des travaux sous rayonnement (catégorie B), font néanmoins l'objet d'une dosimétrie individuelle par film.

Les résultats pour les années 1983, 1984, 1985 sont résumés dans le tableau VI.
6. LE TRANSPORT DES GAMMAGRAPHES EN EMBALLAGES DE TYPE B

Il existe en France environ 400 entreprises effectuant des travaux de gammagraphie et on estime entre 1200 et 1400 le nombre des gammagraphes en service en France.

Les gammagraphes renferment des sources d'émetteurs \( \gamma \) d'activité parfois très importante dans des emballages de type B (U). On distingue à ce propos deux types d'appareils: les appareils dits «fixes», moins nombreux, qui renferment les activités les plus élevées, et les appareils dits «portables». En France, pour les appareils portables, qui constituent la majorité des appareils transportés, l'activité est limitée à 100 Ci (3,7 TBq) et le débit de dose à la surface de l'appareil est limité à 100 mrem·h\(^{-1}\) (1 mSv·h\(^{-1}\)). Ces appareils sont couramment transportés dans des véhicules de service de l'entreprise propriétaire sur les lieux de leur utilisation et on estime à 150 à 200 par jour le nombre de ces mouvements.

Les techniciens affectés à l'utilisation, au transport et à l'entretien des appareils sont obligatoirement titulaires d'une certification et ont suivi un stage de formation. La plupart sont directement affectés à des travaux sous rayonnement (catégorie A). Leur nombre, qui est difficile à évaluer exactement, est de l'ordre de 4 à 5000. Ils sont dispersés entre de nombreuses sociétés privées ou publiques (de 400 à 500).

Nous n'avons pas pu obtenir de résultats relatifs au nombre de transports annuels des appareils de gammagraphie, et au nombre de personnes qui participent à ces transports, ni aux doses effectivement reçues par ce groupe de personnes. Notons cependant que Gelder et Hugues [2], au Royaume-Uni, estiment la dose collective due au transport des gammagraphes à 0,2 homme-sievert en 1983. Ce chiffre devrait être le même en France, les deux pays ayant un développement industriel similaire.

7. ESTIMATION DES DOSES REÇUES PAR LE PUBLIC

A partir des données précédentes, il est excessivement difficile de faire une estimation de la dose collective reçue par le public. Celle-ci nécessiterait l'application de modèles d'exposition ou, dans le cas des transports par avion, l'utilisation de dosimètres placés à l'emplacement des passagers dans les avions transportant des matières radioactives. L'estimation faite par Brenot et al. [1] pour le transport des produits radiopharmaceutiques, se situe entre 0,01 et 0,1 homme-sievert par an.

Sur cette base, si l'on tient compte de l'ensemble des transports radioactifs, la dose collective annuelle reçue par le public pourrait être du même ordre de grandeur que celle reçue par les professionnels.
TABLEAU VII. RESUME DES RESULTATS

<table>
<thead>
<tr>
<th>Matières</th>
<th>Equivalent de dose moyen (mSv)</th>
<th>Equivalent de dose collectif (homme·mSv)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Produits radiopharmaceutiques</td>
<td>0,011</td>
<td>300</td>
</tr>
<tr>
<td>Combustibles irradiés</td>
<td>0,001</td>
<td>15</td>
</tr>
<tr>
<td>Déchets</td>
<td>0,002</td>
<td>40</td>
</tr>
<tr>
<td>Autres</td>
<td>0,0004</td>
<td>9</td>
</tr>
<tr>
<td>Uranium enrichi et plutonium</td>
<td>0,0001</td>
<td>0,7</td>
</tr>
<tr>
<td>Gammagraphes</td>
<td>?</td>
<td>200</td>
</tr>
<tr>
<td>Total</td>
<td></td>
<td>565</td>
</tr>
</tbody>
</table>

8. BILAN DOSIMETRIQUE ET CONCLUSION

Les résultats de cette étude sont résumés dans le tableau VII pour l'année de référence 1984.

Le transport des produits radiopharmaceutiques constitue la principale contribution à la dose collective reçue par les travailleurs. Des efforts entrepris ces dernières années ont permis d'en maintenir constant le niveau et cela malgré un accroissement considérable du trafic de ces matières. Des études plus poussées, permettant notamment une meilleure appréciation des doses, nous paraissent nécessaires pour les déchets et les autres matières radioactives. On peut cependant conclure que l'exposition résultant du transport de l'ensemble des matières radioactives ne représente qu'une fraction très faible (moins de 1%) des expositions liées au cycle du combustible nucléaire.

REFERENCES


LE SYSTEME DE TRANSPORT DES PRODUITS RADIOPHARMACEUTIQUES EN FRANCE*

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Abstract—Résumé

THE SYSTEM FOR THE TRANSPORT OF RADIOPHARMACEUTICAL PRODUCTS IN FRANCE.

Radiopharmaceutical products transported in Type A packages are used for in vivo examinations and for in vitro radiodiagnosis carried out in nuclear medicine services and research laboratories. The activities are low and spread over more than 200,000 packages sent all over France. First, the quantity of radiopharmaceutical products used and their distribution are described, after which there is a description of two aspects of the transport system — conditioning and transport — on the basis of the experience of the Oris-Industrie Company, which is both the biggest French producer and leading company in the French market. Mention is also made of the impact of the ICRP recommendations and the regulations prevailing in France on the design of packaging, dispatch and storage facilities and on handling and transport practices. Each type of transport, by road, rail or air, serves a specific geographical area; hence the importance of one type over the others in terms of the quantities transported or the number of kilometres covered can easily be worked out. The last part of the paper is devoted to the evaluation of risks under normal conditions. At present, the number of transport incidents or accidents is very small and there is no point in making any statistical analysis of them. There is a good record of dosimetric surveys on external radiation exposure of handling and transport personnel: the collective dose equivalent for workers involved in France in transport activities is about 0.42 man-Sv per year. However, as far as the public is concerned, there is need for a large number of assumptions in order to evaluate the exposures resulting mainly from urban road transport: the collective dose equivalent for the public resulting from the transport of unsealed sources for medical purposes probably lies between 0.01 man-Sv and 0.1 man-Sv per year.

* Cette recherche a bénéficié d’un financement de la Commission des Communautés européennes (contrat n° 84/B/7015/11/005/17) [1].
LE SYSTÈME DE TRANSPORT DES PRODUITS RADIOPHARMACEUTIQUES EN FRANCE.

Les produits radiopharmaceutiques, transportés en colis de type A, sont utilisés pour des investigations in vivo et pour le radiodiagnostic in vitro pratiqués dans les services de médecine nucléaire et les laboratoires de recherche. Les activités sont faibles et réparties sur plus de 200 000 colis qui sont envoyés dans toute la France. Dans un premier temps, on précise quelles sont les quantités de produits radiopharmaceutiques employées et comment elles se répartissent. Ensuite, le système de transport est décrit sous ses deux aspects, le conditionnement et le transport, en s'appuyant sur l'expérience de la Compagnie Oris-Industrie qui, premier producteur français, est aussi la première compagnie sur le marché français. On évoque aussi l'impact des recommandations de la CIPR et des règlements en vigueur en France sur la conception des installations de conditionnement, d'expédition et de stockage et sur les pratiques de manutention et de transport. Quant aux modes de transport, chacun d'eux, route, rail ou avion, dessert une zone géographique précise; aussi, l'importance d'un mode relativement aux autres, en terme de quantités transportées ou de kilomètres parcourus, s'en déduit aisément. La dernière partie de ce mémoire est consacrée à l'évaluation des risques en conditions normales. D'ores et déjà, les incidents ou accidents de transport sont très peu nombreux et l'analyse statistique de ceux-ci est sans objet. Il existe un bon historique de relevés dosimétriques pour l'irradiation externe des manutentionnaires et des chauffeurs: ainsi, l'équivalent de dose collectif pour les travailleurs impliqués en France dans les activités de transport est de l'ordre de 0,42 homme-Sv par an. Pour le public par contre, de nombreuses hypothèses sont nécessaires pour évaluer les expositions dues essentiellement au transport routier urbain: l'équivalent de dose collectif du public dû au transport des sources non scellées à usage médical doit être compris entre 0,01 homme-Sv et 0,1 homme-Sv par an.

1. INTRODUCTION

Les radioéléments artificiels peuvent être employés en médecine à des fins thérapeutiques, diagnostiques ou de recherche. Cette étude ne porte que sur les utilisations médicales de ces radioéléments. De plus, les radioéléments distribués seulement sous forme de sources scellées ont été exclus ($^{60}$Co, $^{137}$Cs et $^{192}$Ir); pour ceux qui se présentent sous les deux formes, source scellée et source non scellée ($^{125}$I et $^3$H), seule l'utilisation en source non scellée est retenue. Les radioéléments auxquels on s'intéresse se caractérisent par le fait qu'ils sont l'objet de déplacements plus ou moins longs de leurs lieux de production aux divers sites d'utilisation. De plus, leur livraison n'est possible que si l'utilisateur effectue préalablement une demande de fournitures adressée pour contrôle et agrément à la Commission interministérielle des radioéléments artificiels (CIREA). Après visa de cet organisme, la demande peut être satisfaite par le fournisseur et la livraison peut avoir lieu.

2. LES PRODUITS

Les radioéléments utilisables sont au nombre de 45 environ; une trentaine au plus peuvent être administrés à l'homme (in vivo) dans un but thérapeutique ou de diagnostic. Les autres utilisations sont le diagnostic in vitro ou les travaux de re-
cherche. Les activités délivrées sont celles que l'utilisateur indique dans sa demande de fournitures à la CIREA et qu'il exige au moment où il se sert des produits. Ces activités, pour chaque utilisation et pour les principaux radioéléments, sont données dans le tableau I.

Le radionucléide peut se trouver sous la forme d'un liquide contenu dans un flacon, d'un gaz dans une ampoule scellée, d'une composition, etc. Dans le contexte médical, chaque radionucléide utilisé en source non scellée est de faible activité et dispersable. Seule la valeur limite $A_2$ proposée par l'AIEA est utile. Le tableau II permet d'adapter l'emballage au contenu.

Les préparations radiopharmaceutiques (administrées à l'homme) et radiochimiques (non administrées à l'homme) entrent toutes dans la 3e catégorie (groupe 70301 du classement des matières radioactives). Les trousse d'analyse médicale aux activités très faibles relèvent de la 4e catégorie (groupe 70401 de ce même classement). En conséquence, aucun emballage n'est de type B.

Selon l'activité, une protection en plomb peut s'imposer tant pour des raisons de transport que d'utilisation. Le tout est alors soit conditionné dans une boîte en fer blanc sertie (et c'est un colis de type A), soit dans une trousse en polystyrène expansé (trousses d'analyse médicale). Le produit est transporté dans une boîte en carton. Un cas important est celui des générateurs de technétium 99m; l'appareil est automatique et prêt à l'emploi et se transporte dans un seau. Il constitue un colis de type A.

3. LES CLIENTS ET LES FLUX DE PRODUITS

Les utilisateurs français doivent être enregistrés et agréés par la CIREA. Chaque autorisation est nominative et, en général, restreinte à un seul site. Les gros utilisateurs sont les services de médecine nucléaire qui sont au nombre de 127 en France métropolitaine. Les titulaires CIREA agréés pour des tâches in vitro sont au nombre de 196. Enfin, il y a 375 titulaires CIREA habilités «recherche».

Les pôles importants sont Paris et la région Île-de-France, Lyon et la région Rhône-Alpes, Marseille et la région Provence-Côte d'Azur, Strasbourg et la région Alsace, Montpellier et la région Languedoc-Roussillon, Lille et la région Nord-Pas-de-Calais. Globalement, on peut considérer que les utilisateurs sont peu nombreux et géographiquement très concentrés.

Les flux peuvent être établis pour les quantités de radioéléments ou sur les nombres de livraisons effectuées. Les expéditions se font quasiment toutes à partir de la région parisienne. Les flux essentiels concernent: le tritium, l'iode 123,125 et 131, le technétium 99m, le xénon 133 et le thallium 201.

4. LE SYSTEME DE TRANSPORT DE LA COMPAGNIE ORIS-INDUSTRIE

La compagnie Oris-Industrie développe, vend et distribue environ 80% des produits utilisés en France en médecine nucléaire et 40% de ceux servant au
TABLEAU 1. ACTIVITÉS DELIVRÉES EN FRANCE EN 1984

<table>
<thead>
<tr>
<th>Utilisation in vivo</th>
<th>Activités délivrées (mCi)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Technétium 99m</td>
<td>(T = 6,02 h) 3 812 000</td>
</tr>
<tr>
<td>Xénon 133</td>
<td>(T = 5,24 d) 485 000</td>
</tr>
<tr>
<td>Iode 131</td>
<td>(T = 8,02 d) 316 000</td>
</tr>
<tr>
<td>Thallium 201</td>
<td>(T = 3,04 d) 40 300</td>
</tr>
<tr>
<td>Yttrium 70</td>
<td>(T = 2,67 d) 17 400</td>
</tr>
<tr>
<td>Gallium 67m</td>
<td>(T = 3,25 d) 13 200</td>
</tr>
<tr>
<td>Iode 123</td>
<td>(T = 13,20 h) 11 400</td>
</tr>
<tr>
<td>Phosphore 32</td>
<td>(T = 14,20 d) 11 000</td>
</tr>
<tr>
<td></td>
<td>(...)</td>
</tr>
<tr>
<td>Total</td>
<td>4 732 060</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Utilisation in vitro</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Iode 125</td>
<td>(T = 59,9 d) 3 000</td>
</tr>
<tr>
<td>Tritium</td>
<td>(T = 12,3 a) 1 140</td>
</tr>
<tr>
<td>Chrome 51</td>
<td>(T = 27,7 d) 400</td>
</tr>
<tr>
<td>Phosphore 32</td>
<td>(T = 14,2 d) 35</td>
</tr>
<tr>
<td>Carbone 14</td>
<td>(T = 5730 a) 10</td>
</tr>
<tr>
<td>Total</td>
<td>4 585</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Utilisation en recherche médicale</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Tritium</td>
<td>(T = 12,3 a) 32 200</td>
</tr>
<tr>
<td>Iode 125</td>
<td>(T = 59,9 d) 13 800</td>
</tr>
<tr>
<td>Phosphore 32</td>
<td>(T = 14,2 d) 12 600</td>
</tr>
<tr>
<td>Soufre 35</td>
<td>(T = 76,4 d) 8 100</td>
</tr>
<tr>
<td>Chrome 51</td>
<td>(T = 27,7 d) 3 850</td>
</tr>
<tr>
<td>Carbone 14</td>
<td>(T = 5730 a) 3 400</td>
</tr>
<tr>
<td>Iode 131</td>
<td>(T = 8,02 d) 1 500</td>
</tr>
<tr>
<td>Cuivre 64</td>
<td>(T = 12,84 h) 200</td>
</tr>
<tr>
<td>Total</td>
<td>75 650</td>
</tr>
</tbody>
</table>

radiodiagnostic in vitro. L'étude de son système de distribution avec ses trois secteurs: conditionnement, manutention, transport, permet d'avoir une vision assez complète de la situation présente de ces secteurs en France.

4.1. Description du système

Le bâtiment où sont centralisées les activités de distribution à destination de la clientèle se trouve dans le Centre d'études nucléaires de Saclay, soit à 30 km du centre de Paris. Trois grandes zones y coexistent: la zone de production avec laboratoire et enceintes, la zone d'enlèvement des produits dits «locaux actifs» et le hall d'expédition. À chaque zone correspond un groupe opérationnel: le personnel de production en laboratoire est isolé des produits qui sont élaborés dans les enceintes; les pontiers dans les locaux actifs, les manutentionnaires et chauffeurs dans le hall d'expédition sont physiquement au contact avec les produits élaborés ou les colis.

4.1.1. Conditionnement et expédition

Les contraintes réglementaires portent à la fois sur l'emballage et l'étiquetage indispensables pour le transport ultérieur.

Un emballage de type A n'a pas à être agréé par le ministère des transports. Par contre, le fournisseur de radioéléments doit pouvoir apporter la preuve de conformité aux prescriptions demandées. La compagnie Oris-Industrie a recours au Laboratoire national d'essais pour effectuer la série d'épreuves auxquelles doivent satisfaire les emballages. Le générateur de technétium, qui relève aussi de la catégorie A, subit en plus une épreuve de chute de 9 m et une autre de pénétration où la barre chute de 1,70 m. Les trousses d'analyse médicale sont conditionnées dans des emballages industriels auxquels les tests requis précédemment ne s'appliquent pas.

<table>
<thead>
<tr>
<th>Activité du radioélément</th>
<th>Forme non liquide</th>
<th>Forme liquide</th>
<th>Catégorie</th>
<th>Type de l'emballage</th>
</tr>
</thead>
<tbody>
<tr>
<td>Moins de $10^{-3}$ A$_2$</td>
<td>Moins de $10^{-4}$ A$_2$</td>
<td>4</td>
<td>Industriel ordinaire</td>
<td></td>
</tr>
<tr>
<td>$10^{-3}$ A$_2$ à A$_2$</td>
<td>$10^{-4}$ A$_2$ à A$_2$</td>
<td>3</td>
<td>Type A</td>
<td></td>
</tr>
<tr>
<td>Plus de A$_2$</td>
<td>1 ou 2</td>
<td></td>
<td>Type B</td>
<td></td>
</tr>
</tbody>
</table>
Les règles d'étiquetage sont uniformisées au plan international (règlement de l'AIEA).

La compagnie Oris-Industrie a pour principe d'avoir un emballage pour chaque produit expédié, ceci pour des raisons de standardisation et pour éviter toute confusion. Les opérations de transport, de conditionnement et d'entreposage à l'intérieur du bâtiment diffèrent selon les produits; elles s'effectuent dans des chaînes blindées et automatisées.

4.1.2. Transport et livraison

En France, le transport des produits radioactifs est régi par des dispositions directement issues des recommandations de l'AIEA. Ces recommandations portent sur les débits de dose maximaux; elles ont été adoptées pour chaque mode de transport. L'autorité compétente est le ministère des transports représenté par le président de la Commission interministérielle du transport des matières dangereuses dont un des membres est président de la Commission de sûreté des transports du Groupe CEA. Chaque colis est caractérisé par son indice de transport (IT). Dans les situations usuelles de transport par avion, fer ou route, ou dans le cas des entreposages intérimaires, le IT total des colis ne peut excéder 50. Ces contraintes sur le débit de dose maximal et l'indice de transport ont naturellement de fortes répercussions sur la conception des véhicules spécialisés et surtout sur l'organisation du transport.

Des groupages sont réalisés pour les transports internationaux par route vers les divers pays d'Europe. Les autres trajets routiers sont de courte distance et considérés comme interurbains.

Les chauffeurs constituent un personnel sélectionné, consciencieux et stable; ils on tous suivi le cours sur le transport des matières radioactives donné à l'Institut national des sciences et techniques nucléaires.

Les véhicules, au nombre de dix, sont spécialement équipés. Le chauffeur est isolé du chargement par une paroi de plomb de 10 mm d'épaisseur. Les véhicules subissent un contrôle annuel de radiocontamination, bien qu'il ne soit pas obligatoire pour ce type de transport. Chaque véhicule effectue environ 180 km par jour et, sur l'année, un total de 40 000 km.

Pour les clients hors de l'Ile de France, la compagnie fait déposer les colis aux diverses gares parisiennes. Ils sont repris par le Service national de messageries qui assure le transport par chemin de fer et la livraison chez le client.


La remise directe d'un colis à la poste est possible en France mais dans des conditions contraignantes; la compagnie n'utilise pas actuellement ce moyen.

Quant au nombre de colis, il est de l'ordre de 150 000, dont 100 000 colis de type A et 50 000 trousses d'analyse médicale.
4.2. Les risques

Les risques peuvent être classés en deux grandes catégories selon qu’ils sont inhérents à tout transport ou spécifiques des produits transportés.

4.2.1. Risques non nucléaires

Il s’agit des accidents du travail qui se produisent lors des manutentions et des accidents de type classique qui ont lieu lors des transports. Il n’y a jamais eu d’accident du type choc frontal pour les véhicules assurant la répartition des colis à partir de Saclay. Il n’y a pas eu jusqu’à présent de perte ni de vol.

4.2.1. Risques nucléaires

Les risques d’exposition interne et externe des travailleurs chargés de la manutention existent naturellement dans le bâtiment décrit. Une équipe permanente est chargée de la radioprotection des locaux. La surveillance dosimétrique des personnels est individuelle, au moyen de films dosimètres X + γ poitrine, et collective, grâce à la métrologie d’ambiance.

L’équivalent de dose moyen des pontiers a baissé de 17,3 à 7,95 mSv en vingt ans et le nombre d’agents est passé de 6 à 12. La totale automatisation des chaînes de conditionnement va conduire encore à une diminution des expositions et donc des doses.

Pour les agents du hall d’expédition, dont le nombre est en légère augmentation (de 7 à 10), l’équivalent de dose moyen est passé de 16,03 à 9,06 mSv en treize ans. Dans les études étrangères [2-4], les données ne concernent que les manutentionnaires des centres de marchandises en transit des aéroports et les doses annuelles par agent sont donc normalement beaucoup plus faibles: de 1 à 5 mSv.

Les chauffeurs, dont le nombre est passé de 7 à 9, constituent le groupe de travailleurs pour lequel la diminution en treize ans de l’équivalent de dose moyen annuel est la plus faible: de 16,76 à 12,87 mSv.

Ainsi, pour le groupe des chauffeurs, la dose collective n’a pas vraiment changé. Chargement et livraison sont responsables de la moitié de la dose reçue. L’efficacité des tâches a cependant été multipliée par 6 si l’on rapporte la dose collective au nombre de colis. Les doses individuelles subies sont comparables à celles publiées concernant les chauffeurs spécialisés [3, 5]. Pour les chauffeurs qui ne livrent pas seulement des colis radioactifs, les doses individuelles reçues annuellement semblent être dix fois plus petites [6].

5. ESTIMATION DE LA DOSE COLLECTIVE

Pour la France entière en 1984, le nombre total de colis transportés et destinés au marché français (tous producteurs confondus) s’élève à 200 000 environ, dont
TABLEAU III. DOSES COLLECTIVES POUR LES GROUPES DE TRAVAILLEURS

<table>
<thead>
<tr>
<th>Conditionnement</th>
<th>Homme-Sv</th>
<th>%</th>
</tr>
</thead>
<tbody>
<tr>
<td>Manutentionnaires aéroport/gare</td>
<td>0,177</td>
<td>42</td>
</tr>
<tr>
<td>Chauffeurs</td>
<td>0,081</td>
<td>19</td>
</tr>
<tr>
<td>Total</td>
<td>0,425</td>
<td>100</td>
</tr>
</tbody>
</table>

125 000 sources non scellées en colis A étiquetés et 75 000 trousses d’analyse médicale en emballage industriel.

Les doses collectives observées pour les divers groupes de travailleurs de la compagnie Oris-Industrie ont servi de base à l’évaluation des doses reçues par l’ensemble des travailleurs sur le territoire français qui sont en contact occasionnel ou régulier avec des colis de type A. On obtient le tableau III.

L’évaluation faite est proche de celle publiée par Hamard et Sousselier en 1983 [7].

Quant à l’exposition du public, elle trouve son origine dans le fait que les colis sont transportés dans le trafic routier urbain par des camionnettes. Deux scénarios hypothétiques de circulation urbaine, trafic très fluide et trafic très dense, conduisent à l’intervalle suivant pour l’équivalent de dose collectif du public: de 0,01 à 0,1 homme-Sv.

6. CONCLUSIONS

Pour l’année 1984 en France, les radionucléides en sources non scellées utilisés dans le milieu médical représentent 4812 Ci (4812 × 37 GBq) dont 80% pour le seul technétium 99m; le marché de ces produits a connu en quinze ans une très forte expansion, d’un facteur 6 au moins. Quant aux colis, ils sont au nombre de 200 000 au moins, dont 120 000 colis A étiquetés contenant des préparations radiopharmaceutiques administrées in vivo et des préparations radiochimiques utilisées in vitro ou en recherche. Les utilisateurs sont peu nombreux, 375 en tout, dont 127 services de médecine nucléaire qui sont les principaux usagers et pour lesquels les livraisons se font très régulièrement et selon quelques grands axes.

Les accidents de transport sont rarissimes.
Les données dosimétriques des travailleurs sont de qualité et l'équivalent de dose collectif pour les travailleurs est de l'ordre de 0,42 hommes-Sv; de plus, cet équivalent de dose collectif est sensiblement le même depuis quinze ans malgré l'accroissement du personnel et l'augmentation de la production. Les chauffeurs, qui ne transportent que des colis radioactifs, sont régulièrement exposés à plus de 0,1 homme-Sv/a et constituent le groupe de travailleurs qui sont à surveiller prioritairement; la distribution des doses reçues par les manutentionnaires des gros centres de transit de marchandises est mal connue.

L'équivalent de dose collectif du public, dû au transport de ces sources non scellées à usage médical, doit être compris entre 0,01 et 0,1 homme-Sv.

REFERENCES

TRANSPORT EXPERIENCE
AND PLANNING
ESTIMATED ANNUAL WORLDWIDE SHIPMENTS OF RADIOACTIVE MATERIAL

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Abstract

ESTIMATED ANNUAL WORLDWIDE SHIPMENTS OF RADIOACTIVE MATERIAL.

Data on shipments within or from 35 Member States of the IAEA have been used to estimate the volume of radioactive material shipments worldwide. The data have been extrapolated on the basis of gross domestic product and population, and it is estimated that from 18 to 38 million package shipments were made annually in the early 1980s.

INTRODUCTION

At the first meeting of the International Atomic Energy Agency’s Standing Advisory Group on the Safe Transport of Radioactive Materials (SAGSTRAM) held in Vienna in October 1978, it was recommended that “a system should be developed by the Agency for the collection, storage and retrieval of information on the worldwide volume of traffic in all types of radioactive materials by all modes of transport.” The purpose of this effort was to develop a database which would assist in assessing the adequacy of the requirements set forth in the Regulations for the Safe Transport of Radioactive Material, Safety Series No. 6, and thereby guide the revision of these Regulations.

In considering this recommendation, it was decided that it would be unrealistic to seek such data on a continuing basis. Therefore, the Agency responded to this advice in February 1980 by sending forms to all Member States requesting

* The authors gratefully acknowledge the efforts of Ms. D. Pal of the IAEA, Mr. D.R. Hopkins of the US Nuclear Regulatory Commission and Mr. B.G. Pettersson of the Swedish Nuclear Power Inspectorate.
"summary information based on the best data available to or collected by the competent authorities over an initial one-year reporting period." It was felt that this "would be a reasonable compromise between the desirability of receiving exact data on a continuing basis and the necessity of having adequate, representative, worldwide data." Furthermore, it was planned that these data would be compiled and stored in the Agency's computer for future use and reference by both the Agency and Member States. This paper provides a status report on this activity, an overview of the data provided, and an estimate of the worldwide volume of shipping based on an extrapolation of the data in terms of gross domestic product and population.

DATA COLLECTION PROCEDURES

The methods used in collecting the data were left to the discretion of each Member State. Four sets of forms were provided for summarizing data to the Agency:

(1) **Form A** — Summary of Package Movements for International Transport (exports only). This form included requests for information in terms of total annual number of packages, transport index and distance by package classification (Exempt, LSA or LLS, Type A, etc.) and by transport mode (road, rail, sea, etc.); and number of packages by labelling category (Exempt, I-WHITE, II-YELLOW, III-YELLOW and Fissile).

(2) **Form B** — Summary of Package Movements for Domestic Transport. This form included requests for the same type of data as in Form A.

(3) **Form C** — Summary of Consignment Movements. This form included requests for data on specific types of material consignments (plutonium, radiopharmaceuticals, UF₅, UO₂, etc.) in terms of total activity, number of packages, transport index and distance by transport mode.

(4) **Form D** — Summary of Movements of Substantial Quantities. This form included requests for similar data by radionuclide and chemical form.

In anticipation of the receipt of the data, Mr. D.R. Hopkins of the United States of America worked as a consultant to the Agency in 1982 and established the elements which would be required in a computer code and provided data consistency checks to be used prior to computer input. As the data arrived at the Agency, the checks performed by Agency staff indicated that most data were not internally consistent and further work was needed to clarify the problems. Mr. B.G. Pettersson of Sweden worked as a consultant in 1983 and identified a number of major problems with the data. These problems were of sufficient magnitude that it became clear computerization would not be possible as originally foreseen. It was decided that the inconsistencies with the data could only be resolved by using Agency staff and an additional consultant to assess the data on a case-by-case basis, identify inconsistencies and clarify their causes. The following summarizes the results of these efforts.
OVERVIEW OF DATA

A total of 49 countries responded to the request. Data from 35 countries have been used in the summary contained in this paper. Some countries provided data which could not be included in the database. For example, to prevent 'double-counting' of international shipments, only export shipment data were requested; however, Austria provided only air import shipment data, and Egypt provided only data on packages transiting the Suez Canal. In addition, 12 countries acknowledged the request but did not send data.

The data provided were for a one-year period, centred on 1981. Table I lists: the countries which provided data; whether the data were for domestic or international shipments, or both; the time period for which the data apply; and the Agency's assessment whether the data were complete.

For many of the countries, it was acknowledged or it became apparent that the data were not complete. For example: Italy and Spain provided no data on Exempt packages (now known as Excepted in the 1985 Edition of Safety Series No. 6); Australia, Finland, Hungary, Ireland, Italy and Poland provided no data on packages containing LLS or LSA (known as LSA or SCO in the 1985 Edition of Safety Series No. 6); Norway noted their data were not extrapolated to all shippers and were only about 70% accurate; Switzerland only provided data for fissile packages; the UK data are only for certain establishments; and the US data are for domestic shipments only, are known to underestimate the total packages shipped domestically by at least 500 000, and do not account for multi-modal transport of packages. It has since become apparent that one large commercial shipper was not included in the Netherlands data; inclusion of this shipper could have increased their reported package shipments by 500 000 to as much as 1 000 000.

Of the 32 countries which sent data of the Form A and/or Form B type:

(1) 21 provided data on international shipments, and 10 of these were judged to be incomplete; and
(2) 31 provided data on domestic shipments, and 12 of these were judged to be incomplete.

Data from Belgium, Brazil and Thailand were included in this review by adjusting the data to the Form A/Form B format.

COUNTING OF SHIPMENTS

As the data were reviewed, it became apparent that the statistics were developed differently by different countries.

Safety Series No. 6 defines "shipment" as "the specific movement of a consignment from origin to destination". A consignment can consist of one package or of many packages.
### TABLE I. COUNTRIES PROVIDING DATA USED IN THIS REPORT

**Key to table:** (N = no data, D = data, I = incomplete data provided)

<table>
<thead>
<tr>
<th>Country</th>
<th>International shipments</th>
<th>Domestic shipments</th>
<th>Time period of data reported</th>
</tr>
</thead>
<tbody>
<tr>
<td>Albania</td>
<td>N</td>
<td>D</td>
<td>81/82</td>
</tr>
<tr>
<td>Australia</td>
<td>I</td>
<td>I</td>
<td>80</td>
</tr>
<tr>
<td>Canada</td>
<td>D</td>
<td>D</td>
<td>81</td>
</tr>
<tr>
<td>Chile</td>
<td>D</td>
<td>D</td>
<td>79</td>
</tr>
<tr>
<td>El Salvador</td>
<td>N</td>
<td>D</td>
<td>80-82</td>
</tr>
<tr>
<td>Finland</td>
<td>I</td>
<td>I</td>
<td>80</td>
</tr>
<tr>
<td>France</td>
<td>D</td>
<td>D</td>
<td>80</td>
</tr>
<tr>
<td>Germany, Fed. Rep. of</td>
<td>D</td>
<td>D</td>
<td>80/81</td>
</tr>
<tr>
<td>Greece</td>
<td>N</td>
<td>D</td>
<td>-</td>
</tr>
<tr>
<td>Hungary</td>
<td>I</td>
<td>I</td>
<td>80</td>
</tr>
<tr>
<td>Ireland</td>
<td>I</td>
<td>I</td>
<td>80</td>
</tr>
<tr>
<td>Italy</td>
<td>I</td>
<td>I</td>
<td>82</td>
</tr>
<tr>
<td>Japan</td>
<td>D</td>
<td>D</td>
<td>80</td>
</tr>
<tr>
<td>Jordan</td>
<td>N</td>
<td>D</td>
<td>79/80</td>
</tr>
<tr>
<td>Republic of Korea</td>
<td>N</td>
<td>D</td>
<td>80</td>
</tr>
<tr>
<td>Luxembourg</td>
<td>D</td>
<td>D</td>
<td>80</td>
</tr>
<tr>
<td>Netherlands</td>
<td>N</td>
<td>I</td>
<td>-</td>
</tr>
<tr>
<td>New Zealand</td>
<td>D</td>
<td>D</td>
<td>79/80</td>
</tr>
<tr>
<td>Norway</td>
<td>I</td>
<td>I</td>
<td>81</td>
</tr>
<tr>
<td>Peru</td>
<td>N</td>
<td>D</td>
<td>-</td>
</tr>
<tr>
<td>Philippines</td>
<td>D</td>
<td>N</td>
<td>79</td>
</tr>
<tr>
<td>Poland</td>
<td>I</td>
<td>I</td>
<td>80/81</td>
</tr>
<tr>
<td>Senegal</td>
<td>N</td>
<td>D</td>
<td>-</td>
</tr>
<tr>
<td>Singapore</td>
<td>D</td>
<td>D</td>
<td>80</td>
</tr>
<tr>
<td>South Africa</td>
<td>D</td>
<td>D</td>
<td>80/81</td>
</tr>
<tr>
<td>Spain</td>
<td>I</td>
<td>I</td>
<td>80</td>
</tr>
<tr>
<td>Sweden</td>
<td>D</td>
<td>D</td>
<td>80</td>
</tr>
<tr>
<td>Switzerland</td>
<td>I</td>
<td>I</td>
<td>80</td>
</tr>
<tr>
<td>Syrian Arab Republic</td>
<td>N</td>
<td>D</td>
<td>82</td>
</tr>
<tr>
<td>Turkey</td>
<td>N</td>
<td>D</td>
<td>80</td>
</tr>
<tr>
<td>UK</td>
<td>I</td>
<td>I</td>
<td>80</td>
</tr>
<tr>
<td>USA</td>
<td>N</td>
<td>I</td>
<td>81/82</td>
</tr>
</tbody>
</table>

--- Other data ---

- **Austria**: Import by air only for 1979
- **Belgium**: Reported general data for 1977
- **Brazil**: Forms C & D only for 1979 (assumed to be domestic)
- **Egypt**: Packages transiting Suez Canal in 1979
- **Thailand**: Form C, only for 1980 (assumed to be domestic)
For accounting purposes with this study, two terms were defined as follows:

1. **Package Shipment** (PS) shall mean a single package transported by a single mode of transport from an origin to a destination.

2. **Package Combined Shipment** (PCS) shall mean a single package transported by one or more modes of transport, from its initial point of origin (where it is loaded with radioactive material) to its destination (where its contents are removed or otherwise used).

Thus, a PCS is composed of one or more PS, and therefore

\[ \text{Number of PS} \geq \text{Number of PCS} \]

As an example of the above, consider a single package which may go first by truck from a plant to an airport, next by aircraft from airport to airport, and finally by truck from airport to using facility. In this case, PS = 3 and PCS = 1.

The numbers of PS were requested in the Agency's survey in the Package Classification/Mode data sheets (top parts of Forms A and B), whereas the numbers of PCS were requested in the Statistical Data on Packages by Labelling Category (bottom parts of Forms A and B).

To allow for consistent reporting of the data in this report, the package shipments, PS, were calculated or estimated for each country where possible.

### TOTAL PACKAGE SHIPMENTS

Table II shows the total number of package shipments reported by each country. For the countries shown, when known inaccuracies are considered, the total package shipments per year exceed ten million.

The countries reporting data are 'ranked' from the largest shipper to the smallest shipper in Table III. It can be seen that, in order to gain a perspective on worldwide shipping, only a limited number of countries need be considered. In this case, the top 12 countries account for 99% of all package shipments reported.

While studying the data, and following discussions with experts in various Member States, it was concluded that the greatest uncertainty probably lies with Exempt packages. Table IV lists the 'Top 20 Countries' which are shippers of radioactive material packages other than Exempt packages. Here it can be seen, for example, that the ranking of the USA and Canada is reversed, which is more consistent with the relative populations of the two countries. It was therefore concluded that the Canadian data fairly represent the proportion of Exempt packages shipped, whereas the US data do not. The US data are further complicated in that the numbers are package combined shipments (PCS), not package shipments (PS).
### TABLE II. SUMMARY OF PACKAGE SHIPPMENTS (PS), BY COUNTRY, BASED ON DATA SUPPLIED BY MEMBER STATES TO THE IAEA

<table>
<thead>
<tr>
<th>Country</th>
<th>International package shipments</th>
<th>Domestic package shipments</th>
<th>Total package shipments</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Albania</td>
<td>—</td>
<td>133</td>
<td>133</td>
</tr>
<tr>
<td>Australia</td>
<td>555(^b)</td>
<td>56331(^b)</td>
<td>56886(^b)</td>
</tr>
<tr>
<td>Belgium</td>
<td>—</td>
<td>78422(^b)</td>
<td>78422(^b)</td>
</tr>
<tr>
<td>Brazil</td>
<td>—</td>
<td>(Not Specified) —</td>
<td>11144</td>
</tr>
<tr>
<td>Canada</td>
<td>433088</td>
<td>3947341</td>
<td>4380429</td>
</tr>
<tr>
<td>Chile</td>
<td>1</td>
<td>1463</td>
<td>1464</td>
</tr>
<tr>
<td>El Salvador</td>
<td>—</td>
<td>294</td>
<td>294</td>
</tr>
<tr>
<td>Finland</td>
<td>26(^b)</td>
<td>24502(^b)</td>
<td>24503(^b)</td>
</tr>
<tr>
<td>France</td>
<td>34802</td>
<td>84757</td>
<td>119559</td>
</tr>
<tr>
<td>Germany, Fed. Rep. of</td>
<td>46882</td>
<td>414532</td>
<td>461414</td>
</tr>
<tr>
<td>Greece</td>
<td>—</td>
<td>1338</td>
<td>1338</td>
</tr>
<tr>
<td>Hungary</td>
<td>2099(^b)</td>
<td>23824(^b)</td>
<td>25923(^b)</td>
</tr>
<tr>
<td>Ireland</td>
<td>60(^b)</td>
<td>15500(^b)</td>
<td>15560(^b)</td>
</tr>
<tr>
<td>Italy</td>
<td>8166(^b)</td>
<td>197505(^b)</td>
<td>205671(^b)</td>
</tr>
<tr>
<td>Japan</td>
<td>18459</td>
<td>705055</td>
<td>723514</td>
</tr>
<tr>
<td>Jordan</td>
<td>—</td>
<td>116</td>
<td>116</td>
</tr>
<tr>
<td>Rep. of Korea</td>
<td>—</td>
<td>5977</td>
<td>5977</td>
</tr>
<tr>
<td>Luxembourg</td>
<td>21</td>
<td>37</td>
<td>58</td>
</tr>
<tr>
<td>Netherlands</td>
<td>—</td>
<td>10359(^b)</td>
<td>10359(^b)</td>
</tr>
<tr>
<td>New Zealand</td>
<td>35</td>
<td>21</td>
<td>56</td>
</tr>
<tr>
<td>Norway</td>
<td>6706(^c)</td>
<td>10298(^c)</td>
<td>17004(^c)</td>
</tr>
<tr>
<td>Peru</td>
<td>—</td>
<td>19</td>
<td>19</td>
</tr>
<tr>
<td>Philippines</td>
<td>587</td>
<td>—</td>
<td>587</td>
</tr>
<tr>
<td>Poland</td>
<td>8258(^b)</td>
<td>25048(^b)</td>
<td>33306(^b)</td>
</tr>
<tr>
<td>Senegal</td>
<td>—</td>
<td>168</td>
<td>168</td>
</tr>
<tr>
<td>Singapore</td>
<td>166</td>
<td>836</td>
<td>1052</td>
</tr>
<tr>
<td>South Africa</td>
<td>80</td>
<td>12154</td>
<td>12234</td>
</tr>
<tr>
<td>Spain</td>
<td>107(^b)</td>
<td>48894(^b)</td>
<td>49001(^b)</td>
</tr>
<tr>
<td>Sweden</td>
<td>3760</td>
<td>119151</td>
<td>122911</td>
</tr>
<tr>
<td>Switzerland</td>
<td>774(^d)</td>
<td>284(^d)</td>
<td>1058(^d)</td>
</tr>
<tr>
<td>Syrian Arab Rep.</td>
<td>—</td>
<td>100</td>
<td>100</td>
</tr>
<tr>
<td>Thailand</td>
<td>—</td>
<td>(Not Specified) —</td>
<td>89</td>
</tr>
<tr>
<td>Turkey</td>
<td>—</td>
<td>621</td>
<td>621</td>
</tr>
<tr>
<td>UK</td>
<td>208168(^b)</td>
<td>190013(^b)</td>
<td>398181(^b)</td>
</tr>
<tr>
<td>USA</td>
<td>—</td>
<td>2819308(^e)</td>
<td>2819308(^e)</td>
</tr>
<tr>
<td><strong>TOTAL</strong></td>
<td></td>
<td></td>
<td>~9578459(^f)</td>
</tr>
</tbody>
</table>

\(^a\) Including Full Load  
\(^b\) Data incomplete  
\(^c\) Estimated to account for only ~70\% of shipments  
\(^d\) Fissile shipments only, data incomplete  
\(^e\) Known to be low by at least 500 000, data incomplete, data are package combined shipments, not package shipments  
\(^f\) When known inaccuracies in data from Norway and the USA are added in, the total exceeds 10 000 000 package shipments
TABLE III. SUMMARY OF THE ‘TOP 20 COUNTRIES’, THE LARGEST SHIPPERS OF RADIOACTIVE MATERIAL PACKAGES BASED UPON DATA SUPPLIED BY MEMBER STATES TO THE IAEA (footnotes are explained in Table IV)

<table>
<thead>
<tr>
<th>Country</th>
<th>Number of package shipments</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Canada</td>
<td>4 380 429</td>
</tr>
<tr>
<td>2. USA</td>
<td>2 819 308^a,b</td>
</tr>
<tr>
<td>3. Japan</td>
<td>723 514</td>
</tr>
<tr>
<td>5. UK</td>
<td>398 181^c</td>
</tr>
<tr>
<td>6. Italy</td>
<td>205 671^d</td>
</tr>
<tr>
<td>7. Sweden</td>
<td>122 911</td>
</tr>
<tr>
<td>8. France</td>
<td>119 559</td>
</tr>
<tr>
<td>9. Belgium</td>
<td>78 422^b,e</td>
</tr>
<tr>
<td>10. Australia</td>
<td>56 886^f</td>
</tr>
<tr>
<td>11. Spain</td>
<td>49 001^e</td>
</tr>
<tr>
<td>12. Poland</td>
<td>33 306^f</td>
</tr>
<tr>
<td>13. Hungary</td>
<td>25 923^f</td>
</tr>
<tr>
<td>14. Finland</td>
<td>24 503^f</td>
</tr>
<tr>
<td>15. Norway</td>
<td>17 004^g</td>
</tr>
<tr>
<td>16. Ireland</td>
<td>15 560^f</td>
</tr>
<tr>
<td>17. South Africa</td>
<td>12 234</td>
</tr>
<tr>
<td>18. Brazil</td>
<td>11 144</td>
</tr>
<tr>
<td>19. Netherlands</td>
<td>10 359^b,h</td>
</tr>
<tr>
<td>20. Rep. of Korea</td>
<td>5 977^b</td>
</tr>
<tr>
<td><strong>TOTAL</strong></td>
<td><strong>~ 10 \times 10^6</strong></td>
</tr>
</tbody>
</table>

SHIPMENTS BY PACKAGE TYPE

Table V presents a summary of the data provided from 16 of the ‘Top 20’ countries, showing the percentage, by package type, of each country’s total number of package shipments (data from Belgium, Brazil, Netherlands and the Republic of Korea were insufficient for this assessment). It can be seen that there are wide differences in the reported data for various package types. For example, Canada and Sweden reported more than 80% of their package shipments were Exempt packages, whereas Australia, Finland, France, Hungary, Italy, Norway, Poland, South Africa, Spain and the USA reported less than 30% were Exempt package shipments. For the latter countries it is postulated that many of the Exempt package shipments were not identified.
<table>
<thead>
<tr>
<th>Country</th>
<th>Number of package shipments</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. USA</td>
<td>2 402 429(^b)</td>
</tr>
<tr>
<td>2. Canada</td>
<td>612 632</td>
</tr>
<tr>
<td>3. Japan</td>
<td>353 054</td>
</tr>
<tr>
<td>4. UK</td>
<td>242 268(^c)</td>
</tr>
<tr>
<td>5. Italy</td>
<td>205 679(^f)</td>
</tr>
<tr>
<td>7. France</td>
<td>99 617</td>
</tr>
<tr>
<td>8. Belgium</td>
<td>78 422(^b)</td>
</tr>
<tr>
<td>9. Spain</td>
<td>49 425</td>
</tr>
<tr>
<td>10. Australia</td>
<td>45 208(^f)</td>
</tr>
<tr>
<td>11. Poland</td>
<td>32 208(^f)</td>
</tr>
<tr>
<td>12. Sweden</td>
<td>21 751</td>
</tr>
<tr>
<td>13. Hungary</td>
<td>18 423(^f)</td>
</tr>
<tr>
<td>14. Finland</td>
<td>17 753(^f)</td>
</tr>
<tr>
<td>15. Norway</td>
<td>16 604(^g)</td>
</tr>
<tr>
<td>16. South Africa</td>
<td>11 418</td>
</tr>
<tr>
<td>17. Brazil</td>
<td>11 011</td>
</tr>
<tr>
<td>18. Netherlands</td>
<td>10 359(^b,(^h)</td>
</tr>
<tr>
<td>19. Ireland</td>
<td>8 560(^f)</td>
</tr>
<tr>
<td>20. Rep. of Korea</td>
<td>5 220(^b)</td>
</tr>
<tr>
<td><strong>TOTAL</strong></td>
<td>(-4.4 \times 10^6)</td>
</tr>
</tbody>
</table>

Footnotes to Tables III and IV

- Data are known to be low by at least 500 000 Exempt packages
- Domestic shipments only
- Data are incomplete, data are from selected shippers only
- Data are incomplete, no data on Exempt packages or LSA/LLS shipments
- No data on Exempt packages
- No data on LSA/LLS shipments
- Estimated to be low by \(-30\%\)
- Data are suspected to be low by at least 500 000 package shipments, and data reported are PCS, not PS
### TABLE V. PACKAGE SHIPMENTS BY PACKAGE TYPE — IN PERCENTAGE

<table>
<thead>
<tr>
<th>Country</th>
<th>Total number of package shipments</th>
<th>Exempt</th>
<th>LSA/LLS</th>
<th>Type A</th>
<th>Type B(U)</th>
<th>Type B(M)</th>
<th>Other</th>
</tr>
</thead>
<tbody>
<tr>
<td>Australia</td>
<td>56,886</td>
<td>20.6</td>
<td>0</td>
<td>78.6</td>
<td>0</td>
<td>0</td>
<td>0.8</td>
</tr>
<tr>
<td>Canada</td>
<td>4,380,429</td>
<td>86.2</td>
<td>1.0</td>
<td>9.0</td>
<td>1.8</td>
<td>~0</td>
<td>2.0</td>
</tr>
<tr>
<td>Finland</td>
<td>24,503</td>
<td>27.5</td>
<td>0</td>
<td>72.4</td>
<td>0.1</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>France</td>
<td>119,559</td>
<td>16.6</td>
<td>0.3</td>
<td>49.9</td>
<td>0.8</td>
<td>0.2</td>
<td>32.2</td>
</tr>
<tr>
<td>Germany, Fed. Rep. of</td>
<td>461,414</td>
<td>66.3</td>
<td>0.04</td>
<td>31.7</td>
<td>1.2</td>
<td>0.01</td>
<td>0.75</td>
</tr>
<tr>
<td>Hungary</td>
<td>25,923</td>
<td>28.9</td>
<td>0</td>
<td>58.6</td>
<td>7.1</td>
<td>0</td>
<td>5.4</td>
</tr>
<tr>
<td>Ireland</td>
<td>15,560</td>
<td>45.0</td>
<td>0</td>
<td>6.5</td>
<td>48.5</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Italy</td>
<td>205,671</td>
<td>0</td>
<td>96.7</td>
<td>3.25</td>
<td>0.01</td>
<td>0.04</td>
<td></td>
</tr>
<tr>
<td>Japan</td>
<td>723,514</td>
<td>51.1</td>
<td>0.08</td>
<td>29.35</td>
<td>0.05</td>
<td>0.02</td>
<td>19.4</td>
</tr>
<tr>
<td>Norway</td>
<td>17,004</td>
<td>2.3</td>
<td>0.1</td>
<td>95.9</td>
<td>1.5</td>
<td>0</td>
<td>0.2</td>
</tr>
<tr>
<td>Poland</td>
<td>33,306</td>
<td>7.2</td>
<td>0</td>
<td>50.17</td>
<td>16.0</td>
<td>0.03</td>
<td>26.6</td>
</tr>
<tr>
<td>South Africa</td>
<td>12,234</td>
<td>6.7</td>
<td>1.0</td>
<td>76.7</td>
<td>15.5</td>
<td>0</td>
<td>0.1</td>
</tr>
<tr>
<td>Spain</td>
<td>49,001</td>
<td>0</td>
<td>69.7</td>
<td>27.6</td>
<td>2.6</td>
<td>0.1</td>
<td>0</td>
</tr>
<tr>
<td>Sweden</td>
<td>122,911</td>
<td>82.3</td>
<td>6.9</td>
<td>10.5</td>
<td>0.2</td>
<td>~0</td>
<td>0.1</td>
</tr>
<tr>
<td>UK</td>
<td>398,181</td>
<td>38.9</td>
<td>0.4</td>
<td>57.8</td>
<td>1.0</td>
<td>0.2</td>
<td>1.7</td>
</tr>
</tbody>
</table>

**USA**

| USA corrected for known missing Exempt packages | 2,819,308 | 14.8 | 17.0 | 63.5 | N/A | 3.4 | 1.3 |

**USA corrected for known missing Exempt packages**

| 3,319,308 | 27.6 | 14.5 | 53.9 | N/A | 2.9 | 1.1 |

**ASSESSMENT OF UTILITY OF THE DATA**

Many problems, a few of which have been outlined above, led to the conclusion that the data provided to the IAEA were not of sufficient quality to justify further analysis or effort. The data could not be used, except in a very general sense, to assess the adequacy of the Transport Regulations or to perform risk assessments.

The following demonstrates the extent to which the database appears to be incomplete. The total package shipments by sea which were reported were 16,500; yet Egypt reported a through-country transit in the Suez Canal of 52,000 packages per year. Thus it appears many shipments are not accounted for by the data provided to the Agency.
ESTIMATED WORLDWIDE SHIPPING VOLUME

In addition to the incomplete nature of the data provided by many countries, many of the Agency's Member States which would be significant shippers of radioactive material provided no data whatsoever. This includes Argentina, Austria, Bulgaria, China, Czechoslovakia, Denmark, the German Democratic Republic, India, Indonesia, Islamic Republic of Iran, Iraq, Israel, Mexico, Pakistan, Portugal, the USSR, Venezuela and Yugoslavia. Therefore, the database, corrected for known deficiencies for certain countries providing data, was examined from the point of view of gross domestic product (GDP) and population. Data were provided by countries representing approximately 26% of the world's population, 55% of the world's GDP and 80% of the world's installed nuclear power generating capacity.

If it is assumed that the GDP is the best indicator for extrapolating to obtain a measure of the volume of package shipments worldwide, then approximately 8 million package shipments of non-Exempt packages and approximately 10 million additional Exempt package shipments are made each year. Conversely, if it is assumed that the population is the best indicator for extrapolating, then approximately 17 million package shipments are made per year of non-Exempt packages and approximately 21 million additional Exempt package shipments are made each year.

Thus, considering all types of radioactive material packages, 18 to 38 million package shipments were made worldwide annually in the early 1980s, with 8 to 17 million of these of the non-Exempt types (Industrial Packages, Type A and Type B).

Irrespective of the basis of extrapolation, this is a much larger shipping volume than previously estimated.

CONCLUSIONS

The following conclusions can be drawn about the shipping data provided:

(1) The data are incomplete; many countries provided partial data and many countries provided no data.

(2) The data are inaccurate; many inconsistencies were identified in data sheets and many contributors acknowledged that data were extrapolations from random or partial surveys.

(3) Very few of the data sets are complete.

(4) However, the data do provide a basis for approximating the magnitude of the transport of radioactive material worldwide; and it can be concluded that from 18 to 38 million package shipments were made each year in the early 1980s.

(5) Finally, if similar data collection activities are attempted in the future, it is recommended that: (a) the data collection sheets should be much simpler; (b) the data collection activity should be less ambitious; (c) data need only be collected from approximately 20 countries (the largest shippers) to obtain a good estimate; and (d) clear detailed instructions, with examples, should be provided.
SURVEY OF RADIOACTIVE MATERIAL TRANSPORT IN CHINA

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  Beijing
  China

Abstract

SURVEY OF RADIOACTIVE MATERIAL TRANSPORT IN CHINA.

The paper gives an outline of the transport of radioactive material in China. At present, the annual freight volume of packages of radioactive material is some 100 000 items. The total activity is about 1.8 PBq. The radioisotopes are mainly $^{131}$I, $^{32}$P and $^{198}$Au. The available results show that individual doses to transport workers are rather low. The annual dose equivalent is less than 5 mSv/a. Much attention has been paid to the safe transport of the radioactive material. Hence, no accident with serious radiation effects has ever happened. A working group is preparing Chinese regulations for the safe transport of radioactive material on the basis of IAEA Safety Series No. 6 (1985).

1. CIRCULATION OF PACKAGES

1.1. Number of packages

The transport of radioactive material began in the early 1960s. At that time, the number of packages of such material transported each year was quite limited. There were only some 20 types of manufactured radioisotope articles. With the rapid development in the production and application of radioisotopes in China, the freight volume of packages of radioactive material has greatly increased. For example, packages of radioisotopes and their manufactured articles produced and distributed by the Institute of Atomic Energy of Beijing increased
The number of items and total activity of radioisotopes and related manufactured articles produced and distributed by the Institute of Atomic Energy, Beijing, increased from 200 in 1959 to 81,300 in 1984 (see Fig. 1). The variety of manufactured radioisotope articles has now increased to more than 300. For example, the types of radioisotope produced by the Institute of Atomic Energy increased from 5 in 1959 to 214 in 1984 (Fig. 2). The annual freight volume of different types of packages by all modes of transport is about 100,000 items, of which packages for industry, agriculture, medicine and scientific research account for more than 98%. The total activity is about 1.8 PBq (4.86 × 10⁴ Ci). The radioisotopes involved are mainly $^{131}$I, $^{32}$P and $^{198}$Au.
TABLE I. PROPORTION OF PACKAGES IN TRANSPORT CATEGORIES (1984)

<table>
<thead>
<tr>
<th>Transport category</th>
<th>I</th>
<th>II</th>
<th>III</th>
<th>IV</th>
<th>Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of packages (10^3)</td>
<td>94.9</td>
<td>2.6</td>
<td>0.2</td>
<td>6.5</td>
<td>104.2</td>
</tr>
<tr>
<td>Proportion (%)</td>
<td>91.1</td>
<td>2.5</td>
<td>0.2</td>
<td>6.2</td>
<td>100.0</td>
</tr>
</tbody>
</table>

TABLE II. PROPORTION OF DIFFERENT MODES OF TRANSPORT

<table>
<thead>
<tr>
<th>Transport modes</th>
<th>Rail</th>
<th>Post</th>
<th>Air</th>
<th>Road and sea</th>
<th>Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of packages (10^3)</td>
<td>48.5</td>
<td>46.8</td>
<td>3.2</td>
<td>5.7</td>
<td>104.2</td>
</tr>
<tr>
<td>Proportion (%)</td>
<td>46.8</td>
<td>44.9</td>
<td>3.1</td>
<td>5.5</td>
<td>100.0</td>
</tr>
</tbody>
</table>

1.2. Transport categories

In the past 10 years, the freight volume of radioisotope packages (of which about 60% is used for nuclear medicine) has greatly increased. The proportion of packages in the various transport categories in China in 1984 is listed in Table I. It can be seen from Table I that 90% of the packages belong to category I-WHITE (the radiation level at any point on the external surface does not exceed 0.01 mSv/h). Category III is very rare. For the sake of safety, packages with higher external radiation levels or radioactivity are always transported under exclusive use.

1.3. Transport modes

The proportion of the different modes is given in Table II. It can be seen that transport by rail is the main mode in China at the moment. Of course, most postal packages travel to their destination by rail as well. Hence, packages transported by rail actually amount to more than 90%. Those packages which need auxiliary transport by road when transported by rail, air, vessels and post are not included in the number transported by road in Table II.
2. PACKAGING

2.1. Packaging requirements for all packages

In order to achieve safe transport, different package requirements are prescribed according to the contents (five categories). For instance, for radioactive sources of $^{14}$C, $^{55}$Fe, $^{60}$Co, $^{226}$Ra, etc., it is required to have an inner container made of glass, plastic or metal. This container is tightly sealed (for gaseous radioisotopes, it must be made of metal sealed by welding). Then the inner container is put into an outer container made of metal or plastic. The gap between the two containers is filled with soft or absorbent material. An outer packaging such as a metal box, plastic bag (drum) or cardboard box, is used to wrap the whole arrangement after the lid of the outer container is screwed on.

2.2. Variety of packagings

There are some 20 kinds of packagings used in China at the moment, to transport all kinds of radioisotope products. A shipping flask for irradiated fissile material is under design.

3. RADIATION PROTECTION CONDITIONS AT TRANSPORT STATIONS

The radiation protection conditions for transport stations (including railway stations, airline terminals and ports) require warehouses appropriate for temporary storage of radioactive packages, transport workers specially trained in radiation protection, and radiation protection monitoring. According to some incomplete statistics, about 30% of transport stations have set up special warehouses or goods shelves. However, they are not well equipped, e.g. 'Ionizing Radiation' labels are not put up at some stations. Only a few stations have transport workers who have received special training. The local sanitation and antiepidemic stations are in charge of radiation protection monitoring, but a complete monitoring system has not been established.

4. EXPOSURE DOSE TO TRANSPORT WORKERS

4.1. Exclusive use

The loading and unloading and escort activities are generally undertaken by the consignor, and individual monitoring is provided. Table III shows the annual cumulative dose to radioisotope transport workers from the Institute of Atomic Energy in 1984. It can be said that the individual dose to transport workers
TABLE III. DOSES TO TRANSPORT WORKERS (INSTITUTE OF ATOMIC ENERGY, 1984)

<table>
<thead>
<tr>
<th>Type of work</th>
<th>People monitored</th>
<th>Collective dose equivalent (man·Sv)</th>
<th>Average dose equivalent (mSv)</th>
<th>Maximum dose equivalent (mSv)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Porter</td>
<td>7</td>
<td>$1.37 \times 10^{-2}$</td>
<td>2</td>
<td>5.2</td>
</tr>
<tr>
<td>Driver(^a)</td>
<td>11</td>
<td>$1.66 \times 10^{-2}$</td>
<td>1.5</td>
<td>4.8</td>
</tr>
<tr>
<td>Fork lift driver</td>
<td>1</td>
<td>$5.0 \times 10^{-4}$</td>
<td>0.5</td>
<td></td>
</tr>
<tr>
<td>Total</td>
<td>19</td>
<td>$3.08 \times 10^{-4}$</td>
<td>1.62</td>
<td>5.2</td>
</tr>
</tbody>
</table>

\(^a\) The drivers often carried packages.

is less than 5 mSv per year. The data from the Public Health Department of Jilin Province show that individual doses are: drivers 1 mSv/a, porters 2 mSv/a, escorts 1 mSv/a. These data indicate that the individual doses to workers involved in the transport of radioactive material in China are rather low.

4.2. General transportation

Part-time workers involved in the transport of radioactive packages have not been provided with individual monitoring because the packages transported by these workers are mostly category I-WHITE with a fairly low level of external radiation exposure; the operational frequency is low and working hours short. Hence, the annual dose equivalent to them may be considered as less than 5 mSv/a.

5. TRANSPORT REGULATIONS

5.1. Existing regulations of some departments

For transport by rail, draft regulations for provisional enforcement were completed in 1961. They were revised in 1972 as “Regulations for the Transport of Dangerous Goods” (trial use), in which radioactive material is classified under dangerous goods in 'Category 10'. The regulations prescribe exemption limits and content limits for packages (Table IV), packaging categories and requirements in transit, etc. In 1979, the General Civil Aviation Administration of China issued “Regulations for Chemicals” (trial use), in which Appendix V is “Regulations for the Transport of Radioisotopes”. Besides transport categories and packaging
and shipment requirements, these provide limits for the loading locations and number of items in different types of aircraft. In 1983, the Ministries of Public Health and Posts and Telecommunications jointly issued "Regulations for the Postal Transport of Radioisotopes", which prescribes that packages containing the radionuclide $^{125}$I with an activity of less than 0.2 mCi (surface dose equivalent rate is less than 1.3 Sv/h) may be consigned for shipment at designated post offices. For transport by road, sea and inland waterway, there are not yet any special regulations.

5.2. Problems in existing regulations

The main problems in existing regulations are as follows:

(1) Some limits are different from those in IAEA Safety Series No. 6; these include exemption limits, transport categories, limits for the radioactive contamination on package surfaces and activity limits for a single package, etc. These differences are not conducive to international exchange.
(2) Requirements for the design, manufacture and testing of packaging are not clearly specified.

(3) Administrative measures are not very rigid. For example, no clear requirements for the design of packagings and for transport modes and routes have been established. Also, approval procedures and the organization of radiation protection measures have not been clearly defined.

(4) The contents of the regulations are not perfect. For instance, radiation protection conditions for transport stations, radiation protection monitoring in transit, annual radiation dose limits for transport workers, quality assurance, and optimization procedures in radiation protection during the transportation of radioactive material have not been stipulated.

6. ACCIDENTS

6.1. Types of accident

Because considerable attention has been paid to the safe transport of radioactive material, few accidents occurred within the past 20 years. The data from Beijing, Shanghai, Liaoning province and the Institute of Atomic Energy show that 17 transport accidents with some impact occurred during transport by rail and road. In the transportation by sea, inland waterways, air and post, accidents have never happened. Statistical data from Beijing, Liaoning province and the Institute of Atomic Energy show that radiation accidents in transit (including internal transport within an organization) account for some 8% of all radiation accidents. The proportion of accident types is listed in Table V. It can be seen that most transport accidents are those in which conveyance and working area were contaminated by radioactive material; these amount to 52.9% of the total accidents. Secondly, 'source missing' accidents constitute 29.4%. The contamination areas are mainly the warehouses at stations and the means of transport. However, there was one accident in which the traffic route was contaminated. That was when luminescent powder containing mesothorium (radium-228) was transported by truck; more than 100 km of road were contaminated because the glass ampoule containing luminescent powder was broken and not found in time. Fortunately, contamination was not serious. The total source activity for all 'source missing' accidents was less than $10^8$ Bq.

6.2. Consequences

No accidents have resulted in a serious radiation hazard or environmental contamination because transport accidents involving strong radiation sources have never happened. However, accidents have had a considerable influence on public psychology.
Table V. Proportion of accident types for 17 accidents

<table>
<thead>
<tr>
<th>Accident type</th>
<th>External exposure</th>
<th>Air pollution</th>
<th>Surface contamination</th>
<th>Source missing</th>
<th>Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number</td>
<td>2</td>
<td>1</td>
<td>9</td>
<td>5</td>
<td>17</td>
</tr>
<tr>
<td>Proportion (%)</td>
<td>11.8</td>
<td>5.9</td>
<td>52.9</td>
<td>29.4</td>
<td>100.0</td>
</tr>
</tbody>
</table>

Table VI. Analysis of causes of 17 accidents

<table>
<thead>
<tr>
<th>Cause of accident</th>
<th>Violations</th>
<th>Operating faults</th>
<th>Equipment defects</th>
<th>Natural factors</th>
<th>Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number</td>
<td>5</td>
<td>1</td>
<td>9</td>
<td>2</td>
<td>17</td>
</tr>
<tr>
<td>Proportion (%)</td>
<td>29.4</td>
<td>5.9</td>
<td>52.9</td>
<td>11.8</td>
<td>100.0</td>
</tr>
</tbody>
</table>

6.3. Analysis of accident causes

The main causes of the 17 accidents have been analysed and are listed in Table VI. It can be seen that accidents resulting from equipment defects are the most frequent; this kind of accident accounts for 52.9% of the total. The occurrence of accidents in which a radioactive source fell from a lead container was also high, 5 out of 17 accidents belonging to this category. It seems that the main lesson has not been learned and special precautions have not been adopted.

7. PERSPECTIVES

In order to increase safety in the transit of radioactive material to meet the needs of increasing numbers of radioactive packages and to facilitate international exchange and international through transport, a working group has been instituted. It is investigating national and international conditions pertaining to the transport of radioactive material and developing Chinese regulations for the safe transport of radioactive material on the basis of IAEA Safety Series No. 6 (1985). Subsequently, different departments will formulate appropriate guidance and regulatory measures.

Research will be carried out on raising the safety level of transport (e.g. developing standard containers, optimizing radiation protection in transit, etc.), establishing monitoring systems for different modes of transport, carrying out scientific administration and strengthening the education and training of transport workers, etc.
REGULATORY COMPLIANCE ASSURANCE: TWENTY-FIVE YEARS OF EXPERIENCE IN SHIPPING LARGE QUANTITY, LOW ACTIVITY RAM PRODUCTS

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Abstract

REGULATORY COMPLIANCE ASSURANCE: TWENTY-FIVE YEARS OF EXPERIENCE IN SHIPPING LARGE QUANTITY, LOW ACTIVITY RAM PRODUCTS.

A brief outline of one company's interpretation of the IAEA Safety Series No. 6 Regulations for Type A and excepted quantity package designs is outlined. The design route for new product packagings for these ranges and the controls introduced to ensure safety, consistency and regulatory compliance throughout the packaging range are explained. An outline of past and present regulatory problems that have been encountered is given, to demonstrate why greater conformity within the Regulations and greater understanding of the differing types of radioactive materials movements should be pursued to simplify their transport throughout the world. Statistical data on safety are given and it is confirmed that the standards of packaging for Type A and excepted quantities are more than adequate for normal conditions of transport. Regulatory bodies are encouraged to increase their existing efforts to unify their requirements and to educate the transport industry in the high level of safety inherent in the movement of radioactive consignments.

INTRODUCTION

This paper describes the experience of a UK company which has since 1948 been developing and producing medical, research and industrial products for life sciences and health care and currently exports to a large number of countries. The traffic in radioactive materials from the company has grown steadily over the years. Though estimated annual traffic for 1986-87 is around half a million packages and 1.5 billion (1.5 x 10^9) package kilometres, the number of incidents, all of a minor nature and involving only damage to packages with no release of material, has been maintained at about 5 per year. In the last 15 years only 2 incidents have involved the release of (small) quantities of radioactive material. For many years the design procedures used have been linked with the regulatory requirements.
All major modes of transport are used and often a consignment will involve two or more modes. Such movements are made more complex if not only the international and national requirements are at variance but also if the modal authorities introduce different interpretations of the same requirement.

Initial shipments from this company involved refined radium for luminous compounds and radium sources for clinical use. These were originally shipped only in the UK, but their movement internationally rapidly developed. In time, many other radioactive materials have been added (including labelled compounds for research and radiopharmaceuticals). To meet the needs of the various users of this wide array of materials, having a wide variation of half-lives and covering many countries throughout the world, the transport system used must be reliable and flexible. For example, for radiopharmaceuticals with relatively short half-lives, a combination of road and air transport is commonly used. Local distributors were added in high-use areas and direct deliveries by road from the UK to the European area are now common. Also, to expedite shipments, centralization of regional consignments for redistribution to local distributors is now quite common.

With a wide range of materials to be shipped (in excess of 18000) and a considerable number of packaging configurations (approaching 1000) it is necessary to ensure that each material be allocated to the correct packaging. Therefore, an approval system covering both packaging design and allocation of contents is used.

Because they are used in the health industry, the half-lives of the products are very short. It is therefore essential that these can be shipped quickly and efficiently, without the intervention of regulatory authorities, either governmental or commercial. In some cases even a few hours delay can render the product unfit for its purpose. Direct shipments by air to remote locations will always be required for certain radioactive products while others can use road, rail or sea alternatives. For this reason it is important that commercial organisations continue to maintain a high profile in the international world of radioactive transport and regulatory controls.

DESIGN CONTROLS

As in all design operations, control of standards is essential. A series of procedures has been developed to serve as guidelines to the package designer. The procedures
are structured to provide a clearly identified route through both the regulatory and company safety standards to ensure a consistency of quality throughout the packaging range.

The system is centred on a work flow procedure which ensures that the basic classification questions are asked to define the type of package to be designed. The design route begins with the production of a design effort request (DER) form which clearly specifies the product and associated hazards as well as providing suggestions of existing package configurations that might be employed. The appropriate procedures dictate the steps to be taken to achieve a satisfactory design. In the case of Type A or excepted packages, these procedures take into account, among others, activity and shielding. One of the problems with operating a large international company is that packaging design facilities cannot always be centralized. Hence tests may need to be carried out at subsidiaries and the results forwarded for approval. To ensure consistency of standards throughout the group, we have developed a series of test procedures, outlining precisely how the test is to be carried out, the number and type of samples to be used and, most importantly, the criteria against which the results are to be judged in order to satisfy the regulatory and company safety standards. All tests are witnessed by two people, each of whom signs the report which is included in the approval submission.

All systems need some method of cross checking and in the present case, this is done through a so-called Stage 2 form, which is raised on receipt of the initiating DER form. The Stage 2 form provides a checklist of items to be considered and gives the approving authority an indication of how each requirement has been met.

APPROVAL SUBMISSION

Once the necessary tests have been completed the information is collated into an approval submission. It is often possible to use previous test results and an assessment rationale is included in the submission to guide the approving officer through it. The dossier also includes the DER and Stage 2 forms, engineering drawings and specific radiological requirements if applicable. The submission is reviewed firstly by the Transport Container Officer, whose main responsibility is for the engineering standards, the inherent safety of the package and compliance with regulatory requirements. It is then reviewed by the Safety Controller, whose main concern is for safety who is independent of commercial considerations. To record the acceptance of a package configuration into the company's range, a certificate
is raised and signed by both of them. The certificate outlines the package specification, its construction and refers to the applicable engineering drawings and purchase specifications. It should be made clear at this point that there is no legal requirement for such a certificate and the present paper should be interpreted as a call for such a requirement to be introduced. However, this method has been found suitable for maintaining records and providing evidence of the design and test work in the event of any queries. With the number of new products and packages being generated, some of which may only have a short life cycle, it is appropriate to limit the validity of a certificate to a determined period, currently five years. This ensures a continual review of designs. Finally, to define the package, a unique design number is allocated. To assist in the packing operation, the package is also given a unique alphanumeric code. The information is then logged into the packaging and order processing database of the computer system which allocates products to packages as orders are received.

PRODUCT ALLOCATION (EPM3)

A package that is suitable for a certain quantity of one nuclide may not be suitable for another. Checks of activity levels, surface dose rate and transport index must be carried out before a product can be approved to be shipped in a given package. This product allocation is controlled by a system of engineering package memoranda, known as EPM3's.

For every new product an EPM3 is produced and the necessary checks carried out to ensure the most appropriate package is used. The EPM3 is checked by the company health physics organisation and finally signed by the Transport Container Officer. The information is then entered into the computer system to enable the product to be allocated to a package. Without an EPM3, a product cannot be shipped. Thus an independent assessment is available for all packages and their products.

QUALITY ASSURANCE AND CONTROLS

It is all very well developing procedural and computerized systems but no system is infallible. To ensure high quality products and associated hardware, documented quality assurance inspections and audits are carried out throughout the packaging and transport operations from incoming raw materials through the entire packaging process to final despatch.
LOGISTICS

It is preferable that radioactive materials for in vivo medical diagnoses have short half-lives to ensure that after use they quickly reduce to safe levels within the body. Thus, a rapid transit time to the customer is desirable. The majority of products involved here contain only small quantities of radioactivity and fall mostly in the excepted or lower Type A ranges. Unfortunately, the transport world is relatively ignorant of such distinctions and treats all radioactive shipments with the same suspicion. The range of materials includes labelled products for biomedical research, technetium generators and assay kits for diagnostic use and sources for use in radiotherapy or industrial processes. Radioactive consignments could therefore include various types of packaging in any number of different combinations.

To give some idea of the problems encountered on a normal shipment, we may consider a delivery to a customer in Australia. This is a reasonably complex shipping route but the basic elements are similar for the majority of movements. On leaving the warehouse, the package is transported by road (under UK regulations) to the freight forwarder's depot where it is checked and taken by road (under UK regulations) to Heathrow airport. It is then checked in for acceptance (under the internationally accepted regulations, of ICAO, the International Civil Aviation Organization, and the airline industry requirements of IATA, the International Air Transport Association). On arrival in Australia, it is taken, after customs clearance (under Australian regulations) to the subsidiary by road (under Australian regulations) and from there either by road (under the relevant state authority regulations) to its final destination or, again, by air (under both the relevant state regulations and/or the international regulations of ICAO and the requirements of IATA, both of which operate for internal domestic movements and both of which may be subject to different local derogations). Further complications may arise if part of the shipment needs to be transported by rail as several state variations may need to be considered.

Another example would be transport to customers via subsidiaries in Europe. The packages are loaded on to a lorry and driven by road (under UK regulations) to Dover, where they are accepted (not under IMO regulations as one might expect but under a special provision of the ADR/RID regulations). On arrival in France, the vehicle is then driven across the frontiers of Belgium, the Netherlands and the Federal Republic of Germany before reaching its final
destination. In each country, part of the load is removed and delivered to the local subsidiary (under the appropriate national regulations). Onward movement is then normally by road or rail (under the appropriate national regulations) to the customer in question.

PROBLEMS

It is now possible to understand the complexities involved in organizing a simple movement from the laboratories to the end user. What does it mean in practice? Here are some examples of problems that have been experienced in the past and that still exist today.

1. HAZARD WARNING LABELS. The minimum package dimension in the IAEA Regulations is 100 mm. The size of the side of the hazard warning label is also 100 mm. To fit on the package the label must be affixed with the sides parallel to the edges of the package or with the corners of the label folded around the package. According to ICAO, this second option is not permissible as "labels may not be folded". However, the IATA regulations clearly state that the labels must be fixed at 45 degrees, which means they must be folded if they are to fit a minimum dimension package. Unfortunately, it does not end there as the situation is further confused by IATA requiring that labels must not be folded.

Requests to IATA have subsequently resolved this anomaly.

2. MINIMUM DIMENSIONS. As stated above the minimum dimension for a package is 100 mm Except, of course, on a particular European railway system where the minimum standard is 150 mm! An insignificant difference perhaps, but the ramifications may be quite complex. Some products are prepacked in anticipation of incoming orders and subsequently labelled. Others are packed when the computer generated labels, based on the order processing system, are received. The packaging instructions are taken from the approved EPM3 listing, which is also held on the computer and which only recognizes product and activity, not destination. Thus, until recently, all products were packed and controlled irrespective of the customer's location. An insignificant change in one country's regulations has made it necessary to amend the entire computer system to recognize individual consignments (i.e. not overpacked or consolidated) which will at some time during their journey use the rail system.

3. REGULATION PRESENTATION. It is useful to be able to locate things quickly and efficiently and this is helped if layouts and presentation styles follow a similar format.
It is therefore disturbing to see how the various modal organizations have dealt with the incorporation of the IAEA Regulations. It is difficult enough confirming a complex regulatory issue within a single mode if one is not entirely confident with the regulations but when there are two sets for intermodal journeys, each laid out in its own way for its own way for its own historical reasons, the difficulties increase considerably. It is frustrating to receive annual copies of the ICAO and IATA regulations, each slightly different from the previous year and each approaching its interpretation of the radioactivity regulations in a slightly different way. The modal regulatory bodies are of course aware of the problem and some excellent work has been carried out by the various working parties that were set up by ICAO, ADR/RID and IMO in response to the publication of the 1985 Edition of Safety Series No. 6. However, more work could perhaps be done within the IAEA itself to reschedule the Regulations to conform more closely to the modal layouts.

4. TRADE ASSOCIATION INFLUENCE. It is only right and proper that trade organizations exist and the work carried out by IATA is excellent. However, with the appearance of the internationally recognized ICAO regulations, the presence of two sets of regulations can only lead to misunderstandings and problems for commercial organizations transporting dangerous goods. Hazard labelling is a clear example of the problems that can arise. What stand should be taken when, for example, the legal requirement is that called for in ICAO but the airline operates with the IATA manual?

5. IMPLEMENTATION DATES. During the next few years there will be changes throughout the world as each country moves towards accepting the new requirements of Safety Series No. 6. There will also be a similar operation in the modal area as each authority sets out its rules for accepting both the new and the old standards during the period of transition. Following the introduction of the 1973 Regulations, major problems were encountered with differing package requirements for certain activity levels of nuclides and one had to be constantly aware of each country's and each mode's demands. Whilst the majority of changes in the 1985 regulations are not as fundamental as those set out in 1973 and whilst moves have been made by the IAEA to involve actively all modes in determining a suitable implementation date, problems are still foreseen as a result of the timetabling of meetings of the various approving bodies. The change most likely to cause concern for this organisation is undoubtedly the introduction of the SI system of units. Not only has the rounding process for curies apparently introduced two levels for the A1 and A2 values which are likely to be reproduced directly into all the
modal regulations, but the acceptance of these units internationally is likely not only to vary from one country to another but also to depend on the rate of acceptance of the IAEA Regulations within those countries. However, efforts have been made by the IAEA to unify implementation and it can only be hoped that these efforts will be continued.

6. NATIONAL DIFFERENCES. Rather like entropy, that ever increasing quantity, the influence of national bodies on international regulations is to make them ever more restrictive or complex. With the Australian shipment described above, this influence is extended, in a way similar to that in the United States of America, to the various states. It is difficult enough to be subjected to the vagaries of the modal derogations but when Member States of the IAEA cannot maintain the fundamental philosophies of the regulations within their own national boundaries it is remarkable that one can manage to ship anything anywhere. Why, for example, do some countries insist on radioactive labels on the outside of excepted packages? Or others restrict passenger aircraft to carrying medical RAM only? Why do some have different A1 and A2 values for certain nuclides? A recent example of the problems that arise was a consignment to Brazzaville from Heathrow which was delayed because the US air cargo company was trying to apply the US regulations in the UK!

EFFECTS

As each new regulatory change is introduced it is monitored, assessed and its impact determined. A decision has to be taken on how best to achieve compliance and invariably this involves changes to the order processing computer network. The system database has been historically structured around the IAEA requirements and package allocation is controlled by several different factors including:

(a) A1 and A2 values
(b) Transport index
(c) Package type
(d) Surface dose rate

In addition, categorization and labelling (both hazard and destination) are controlled. Transport Index accumulation on aircraft is automatically monitored and local regulatory differences are built in where possible. The system is understandably complex and the simplest changes create problems.

PERFORMANCE

The most telling test of quality is the effective track record achieved, so let us just recap. The total
number of packages shipped by this organisation per year has increased steadily from 100,000 in 1970, to the current total of over 500,000 for the financial year. During the same period, the distribution network has increased. Therefore, the rise in package kilometres has increased at a significantly higher rate than package numbers. This has the effect of increasing the probability of incidents occurring. Yet, despite this increasing probability, the actual number has been maintained at about 5 minor incidents per year in recent years. This effectively means that the safety record for packages of this type when expressed as a percentage of the package kilometers covered has continually improved in recent years.

CONCLUSIONS

The problems involved in the international transport of commercial radioactive products would be greatly reduced if the quality of inspection and policing were more consistent throughout the world and apparent dual standards were avoided. However, the main conclusions from the present paper are as follows:

(a) The evidence clearly shows that the standards of packaging for Type A and excepted packages are more than adequate for normal conditions of transport.
(b) The IAEA can and should try to influence modal authorities wherever possible to ensure that other regulations are both consistent and interpretable.
(c) The modal authorities themselves can and should create closer links to help reduce intermodal problems in international movements.
(d) The national authorities should involve themselves in and carefully consider the commercial implications of introducing variations to internationally accepted regulations.
(e) A greater awareness of the high level of safety in the movement of radioactive consignments should be induced into, at the very least, the transport industry in an attempt to educate workers and prevent problems caused by a basic mistrust of these materials.
(f) Such an awareness should be actively promoted not only by the IAEA but also by commercial organizations and the modal authorities who have a responsibility to their industry to help to ensure the continuing smooth and safe distribution of radioactive materials worldwide.
В ЧССР имеется три центра распределения, два из которых являются центрами промышленных организаций, изготавливающих узкий ассортимент изделий с ограниченным количеством потребителей, для которых предназначен данный ассортимент продукции.

Третий центр является изготовителем целого ряда изделий, например, закрытых радионуклидных источников, меченных соединений $^{14}$C и $^{3}$H, эталонов радионуклидов и избранных типов РИА и РЭА наборов. Этот центр исполняет одновременную функцию поставщика импортных радиоактивных материалов для всех рабочих мест в ЧССР.

В стране находится 730 постоянных мест, где имеются радиоактивные материалы, поставка которых осуществляется следующим образом:

1. импорт (прибл. 1050 видов препаратов и источников излучения из 20 стран мира):  

<table>
<thead>
<tr>
<th>Год</th>
<th>Открытые источники радиоактивного излучения (ГБк)</th>
<th>Радиоактивные источники излучения (ГБк)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1982</td>
<td>47 500</td>
<td>6 500 000</td>
</tr>
<tr>
<td>1983</td>
<td>61 500</td>
<td>3 500 000</td>
</tr>
<tr>
<td>1984</td>
<td>66 500</td>
<td>5 710 000</td>
</tr>
<tr>
<td>1985</td>
<td>72 110</td>
<td>4 305 500</td>
</tr>
</tbody>
</table>

* В сборнике опубликована только развернутая аннотация доклада.
2. Отечественная продукция (например, в 1985 году):

радиофармацевтические препараты
$^{131}$I, $^{99m}$Te, $^{125}$I, $^{67}$Ga
диагностические препараты
$^{125}$I, $^3$H (10 видов)
меченые соединения ($^{14}$C, $^3$H) -
прибл. 250 соединений
наборы RIA•REA ($^{125}$I, $^3$H) -
7 видов
светосоставы ($^{147}$Pm - 18 - 166 ГБк/кг)
эталоны разных нуклидов
источники излучения
($^{241}$Am, $^{226}$Ra, $^{90}$Sr и т.д.) -
12 видов

8500 шт.
280 ГБк
1200 шт.
171 кг
2000 шт.
64 750 шт.

Способы поставок изотопной продукции на отдельные рабочие места — прибл. 41 000 поставок/год:

<table>
<thead>
<tr>
<th>Способ</th>
<th>%</th>
<th>(детали)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Скорые поезда</td>
<td>25%</td>
<td>(23 станции назначения)</td>
</tr>
<tr>
<td>Самолетом</td>
<td>20%</td>
<td>(4 аэропорта назначения)</td>
</tr>
<tr>
<td>По почте</td>
<td>2%</td>
<td>(только некоторые вещества)</td>
</tr>
<tr>
<td>Прямая поставка</td>
<td>15%</td>
<td>(только в месте дистрибуционного центра или производственного завода — 75 000 км)</td>
</tr>
<tr>
<td>дистрибутора</td>
<td></td>
<td>(из дистрибуционного центра или у изготовителя)</td>
</tr>
<tr>
<td>Собственными средствами получателя</td>
<td>38%</td>
<td></td>
</tr>
</tbody>
</table>

Поставка производится в транспортных упаковочных комплектах, которые своей конструкцией гарантируют безопасное обращение и транспортировку радиоактивных материалов.

Конструкция транспортных упаковочных комплектов отвечает стандартам безопасной транспортировки радиоактивных материалов. Она подвергалась проверке контрольного органа. За последние три года в ЧССР не произошло ни одной аварии, связанной с последующим радиоактивным загрязнением. Зарегистрированы только три чрезвычайных случая при разгрузке авиапоставок, при которых произошло механическое повреждение упаковок без последствий для их содержимого. При транспортировке по железной дороге были утеряны две упаковки, содержащие медицинские препараты, которые были вскоре найдены соответствующими органами.
The spent fuel from Loviisa's two WWER-440 units is returned to the Soviet fuel supplier, V/O Atomenergoexport (AEE), under separate contracts. The return is carried out using Soviet transport casks and container wagons designed for WWER-440 spent fuel [1, 2]. In addition, road transport trailers owned by Imatran Voima Oy (IVO) are used to transfer the casks from Loviisa railway station to the power plant and back. So far, three shipments from Loviisa-1 have successfully been carried out, in 1981, 1982 [3] and 1985.

The main data of the type TK-6 cask are shown in Fig. 1. The capacity of one cask is 30 assemblies, i.e. about 3.6 t of uranium. The loaded cask weighs about 90 t. The body of the cask is made of forged steel and equipped with cooling fins. The liner as well as the lid are made of stainless steel. The competent Soviet authority (GKAЕ) has classified the cask with gas coolant and fuel with burnup below 24 MW·d/kg U as a Type B(U) package and with water coolant and fuel with burnup between 24 and 40 MW·d/kg U as a Type B(M) package.

For approval for spent fuel transport in Finland the utility needs licences from the Ministry of Trade and Industry, an approval certificate for the package type and approved plans for transport, emergency situations and physical protection. In addition, special transport licences from the Finnish State Railways and from the Finnish road authorities are needed.

* Only a summary is published here.
To increase the transport safety, a risk assessment study for spent fuel transport from Loviisa nuclear power plant was made in 1979 by the Technical Research Centre of Finland. An inspection programme for the casks was compiled in 1981 by IVO.

The Soviet special train is taken over by IVO at Vyborg (USSR), 30 km from the Soviet-Finnish border. The train goes 255 km via Vainikkala and Lahti to Loviisa, where the empty casks are lifted onto IVO’s road transport trailers and transferred 15 km to the power plant. There the casks are inspected and packed. Depending on burnup, the packages are gas filled or water filled as in 1985. After temperature stabilization, the casks are transferred to the train, which goes via the same route to Vyborg, where the whole train is handed over to AEE. So far this procedure has taken 12 to 21 days but it is clear that the next transports will be carried out in shorter times.

Especially in 1981 the media were very interested in the transport. The information given by IVO was not sufficient for them and that is why the newspapers and the Finnish TV observed the event very closely and dramatized it unnecessarily. In 1985, when a film group and journalists were allowed to observe the packing work on site, the attitude of the mass media became positive.
REFERENCES


NUCLEAR FUEL TRANSPORT: THE SPECIAL CASE OF SPENT FUEL TRANSPORT

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Abstract

NUCLEAR FUEL TRANSPORT: THE SPECIAL CASE OF SPENT FUEL TRANSPORT.

Nuclear transport constitutes for Cogéma an essential activity linking the various components of the fuel cycle. It is thus extremely important for Cogéma to have viable transport systems, in terms of safety of course, but also as regards the reliability of the services granted to its customers. The paper deals in particular with spent fuel transports for which, as a result of the development of its reprocessing plants, particularly that of La Hague, Cogéma already has significant means available. The following aspects are presented: (a) industrial aspects (industrial policy, industrial arrangements); (b) technical aspects (qualification of the fuel, technical options adopted, developments presently under consideration); (c) safety aspects; (d) economic aspects (quantities transported, personnel involved, transport equipment and associated facilities).

1. INTRODUCTION

Cogéma, which is a company involved in the field of the nuclear fuel cycle, is of course deeply concerned with nuclear transport activities since nuclear transports constitute essential links between the various industries in the fuel cycle: mining, refining, conversion, enrichment, fuel fabrication, reprocessing, etc.

It is quite natural in this context that Cogéma has a great interest in transport activity but it is also extremely important because Cogéma needs to have viable transport systems in terms of both safety and also the reliability of the services granted to its customers and to its plant operators, in different parts of the world, for the smooth supply of their plants.

There is a great variety of forms of uranium and other materials to be transported throughout the fuel cycle and it is impossible to review them all in
a short presentation. This paper deals more particularly with spent fuel transports. There are two main reasons for this choice:

(1) Cogéma has important means (assets, organization, etc.) for the transport of spent fuel as a consequence of the development of its reprocessing plants, particularly that of La Hague; and

(2) The nature of the material transported results in particular features (complexity of the equipment, safety aspects, etc.) which make this activity worthy of special attention.

2. INDUSTRIAL ASPECTS

2.1. Industrial policy

Cogéma does not intend to operate by itself, directly and systematically, all transports of all kinds in the fuel cycle and likes to rely upon qualified companies working in this field. It likes, however, to supervise all those involved in the transport field, especially when they have a monopoly or when the particular transports considered present some special features (political and industrial implications, important safety risks, etc.).

This is true specially for spent fuel transports to the La Hague and Marcoule reprocessing plants, which Cogéma wants to control extremely closely since a spent fuel transport accident, or even an incident not resulting in nuclear hazards, might totally jeopardize a complete sector of the nuclear industry since:

— owing to the high nuclear risks involved, an accident during a spent fuel transport might result in the prohibition of all such transports for an unlimited period of time, which would not only damage the reprocessing industry but, as a consequence, could damage the operation of all power plants served by the reprocessing plant;

— owing to the extreme sensitivity of public opinion, an incident on roads or rail lines might result in similar effects.

2.2. Industrial arrangements

As indicated above, Cogéma likes to supervise all spent fuel transports, the operation being performed either by Cogéma itself or by an affiliated company, either NTL (Nuclear Transport Ltd) or PNTL (Pacific Nuclear Transport Ltd).

In practice, the spent fuel transports from French power stations are operated directly by Cogéma using its own equipment (casks, wagons, trucks, etc.) and assisted only on some sites by Transnucléaire (TN).

Transports arising from other European power plants are operated by NTL, an indirect subsidiary of Cogéma through Transnucléaire, acting for trans-
ports to La Hague as a subcontractor of Cogéma and operating for some transports with its own fleet of casks and wagons and using those of Cogéma for the other transports.

Transports from Japan to La Hague, like those to the BNFL plant of Sellafield (UK), are operated by PNTL (in which Cogéma is a shareholder) acting on a contract basis for the Japanese utilities. Cogéma can supervise the conditions under which the fuel is transported to La Hague.

This system makes it possible to have at the same time
- a very good service for all partners involved
- an organization fully adapted to the context (domestic, international, land or maritime transports)
- a high degree of safety and security.

3. TECHNICAL ASPECTS

3.1. Qualification of the fuel

It is a matter of fact that when, after their discharge from the reactor core and a preliminary cooling in the reactor pond, the fuel elements are made available for transport, their quality (soundness of the cladding, strength of the structure, occurrence of deposits on the surface, deformation, etc.) needs to be assessed so that risks are not taken by the transporter and later on by the reprocessor when cask unloading takes place.

This is necessary from both the transport and cask unloading points of view. In practice, Cogéma has defined, for example, various procedures depending on the power plants (PWR, BWR) and their equipment (in-core sipping, out-of-core sipping), the implementation of which is audited by Cogéma. These procedures make it possible to estimate the degree of soundness of the cladding and/or the existence of possible leaks from the fuel when it is leached by water during transport or, in the case of dry transport, during the water filling of the casks before unloading.

3.2. Technical options and design criteria adopted

3.2.1. Big casks

Taking into consideration the experience gained in the mid-sixties for gas cooled reactor fuel and the mid-seventies for LWR fuel, Cogéma has decided to use only big casks instead of small or medium sized casks: reduction of exposures to personnel, reduction of the number of personnel required and also, for new units, reduction of the sizes of the cask unloading facilities — all these are factors
leading to substantial cost savings at the reprocessing plant and, as shown by experience, on the transport side too.

3.2.2. Rail or sea transports

As a rule, Cogéma and its associated partners (NTL and PNTL) use mainly rail and sea transport and, with a few exceptions, use road transport only for very short distances between certain power plants and a rail/road terminal built in their vicinity or a port terminal for stations, built along the seashore and served by sea.

3.2.3. Standard packagings

Cogéma has used and is still to some extent using a large variety of casks. When the quantities to be transported were limited and the reactor equipment and fuel types were far from being standardized, the transports were mainly made in relatively small casks. With the increase in the quantities involved, a certain standardization of plants and fuel around five or six standards, a growing concern about the need to develop remote operations and, if feasible, automatic operations at the reprocessing plant where large numbers of casks have to be handled, Cogéma has decided to impose a number of standards where cask design is concerned.

For acceptance of casks at La Hague there are four standards based on identical design features and differing from one another only by the sizes (weight, capacity, length, diameter): two different diameters and two different lengths are defined but the combination of the small diameter and the long length is not used.

The main design features are as follows:

1. Dry type, i.e. no water in the cavity during transport
2. Flask suitable for dry unloading and pond unloading
3. Technical requirements:
   - ability to fit standard protective skirts
   - use of stainless steel coating of a minimum thickness
   - capability of internal and external decontamination by automatic equipment
   - similar unloading procedures
   - maximum capacity
4. Adoption of many standards for shapes, materials (for trunnions, orifices, covers and lids, screws and bolts) and, in general, all dimensions of interest for the cask operations.

The main advantages of these packagings are their large payloads, their moderate costs, their reliability resulting from extensive experience, and the consequences of standardization for fabrication, operation and maintenance, i.e. a series of factors extremely beneficial from a safety standpoint.
3.3. Developments presently under consideration

The evolution of the designs in the years to come will be dictated by two main factors:

(1) Fuel evolution: the gradual increase in the burnup and the progressive introduction of plutonium recycling (MOX fuel) are likely to impose higher shielding and heat dissipation capability partly offset, however, by longer fuel cooling times before transport;

(2) The increase in the transport demand following the launching in the Federal Republic of Germany and Japan of large reprocessing programmes (Wackersdorf and Shimokita) will probably result in the entry into service of large fleets of casks with designs adapted to specific requirements, mainly the cooling time of the fuel transported and the residence time of the fuel in the casks.

The first factor will of course affect the design of the casks used by Cogéma as many customers of La Hague reprocessing plant intend to deliver high burnup fuel and MOX fuel. The second will probably not affect the criteria adopted by Cogéma as Cogéma customers wish to have their spent fuel transported after a rather short cooling time and are rather satisfied by the present systems.

4. SAFETY ASPECTS

Transporters have to comply with many different regulations: French regulations for the transport of radioactive materials within France, ADR, RID and IMCO regulations for international transports by road, rail and sea and all other regulations specific to each country, such as, for example, the French steam pressure rules as the casks may be pressurized during the unloading process at La Hague.

The basic contents of all these various texts are largely common, their respective authors having always followed in the drafting of the texts the recommendations laid down by the IAEA and having mainly adapted the presentation to the structure of the regulations without significant alteration. This situation is very fortunate and worth noting: the universality and the relative stability of the extremely severe criteria that the transporters have to apply in their activity, and in particular in the design of the packagings, are great safety factors. Packagings licensed in a particular country have a very good chance of having their certificates validated in other countries when the need arises.

Experience and analysis have shown that the casks actually used for the transport of spent fuel have a large margin of safety with respect to the regulatory tests and the real conditions they may meet.
TABLE I. SPENT FUEL TRANSPORT ACTIVITY

<table>
<thead>
<tr>
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<th></th>
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<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>GCR</td>
<td>4780 t U</td>
<td>10.6 t U +Pu</td>
<td>2440 t U</td>
<td>10.6 t U +Pu</td>
</tr>
<tr>
<td></td>
<td>1580 casks</td>
<td>125 casks</td>
<td>510 casks</td>
<td>125 casks</td>
</tr>
<tr>
<td>LWR</td>
<td>4120 t U</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>1540 casks</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>FBR</td>
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</tbody>
</table>

The impact test at a speed of 50 km/h on to an unyielding surface is equivalent to a highway or rail crash well in excess of 100 km/h owing to the fact that there is no unyielding surface in highway or rail transport conditions. Crash tests have demonstrated the ability of the currently utilized casks to withstand conditions more severe than those of the regulatory tests.

Evaluations of the behaviour during fire tests of portions of casks have indicated that the cask seals reach a temperature where there is a possibility of leakage only after a period much longer than the 30 minutes of the regulatory test.

5. ECONOMIC ASPECTS

5.1. Quantities transported

The figures in Table I illustrate the economic impact of the spent fuel transport activity correspond to a programme of more than 200 GW(e)-a from a combination of GCR, LWR and FBR plants.

Considering only LWRs, the coming 10 years will lead to the transport to La Hague of about 13 000 t U in about 3000 cask movements to serve approximately 110 reactors, half of them being French. In 1986 for example, it is expected that 300 cask movements to La Hague will be carried out — 200 by rail, 50 by sea and 50 by road.

5.2. Personnel involved

The personnel involved in the spent fuel transport is small: Cogéma, NTL and TN staff dealing with spent fuel transports to La Hague amount to less than 100 people for design, procurement and purchase, operation, maintenance, supervision and management. Of course other partners contribute to this activity to various extents — railway personnel, ship crews, port personnel and truck drivers, for instance. However, in total, as compared to other areas of the fuel cycle, the personnel working in this field is relatively small.
5.3. Transport equipment and associated facilities

The transport of spent fuel is not a very spectacular activity: no big facilities, no remarkable equipment, everything being relatively small and dispersed in many different places (roads, railway lines, sea routes, etc.). However, the equipment is quite extensive and represents very significant assets. Considering again only LWR spent fuel, we can list the following:

- about 90 big casks owned by Cogéma, NTL and PNTL
- about 30 rail wagons owned by Cogéma and NTL
- several special ships (five ships owned by PNTL operate between Japan and Sellafield and La Hague)
- a few very special trucks owned by Cogéma
- a private rail terminal near Cherbourg owned by Cogéma, with facility to maintain wagons and trucks, and a port terminal at Cherbourg
- a facility for the maintenance and reinspection of all casks at La Hague.

6. CONCLUSION

Transport of spent fuel represents a comparatively small turnover as compared with the other services of the nuclear fuel cycle (about 5 to 10% of the reprocessing costs, for instance, and in general not more than 0.4 to 0.7% of the cost of nuclear electricity generation). Nevertheless, transport is indeed an essential part of the fuel cycle and partly controls the operation of the downstream facility (reprocessing plant) and of the upstream ones (power plants).

All matters related to spent fuel and spent fuel management are quite sensitive, as everyone knows: this is due not only to political factors but also to the particular risks involved and the very serious impact on public opinion of any incidents. This is especially true of the transport of spent fuel since transport is the only step in the fuel cycle which takes place outside a nuclear facility — on public roads, railways or open seas, where incidents or accidents would eventually involve not only the environment but also the public and subsequently might result in the risk of jeopardizing all or part of the nuclear industry.

Despite the large number of shipments already carried out to Cogéma plants and to other reprocessing plants throughout the world, no accidents have happened in which persons have been injured or property damaged as a result of the radioactive nature of the materials. The reason for this lies in the very high level of safety which, as in all areas of the nuclear industry, applies to the transport of irradiated nuclear fuel.

The figures given above show that considerable experience has been gained in this field. In summary:
— 300 transports to La Hague, LWRs only, in 1986, i.e. approximately 1200 t U.
— every day, on average, two ships, a dozen rail wagons and two trucks carrying casks loaded with irradiated fuel are moving in various countries, bound for the La Hague reprocessing plant.

Skilled personnel are working so that this activity continues, as in the past, safely and with discretion.
ACCIDENTS INVOLVING RADIOACTIVE MATERIAL IN TRANSIT STORES AT PORT DEPOTS: OPERATIONAL AND REGULATORY ASPECTS.

The storage of hazardous goods in transit at port depots is an aspect of the transport of radioactive material requiring particular attention. The IAEA Regulations for the Safe Transport of Nuclear Material leave the competent authorities of each country free to impose standards relating to the storage of such material. Experience gained from events that have occurred indicates that operational instructions are clearly insufficient. More specifically, an analysis is made of a large-scale fire which occurred on 24 June 1975 in a depot for goods in transit at the port of the city of Buenos Aires, in which a package containing a cobalt-60 source with an activity of the order of 2000 Ci was involved. In addition, certain aspects are discussed from two different points of view: first, a brief analysis shows that it is not certain whether the arrangements laid down by the Regulations for the testing of packages are such that they will withstand the "fire-impact" sequence shown to take place by the event mentioned above; second, rules of a general nature are given for operational planning by emergency action teams, with a view to reducing the probability of radioactive material being released or radiation leaking out, and to limiting the hazards for such personnel in the event of a fire during storage in transit.

ACCIDENTES QUE INVOLUCRAN MATERIAL RADIACTIVO EN ALMACENAJES TRANSITORIOS EN DEPOSITOS PORTUARIOS: ASPECTOS OPERACIONALES Y REGULATORY.

Durante el transporte de material radiactivo se observa la particularidad que presenta el almacenaje transitorio de mercaderías peligrosas en depósitos portuarios. El Reglamento para el transporte seguro de materiales radiactivos del OIEA deja a las autoridades competentes de cada país la facultad de dictar las normas relativas a la guarda de tales materiales. La experiencia de hechos ocurridos pone de manifiesto una aparente insuficiencia de instrucciones operacionales. Concretamente, se analiza un incendio de grandes proporciones, ocurrido el 24 de junio de 1975 en un depósito de mercaderías en tránsito ubicado en el puerto de la ciudad de Buenos Aires, en el que estuvo involucrado un bulto que contenía una fuente de cobalto 60 con una actividad del orden de los 2000 Ci. Asimismo, se hacen algunas consideraciones desde dos puntos de vista: por un lado, en un breve análisis se plantea la incertidumbre de si el orden establecido para ensayos de bultos en el Reglamento avala que éstos soportan la secuencia "incendio-impacto" que muestra la experiencia mencionada; por otro lado, se brindan reglas de carácter general destinadas a la planificación operativa por parte de las fuerzas de intervención primarias, con el objeto de reducir la probabilidad de liberación de material radiactivo o fuga de radiación y disminuir los riesgos que para dicho personal ocasionara un incendio durante el almacenaje transitorio.
1. ANTECEDENTE HISTORICO

El 24 de junio de 1975, en una de las dársenas del puerto de la ciudad de Buenos Aires, se produjo un incendio, en el cual estuvo involucrado un bulto del Tipo B que contenía una fuente de cobalto 60 con una actividad del orden de los 2000 Ci. Este se encontraba ocasionalmente en un depósito de mercaderías en tránsito, por un error administrativo, ya que debía recibir urgente despacho.

El siniestro fue detectado por personal de vigilancia a las 7.35 de la mañana, quien reclamó la inmediata intervención del cuerpo de bomberos, así como de refuerzos debido al incremento de las llamas.

Se trataba de un galpón de construcción de mampostería, con techos de losa, compuesto de una planta principal y un piso elevado, cuyas dimensiones eran de 340×30×10 m. En su sector medio, el fuego alcanzó una extensión de 50×30×10 m, sobre bobinas de papel, caucho, un camión estacionado en la planta baja, y el embalaje de bultos conteniendo repuestos para locomotoras situado en la planta alta; el bulto con la fuente de cobalto 60 se encontraba en las cercanías del vehículo.

El proceso ígneo, que demandó a los bomberos más de 35 horas de ininterrumpida labor, tuvo su máxima expresión a las 2 horas de iniciado, al ceder la losa principal del depósito en cuestión, derrumbándose la totalidad de su estructura sobre los citados materiales. Afortunadamente, el impacto sobre el bulto radiactivo fue amortiguado por el mencionado camión. Es de destacar que el blindaje en ningún momento perdió su estanqueidad.

2. CONSIDERACIONES TECNICO-TEORICAS

Al nivel actual de la ciencia y la técnica, el incendio no debe considerarse como una fatalidad, sino como un fenómeno superable en base a la experiencia y al trabajo de laboratorio.

2.1. Análisis de la curva característica tiempo-temperatura

Partiendo de la observación de numerosos hechos reales, ensayos y discurrimento teórico-práctico, en distintos países se han compilado datos sobre las temperaturas alcanzadas en incendios en función del tiempo. La representación gráfica de estos procesos aparece en la Fig. 1, en la que se observa que el incendio muestra en general tres fases expresadas en términos de tiempo, a saber:

Fase 1. Se corresponde con el período preliminar, durante el cual las temperaturas crecen lentamente en función del tiempo, debido a que el calor generado se utiliza para elevar la temperatura del combustible y del aire circundante.
Fase 2. Coincide con el período de propagación (1-2), donde la temperatura aumenta rápidamente, al ser mayor la cantidad de combustible en actividad, al aumentar la velocidad de combustión, etc. La duración de esta fase está vinculada con la cantidad de material combustible presente y la alimentación de aire en cantidad suficiente para asegurar la combustión. En el punto 2, se corresponde con el equilibrio entre el calor desarrollado y el que se disipa; esta temperatura máxima del evento se produce en un tiempo que es el denominado "duración del incendio".
**Fase 3.** Representa al período de decrecimiento y extinción.

Por supuesto que cada incendio presentará su propia curva “tiempo-temperatura”, variando de uno a otro la forma de los períodos indicados. Del examen de una familia de curvas surge la curva característica, que puede obtenerse con un criterio de media, basándose en consideraciones probabilísticas y trazando la curva descripta por los valores medios de las integrales de la familia. Así, por ejemplo, la curva característica que presenta la Fig. 2, utilizada en la Argentina, responde a la fórmula:

\[ T = 290 \log (29 t + 1) \]

donde \( T \) es la temperatura alcanzada por el incendio, en °C
\( t \) es el tiempo, en minutos.

**2.2. Estudio analítico de la carga de fuego**

La cantidad de calor desarrollado es otro de los parámetros fundamentales a considerar en el estudio del fenómeno térmico que constituye el incendio. Esta variable depende esencialmente de la cantidad y poder calorífico de los materiales presentes en el sector afectado.

La máxima cantidad de calor desarrollado estaría dada por la sumatoria de los productos de los pesos de los materiales combustibles presentes por sus respectivos poderes caloríficos, es decir, considerando la combustión completa y en ausencia de dispersiones; su expresión matemática es

\[ Q = \sum_{i=1}^{n} K_i P_i \]

donde \( Q \) es la cantidad de calor desarrollado
\( K_i \) el poder calorífico del elemento \( i \)
\( P_i \) el peso del material \( i \).

Podemos definir la *carga combustible* (q) del sector de incendio como la relación entre la cantidad de calor desarrollado y la superficie de piso acumulada del local en estudio (S):

\[ q = \frac{Q}{S} \]

Con miras a simplificar, resulta conveniente referir los materiales presentes a un combustible standard, adoptándose a tal efecto la madera, con un poder calorífico de 4400 cal/kg (18,41 MJ/kg). Esta consideración da origen a la carga de fuego, que representa el peso de madera ideal, supuesto uniformemente distribuido, capaz de desarrollar una cantidad de calor equivalente a la que produciría la combustión
completa de los materiales contenidos en el sector de incendio, representada matemáticamente por la expresión:

$$q_f = \frac{\sum_{i=1}^{n} K_i P_i}{4400 \text{ cal/kg} \cdot S}$$

2.3. Curva tiempo-carga de fuego

Existe una correlación entre la carga de fuego y la duración probable del incendio, entendiéndose por “duración” el tiempo empleado en alcanzar las máximas temperaturas. Esta predicción puede hacerse a través de curvas resultantes de ensayos, basadas en la cantidad ideal de madera cuya combustión completa alcanza para cada par de valores tiempo-temperatura de la curva standard.

Una expresión aproximada (utilizable para cargas de fuego de hasta 150 kg/m²) y que es confirmada a diario, es la siguiente:

$$D = 0,02 q_f$$

donde $D$ es la duración del incendio, en horas

$q_f$ es la carga de fuego, en kg/m².

3. INFLUENCIA DE LA TEMPERATURA SOBRE LA RESISTENCIA DE LOS MATERIALES

Interesa el estudio de las características mecánicas de los metales, porque de ellas surge el comportamiento del material utilizado necesario para garantizar la integridad del bulto. Las altas temperaturas provocan un notable decaimiento de esas propiedades. En nuestro caso, analizaremos el comportamiento del acero por su uso corriente como elemento de protección mecánica de los bultos.

En la Fig. 3 se han consignado los valores del módulo de elasticidad y tensiones de proporcionalidad, fluencia y rotura, con sus variaciones relativas en función de la temperatura.

4. EVALUACION DEL INCENDIO ASOCIADO AL SECTOR

Se procederá a continuación al estudio teórico del incendio esperado en un hangar similar al siniestrado.
4.1. Descripción del depósito

El local está destinado al almacenamiento de mercaderías, que se realiza inmediatamente al ser bajadas éstas de los buques, clasificándolas y distribuyéndolas por sus distintos sectores.

Se trata de un galpón que consta de planta baja y un piso superior, dividido por paredes de ladrillos macizos en tres sectores de 90 m de largo y dos de 35 m; éstos se encuentran abiertos en la planta baja para permitir la circulación a través del hangar, separando los sectores precitados. Cabe señalar que fue en uno de estos pasajes donde se originó el siniestro de referencia.

4.2. Materiales involucrados

De acuerdo con estadísticas realizadas, se llegó a determinar una cantidad promedio de material combustible presente en el depósito, teniendo en cuenta exclusivamente los materiales constitutivos de embalajes. Los resultados obtenidos
son sumamente conservadores: madera, $3,4 \times 10^5$ kg; sintéticos, $3,1 \times 10^5$ kg; y papel y cartón, $2,1 \times 10^5$ kg.

4.3. Carga de fuego

Tomando como base una superficie de piso acumulada de 15 000 m$^2$, obtenemos el valor de la carga de fuego:

$$q_f = \frac{5,436 \times 10^9 \text{ cal}}{4400 \text{ cal/kg} \times 15 \times 10^3 \text{ m}^2} = 82,4 \text{ kg/m}^2$$

4.4. Duración del incendio

De acuerdo con la expresión correspondiente, la duración del incendio fue de:

$$D = 0,02 \times 82,4 = 1 \text{ hora y 39 minutos}$$

4.5. Temperatura máxima del incendio

Aplicando la pertinente fórmula, se obtiene:

$$T = 290 \log (29 \times 99 + 1) = 1004^\circ \text{C}$$

5. COMPORTAMIENTO DE LOS MATERIALES DE CONSTRUCCION ANTE LA ACCION DEL FUEGO

Este análisis puede realizarse desde el punto de vista de la reacción o el de la resistencia al fuego. Interesa aquí el segundo enfoque dado que contempla la determinación del tiempo durante el cual los elementos constructivos conservan sus cualidades funcionales asignadas. El fenómeno es tan complejo y particularizado que los ensayos sólo contemplan aspectos parciales del problema, cuya evaluación final no puede independizarse de la experiencia.

Resulta evidente, luego del análisis precedente, que las condiciones a las que se vió sometido el edificio causaron el colapso de su estructura.

6. ASPECTOS OPERATIVOS

Es menester brindar una serie de pautas operativas destinadas a las fuerzas de bomberos en su carácter de encargadas de la intervención primaria, lo cual es necesario por las falencias que se habrían detectado en la reglamentación vigente. Asimismo, se considera conveniente que estas normas sean conocidas por el personal
que tiene relación directa con la vigilancia y/o guarda de los bultos. No se trata de normas generales de ataque al fuego, ya que se supone conocidas por los bomberos y no hacen a la especificidad de este trabajo.

En el antecedente citado se observan dos prioridades para combatir el incendio en forma segura. La primera consiste en garantizar el mantenimiento de la funcionalidad de la estructura edilicia; la segunda está ligada a la conservación de la integridad del bulto.

En cuanto a este último aspecto, es menester limitar la zona cercana al bulto de material radiactivo mediante un vallado ubicado en un radio de 30 m. Este será provisto de leyendas que prohíban el acceso, a excepción del personal autorizado y con la debida anuencia del oficial de radioprotección.

Deberán usarse indefectiblemente equipos respiratorios independientes de la atmósfera, con el objeto de evitar los riesgos derivados de la contaminación interna, y se darán instrucciones precisas para impedir el contacto con elementos ubicados en el sector vallado que pudieran estar contaminados.

En lo que se refiere a las tareas de extinción propiamente dichas, existen limitaciones operativas tales como el tiempo de exposición y la distancia entre la fuente radiactiva y el bombero. Este último objetivo puede lograrse mediante un ataque al fuego empleando monitores y/o boquillas que permitan un gran alcance.

Asimismo, será conveniente reducir el tiempo de permanencia en la zona afectada, practicando para ello relevos permanentes de personal.

En cuanto a la atenuación de las dosis mediante el empleo de blindajes, es de destacar que se pueden utilizar como tales las paredes propias del inmueble u otros bultos no combustionados y que brinden seguridad desde el punto de vista del incendio, con el objeto de permitir un mayor acercamiento del personal a la fuente mientras duren las tareas de extinción.

7. ASPECTOS REGLAMENTARIOS

Como es sabido, existe una serie de ensayos a soportar por los bultos de material radiactivo para su aprobación.

Dichos ensayos tienen por objeto brindar una garantía de carácter práctico de que el bulto conservará su integridad en condiciones normales de transporte y luego de un accidente. Para este último caso se entiende que los ensayos son una representación de algunos de los parámetros del accidente (no por ello menos exigentes que el propio accidente, sino generalmente al contrario), derivada del análisis de una hipótesis establecida como la más probable.

Evidentemente, con la secuencia establecida para los ensayos se pretende avalar la resistencia del bulto ante un accidente derivado de una colisión, descarrilamiento, vuelco, etc. en el que se produzca un fuerte impacto y posterior incendio. Entendemos que es bastante riguroso, dada la baja probabilidad de que luego del choque el bulto se encuentre totalmente envuelto por el fuego durante tanto tiempo.
Es aquí donde surge nuestro interrogante sobre la resistencia del bulto en un hecho como el citado. En este incendio, el orden de los sucesos se invierte: primero el bulto es sometido a un fuego y luego al esfuerzo de choque, agravándose la situación dado que puede considerarse que casi siempre se verá sometido a una mayor temperatura que la del ensayo y totalmente rodeado por llamas durante mayor tiempo, disminuyéndose así notoriamente las tensiones admisibles y aumentando la probabilidad de rotura.

Ejemplificando lo antedicho, los análisis del incendio asociado a un sector similar al afectado arrojan valores de tiempo y temperatura que superan sustancialmente los requeridos por la prueba termal reglamentada.

Aparentemente, los riesgos derivados del almacenaje transitorio no estarían contemplados en el Reglamento para el transporte seguro de materiales radiactivos, siendo esta situación bastante difícil de eliminar dentro de los pasos necesarios para efectuar el transporte, ya que sería prácticamente imposible suprimir el tránsito por los depósitos portuarios.

Para poder obtener una solución a este inconveniente, se propone lo siguiente:

1) Realizar los estudios correspondientes, tendientes a verificar si los bultos que superan los ensayos reglamentarios también soportan la secuencia del accidente del incendio en un ambiente cerrado y el consiguiente impacto debido al colapso de la estructura del depósito, con la sobrecarga del material almacenado en pisos superiores; y

2) Hacer algún tipo de consideración de carácter más específico sobre el almacenamiento transitorio de material radiactivo, haciendo especial hincapié en la limitación de la carga de fuego y la protección estructural de los depósitos.

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INTRODUCTION

The purpose of this paper is to provide a technical summary of RAM transport experience in the United States of America. This summary describes the US RAM transport experience and emphasizes three areas: (a) US RAM transport shipment volume; (b) RAM transport accident/incident experience in the USA; and (c) commercial spent fuel transport that has occurred in the USA from 1964 through 1984.

RAM TRANSPORT SHIPMENT DESCRIPTION

There have been two estimates made of the number of annual shipments of RAM in the United States of America. The first of these estimates occurred in 1975 when a study was commissioned by the US Nuclear Regulatory Commission to survey USA RAM shipments on an annual basis; this work was performed by Battelle Pacific Northwest Laboratory. The second occasion to make such an estimate took place in 1981 when a study was undertaken to update the 1975 survey. This update study was jointly funded by the US Department of Energy, the US Nuclear Regulatory Commission, the US Department of Transportation, and the US Federal Emergency Management Agency. This most recent estimate [1] was conducted by SRI-International and determined that during a sample period from September 1981 to September 1982 there were approximately 511 shipments per year.
1.96 million annual shipments of RAM in the USA. It was further determined that these shipments involved 2.79 million packages or approximately 1.4 packages per shipment. These shipments contained approximately $3.32 \times 10^{17} \text{ Bq}$ (8.97 million curies) of radioactive material.

**RAM TRANSPORT ACCIDENT/INCIDENT EXPERIENCE**

The Transportation Systems Development Department at Sandia National Laboratories has monitored US RAM transport accident/incident experience since 1979. This monitoring activity has led to the development of the Radioactive Material Incident Report (RMIR) data base. The management of the RMIR data base is conducted on a daily basis by the Joint Integration Office of the Department of Energy, Albuquerque Operations Office. Presently, there are approximately 1034 reported events in the RMIR data base for the period from 1971 to March 1985. These events consist of 167 transport accidents, 203 handling accidents, and 664 other reported events that do not involve accident conditions. The RMIR data base can provide detailed summaries and a special search capability which allows a national perspective of RAM transport accidents and incidents and the packaging response to these events [2].

**COMMERCIAL SPENT FUEL SHIPMENTS IN THE USA**

A major fraction of the commercial spent fuel that has been generated in the USA is still resident in the storage pools at the reactors. As the USA waste management programme proceeds there will be transport of these spent fuel elements to their final point of disposal. Despite the major role of storage at reactors, some transport of spent fuel has occurred in the USA since 1964. As of 31 December 1984, approximately 4141 commercial spent fuel assemblies have been transported by road to locations away from the reactor sites [3]. An additional 1656 spent fuel assemblies have been transferred between storage pools at the reactor sites. This 21 years of experience has totalled approximately 5797 fuel elements that have been transported away from or between the reactor sites.

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ASSESSING COSTS AND EFFECTIVENESS
OF SAFETY MEASURES FOR
THE TRANSIT OF SMALL TYPE A
PACKAGES THROUGH ROAD TUNNELS

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Abstract

ASSESSING COSTS AND EFFECTIVENESS OF SAFETY MEASURES FOR THE TRANSIT
OF SMALL TYPE A PACKAGES THROUGH ROAD TUNNELS.

The Mont Blanc Tunnel is situated under the highest mountain in Europe. Being 12 km
long, it is also one of the longest road tunnels in the world. Local authorities have to state whether
the general regulations for the road transportation of radioactive materials, as defined by the
IAEA, apply, or whether additive measures need to be taken. Whereas an activity limit — \( A_2 \) —
applies only to the content of a Type A package containing dispersible materials, a derived limit
applying to the whole cargo of a truck has been in use in the tunnel and can be redefined. The
present paper deals with the question of the choice of a proper figure for such a limit that might
regulate the transit for technetium generators (Elumatic III from Oris France). The first step of
the study is a risk assessment, with the truck content as an explicit parameter. The yearly traffic
is of 150 trucks, carrying, on the average, 26 Ci of technetium-99 m in Elumatic generators at the
time of the crossing. On a yearly basis about \( 5 \times 10^{-6} \) road accidents might be expected, while
the expected radiological fatalities would amount to approximately \( 2 \times 10^{-8} \) and the expected
monetary loss would be US $10. The second step is the implementation of decision aiding
techniques based on the previous estimates. As the mathematical expectations of such risk
indices were not dependent on the shipped activity, a classical approach, the cost effectiveness
curve, did not lead to an optimum. Other approaches and other criteria were investigated, such
as the comparison with other hazardous materials, the likelihood of lethal or morbidity effects
and ground contamination. Should the latter criterion be considered pertinent, it would lead to
a limit of 130 Ci of technetium at the time the truck crosses the tunnel.

1. SCOPE OF THE STUDY

The transportation of small quantities of dispersible radioactive materials is
allowed on European roads according to the IAEA standards [1] in the so called
'A package', up to a certain limit in activity for a single package (the 'A_2 limit',
which depends on the radionuclide). Under specific traffic conditions, namely when crossing the 12 km long tunnel under Mont Blanc, more restrictive standards can prove necessary. A possibility is to prohibit the crossing when the content of a whole cargo is above a certain activity limit. This measure was applied, with a very restrictive limit, until recently. The question is then to determine the authorized activity in the tunnel, and the purpose of this paper is to show the analyses supporting a decision in this field.

2. TRANSPORTATION SYSTEM

Traffic of technetium generators

There are three transits weekly through the tunnel. The vehicle in use is generally a light truck. Its cargo content, expressed in actual activity, averages to 54 Ci, ranging from half to twice this figure. It consists of various radioisotopes, but technetium generators account for 99% of this activity (28 Ci of molybdenum and 25.5 Ci of technetium).

This device shipped in a Type A package, contains molybdenum-99 (half-life 66 h), which is transformed gradually into technetium-99 m (half-life 6 h). Generators of this kind can provide technetium during a week for medical scanning purposes. Although its content in activity can vary, the generator itself, the Elumatic III from Oris, is always the same. It contains, within a parallelepiped plastic box of about 20 cm, a system to extract the required solution of technetium out of a small glass column in which both isotopes are contained and a biological shielding of 13 kg of lead (see Fig. 1).

Tunnel environment

The total length of the tunnel is 12 km and there is one lane in each direction. On average there are about 30 vehicles in the tunnel at a given time [2]. Should an accident occur that would shut one lane, about 90 people might be subjected to potential consequences, and possibly 10 vehicles might be trapped behind the truck. The important physical parameters of the tunnel are its shape and its ventilation system. The cross-section is the typical horseshoe; however, the air ducts are underneath the roadway.

The emergency response system comprises fire extinguishers distributed in the tunnel; this has been used to extinguish 13 of the 14 fires which took place in the tunnel. At the portals there is an emergency vehicle equipped with more powerful extinguishers and breathing apparatus. With regard to radiation hazard there is

\[1 \text{ Ci} = 37 \text{ GBq}.\]
no monitoring device available at the tunnel site. The decontamination teams would have to come from Lyon, which is located 200 km from the tunnel [2, 3].

3. RISK ASSESSMENT

Possible consequences of an accident

There are two main categories for the possible consequences of an accident. First, the economic impacts; these can include the cost of monitoring, the cost of decontamination and the loss of earnings due to the shutting down of the tunnel. Second, the radiation health effects can be either short term effects or long term stochastic effects. Table I summarizes the main impacts of an accident and the way they will be quantified. They are not of the same importance. Some are very unlikely, others are almost certain. The risk assessment comprises two steps: computation of the consequences of an accident and probabilistic assessment.
TABLE I. IMPACTS OF THE LOSS OF PACKAGE CONTENTS AND QUANTITATIVE INDICES

<table>
<thead>
<tr>
<th>Impact</th>
<th>Index</th>
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<tbody>
<tr>
<td>Immediate death</td>
<td>Probability of occurrence</td>
</tr>
<tr>
<td>Morbidity</td>
<td>Probability of occurrence</td>
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<tr>
<td>Late radiation effects</td>
<td>Collective dose</td>
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<tr>
<td>Shutdown of tunnel</td>
<td>Monetary unit</td>
</tr>
<tr>
<td>Radiation control</td>
<td>Monetary unit</td>
</tr>
<tr>
<td>Decontamination</td>
<td>Monetary unit</td>
</tr>
</tbody>
</table>

Computation of the consequences of a release

The packages contain molybdenum and technetium, its daughter product. For 1 Ci guaranteed to the customer (one can speak of 'nominal' activity), there is, at the time the truck reaches the tunnel, 2.1 Ci of technetium and 2.3 Ci of molybdenum. The radiological hazard can arise from external irradiation and inhalation. External irradiation results from a loss of shielding. Neglecting its contribution to the collective dose, a lethal area ($10^{-3} \text{ m}^2 \text{ per nominal Ci at the 5 Sv threshold}$) and a morbidity area ($10^{-2} \text{ m}^2 \text{ per nominal Ci at the 0.5 Sv threshold}$) correspond to the hypothesis that a bystander would stay half an hour. Inhalation occurs when the products are airborne. In this case a model must be implemented for atmospheric transport. Before it reaches the lining of the tunnel (about 15 s after the release) the initial puff can be assumed to be Gaussian and immediate effects may be observed. Lethal (25 Gy to the lung and 30 Gy to the intestine) and morbidity areas can be computed as previously, but they are five times larger. For distances larger than about 50 m a box model is applied. Only delayed effects are expected in this case and a collective dose ($1.5 \times 10^{-3} \text{ man} \cdot \text{Sv per airborne nominal Ci}$) for an average location of the release and an average number of people in the tunnel accounts for them.

The loss of toll fees is directly linked to the duration of the closure of the tunnel. Any accident involving a truck would lock the tunnel for about one hour. An average figure of US $2000 can be assumed for the loss of earnings. Should there be any doubt about the integrity of the cargo, a radiological survey team would be called upon and radiological monitoring expenses would follow. The work of the team would be to check the cargo and the cars which were behind it. In addition, the roadway and walls would be monitored. These costs are almost
insensitive to the amount of damage to the packages. The work would last about five hours, since three are needed for the team to get to the tunnel location. The total cost is estimated at US $17 000, and the loss of earnings remains the main component.

The previous calculations performed with the box model allow one to compute the ground contamination. It requires the definition of an acceptable level: 50 mCi·m⁻² of molybdenum should be acceptable for a location that is not a working place. The tunnel is divided into 40 sections of 300 m, corresponding to the ventilation system. The probability of having one of these sections contaminated is dependent on the released activity. It vanishes when the release is below a 'nominal' 60 Ci. If the contamination is very slight (little release, or simple loss of biological shielding) it can be assumed that the control team might handle the problem within one hour. This implies one hour more of tunnel shutdown. When a whole 300 m section is to be decontaminated, one other team is necessary, and the operation would take about eight hours.

Various impacts have been computed (see Table II) that can or cannot be observed according to the type of the accident. In every case but the last one (the probability of one whole action to decontaminate depends on the activity), the economic impacts are not dependent on the activity carried. On the other hand, the health impacts are proportional to this parameter.

Probabilistic assessment

The aim of the probabilistic part of the assessment is to establish the accident scenarios that can result in the 'consequence scenarios' stated above, and to compute their probabilities.

Although some statistics are available on Type A package accidents [4, 5], they are not specific to technetium generators. A crush and fire experiment has been performed in the Amersham Centre with a light truck containing a mixed cargo of Type A and B packages [6]. An interesting feature was the very short time which was necessary for a fire to encompass the whole vehicle. However, the results of the regulatory tests, the analyses of a train accident, and the destructive fire test performed in June 1985 are specific to the French Elumatic technetium generator.

A review of these data and of the tunnel accident records has allowed us to focus on four accident scenarios, with the following consequences:

- light crash: no loss of shielding,
- frontal collision (i.e. about 120 km·h⁻¹): loss of shielding, 1% airborne material,
- short fire: no effect,
- strong fire (i.e. destroyed vehicle): 75% airborne material.

The probability of light crashes is $3.5 \times 10^{-6}$ at each crossing of the tunnel, half of them requiring monitoring. This probability is $4 \times 10^{-7}$ for a collision,
### TABLE II. MAGNITUDE OF THE IMPACTS FOR VARIOUS ACCIDENT SCENARIOS

<table>
<thead>
<tr>
<th>Scenario</th>
<th>Cost of traffic interrupt (US $)</th>
<th>Cost of control (US $)</th>
<th>Cost of decontamination (US $)</th>
<th>Probability of morbidity</th>
<th>Probability of mortality</th>
<th>Collective dose (man·Sv)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Trivial accident</td>
<td>2000</td>
<td></td>
<td>2000</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Suspected loss of content</td>
<td>10000</td>
<td>7000</td>
<td>1000</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Loss of biological shielding</td>
<td>12000</td>
<td>6000</td>
<td>1000</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Actual airborne release</td>
<td>12000</td>
<td>6000</td>
<td>1000</td>
<td>2.5×10⁻³ X A.f</td>
<td>2.5×10⁻⁴ X A.f</td>
<td>1.5×10⁻³ X A.f</td>
</tr>
<tr>
<td>Actual airborne release and decontamination</td>
<td>28000</td>
<td>7000</td>
<td>14000</td>
<td>2.5×10⁻³ X A.f</td>
<td>2.5×10⁻⁴ X A.f</td>
<td>1.5×10⁻³ X A.f</td>
</tr>
</tbody>
</table>

A.f: Released fraction expressed in 'nominal' curies (1 'nominal' Ci = 2.3 Ci of molybdenum).
$8 \times 10^{-7}$ for a light fire and $5 \times 10^{-8}$ for a severe fire. On the basis of the actual traffic of 150 passages with 12 Ci of nominal activity (25.2 Ci of $^{99}$Tc$^{m}$ and 27.5 Ci of $^{99}$Mo) the risk is as follows:

- accident probability $7.1 \times 10^{-4}$
- expected monetary loss US $6.5$
- expected collective dose $1.11 \times 10^{-7}$ man$\cdot$Sv
- probability of a lethal effect $1.85 \times 10^{-8}$
- probability of reversible effect $1.85 \times 10^{-7}$

The level of risk appears to be low, and this not only due to the small amount of traffic. For instance, the number of health effects is one thousand times lower than the expected number of deaths due to the traffic accidents themselves.

4. ELEMENTS FOR THE DECISION MAKING PROCESS

Cost benefit analyses

The question is whether there is an optimum in a possible allowed nominal activity for the technetium generators. One must therefore look at the costs and the benefits of an increase in this level. The benefit arises from the reduction in the number of shipments. The 'cost' of the measure was expected to be an increase in the risk level. In principle, there should be an optimum when balancing these figures. It has already been stressed that most of the costs of the accident are not dependent on the activity carried while the health effects are linearly connected with it. Increasing the allowed limit means decreasing the number of shipments and therefore the accident probability. The conclusion (see Fig. 2) is that the expected number of health effects remains constant, while the monetary cost of the accidents decreases.

This is a situation in which cost benefit analysis does not lead to an optimal level. Thus a limit must be searched for among the constraints that might apply to this kind of transportation. The analysis was however of some interest. It illustrated the orders of magnitude of the impacts and it showed that increasing the limit is sound.

Other criteria

A regulatory constraint, the Transportation Index, makes it difficult to reach figures higher than 100 Ci, but it is technically feasible. Another criterion arises from the comparison with other hazardous materials. It would lead to allowed amounts well beyond plausible figures. Looking at the consequences of the major
event, here a large fire, two other criteria appear. For about 1000 Ci, the likelihood of inducing a lethal effect becomes of some magnitude. The same was computed for 100 Ci looking at morbidity effects. Finally, an interesting figure corresponds to the amount above which, still in the worst case accident, it would be likely to have to decontaminate a whole 300 m section of the tunnel. This quantity is around 60 Ci of 'nominal' activity. This criterion is worthwhile considering since such work would have a considerable impact on public opinion.

5. CONCLUSION

This study has a clear result. It demonstrates the low level of risk associated with the transportation of medical sources under the tunnel, both from a probabilistic and a worst case viewpoint. However, the use of a traditional cost benefit approach is not possible because there are only advantages, when dealing with the mathematical expectation of the cost and benefits, in raising the limits. Owing to the difficulty with that objective criterion, other criteria of a more subjective nature have been examined.

The limit above which important decontamination work would have to be undertaken after a very serious accident was found to be a criterion of interest. This is due to the economic impact, but especially to the potential effect on public opinion of a long shutdown of the tunnel attributable to a radioactive material inci-
dent. It would lead to a value of about 130 Ci of technetium at the time the truck passes through the tunnel (60 Ci of 'nominal' activity). Although it clearly appears that the last figure relies on a subjective judgement and that the final decision should carefully weigh these subjective factors, this study has illustrated how a quantitative assessment and a formal approach prove useful when dealing with decision related problems of this kind.

REFERENCES


COMPETENT AUTHORITY DATA BANK
ON THE TRANSPORT
OF RADIOACTIVE MATERIALS IN ITALY*

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Rome, Italy

Because of the importance of gathering data on the transport of radioactive materials – according to IAEA recommendations – the technical Competent Authority in Italy, ENEA-DISP, has organized a specific data bank. This is based on the particular administrative requirements of the domestic regulations, which prescribe that authorized carriers are obliged to prepare a summary record of transports carried out every three months. The data gathering process was complicated up to 1982 because of the large amount of shipments recorded by hand. Since 1983 it has been possible by agreement with the carriers to computerize the data.

The number of authorized carriers is at present 130 and is increasing. The number of shipments in the data bank is about 250,000 per year.

For each shipment it is possible to check the factory of departure, the arrival point, the number of kilometres per shipment, the last shipment, package characteristics, the radioactive contents and the labels used.

Results for the years 1983 and 1984 have been processed to give the most important conclusions, such as the zones with high traffic density, the percentage of various modes of transport, and the amount of radionuclides transported. An assessment of the collective dose to the public during recent years has also been made.

* Only a summary is published here.
RECENT US SPENT FUEL RAIL SHIPPING CAMPAIGN*

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In February 1984, the transfer of the 1058 fuel assemblies from Northern States Power's (NSP) Monticello Station near Minneapolis, Minnesota, to General Electric's spent fuel storage facility near Morris, Illinois, was authorized. The first shipment was planned to occur before November 1984, and proceed at a guaranteed rate of six shipments per year; each of two IF-300 Casks (36 fuel assemblies per shipment) for an annual target of 216 assemblies.

A 'Campaign Plan' and 'Monticello Campaign Task Description' were initiated to describe the detailed activities required to assure that equipment, personnel, shipping and receiving sites, transportation details, procedures, regulatory, and public relations aspects were properly considered.

The first phase of the campaign actually began with receipt of the shipment on 22 November 1984. It proceeded routinely and concluded on 7 October 1985, with a total of 19 two-cask shipments. Transported was a total of 684 BWR fuel assemblies (65% of the total), containing 124.4 t of heavy metal (uranium plus plutonium).

Key considerations of the campaign included:

- route selection
- cask handling procedures
- interfaces between programme participants.

Data generated by the programme included:

- cask transport cycle times
- cask surface contamination levels
- cask radiation levels.

Phase one of this programme was felt to have been accomplished quite efficiently; there were no incidents during the campaign which impacted on safety — either industrial or radiological. The few delays experienced were due to normal railroad maintenance activities.

* Only a summary is published here.
TRANSPORT EXPERIENCE AND BASIC REGULATIONS FOR ENSURING SAFETY AND PHYSICAL PROTECTION DURING THE TRANSPORT OF IRRADIATED FUEL IN THE USSR.

The paper considers the main principles involved in the organization in the USSR of the transport of irradiated nuclear power plant fuel by special trains, with account taken of experience already gained. Certain provisions of the current USSR regulations for safety and physical protection in the transport of nuclear materials are discussed.

ОПЫТ ПЕРЕВОЗОК И ОСНОВНЫЕ ПРАВИЛА ОБЕСПЕЧЕНИЯ БЕЗОПАСНОСТИ И ФИЗИЧЕСКОЙ ЗАЩИТЫ ПРИ ПЕРЕВОЗКЕ ОТРАБОТАВШЕГО ТОПЛИВА В СССР

В докладе рассмотрены основные принципы организации перевозок в СССР отработавшего ядерного топлива атомных электростанций с использованием специальных железнодорожных поездов с учетом накопленного опыта перевозок. Обсуждены некоторые положения действующих в СССР "Основных правил безопасности и физической защиты при перевозке ядерных материалов".

Регулярные перевозки отработавшего ядерного топлива (ОЯТ) от атомных электростанций в СССР осуществляются сравнительно недавно. В 1974 году с Нововоронежской АЭС было перевезено ОЯТ реакторов ВВЭР-210 и ВВЭР-365, в дальнейшем выполнены перевозки ОЯТ серийных реакторов ВВЭР-440. Для первых перевозок использовались два вагона-контейнера ТК-НВ в составе специального железнодорожного поезда, в который кроме вагонов-контейнеров входили два вагона прикрытия и вагон сопровождения. В вагоне-контейнере размещался один контейнер вместимостью 30 отработавших ТВС (около 4 т по UO₂).

Примерно с этого же времени начались постоянные перевозки в СССР ОЯТ реактора ВВЭР-70 АЭС "Райнсберг", ГДР. При этом также использовали специальный железнодорожный поезд в составе вагона-контейнера (разработки ГДР), двух вагонов сопровождения и вагона прикрытия. В вагоне-контейнере размещался контейнер ТК-Райнсберг вместимостью 31 отработавшая ТВС. Количество рейсов с использованием указанных транспортных средств было небольшим и составляло приблизительно 1–2 рейса в год каждого из вагонов-контейнеров.
С 1979 года, после ввода в эксплуатацию вагонов-контейнеров ТК-6 новой конструкции, начались перевозки ОЯТ реакторов ВВЭР-440 не только от советских АЭС, а также АЭС стран-членов СЭВ (НРБ, ГДР, ЧССР) и Финляндии. Для этих перевозок были сформированы специальные поезда, в состав каждого из которых входили 4 вагона-контейнера ТК-6, два вагона сопровождения и вагоны прикрытия. В вагоне-контейнере размещается контейнер ТК-6 вместимостью 30 отработавших ТВС.

Уже при первых перевозках ОЯТ от АЭС в СССР были определены некоторые принципиальные решения по общей организации таких перевозок, которые, в основном, остаются в силе и в настоящее время. Первенствующее, это использование для перевозок железнодорожного транспорта в качестве основного вида транспорта. Такое решение обусловливалось широким развитием железнодорожных дорог в СССР, наличием подъездных железнодорожных путей почти на всех АЭС. Использование железнодорожного транспорта позволяет осуществлять перевозки в больегрузных контейнерах, и уже первые контейнеры имели практически максимально возможные габариты, а соответственно и вместимость по ОЯТ.

Другим принципиальным решением по организации перевозок ОЯТ являлось использование специальных железнодорожных поездов и выполнение перевозок только на условиях полного груза (исключительного использования) в соответствии с Правилами МАГАТЭ. В состав специальных поездов входит только вагоны-контейнеры, вагоны сопровождения и вагоны прикрытия. В вагоне-контейнере обычно размещается один контейнер с ОЯТ и системы поддержания необходимых температурных условий (системы вентиляции, отопления и т.д.). Следует отметить, что при перевозках специальный поезд может проходить через территории с довольно значительным колебанием температуры окружающей среды. Так АЭС с реакторами ВВЭР-440 находятся и на Кольском полуострове и в Армении. Поэтому системы принудительного охлаждения и отопления, теплоизоляция кузова являются во многих случаях необходимыми, так как позволяют свести к минимуму влияние внешних условий на тепловой режим транспортного упаковочного комплекта. Вагоны сопровождения предназначены для размещения технического персонала сопровождения перевозчика груза ОЯТ, персонала охраны, а также для размещения средств дистанционного контроля и поддержания температурных условий. Количество вагонов-контейнеров в составе специального поезда определяется, в основном, из задачи вывоза за один рейс годовой вырузки ОЯТ от 1—2 реакторов. Например, в состав поезда может входить до восьми вагонов-контейнеров ТК-6, обеспечивающих вывоз от двух реакторов ВВЭР-440. На это количество вагонов-контейнеров рассчитываются и технические средства в вагонах сопровождения.

Решение об использовании для перевозок ОЯТ специальных поездов определялось с одной стороны необходимостью обеспечения высокой степени безопасности перевозок ОЯТ, а с другой стороны специфическими особенностями перевозок и требованиями к таким перевозкам, которые обуславливают целесообразность использования специальных поездов и с технико-экономической точки зрения.

Использование специальных поездов для обеспечения безопасности естественно предпочитительнее. Это позволяет особенно тщательно проводить подготовку подвижного
состава всего поезда к рейсу и контролировать его состояние в ходе перевозки, что практически исключает возможность аварий из-за неисправности подвижного состава. Вследствие отсутствия в составе специального поезда других опасных грузов (горючих, взрывчатых веществ и т.п.) даже при наиболее частых железнодорожных авариях, как, например, сход с рельсов, последствия будут минимальными. Пропуск специального поезда осуществляется в первую очередь и за его движением осуществляется постоянный контроль, что приобретает особое значение для оперативной организации спасательных работ в случае аварии, даже при недееспособности сопровождающего персонала вследствие аварии. Большое значение имеет предупреждение случайного облучения лиц из населения по пути следования, и, в особенности, железнодорожного персонала на постоянных маршрутах перевозок ОЯТ, в частности, в пунктах перестановки колесных пар и передачи транспортных средств на границе СССР, при международных перевозках.

С технико-экономической стороны целесообразность использования специальных поездов определяется следующими особенностями.

Во-первых, маршруты перевозок являются определенными, в частности, они не должны проходить через крупные города, и требования по маршрутам с использованием специальных поездов выполняются.

Во-вторых, расстояния перевозок довольно велики, особенно при международных перевозках, и время нахождения вагонов-контейнеров в пути составляет значительную часть времени всего цикла-рейса. При международных перевозках время нахождения в пути дополнительно увеличивается вследствие необходимости выполнения процедур передачи груза ОЯТ из одной страны в другую и перестановки колесных пар. Исходя из этого, оборачиваемость транспортных средств при использовании специальных поездов не на много отличается от повагонных отправок.

Далее, технологические операции по приемке на АЭС вагонов-контейнеров, подготовке и загрузке контейнеров ОЯТ, подготовке контейнеров с ОЯТ к отправке, включая выход на стационарный тепловой режим, дезактивацию, радиационный контроль, оформление документации, дают возможность параллельно обслуживать сразу несколько контейнеров. С учетом возможности наличия на АЭС нескольких реакторов (для ВВЭР-440 два реактора размещены в одном здании, а на одной площадке до четырех реакторов) может вестись одновременная загрузка контейнеров специального поезда. Все это значительно уменьшает простой вагонов-контейнеров.

Так из опыта перевозок ОЯТ ВВЭР-440 следует, что если все операции по загрузке и отправке одного вагона-контейнера занимают 5—6 суток, то для четырех вагонов-контейнеров это время составляет 10—12 суток.

Особое преимущество, а практически необходимость, использования специальных поездов, обусловливается возможностью перевозок ОЯТ в контейнерах, требующих контроля и обслуживания в ходе перевозки, например, таких как некоторые упаковки типа В(М) в соответствии с Правилами МАГАТЭ. Так для вагонов-контейнеров ТК-6 при перевозках ОЯТ в водозаполненных контейнерах необходимо поддержание определенного теплового режима с помощью принудительного охлаждения или в некоторых случаях дополнительного обогрева контейнеров. В то же время, можно
отметить, что использование принудительного охлаждения в ходе проведения перевозок было незначительным. Это объясняется тем, что по требованиям Правил МАГАТЭ расчет теплового режима проводился для более жестких условий, чем при реальных перевозках. Например, расчет проводился на условия стоянки поезда, максимальную температуру окружающего воздуха +38°С, солнечную инсоляцию в течение 12 часов.

Исходя из принятых в СССР и в других странах—членах СЭВ требований по физической защите груза ОЯТ, как материала, попадающего под действие "Международной конвенции о физической защите ядерного материала", при перевозках ОЯТ необходима охрана. Это требование также легко выполняется при использовании специальных поездов.

В чисто экономическом отношении стоимость пропуска по железной дороге специального поезда из достаточно большого количества вагонов-контейнеров оказывается не больше, чем при их повагонных отправках. Следует учесть, что при повагонных отправках стоимость определяется количеством осей вагона, а используемые для перевозок ОЯТ вагоны-контейнеры имеют до 12 осей. То есть, например, специальный поезд из 8 вагонов-контейнеров, двух вагонов сопровождения и вагонов прикрытия имеет не менее 108 осей, что в пересчете на обычные грузовые вагоны составит 27 вагонов.

Таким образом, указанные особенности и требования к перевозке ОЯТ определяют технико-экономическую целесообразность использования для перевозок ОЯТ специальных поездов. В то же время, возможность долговременного планирования перевозок ОЯТ по конкретным срокам позволяет облегчить необходимую подготовку на железных дорогах к пропуску таких специальных поездов.

Опыт организации перевозок ОЯТ нашел отражение в разработанных в СССР в 1983 году и внедренных в действие "Основных правилах безопасности и физической защиты при перевозке ядерных материалов". В этих правилах определены требования безопасности к упаковкам и их испытаниям (в основном в соответствии с Правилами МАГАТЭ), а также даны указания по особенностям организации перевозок с точки зрения обеспечения безопасности, которые были рассмотрены выше.

Среди других особенностей по организации в СССР перевозок ОЯТ, которые отражены в указанных правилах, хотелось бы отметить требование по отсутствию снимаемого радиоактивного загрязнения наружной поверхности транспортных средств, выполнение которого исключает распространение радиоактивных веществ, в окружающую среду при перевозках. Накопленный опыт перевозок ОЯТ показал, что при организации перевозок имеется возможность без больших затрат предотвратить загрязнение наружной поверхности транспортных средств.

Особое внимание при разработке правил было уделено определению необходимых мероприятий по ликвидации последствий аварий, связанных с перевозкой ядерных материалов. С целью оперативного определения уровня радиационной опасности, возникающей при аварии, введена классификация аварий по трем степеням опасности. Кратко они могут быть определены следующим образом:
1 степень опасности — аварии, когда не происходит увеличения радиационного воздействия от упаковок с ядерным материалом, и как таковой радиационной аварии нет;

2 степень опасности — аварии, когда уровень излучения и выхода активности из упаковок возрастает и находится в пределах установленных для предвидимых аварийных ситуаций (в соответствии с Правилами МАГАТЭ);

3 степень опасности — аварии, когда уровни излучений и выход активности из упаковок превышают пределы, установленные для предвидимых аварийных ситуаций.

Учитывая, что в соответствии с правилами, перевозки осуществляются с сопровождающим персоналом, при аварии 1 степени опасности ее ликвидации осуществляется сопровождающим персоналом с участием соответствующих сил транспортных организаций. На крайний случай (например, выход сопровождающего персонала из строя) предусматривается использование аварийной карточки, которая в металлическом пенале находится в транспортном средстве, а также имеется у транспортных организаций.

Для ликвидации последствий аварий 2 и 3 степени опасности наряду с спасательными силами транспортных организаций требуется вызов специальных аварийных бригад. С этой целью все предприятия, осуществляющие отправку или приемку ядерных материалов, а также другие предприятия, определяемые министерствами, имеют аварийные бригады и соответствующие средства их технической оснастики.

Для аварий каждой степени опасности в правилах определены конкретные меры по обеспечению безопасности, порядок проведения спасательных работ, лица и учреждения, ответственные за их проведение.

В разработанных правилах определены требования по физической защите ядерных материалов в ходе перевозок, включая и международные перевозки. Так для всех перевозок ядерных материалов требуется наличие сопровождающего персонала, для некоторых видов ядерных материалов требуется охрана. При организации перевозок с точки зрения физической защиты, должны выполняться следующие требования: ограничение времени нахождения ядерного материала в пути; ограничение перевалок груза с одного вида транспорта на другой; избегание регулярных графиков движения; использование кодирования; подготовка транспортных средств; ограничение круга лиц, осведомленных о маршруте и сроках перевозки и др. В правилах определены и другие меры по физической защите при перевозке ядерных материалов, а также организации ответственные за их выполнение, что позволяет обеспечить уровень физической защиты не ниже, чем это требуется "Международной конвенцией о физической защите ядерного материала".

В заключении доклада следует отметить, что накопленный к настоящему времени опыт перевозок ОЯТ ВВЭР-440 подтверждает правильность основных принципов организации перевозок и положений действующих правил. С помощью специальных железнодорожных поездов было выполнено более 150 вагон-контейнерных перевозок. При перевозках не было зафиксировано ни радиационных аварий, ни радиационных инцидентов, ни ситуаций, которые могли бы привести к авариям, например, неисправности подвижного состава. Все это позволяет и на будущее рассматривать проведение
перевозок ОЯТ от АЭС СССР и других стран—членов СЭВ с использованием специальных железнодорожных поездов в качестве основного варианта перевозок. В то же время опыт перевозок определил и целесообразность некоторых изменений, например, переход с водозаполненных на газозаполненные контейнеры, которые более просты в эксплуатационном отношении. Эти изменения учтены в разработках конструкций контейнеров для перевозки ОЯТ нового поколения реакторов ВВЭР-1000.
OPERATION OF THE SWEDISH SEA TRANSPORTATION SYSTEM FOR SPENT FUEL

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Abstract

OPERATION OF THE SWEDISH SEA TRANSPORTATION SYSTEM FOR SPENT FUEL.

At PATRAM 83, the Swedish Nuclear Fuel and Waste Management Co. (SKB) presented a paper on "The realization of a sea transport system for radioactive material". Some information was also given on the very first experience of the transport of some 35 t of uranium between Sweden and France. Since 1 March 1985, M/S Sigyn has been sailing under the Swedish flag and operated by the Swedish shipping line, Rederiaktiebolaget Gotland. The Swedish AFR facility CLAB was put into active operation in July 1985, with a receiving capacity of 300 t of uranium per year and a storage capacity of 3000 t of uranium at present. During the first year of operation CLAB received 250 t of spent fuel. The paper describes experience based on 16 sea voyages and the reception in CLAB of about 80 casks with spent fuel from Swedish nuclear power plants. The CLAB facility is equipped with a cask maintenance workshop, where preventive maintenance, re-inspection and repairs are carried out.

GENERAL

The Swedish Nuclear Fuel and Waste Management Co. (SKB), which is owned by the four nuclear power utilities in Sweden, is in charge of the back-end of the nuclear fuel cycle. The company's responsibilities include research, planning, construction, operation and ownership of facilities and the transport system.

A central interim storage facility for spent nuclear fuel (CLAB) (Fig. 1) next to the Oskarshamn nuclear power plant, has been in operation since 1985. A final storage facility for low and intermediate level waste (SFR) is being constructed next to the Forsmark nuclear power plant and will be put into operation in 1988.

A complete transport system has been built up by SKB for carrying spent nuclear fuel and radioactive waste from Swedish nuclear power plants. As all these plants have their own harbour facilities, the spent nuclear fuel and radioactive waste can be transported by sea. The M/S Sigyn handles all sea transport duties from the stations to CLAB at Oskarshamn. As of 1988, it will also be used for transporting waste to the SFR at Forsmark. The ship will then be making about 30 trips a year between the power plants, CLAB and the SFR.
CLAB was commissioned in July 1985. Spent fuel and core components will be stored there for a period of about 40 years prior to disposal in a final repository somewhere in the Swedish crystalline bedrock. During interim storage in CLAB, the radioactivity and heat generation of the spent fuel decline considerably, facilitating handling prior to final disposal.

During transport, the fuel assemblies are stored in forged steel shielding casks, each weighing about 80 t and with a capacity of 3 t of fuel (uranium).

TRANSPORTATION SYSTEM

At present, the sea transportation system consists of ten transport casks for spent fuel, two casks for core components, three terminal vehicles for local road transport at the sites, and a specially designed ship, the M/S Sigyn (Fig. 2).

The system was put into operation in January 1983. Six transports (57 t) of spent fuel from Swedish reactors to La Hague (Cherbourg) were made during 1983, the central storage facility, CLAB, then still being under construction.
During its first years the ship sailed under the French flag, and was operated by a French shipping line, Chargeurs Réunis. Since 1 March 1985, it has been sailing under the Swedish flag and is operated by the Swedish shipping line Rederiaktiebolaget Gotland. M/S Sigyn is wholly owned by SKB.

Since the storage facility was put into operation in the middle of 1985, the transport system has been operated continuously. About 250 t of spent fuel has been transferred to CLAB, corresponding to over 1200 fuel elements, most of them from BWR plants. About 20% of the casks have been transported from the nearby Oskarshamn plant, and about 80% by sea from the other plants.

The transport cask used is the TN17/Mk2, which can hold 17 BWR assemblies or 7 PWR assemblies. It is positioned on a transport frame during transport. In the cargo hold of the ship there are lashing fittings and corner pieces in ten fixed positions for the transport frames. On land, the transport is handled by a terminal vehicle, which drives in under the transport frame and lifts the entire cargo unit hydraulically.

There are at present ten TN17/Mk2 casks in the system, which, according to the experience gained so far, provide sufficient transport capacity. There are also two special casks for core components (TN17/CC, a simplified version of...
the TN17/Mk2, without cooling fins and neutron shield). These are used for such items as spent fuel channels, boron temporary absorbers and BWR control rods, which are also to be stored in CLAB. However, most BWR fuel channels, and some PWR fuel components such as boron glass rod assemblies, are being transported and stored together with the fuel assemblies.

RECEIVING CAPACITY FOR SPENT FUEL

The nominal yearly receiving capacity of CLAB is 300 t. The working schedule of CLAB is 2 shifts, 5 days a week. To fulfil the capacity requirements with some margin, about 5 casks must be emptied and prepared in 2 weeks (10 working days). After less than one year of operation, it is obvious that the capacity requirements of the CLAB facility are well satisfied.

There are 12 reactors in Sweden, 8 of which have delivered spent fuel to CLAB. (The last 4 have not been in operation long enough.) The immediate need is thereby taken care of, i.e. there is now enough space in the plants' own pools to guarantee next year's refuelling. When all 12 plants produce spent fuel, about 250 t will be discharged every year. With the reception of 300 t in CLAB, there will still be capacity for a further reduction in the accumulated amounts of fuel at the plants.

TRANSPORT LOGISTICS AND TIME SCHEDULE

Normally, about 5 casks at a time are delivered to a power plant. These are filled with fuel in 8–12 days and returned to CLAB with the ship. In the meantime the other 4 or 5 casks are emptied in the CLAB facility and prepared for transport. Thus, directly after unloading of the filled casks at the Simpevarp harbour (CLAB site), the empty casks can be loaded on board, and the ship is ready to sail. One terminal vehicle at the power plant and at least one vehicle at CLAB must be available (Fig. 3).

The transport time schedule is prepared on an annual basis. During the first few months of CLAB operation some spare weeks were included, but today a schedule with one transport every 2 weeks is applied, with the exception of holidays, etc. This means a cycle time for each individual cask of maximum 4–5 weeks and each cask is used about 10 times a year.

Since the actual time at sea is about 24 hours in each direction, it is small compared to the total cask handling time.

In total, 20 of the 21 scheduled transports had been made by the end of May 1986, comprising 82 casks (Fig. 4). During the same period of time, 13 casks with used core components (compressed BWR channels) were received. Thus there are 70 PWR (about 32 t) and 1200 BWR (about 220 t) assemblies, and
13 component canisters stored in the underground storage pools of CLAB at the time of writing (May 1986).

In 1988 the final repository for low and intermediate level waste, SFR, will be in operation. The shielding waste containers will be transported with M/S Sigyn to Forsmark from the other power plants. There are (at least) three different kinds of waste containers, for waste of different sizes and radiation levels. The time schedule will be co-ordinated with the fuel transports to CLAB.

CASK HANDLING

Nuclear Transport Limited, NTL, was contracted by SKB to provide services and advice regarding cask handling and loading operations, and NTL has been present at the power plants during cask loading, thus helping the plant staff to build up their own experience. During the first months of operation in CLAB, NTL was also present during unloading.

There is a maintenance workshop for transport casks in CLAB where regular inspection and maintenance operations in accordance with the ‘Green Book’ are performed. So far 5 casks have passed their first ‘10-cycle-service’.
FIG. 4. Cask handling in the CLAB reception building.
The routines for the cask handling, i.e. the chain of operations from CLAB to ship to harbour to power plant and vice versa, have worked adequately. Each transport is governed by a transport message and there are people responsible for the casks and their documents during each step.

The radiation shielding and protective measures have also proved adequate. The radiation doses to the ship's crew, which are evaluated once a month, have been below the registration limit without exception (July 1985–April 1986). However, the fuel transported in the casks has been, on average, of relatively low burnup/long cooling characteristics, thus giving surface dose rates and especially cask temperatures far below design values for the TN17/Mk2 — typically, 2-14 kW heat generation (compared to a design value of 43.5 kW) and 0.03-0.15 mSv/h (compared to 2 mSv/h). Sometimes contaminated spots (i.e. spots with more than 40 kBq/m²) are found on smear tests after arrival. Typical surface temperatures during transport have been around 30°C.

The total time taken for transportation with the ship is far below the capacity of the ship. When waste transports to the SFR start in 1988, the utilization of the ship will increase, but the capacity will still be sufficient.

There have been no delays caused by equipment problems on the ship, and only minor delays caused by severe ice or wind conditions.
ADMINISTRATIVE ISSUES
LEGAL AND INSURANCE PROBLEMS RELATED TO THE SALVAGE OF RADIOACTIVE MATERIALS FROM THE SEA

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Abstract

LEGAL AND INSURANCE PROBLEMS RELATED TO THE SALVAGE OF RADIOACTIVE MATERIALS FROM THE SEA.

The costs involved in salvaging radioactive materials from the sea are usually not covered by nuclear third party liability insurance, whereas transport insurance normally covers such costs. However, in specific cases, such as high level waste transports, even transport insurance might not provide sufficient protection. The issue then for the party concerned, which may be obliged to salvage, becomes complex. According to administrative laws concerning public order and security in the Federal Republic of Germany (FRG), the owner of cargo which represents a disturbance to the public order can be obliged to remove such a disturbance (in this case, salvaging the radioactive materials). These laws are applicable not only when the cargo is to be salvaged within the territorial waters of the FRG, but also for salvage operations on the open seas when the public order and security of the FRG is, or might be, negatively affected. The latter application does not conflict with international law. However, it should be mentioned that according to international law, no state is allowed to enforce its own administrative laws against persons or organizations residing in a foreign state. In such cases, the foreign state must be asked to enforce the salvage obligation. The owner of the cargo usually cannot take recourse with respect to the salvage costs against such third parties as the shipping companies involved in the accident or the shipping agent because, according to international treaties, the liability of these parties for nautical negligence that causes damage to the transport goods is excluded. Recently, it has become possible to buy special insurance coverage for salvage costs.

1. INTRODUCTION

The sinking of the French freighter Mont-Louis near the Belgian coast in August 1984 raised awareness of the existence of a specific transport risk, i.e. the salvage of radioactive materials from the sea. As long as the salvage costs are covered by transport or nuclear third party liability insurance, this risk does not cause any special problems. However, these types of insurance (as will be shown in more detail) either might not provide coverage for the salvage at all or might provide insufficient coverage.
If, in such a situation, the cargo can and must be salvaged, the question that arises is which party, on what legal basis, is obliged to salvage. In addition, it must be examined how far the responsible party has recourse with respect to the salvage costs against third parties. The answers to these questions depend very much on the national laws applicable to the specific case. This paper will discuss these questions mainly from the standpoint of laws in the Federal Republic of Germany (FRG) and international law.

2. NORMAL INSURANCE COVERAGE

Normally the transport of radioactive materials is covered by two types of insurance, nuclear third party liability and transport insurance.

2.1. Nuclear third party liability insurance

The legal provisions of nuclear third party liability insurance, as well as the corresponding insurance, only apply when a nuclear ‘incident’ has occurred. A nuclear incident, in turn, only occurs when the containers in which the radioactive materials are transported are so damaged in the course of the accident that they start to leak. However, that occurrence is rather unlikely. Thus, one can say that nuclear third party liability insurance most probably will not provide adequate coverage for salvage costs. This situation has also been found to hold when the salvage is performed to prevent a nuclear incident [1].

2.2. Transport insurance

Transport insurance provides coverage in case of damage to or loss of the insured transport goods. The coverage, at least ‘all risk cover’, also includes salvage costs. The payment obligation of the insurer is normally limited to the replacement value of the radioactive materials, plus the container value and all freight and transport costs, including other dispatching expenses. However, with respect to salvage costs, the payment obligation might even go beyond that limit because salvage costs which are reasonable and which do not lead to success are insured in principle without limitation. The restriction ‘reasonable’ salvage cost means that at the beginning of the salvage any exceeding of the insurance limit is not expected. If, during the course of the salvage, additional costs arise which exceed the insurance limit, the insurer must also bear these costs. However, the insurer takes part in the decision on whether to continue the salvage operation. However, should a salvage operation be broken off without success, the insurer has to pay not only for the costs of the salvage, but also pay full compensation for the total loss in materials, the container, etc. The insurance limit might also be exceeded if a second event on the same journey leads to damage of the shipment. On the other hand, costs which have nothing to do
with the actual salvage of the insured shipment are not the liability of the insurer, e.g. when the insured shipment must be salvaged only because a sunken ship must be removed from navigable waters.

Transport insurance normally should provide sufficient protection with respect to eventual salvage costs, especially in view of the higher value of radioactive materials. However, in specific cases the salvage costs, which will always be of a certain minimum amount, might exceed the insurance coverage, as for example when the insurance value of a shipment is low because only relatively small quantities are transported. Another example is the transport of spent fuel or high level wastes, where the insurance value could also be quite low.

It should be pointed out that the actual danger involved in the handling of the different radioactive materials is not discussed. For the purposes of this paper, only those radioactive materials will be taken into account which, by general understanding, represent such a danger that they must be salvaged (for example, high level wastes or spent fuel). Furthermore, the question of the water depth up to which a salvage operation can and must be performed will also not be discussed, though it will be assumed that the salvage is possible and necessary to eliminate a dangerous situation. The scenario described in the following does not assume that there has been a sea accident. It also includes cases of force majeure, where no party is responsible for the sinking of the cargo.

3. THE OBLIGATION TO SALVAGE

The obligation to salvage is investigated in a threefold manner. First, it will be determined how far such an obligation can be imposed within the territorial waters of a state — in particular the Federal Republic of Germany. The same question will then be answered with respect to the open seas outside territorial waters. Finally, the problem of enforcing this obligation outside the territory of the enacting state will be discussed.

3.1. Salvage within territorial waters

Territorial waters vary from state to state, extending anywhere between 3 and 12 miles from the coastline (the territorial waters of the FRG are 3 miles), and are considered by all states as being part of their territory and consequently governed by their national laws [2]. Thus, the obligation to salvage must be determined on the basis of the laws of that state in whose territorial waters the salvage is to be performed. However, the specific legal basis, upon which such an obligation can be imposed, must be determined. The Atomgesetz (Atomic Law) of the FRG does not contain any such basis. Since the salvage of dangerous goods is a question of public order and security, the obligation to salvage can only be found in the corresponding administrative laws.
'Public order and security' of the territorial waters of the FRG is regulated in the Bundeswasserstraßengesetz (Federal Law for Waterways). According to Section 25 of this law, a party which has caused a disturbance to public order and security — an example could be the owner of a vessel involved in an accident with the consequent sinking of the cargo — can be obliged to cease causing the disturbance, i.e. in this case to salvage the radioactive materials. In addition, the owner of the property (here the nuclear cargo) which represents a disturbance to the public order is obliged to end the disturbance, regardless of whether or not the disturbance has been caused by another party.

As result of the foregoing, one can say that in the case of a cargo sinking by reason of force majeure, or for similar reasons, the owner of the cargo will be the only party to which the obligation to salvage can be assigned, whereas in the case of an accident at sea, there will be a question as to which of the two parties, the owner of the vessel or the owner of the cargo, has to salvage the cargo. In principle, the party which has caused a disturbance of public order and security should be addressed first. Only if such a party cannot be found, or if the party is financially not able to support the cessation of the disturbance, can this obligation be imposed on the owner of the property [3]. However, as far as the salvage of dangerous goods, in particular of radioactive materials, is concerned, it would seem appropriate, considering the specific nature of the goods and the danger deriving from these goods, that the owner of the cargo should, in the first instance, be liable for salvage. Moreover, the owner of the cargo will usually have the best knowledge of the real danger of the cargo and will thus be in the best position to arrange appropriate salvage measures.

It should also be mentioned that pursuant to Section 28 of the Bundeswasserstraßengesetz, the state, by itself, in the event of an emergency, can perform the salvage and then take recourse with respect to the salvage costs against the responsible party (the owner of the cargo) [3]. A final comment is that according to international law, no state is entitled to enforce its own laws against persons or organizations outside its own territory.

3.2. Salvage outside territorial waters

Since the open seas do not belong to any national territory, there is a question whether a salvage on the seas is required by international law. In addition, it is necessary to discuss how far national laws can be applied to such salvages. However, it must first be noted that obligations arising from international law are directed only against states since only they are legal subjects under international law.

Traditionally, international law did not contain an obligation to salvage. However, in recent times, particularly in the area of environmental protection, the liability of states has become a topic of increasing discussion in legal literature [4, 5]. In a very recent study, the issue of whether a state that lawfully provides permission to transport might be strictly liable for the harmful consequences arising in another state as a result of that permission was discussed [5]. It was concluded that while
there was a due diligence obligation of states to prevent or mitigate environmental damages, there was no strict liability of states for environmental damages. Although the author of this study has convincingly demonstrated that, at least at the present time, the liability of states for environmental damages is still quite limited, his study has also shown that in the area of environmental protection, international law is moving in the direction of state liability.

The application of national administrative laws to a salvage operation on the open seas is now considered. Such an application of national administrative laws to events occurring outside the national territory does not conflict with international law; in particular, it does not violate the principle of the territorial limitation of administrative law. In the past, this principle was, in fact, understood as being a limit on such application. Today such a limitation is no longer accepted, as is demonstrated by state enforcement of antitrust laws [6].

A different question is whether a limitation of such an application arises from the national law itself. Although Section 1 of the Bundeswasserstraßengesetz defines as waterways only internal bodies of water and the territorial waters [3], this does not exclude an application of the law to a salvage on the open seas when it is deemed to be protecting public order and security within the territory of the FRG. Even if the Bundeswasserstraßengesetz is considered to be inapplicable, that would still not mean that a salvage on the open sea would in no event be covered by the laws of the FRG.

Indeed, instead of the Bundeswasserstraßengesetz, which is a special law, general laws for public order and security could be applied. Since the rules of interest here are the same in both laws [7], these general laws do not have to be discussed in detail. However, the sole difference that should be mentioned concerns the public authority responsible for imposing the obligation to salvage. For enforcing the Bundeswasserstraßengesetz, a special water police authority is responsible [3], whereas the regular police authorities enforce the general laws.

3.3. Enforcement of salvage obligations

As long as the salvage obligation is enforced against a person or organization within the state which imposed the obligation, there are no special problems. However, if, for example, radioactive materials from a foreign power station sink in or just outside the territorial waters of the FRG, the competent authority of the FRG cannot enforce salvage obligations against the foreign power station. Only the competent authority of the foreign state to which the power station belongs can assign this obligation. This restriction derives from international law, specifically the principle of the territorial limitation of administrative law, according to which no state is allowed to enforce its own law within the territory of a foreign state [6].

Although in practice it can be expected that states will assist each other in the case of a reasonable salvage order, there is still the question of whether states are legally obliged to enforce a foreign salvage order. As already mentioned in connec-
tion with the discussion of the strict liability of states in the area of environmental protection, international law is moving in the direction of some state liability. Therefore, it could be argued, bearing in mind the due diligence obligation of states to prevent or reduce environmental damages, that the enforcement of a foreign salvage order (provided it does not conflict with national public order) is required by international law. Such a position should prevail at least in the case of the salvage of high level radioactive wastes, since here it is also possible to refer to the Convention on the Prevention of Marine Pollution by Dumping of Wastes and Other Matter, 1972 (commonly known as the London Dumping Convention), which prohibits the sea dumping of high level radioactive wastes [8].

4. RECOVERY

In cases where a third party, e.g. the owner of the vessel the cargo is transported in, the other vessel involved in a collision, or the shipping agent, is responsible for the sinking of the cargo, the owner of the cargo would most probably have to take recourse against one of these parties. However, the liability of these parties is so limited that a recourse would not make very much sense.

The liability of the owner of a vessel is determined by the Hague Rules and has been taken over into national laws, for example in the Handelsgesetzbuch (Trade Law) of the FRG since 1936 [9]. According to Section 485 of the Handelsgesetzbuch, the liability of the owner of a vessel for the cargo is limited in any event to DM 1250 per packaging or unit. In the case of nautical negligence by his employees even this liability is excluded. According to Section 607 of the Handelsgesetzbuch, the liability of the shipping agent is also restricted to these limits.

5. SALVAGE INSURANCE

Legal analysis has shown that for the payment of salvage costs of radioactive materials, a pragmatic solution is necessary. Only comprehensive coverage, in connection with the transport insurance, can offer such a solution at acceptable conditions.

It is recommended that transport insurance be extended to salvage costs that exceed the value of the shipment when the insurance value of the shipment is below DM 30 million. Under current transport insurance, such an amount is considered to be sufficient protection against salvage costs. According to several discussions with insurance companies, the insurance terms and conditions would be the following. For goods which are insured under such a contract, salvage expenses, including costs for salvage from the seabed, can be additionally insured. The preconditions are that the costs should be incurred as the result of an insured case of damage and are necessary because of government or international regulations and/or the instructions of public
authorities. If such regulations do not exist, then these costs will be the policyholder's own responsibility, subject, however, to prior consultation with the insurer. In fact, such coverage is in force for some transports by companies in the FRG and the United Kingdom.

REFERENCES


Abstract

RECENT DEVELOPMENTS IN THE AREA OF INSURANCE AND INDEMNITY COVERAGE FOR TRANSPORTATION OF RADIOACTIVE MATERIALS IN THE UNITED STATES OF AMERICA.

A continuing concern surrounding the packaging and transportation of radioactive materials is the availability of adequate compensation in the unlikely event of a serious accident. The paper will discuss recent developments in the area of insurance and indemnity coverage for such in the United States of America. At two earlier PATRAM symposia, held in 1980 and 1983, descriptions of the US Price-Anderson insurance-indemnity system were presented. Since then, there have been several developments of interest. Foremost among these is the fact that the United States Congress is considering whether to again extend Price-Anderson Act authority. Thus, 1986 is a pivotal year in terms of whether Congress will re-authorize this legislation before it expires on 1 August 1987.

Introduction

The Price-Anderson Act of 1957 established a comprehensive and unique system of private insurance and Federal Government indemnity for public liability that might arise from the use of radioactive materials in the United States. This system, which provides broad coverage for public liability associated with fixed nuclear facilities and associated transportation, expires on August 1, 1987. The United States Congress now is considering whether to extend the Act again. What expires on August 1, 1987 is only the authority to extend nuclear hazards liability coverage to new power plants licensed by the U.S. Nuclear Regulatory Commission (NRC) and new U.S. Department of Energy (DOE) contracts. Thus, with no new nuclear power plants now being ordered in the United States, the expiration date is of more immediate concern to DOE contractors than electric utilities whose existing plants would continue
to be covered if Congress fails to act. In fact, four important DOE prime contracts expire on September 30, 1987 (i.e., those for Hanford, Los Alamos, Lawrence Berkeley and Lawrence Livermore). Entities considering bidding on these contracts already are concerned that Price-Anderson may not be extended by the date the new contracts will be entered into. Without Price-Anderson coverage (or its equivalent), DOE contractors and especially subcontractors would be very reluctant to furnish transportation services and packaging materials or otherwise do nuclear business with DOE.

Scope of Risk From Nuclear Transportation Activities

Before examining liability coverage, it is useful to review the scope of risk from nuclear transportation activities. Even with the stringent transport safety requirements applicable to nuclear materials, there is a definite—and increasing—risk of liability for anyone involved with their transportation in the United States. This is not to suggest that the risk of a transportation accident is increasing, but only that the risk of liability in the unlikely event of an accident is increasing. Compensatory damage awards and environmental clean-up costs associated with the hazardous properties of various materials have been increasing dramatically in the United States over the last several years. The number of radiation injury claims likewise has been increasing. With the application of new latent toxic tort concepts that make damage recoveries easier, such as probability of causation, even more are expected. The January 1984 Supreme Court of the United States ruling in Silkwood v. Kerr-McGee Corp. portends an exposure to even greater punitive damages. (The U.S. Supreme Court's 5-to-4 decision in Silkwood allows juries of laymen to impose punitive damages—in effect, fines—even where an entity has been operating in full compliance with applicable Federal safety regulations.)

Price-Anderson Act

The Price-Anderson Act provides for coverage for public liability associated with nuclear material while at a covered nuclear facility or in the course of transportation to or from such a facility. Substantive tort law is left to the States (except when an incident rises to the level of an "extraordinary nuclear occurrence"). The United States Congress originally enacted the Price-Anderson Act for the dual purpose of (1) assuring that funds would be available in the unlikely
event of a serious nuclear incident, and (2) encouraging private industry to participate in the nuclear field. The Price-Anderson Act has been re-enacted twice since 1957 and amended several other times. The most recent re-enactment was at the end of 1975.

"Omnibus" Feature

The unique feature of the Price-Anderson system that makes coverage under it most desirable is that, when it applies, it covers "anyone liable" (except the United States Government) for "any legal liability arising out of or resulting from a nuclear incident." This so-called "omnibus" feature is similar to the channeling of all liability to the power plant operator in Western European countries. The omnibus feature would facilitate the handling of lawsuits and reduce costs by allowing for consolidation of the defense and avoiding cross-claims among defendants. This would be of great advantage to claimants (as has been demonstrated by the litigation arising from the Three Mile Island accident). There is coverage regardless of how liability of particular defendants (any one of whom might have very limited assets) is allocated by U.S. tort law, a system unique to nuclear applications.

Limitation on Liability

Additionally, the Price-Anderson Act now provides that the liability of all entities covered by it is limited to the amount of coverage provided by the system. This limitation-on-liability provision was upheld unanimously by the U.S. Supreme Court in 1978 in Duke Power Co. v. Carolina Environmental Study Group.

Section 170c Licensee Coverage

Historically, most attention in the United States has focused on Price-Anderson Act coverage for commercial nuclear facility licensees, especially nuclear power plant operators. That portion of the Act (principally Section 170c licensee coverage) is administered by the NRC. The total liability coverage for power plants and their associated shipments of nuclear materials is $665 million. (This is the amount as of June 1986; and, as described below, increases by $5 million each time a new power plant is licensed to operate.)
Nuclear Insurance Pools

NRC indemnity agreements (under Section 170c) or DOE indemnity agreements (under Section 170d), as discussed below, may be the sole source of funds for public liability for nuclear risks where there is not insurance from private sources. Private insurance, when applicable, can furnish either underlying or exclusive coverage. It is provided by either the two nuclear insurance pools (American Nuclear Insurers, the pool of stock companies, and the Mutual Atomic Energy Reinsurance Pool, the pool of mutual companies) or the conventional insurance market. As a general rule, the pools cover nuclear fuel cycle activities, while non-fuel cycle activities (which are not considered to involve a level of risk requiring a pooling arrangement) are covered by the conventional insurance market. The pools issue two principal types of nuclear energy liability policies both in amounts presently up to $160 million: the Facility Form, and the Supplier's and Transporter's Form (which is not part of the Price-Anderson system). A large portion of this insurance capacity is provided by reinsurers worldwide.

Utility Industry Retrospective Premium

In the case of liability associated with NRC licensed power plants, if the primary level of financial protection afforded by the plant's Facility Form ($160 million) were insufficient to pay all claims, power plant operators would be assessed up to $5 million per incident retrospectively for each reactor. This Industry Retrospective Premium provision (a form of enterprise liability) was added to the Price-Anderson Act in 1975 at the utilities' suggestion for the purpose of substantially increasing the amount of financial protection (for power plants only) afforded by private sources. The amount of power plant coverage now is $160 million under the Facility Form plus $505 million under the Retrospective Plan (i.e., 101 nuclear power plants operating as of June 1986 times $5 million each) for a total of $665 million.

Section 170d Contractor Coverage

The other principal kind of Price-Anderson coverage is that issued by DOE under the Section 170d contractor provision. That subsection expressly authorizes DOE to indemnify its contractors against public liability in the event of a "substantial" nuclear incident. The Section 170d indemnity can be for
up to $500 million for each nuclear incident occurring inside the United States and up to $100 million for each incident occurring outside the United States, and, when extended, brings into play the limitation-on-liability provisions of Section 170e of the Price-Anderson Act. Coverage under a DOE nuclear hazards indemnity agreement is substantially the same as that afforded under the pools' Facility Form policy.

Incidents Outside the United States

Indemnification of DOE contractors for a nuclear incident occurring outside the United States specifically is authorized by Section 170d, but, as noted above, the maximum amount of Federal Government indemnity under Price-Anderson for such an incident now is $100 million. There also are other differences about coverage under Section 170d (not all of which are applicable to Section 170c licensee coverage) for incidents outside the United States: The class of persons eligible for indemnity coverage is smaller. There is not omnibus coverage for "anyone liable". Coverage for incidents outside the United States extends only to the prime DOE contractor with the indemnity agreement, subcontractors, suppliers of any tier, and others whose liability arises by reason of activities connected with such contract or subcontracts. The special features of the "extraordinary nuclear occurrence" provision (discussed below) do not apply to a nuclear incident occurring outside the United States.

"Extraordinary Nuclear Occurrence" Provision

An often misunderstood feature of the Price-Anderson system is the "extraordinary nuclear occurrence" (ENO) provision. The ENO provision was added to the Price-Anderson Act in 1966 for the purpose of further assuring prompt compensation to the public for serious nuclear incidents without at the same time totally displacing state laws by the creation of a "federal tort". The 1966 amendment provides that, in the event of an ENO, certain ordinarily available state law defenses are waived. Congress did not wish to make these provisions applicable to all nuclear incidents for fear of encouraging nuisance suits. Determination as to whether an incident was an ENO is made by NRC or DOE on the basis of predetermined criteria. (The only case in which an ENO determination previously has been made was the Three Mile Island accident. NRC determined that, while that event was "extraordinary" in ordinary parlance, it was not an ENO.) It is not necessary that
an ENO determination be made for coverage under the Price-Anderson system to apply.

Current Congressional Activities

Price-Anderson extension now is being considered actively by the United States Congress where at least seven committees have jurisdiction over such legislation. Price-Anderson is probably the single most important and controversial nuclear energy issue facing the 99th Congress. 1985 was a record-building year, with Congressional committees holding several hearings. 1986 is a pivotal year in terms of whether Congress will reauthorize the legislation before its provisions for new coverage expire. However, the 1986 Congressional schedule includes a number of recesses and the early adjournment target typical of election years. The Federal Government budget still is occupying much of the Members' attention. Additionally, there now is a new factor that may have important effects on Price-Anderson extension, i.e. the Chernobyl power reactor accident. At this point, however, it is not possible to predict precisely what impacts the recent Soviet accident will have on public perceptions in the United States or on American Congressional actions.

Most of the Price-Anderson bills before Congress do not address transportation coverage directly (except, in certain cases, with respect to DOE waste management activities), but all would affect such coverage. In the hearings and bill markup sessions to date, the key issues have been the limitation on liability, subrogation (e.g., a right of action back against an entity whose "gross negligence" or "willful and wanton misconduct" contributed to a loss), punitive damages, coverage for DOE waste management activities and use for the latter of the Nuclear Waste Fund (paid by utilities to DOE for spent fuel disposal).

In the Senate, most attention has focused on S.1225, which was introduced by Senators Alan K. Simpson of Wyoming, Majority Whip and Chairman of the Nuclear Regulation Subcommittee of the Senate Environment and Public Works Committee, and James A. McClure of Idaho, Chairman of the Senate Energy and Natural Resources Committee. The Simpson-McClure bill would retain the basic features of the present system, but increase the limitation on liability to over $2 billion. On April 24th, the Senate Energy Committee reported out a modified version of S.1225. The bill as reported by the Energy Committee would provide coverage of about $2.4 billion for both NRC power plant licen-
sees (up from the present figure of $665 million) and DOE contractors (up from $500 million). For power plants, the bill would require a retrospective premium of up to $20 million, with an annual adjustment for inflation. It would make coverage for DOE contractors mandatory for all "nuclear incidents", whether or not the risk is "substantial". The bill also contains an amendment that creates a new discretionary civil penalty of up to $10 million for DOE contractors, if an incident is the result of "gross negligence or willful misconduct". The bill would make more explicit DOE's authority to cover waste management activities (specifically including transportation) under the Nuclear Waste Policy Act of 1982 and the Waste Isolation Pilot Plant authorizing legislation. The Nuclear Waste Fund would be used for incidents involving DOE waste management activities (including transportation of spent fuel from power plants). The Energy Committee bill also includes an amendment that would bar awarding punitive damages "under State law" in actions against persons indemnified by the Federal Government (i.e., DOE contractors, indemnified NRC licensees required to maintain less than the maximum amount of financial protection (now only plutonium fuel fabricators), and nonprofit educational institutions). There is a new provision requiring coverage for "precautionary evacuations", but only if certain conditions are met. (Whether there would be coverage in the event the conditions were not met is an open question.) The "extraordinary nuclear occurrence" provisions are made applicable to DOE waste management contracts. The Senate Environment Committee now must complete its action by August 21st or lose jurisdiction over the bill.

In the House of Representatives, the Interior and Insular Affairs Committee on May 21st reported out a modified version of a bill (H.R.3653) introduced by its Chairman, Morris K. Udall of Arizona. H.R.3653 as reported would increase coverage and the liability limitation to about $6.5 billion. This "compromise" figure would be raised by increasing the first layer of private insurance to $200 million and then by assessments of $63 million per plant per incident (with no more than $10 million payable in any one year). The reported bill provides for unlimited liability for various DOE nuclear waste activities, with the first $6.5 billion coming from the Nuclear Waste Fund. Other House Committees, such as Energy and Commerce, and Science and Technology, probably will receive sequential referrals of this legislation. In this regard, note that nuclear power plant operators have been strongly opposing any new Price-Anderson limit much
above $2 billion. Thus, with the legislative focus now shifting to Committees generally less sympathetic to nuclear issues, strong efforts will be needed for Congress to adopt a favorable Price-Anderson bill this year.

A possible schedule for further Congressional action on Price-Anderson extension this year is as follows: Depending upon whether other House of Representatives Committees (including the Energy and Commerce Committee, and perhaps the Science and Technology Committee, the Armed Services Committee, and the Ways and Means Committee) each seek sequential referrals of the House Interior Committee bill (presumably with time limits), House floor action and passage could take place this Summer. The Senate Energy Committee reported its bill on April 24th, so the Senate Environment Committee now has until about August 22d to complete any action on a Price-Anderson bill. As a practical matter, this means it must complete its action by August 15th, the day on which the Labor Day recess now is scheduled to begin. Following its second hearing in mid-May, the Environment Committee's Nuclear Regulation Subcommittee has been planning to begin to markup a bill this month. Action by the full Senate Environment Committee and the full House this Summer could allow for final Congressional action before both Houses of Congress rush to adjourn in October. (Or, if there is a "lame-duck" session after the November elections for the new Congress, there will be more time for action this year.) The President could sign the bill (which would enact it into law) shortly thereafter. If this schedule slips, extension of Price-Anderson could be in serious jeopardy. Next year's Congress would not have to begin over, but might not be disposed to act quickly enough to have an extension in place by August 1, 1987.

Conclusions

The changes in the Price-Anderson Act being considered by Congress could have significant impacts on nuclear transportation activities in the United States. In addition to contentious issues of coverage for DOE nuclear waste management activities, punitive damages and subrogation, renewed consideration by the Congressional committees yet to act is expected to be given to other controversial issues such as covering acts of sabotage, lowering the "extraordinary nuclear occurrence" threshold to make the waiver of defenses apply to all nuclear incidents, and extending the present 20-year statute of limitations applicable to
ENO's. Efforts to increase the liability limitation beyond even the $6.5 billion figure recently passed by the House Interior Committee are expected. This could cause utilities to withdraw support for any extension. Potential impacts of the Chernobyl accident on Congressional actions also must be considered. It thus is important to remember, in evaluating present and future nuclear transportation programs in the United States, that major modifications actively are being promoted and extension of the Price-Anderson Act is not assured (even for government contractors).
RAILROADS AND SPENT FUEL
IN THE UNITED STATES OF AMERICA

A new era

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Abstract

RAILROADS AND SPENT FUEL IN THE UNITED STATES OF AMERICA: A NEW ERA.

Passage of the Civilian Nuclear Waste Act of 1982 is perceived by the railroad industry in the United States of America as the beginning of a new era in co-operation between the 'nuclear' and railroad industries with regard to shipping of spent fuel by rail. Despite a decade and a half of meetings and legal disputes, the railroads continue to have concerns about moving spent fuel by rail. However, the railroad industry is optimistic that it will be able to work effectively with the U.S. Department of Energy's Office of Civilian Radioactive Waste Management (OCRWM) in establishing a rail transportation system capable of safely moving the spent fuel from the reactors to the repository.

INTRODUCTION

Passage of the Civilian Nuclear Waste Act of 1982 represents the beginning of a new era in co-operation between the nuclear and railroad industries in shipping spent fuel by rail. Over the last decade and a half attempts to work out the differences between the two industries have led to numerous meetings, hearings before the Interstate Commerce Commission (ICC) and legal disputes in the court system. Despite all of this, the basic problem has not been solved, for the railroad industry still has reservations and concerns regarding the safe movement of these materials by rail. The railroad industry does, however, perceive the 1982 Act as an opportunity to lay to rest the conflicts of the past and to work with the U.S. Department of Energy's (DOE) Office of Civilian Radioactive Waste Materials (OCRWM) to develop a safe system for transporting spent fuel, a system that will meet the needs of all concerned parties.

BACKGROUND

The railroad industry in the United States of America is almost entirely privately owned and operated. It is an industry of huge proportions, with assets including approximately 26 000 locomotives, 1.5 million freight cars and over 410 295 km
(255 000 miles) of track. The investment in plant and equipment is in excess of US $43 thousand million. In 1984, the railroad industry generated over US $30 thousand million in revenues; unfortunately the return on investment was only 5.71 per cent.

In 1984, the US railroads moved over 20 million carloads of freight. By contrast, it is expected that only between 500 and 600 carloads of spent fuel will be tendered per year beginning in 1998. Whatever the number of carloads, it is a very small potential market. However, regardless of market size, the railroads are interested in each and every carload of traffic, provided it is possible to:

— Recover the actual cost of the move
— Make a reasonable profit
— Cover the risk involved in the move.

It is the uncertainty regarding the determination of risk that has caused so much concern and anxiety within the railroad industry.

The railroad industry first expressed its concerns regarding the transport of spent fuel by rail in the late 1960s. In the early 1970s, the Board of Directors of the Association of American Railroads (AAR) recommended to the member railroads that spent fuel shipments be carried only in special trains travelling not faster than 56 kph (35 mph). They believed that these restrictions would keep the thermal and mechanical forces generated in a train accident within the limits established by the cask design criteria.

When the railroad industry refused to carry spent fuel shipments unless the move was in agreement with these restrictions, they were ordered to appear before the Interstate Commerce Commission (ICC). The ICC determined that it could not hear questions of safety, the domain of the Department of Transportation (DOT), and consequently found the tariffs published by the railroads to be excessive and rescinded them. As a result, the railroad industry is allowed to charge only regular train rates for the movement of spent fuel, unless the shipper requests special or dedicated nuclear train movements.

CONCERNS

The concerns of the railroad industry are really not that much different today than they were ten or fifteen years ago. Despite all of the meetings, workshops, and discussions, little has changed, with perhaps one exception: the Civilian Radioactive Waste Act of 1982 and the consequent establishment of the Office of Civilian Radioactive Waste Management (OCRWM). Passage of the Act and the direction that OCRWM is taking are viewed as positive measures in solving the problems facing the railroad industry in moving spent fuel.

The Act mandates that a repository for the spent fuel be established and that OCRWM be responsible for the development of a transportation system to safely
deliver the spent fuel either to a monitored retrieval storage area (MRS) or directly to the repository. To assist in the development of the transportation plans, OCRWM has established a working relationship with the railroad industry. So far this working relationship has proved to be productive and it is expected to help alleviate many of the railroads' concerns regarding the movement of spent fuel by rail.

The AAR has recently established the Committee on the Movement of Spent Fuel by Rail, consisting of experts from major railroads. Their assistance will be invaluable in defining the concerns of the railroads and in assisting in the evaluation of proposed railcar/cask designs. While the railroad industry recognizes and applauds the effort that has been directed toward preventing spent fuel accidents, they still have concerns regarding the safe movement of spent fuel by rail. For the purposes of this paper, these concerns are:

1. Accident prevention
2. Accident response
3. Public perception
4. Liability.

**Accident prevention**

There are two primary areas of concern. (1) How the railcar/nuclear cask will behave as a system when operating within the railroad environment and (2) the ability of the cask to withstand the mechanical and thermal forces generated in a railroad accident.

Whereas the railroad industry does not claim expertise in the design of spent fuel casks, it does have experience in the design, construction and operation of loaded and empty railcars. Thus, the OCRWM discussed the possibility of having the AAR's Transportation Test Center (TTC) become involved in a research and test programme involving the proposed new railcars. These discussions led to the signing of a contract earlier this year.

Under this contract, TTC will furnish technical reviews of the proposed railcars and of the railcars/casks as a system, develop railroad-related performance specifications, evaluate potential railcar surveillance and alert systems, study the feasibility of 'overweight' railcars and assist in the certification programme. Involving personnel from the railroad industry in the planning stage should eliminate potential high-risk problems associated with the development of a new railcar/cask system.

Another major concern is the integrity of the cask itself. Will it be able to withstand the mechanical and thermal forces generated in a train accident? Although the spent fuel cask is perhaps the best shipping container that the railroads currently transport, there is concern about the integrity of the cask. The railroads are aware of the test criteria and are confident that the casks will pass the certification requirements. However, two problems remain. First, how can the railroads be sure that the test requirements accurately portray the actual forces that might be generated in a train accident? Second, what are the failure thresholds, both mechanical and thermal,
of the cask? The railroads would be less apprehensive about moving spent fuel by rail if there were answers to such questions as the relationship between the mechanical and thermal forces generated in drop/fire tests and 'real world' accidents and how these forces compare with the failure threshold of the cask.

Will the new generation of casks, designed to carry only fuel that is at least five years old, be as safe as the older casks that were designed to carry 90-day old spent fuel? Knowledge of the failure thresholds, and not just that the cask passed a 9 m (30 ft) drop onto an unyielding surface, would provide additional valuable information for the railroad industry to use as a basis for making important operating decisions. Comparing actual forces required to breach a cask with the forces that could be generated in a train wreck would provide the operating railroad company with the information needed to move the spent fuel with confidence.

Accident response

The railroad industry is the only transportation mode which owns the right-of-way over which it operates. Consequently, unlike truckers who can easily detour around a highway closure, railroads are faced with the problem that any obstruction to the track will stop the movement of trains through that territory and can result in large losses in revenues. In addition, since it owns the right-of-way, the cost of clearing obstructions or cleaning up spills must be borne entirely by the railroad.

Concern over emergency response is not a new issue. It has been the subject of much discussion over the years and was formally identified as a major issue by the 1981 DOE/AAR workshop on the shipment of spent fuel by rail. As a result, the following DOE/AAR workshop, held in 1983, had, as its sole subject, emergency response for rail accidents involving nuclear materials. At that workshop, representatives from Federal, state and local governments worked with railroad officials and shippers to define existing emergency response capabilities and where additional capability was needed. It became apparent at that workshop that the responsibilities and capabilities of the various governmental agencies are still not clear and considerable work remains to be done to clear up this confusion.

Past experience with emergency response to accidents causes railroads to pose the following questions:

1. Who from the government would respond to an accident involving nuclear materials?
2. Who would be in charge?
3. How long would it take for meaningful decisions to be made?
4. What kind of training should be provided to first responders?
5. Who is qualified to provide that training?
6. Are there firms both willing and capable of clearing a train wreck that has been contaminated by radioactive materials?
Public perception

Concerns regarding public perception can be categorized into two major areas: how the public, management and labour perceive the problems associated with radioactive materials and the impact public perceptions can have on the operation of the railroad following an accident.

The railroad industry works with local and state authorities on a daily basis. To be credible spokesmen regarding the transport of radioactive materials requires having access to information that is relevant and easily understandable. It is thus important that measures of the real risk associated with the movement of these materials be provided in terms that can be easily understood by all. By knowing what the real risks are, ways can be found so that shippers, carriers and the public can work together to develop a transportation system that will meet everyone's needs.

The second concern of the railroad industry regarding public perception has to do with what it can end up costing the carrier because of actions by governing bodies. How does this work? Since the railroads in the USA are privately owned, this also includes ownership of the right-of-way. Any loss of the use of right-of-way, for any reason, natural or man-made, will result in that railroad losing large amounts of revenues if it cannot detour around the problem area. The concern here is that long term loss of the use of right-of-way may come about because of public perception of the risks involved with nuclear materials.

This concern of loss of the use of the right-of-way is not just an idle thought, for there are several case histories. The most recent occurrence involved the Burlington Northern (BN) Railroad in October 1985. The accident occurred on a Tuesday afternoon, when a truck carrying uranium hexafluoride collided with the side of a BN locomotive. The driver was killed and, although the tanks were thrown from the truck, no release of product occurred. However, because the accident involved radioactive materials, it was late Friday before Burlington Northern was allowed to enter its own right-of-way and return it to service. Consequently, that section of the BN track system was out of service from Tuesday through most of Friday because of public perception regarding the extent of the accident. Few, if any, railroads would be able to financially withstand an extended shutdown of their mainline.

Liability

The railroad industry is still very uneasy about the problem of liability associated with the transportation of spent fuel. The railroads have been hit by all of the problems associated with obtaining liability insurance today. Most railroads are self-insured and carry only major catastrophic insurance. Apparently, even that is becoming difficult to obtain.

It is claimed that the railroad industry is covered by the Price-Anderson Act, yet some people contend that this is not so. Even if covered, Three Mile Island has
taught that large expenses can be incurred in the event of a ‘nuclear accident’ even if no release occurs and (therefore) Price-Anderson does not apply. If, as in the BN case, there is an accident with no release of product and Price-Anderson is not actuated, the carrier would still be liable for all of the costs. Until such time as all of these factors are clear, a railroad cannot reasonably be expected to prepare a rate for transporting spent fuel. As stated earlier, the cost of each movement must contain an amount to cover the actual cost of the move, an amount for profit and an amount to cover the risk associated with the move. We need to know more about the risks, but we also need to know exactly what type of liability coverage is provided by the shipper when moving spent fuel by rail. To cover their responsibility to the shareholders, the managements of each railroad must know the extent of the liability being assumed whenever a spent fuel shipment is made over their railroads.

SUMMARY AND CONCLUSIONS

If the railroad industry was confident that it could move spent fuel and make a profit doing so, there would have been no need to list all of the concerns and reservations regarding such movements. However, the railroads do have legitimate concerns and reservations regarding the movement of spent fuel. Four primary areas of concern briefly discussed in this paper are:

1. Accident prevention. The primary concerns are how the railcar/cask will behave as a system and what are the failure thresholds for both the mechanical and thermal forces required to breach the cask.
2. Accident response. The responsibilities of various governmental bodies and agencies need to be clarified and assistance is needed in finding firms willing to clean up a train wreck that has been contaminated by radioactive materials.
3. Public perception. The railroad industry must be provided with clear answers to questions in order that a credible case can be presented to the public.
4. Liability. What type of liability is covered by others and what liability is assumed by the railroad when it is transporting spent fuel?

The railroad industry has legitimate concerns regarding the movement of spent fuel by rail. These concerns must be addressed, if the railroads and shippers are to become willing partners in moving the spent fuel to the repositories. The railroad industry is optimistic that it will be able to work effectively with the DOE Office of Radioactive Waste Management in helping to develop a safe transportation system capable of meeting the needs of the permanent waste disposal system.
EXPERIENCE OF TRANSPORTATION OF RADIOACTIVE SAMPLES FOR ANALYSIS

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Abstract

EXPERIENCE OF TRANSPORTATION OF RADIOACTIVE SAMPLES FOR ANALYSIS.

Independent verification of the concentration of special nuclear materials in the input solution, plant discards and product samples constitutes an important element of the state system of accounting or international safeguards for a reprocessing plant. This requires that the samples collected at the reprocessing plant are sent to the referee laboratory for verification analysis. Safe handling procedures and packaging codes have been evolved and are being kept constantly reviewed. During the reprocessing campaigns carried out at Prefre, India, under IAEA safeguards, a number of samples from stipulated stages of the process were sent to the Agency in discrete consignments. It was observed that verification analysis at the Agency's Analytical Laboratory involves considerable delay due to various procedural and logistic constraints in the transportation and co-ordination. This delay has certain implications for the verification system, the most notable being the problem of ageing of the samples. The paper describes experience in various aspects of transportation of samples, highlighting detrimental effects of ageing on the determination of elemental and isotopic composition in such samples. Corrective measures to be taken to compensate for these detrimental effects are also described in the paper.

1. INTRODUCTION

Reprocessing involves handling of special nuclear materials such as uranium and plutonium. Because of their extreme toxicity and also because of their strategic importance, the facility handling these materials maintains an accurate accounting system. In order to make the facility's material balance statement based on its measurements more credible, an independent verification of the plant's inventory is used. In the reprocessing campaigns involving IAEA safeguards, the independent verification is provided by the Agency's Laboratory. For this purpose, samples of the plant's input, discards and products are shipped to the safeguards Analytical Laboratory in Vienna. Air shipment is the commonly accepted mode of transportation of these samples. In spite of this, it is found that there are considerable delays...
in getting the samples analysed at the verification laboratory. These delays have two major implications:

(1) Because of the delay, a timely verification of the plant’s inventory is not possible.

(2) Delays introduce uncertainties in the verification procedure due to the problem of ageing of the samples.

Various aspects of the transportation of radioactive samples to the IAEA with special emphasis on the problem of ageing of the samples due to delays in transportation are discussed in this paper.

2. SHIPMENT OF SAMPLES

During the reprocessing campaigns carried out at Prefre, India, under IAEA safeguards, a number of samples from stipulated stages of the process were sent to the Agency in discrete consignments. The steps involved in the shipment of these samples are shown in Fig. 1.

2.1. Sampling and aliquoting

Samples drawn from different key measurement points of the plant are aliquoted and given chemical treatment in shielded cells and glove boxes or fume-hoods. Details of the samples shipped are given in Table I. The input samples are diluted to nearly 150 times to bring down the associated fission product activity and hence the personnel exposure while handling the package. While the product samples are already in solid form, input and discard samples are also dried before being packed.
TABLE I. DETAILS OF SAMPLES SHIPPED FOR VERIFICATION ANALYSIS

<table>
<thead>
<tr>
<th>No.</th>
<th>Inventory stratum</th>
<th>Nature of sample</th>
<th>Approximate amount of material in the vial</th>
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<td>Uranium</td>
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<tr>
<td>1.</td>
<td>Input</td>
<td>Dried liquid</td>
<td>1.5 mg</td>
</tr>
<tr>
<td>2.</td>
<td>Discard</td>
<td>Dried liquid</td>
<td>0.1 mg</td>
</tr>
<tr>
<td>3.</td>
<td>Product (Pu)</td>
<td>Solid</td>
<td>300 µg</td>
</tr>
<tr>
<td>4.</td>
<td>Product (U)</td>
<td>Solid</td>
<td>10 g</td>
</tr>
</tbody>
</table>

Radiation level outside TNB-0145 container with about 60 samples: 1–5 mR/h (on contact).

2.2. Packaging

Suitable aliquots of samples are collected in glass vials which are loaded into the canisters. Each loaded canister is sealed in a PVC bag before being removed from the glove box to keep its external surface free of contamination. The canisters are packed into the container drums and sealed. Packaging is done in accordance with the codes and practices prescribed for each type of container. The sealed container is checked for loose contamination and radiation level before it is cleared for transportation. Packages come under yellow category II and transport index 0.5. Certified containers provided by the Agency, namely TNB 0145 and PAT-2 are used for transporting the samples. The TNB-0145 design conforms to the IAEA Safety Series No. 6, 1973 (revised 1979). The package is transported as Type B(U) fissile Class 1. This package can accommodate 50–60 vials in layers one over the other in fixed grooves in the inner container. The PAT-2 design meets all the requirements of NUREG-0360 under stringent accident modelling conditions and qualifies as fissile Class I package. Certification of both the containers for national regulatory requirements is carried out by the Division of Radiological Protection, Bhabha Atomic Research Centre.

2.3. Documentation

All relevant information pertaining to the shipment is carefully documented for reference by the facility and the Agency. Necessary freight documents such as the government clearance certificate, duty excise exemption certificate, dangerous goods certificates, etc., are also prepared.
2.4. Transportation

As the facility is located about 100 km away from the international airport, shipment involves transportation by road before the consignment is air freighted to Vienna. During inland transportation the consignment is provided with a security escort for the purposes of physical protection. The package is handed over to the airlines after the necessary shipment formalities have been completed.

3. DELAYS IN SHIPMENT

It has been observed that verification analysis at the Agency’s Analytical Laboratory involves considerable delay due to various procedural and logistic constraints in the transportation. The factors contributing to the delay can be traced to three sources.

3.1. Pre-shipment delay at the facility

The economics of transportation dictate that a sufficient number of samples are collected for each consignment. This results in delays in sending the samples. Non-availability of transport containers at the proper time also contributes to the pre-shipment delays. Compared to these delays, the time taken for sampling, packaging and clearance is insignificant.

3.2. Transit delay

The major factors contributing to the delays in transit are procedural problems associated with: (a) awarding inland transport contract; (b) getting necessary clearances from various national and international authorities; (c) finding an air carrier cleared to accept the freight.

3.3. Post-arrival delay at the Agency

The long delay at the Agency is presumed to be due to the excessive work load of the verification laboratories and to problems in co-ordination with network laboratories.

3.4. Prefre experience

Results of a study conducted at random on three consignments sent from Prefre to the Agency are shown schematically in Fig. 2. It can be seen that on average it takes 40–50 weeks to get the results of verification analysis from the Agency. Of this, 18% of the delay is due to shipment problems and 52% is at the Agency end.
3.5. Consequences of delays

The delay in getting the analysis results from the Agency has two major implications for the verification system. Firstly, the important objective of the verification system, namely, timeliness of detection of discrepancies in the inventory, is not met. The facility is forced to preserve a large number of archive samples till the reconciliation of results. This creates storage and radiation problems for the facility. Secondly, delays introduce uncertainties in the accuracy of the verification procedure due to sample ageing.
4. AGEING OF SAMPLES

During the long time interval between sampling at the facility and analysis of the samples at the Agency Laboratory, the samples can undergo several changes collectively referred to as ageing. Unless proper corrective measures and precautions are taken, changes taking place in the samples due to ageing can detrimentally affect the accuracies of verification analysis. The corrective measures to be taken depend on the nature of changes in the sample. They include special treatment to the samples before analysis and application of correction factors to the results obtained to nullify the systematic error caused by changes in the sample.

4.1. Physical changes

This effect is most pronounced in the case of product samples which are shipped in oxide form (U₃O₈, PuO₂). Since these are reactive compounds they have a tendency to absorb moisture. This results in a change in the weight of the sample. As only a portion of the shipped sample is taken for analysis, any change in the weight of the sample in transit introduces a systematic error in the results, the magnitude of which is directly related to the time delay. This error can be taken care of in one of the following three ways:

(i) Bring into solution the entire contents of the vial. Accurately measure the weight or volume of the solution obtained. Use an accurately aliquoted portion of this solution for analysis.

(ii) Heat the contents of the vial to a constant weight before aliquoting the sample for analysis.

(iii) Accurately find the weight of the vial with the sample before analysis and apply a correction factor to the result based on the change in weight of the sample. For example,

\[
\text{Actual Pu content in a sample} = \frac{\text{Pu content at the time of analysis}}{\frac{\text{Wt. of sample at verification laboratory}}{\text{Wt. of sample at the facility}}}.
\]

4.2. Chemical changes

As a result of radiolysis, the chemical state of plutonium in liquid samples can undergo several changes. These include changes in the oxidation state of plutonium (from the Pu⁴⁺ state to Pu³⁺ or Pu⁶⁺ state) and changes in its chemical identity (from the ionic to hydrolysed or polymeric form). Traces of organic material present (in waste samples) can also complex the plutonium ions. These changes become more pronounced as time passes.

The mass spectrometry technique, which is the method normally applied for the accurate determination of plutonium in these samples, involves a vital step in
which the plutonium ion in the sample undergoes isotopic equilibration with a plutonium spike tracer added to the sample. Any change in the chemical composition of plutonium in the sample will adversely affect this isotopic equilibration and will lead to a bias in the measurement results.

Corrective measures for this effect include:

(i) Shipping the sample in dry form after mixing it with the spike tracer.
(ii) Repeatedly evaporating to dryness the mixture of sample and spike with concentrated nitric acid before shipment.
(iii) Treating the dry sample spike mixture with a mixture of nitric, perchloric and hydrofluoric acids before analysis at the verification laboratory. This treatment helps in restoring the chemical identity of plutonium in the sample by effectively depolymerizing or dehydrolysing or decomplexing it.

Some results of assay of a process sample, analysed immediately after and one month after sampling with and without such treatment (Table II) amply illustrate this point.

4.3. Radioactive decay

Radioactive elements undergo decay and in course of time change into entirely different species. Therefore, concentrations of these elements in a sample measured at different times significantly differ. The magnitude of the difference depends on the abundance of the particular isotope in the sample, its half-life and the time delay in measurements. The process solution sample in a reprocessing plant treating a PHWR fuel irradiated to about 6000 MW·d/t contain, among other isotopes of plutonium, significant amounts of $^{238}$Pu and $^{241}$Pu (0.1 and 3.85 wt.%, respectively); $^{238}$Pu decays to $^{234}$U by alpha emission and $^{241}$Pu decays to $^{241}$Am by beta emission. In view of their very short half-lives (87 and 14.1 years) the $^{238}$Pu content of the sample decreases from its original value by 0.8% and that of $^{241}$Pu by 5.2% within a short span of one year. Since the mass spectrometric determination of the total plutonium content of the sample is based on the accurate assay of the isotopic abundance, this decay will lead to a negative bias in the result. Change of $^{241}$Pu content alone can lead to an error of −0.2% if the measurement is made after one year.

Hence, if the sample is analysed in the verification laboratory after a long time, a decay correction should be applied to the results:

$$\text{Actual content of isotope A} \times \frac{T}{2 \cdot t_{1/2}}$$

where $T$ is the time delay in the analysis and $t_{1/2}$ is the half-life of isotope A (both in same units).
TABLE II. EFFECT OF AGEING OF PLUTONIUM SAMPLE ON ASSAY

<table>
<thead>
<tr>
<th>No.</th>
<th>Assayed&lt;sup&gt;a&lt;/sup&gt;</th>
<th>Conc. of Pu in sample (μg/g)</th>
<th>Random error in assay (%)</th>
<th>Relative deviation (%)&lt;sup&gt;c&lt;/sup&gt;</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Without treatment</td>
<td>With treatment&lt;sup&gt;b&lt;/sup&gt;</td>
<td>Untreated sample</td>
</tr>
<tr>
<td>1.</td>
<td>Immediately</td>
<td>4.335 (2)</td>
<td>4.353 (2)</td>
<td>0.14</td>
</tr>
<tr>
<td>2.</td>
<td>After 1 month</td>
<td>4.014 (14)</td>
<td>4.347 (1)</td>
<td>10.25</td>
</tr>
</tbody>
</table>

<sup>a</sup> With respect to sampling.

<sup>b</sup> Treatment with a mixture of HClO₄ + HNO₃ + HF.

<sup>c</sup> With respect to the value of a treated sample analysed immediately after sampling.
5. CONCLUSIONS

The present system of transportation of radioactive samples from the reprocessing facility to the analytical laboratory at the Agency's Headquarters involves considerable delay in the verification analysis. The delays introduce the problem of ageing of the samples which has a detrimental effect on the verification analysis. Though these effects can be compensated for by appropriate corrective measures the verification system is still extremely susceptible to inaccuracies. It is therefore necessary to review the entire transportation system with a view to substantially reducing the delays and making the verification procedure more meaningful.

REFERENCES

Abstract

Packaging and transport of small quantities of nuclear materials for safeguards analysis in Japan.

Timely evaluation of measurement results as well as accuracy and precision are very important in safeguards implementation. Though no delays in the evaluation have been observed for uranium bearing samples at present, the criteria of timely verification are still not met for plutonium bearing samples because of delay due to sample pretreatment and shipment. Uranium bearing samples are taken by the facility operator under the observation of government and IAEA inspectors and shipped from facilities to SAL-NMCC (Safeguards Analytical Laboratory, Nuclear Material Control Centre) and to SAL-IAEA, respectively. The shipments of these samples take place in Type L packages by surface in the case of domestic and by air in the case of international transport. Plutonium bearing samples are also taken by facility operators under the observation of inspectors and are shipped in Type A rather than Type B packages, which involves a considerable amount of labour and expense. With a Type A packaging, up to approximately 4 mg of plutonium can be transported. To take such small amounts of plutonium accurately, sample pretreatments, including dissolution, precise aliquoting and drying procedures, are generally necessary before shipment. Since the inspectors are not permitted to handle these samples, all preparatory work must be performed by the facility operators and this becomes a considerable burden. To shorten the time necessary before analyses, the possibilities of using future plant space and instruments for this type of work by safeguards authorities might be suggested as a long term solution.

1. Introduction

The IAEA document INFCIRC/153, paragraph 28, states that the technical objectives of safeguards are “timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or other nuclear explosive devices or for purposes unknown, and deterrence of such diversion by the risk of early detection”. According to the current guidelines, the timelyness of detection should correspond to the conversion time, i.e. 1 to 3 weeks for plutonium compounds and their mixtures and the order of one year for enriched uranium (<20%) and its compounds.
Consequently, timely evaluation of measurement results as well as accuracy and precision are very important when a DA (destructive assay) technique is used as one of the safeguards measures. In safeguards implementation, nuclear material samples taken at facilities in the nuclear fuel cycle are sent to the Safeguards Analytical Laboratories at the Nuclear Material Control Centre and the IAEA (SAL-NMCC and SAL-IAEA) and the concentration and isotopic composition of uranium and plutonium in samples are determined by DA techniques such as chemical methods and mass spectrometry.

No delays in the evaluation of results obtained by the above procedures have been observed for uranium bearing samples because of the ease of shipment and the longer detection time. On the other hand, timely verification criteria are still not met for plutonium bearing samples because of the delay resulting from sample pretreatment and shipment.

A seminar and a consultants meeting on these matters were held in Vienna to discuss how to overcome difficulties experienced in the process of safeguards sample preparation and shipment. During the seminar on practical problems encountered in the transport of radioactive materials for safeguards related activities, which was held in 1984, the recommendations made by the participants were mainly directed to simplifying procedures by, at least, adopting an approval system either on generic or on an annual basis for the package type to be used and for shipment, export and import approvals.

In the consultants meeting on practical problems in the preparation of safeguards samples from plutonium products and spent fuel solutions, which was held in 1985, the consultants discussed and gave valuable comments and recommendations on sampling, sample treatment, transport and analytical procedures. Among these recommendations, it is worth noting that the consultants recommend the Agency to increase its efforts to investigate the possibilities of using on-site measurements for verification.

This paper describes the status of and experience in packaging and transport of safeguards samples in Japan.

2. URANIUM BEARING SAMPLES

In the bulk handling facilities where nuclear material is held, processed or used in loose form, as liquids, gases, powders, or large numbers of non-identifiable units, the flow and inventory of nuclear materials declared by the facility operator are verified by the Government and IAEA through their independent measurement and observation as a part of inspection activities. The independent measurement is generally performed using both DA and NDA (non-destructive assay) methods in addition to bulk measurements such as weight, volume, etc. When the Government and Agency inspectors visit bulk uranium handling facilities in Japan for inspection, uranium bearing samples for DA and NDA are taken at key measure-
ment points (KMPs) by a facility operator under observation of the inspectors. After sampling, the weight and enrichment of samples taken are determined in the field by means of NDA instruments and so on. On the other hand, DA samples such as uranium oxides (UO₂, UO₃ and U₃O₈) powders, uranium dioxide pellets, U-Al alloy, and scraps, are put into glass bottles with a metal screw cap and are sent to the SAL-NMCC by surface using Type L¹ packaging in the case of samples for domestic safeguards. Details of this Type L package are as follows; a carton box contains a metal can in which 14 glass bottles containing samples are packed. The approximate weight of each uranium sample in the glass bottle is 5 g for uranium oxide powder, 0.5 to 1 g for U-Al alloy, 10 g for scraps, and about 10—15 g for uranium dioxide pellets. In the last three years, a total of approximately 400 uranium bearing samples have been sent to the SAL-NMCC each year from facilities for DA. It takes a few weeks on average (one month as an exception) from sampling to shipment including related document preparation. Almost the same types and numbers of uranium bearing samples are also taken by IAEA inspectors each year and shipped from facilities to the SAL-IAEA by air in general a few weeks after the time of sampling.

It may be concluded that no delays in evaluation due to transport have been observed for uranium bearing samples.

3. PLUTONIUM BEARING SAMPLES

Because of the shorter conversion time of plutonium compared with that of uranium, the criteria for timely verification are still not met for plutonium bearing samples. This situation has been said mainly to result from delays in sample shipment.

When the Government and IAEA inspectors visit facilities in PNC (Power Reactor and Nuclear Fuel Development Corporation) for inspection, plutonium bearing samples such as spent fuel solution, plutonium nitrate solution, plutonium dioxide powder, MOX powder and pellets, and scraps are taken for DA and NDA by facility operators under the observation of inspectors. The samples taken for DA are shipped from PNC to the SAL-NMCC and the SAL-IAEA by surface and by air, respectively, using Type A packaging. As shown below, the transport of plutonium bearing samples by a Type A packaging can reduce considerably licensing and administrative procedures compared with shipments by Type B packaging. However, the total amount of plutonium which can be transported with a Type A packaging is limited approximately to 4 mg in the case of plutonium recovered from LWR spent fuel because a Type A package shall not contain activities greater than the A₂ values.

¹ The Type L package is the package provided in Japanese regulations for nuclear material transport. This package corresponds closely to the exempt package provided in the Regulations for the Safe Transport of Radioactive Material (1973 Edition).
Such small amount of sample excludes application to the determination of plutonium concentration by standard routine methods and forces inspectorate laboratories to develop microanalytical techniques for determination. In addition, sample pretreatments including dissolution, precise aliquoting and drying procedures are necessary before shipment. Details of the pretreatment for IAEA spent fuel solution samples are as follows. After a spent fuel solution sample is received from the sampling bench, 1 mL of the sample solution is diluted with 150 mL of nitric acid and mixed until it is homogeneous. Three 1 mL lots of the diluted sample solution are taken into three penicillin vials (two vials contain dried spike) provided by the IAEA. After mixing well, each vial is heated to evaporate the solution nearly to dryness. It takes about 12 to 15 hours for this work. A set of the sample vials consisting of two spiked and one non-spiked is packed into a metal can and sealed after the vials have been installed into three lead cans for radiation shielding. Then the metal can is packed again into a larger metal can with a polystyrene foam cushion. Finally the metal can containing the samples is packed in a carton box. In parallel with this pretreatment, the operator analyses the concentration and isotopic composition of uranium and plutonium by the isotopic dilution method and calculates total activities in each vial on the basis of the analytical results obtained and the amounts of fission products estimated by the ORIGEN code. In the case of an MOX powder sample, a dissolution step is necessary prior to the procedure described. In order to reduce errors due to sampling, dilution and aliquoting during the pretreatment, an automated gravimetric sampling system is now being developed at the PNC reprocessing plant within the framework of JASPAS (Japan Support Programme for Agency Safeguards).

In 1985, a total of approximately 510 and 950 plutonium bearing samples were sent from PNC to SAL-NMCC and SAL-IAEA, respectively, and the number of spent fuel solution samples comprised more than 60% of the total number of samples pretreated and shipped. Therefore, the difference between the total number of samples shipped to the two laboratories was mainly due to the difference in spent fuel solution sampling, i.e. the IAEA takes three samples per batch and the Government takes two. It takes about one month at a minimum (three months as an exception) from sampling to shipment of plutonium bearing samples by using Type A packaging in spite of the operator’s efforts. As one of the possibilities to resolve this situation, the IAEA has proposed to the Japanese Government to study Type B package air transport of plutonium bearing samples, especially transport using the PAT-2 container developed for this purpose.

Though we have no experience of sending Type B packages by air in Japan, the Special Committee on the Safe Transport of Radioactive Materials operated by the competent authority of the Japanese Government organized a task force in 1984 for preparing basic safety criteria for plutonium transport by air, which is now under consideration. It can be recognized that the PAT-2 has merits, that is, simplification of sample pretreatment, reduction in the number of packages and
FIG. 1. Flow diagram of licensing and administrative procedures.
so on, for the transport of IAEA safeguards samples. However, as the Type B package will be subject to more severe safety regulations than those for the Type A package, the acquisition of the necessary licensing will take longer. Current procedures for Type A packages and prospective licensing and administrative procedures for the PAT-2 as a Type B package for air transport of safeguards samples from Japan to the IAEA are summarized in Fig. 1. It still takes more than one month for the necessary administrative procedures on every shipment if the PAT-2 is approved as a Type B package. In addition, the transport costs will be significantly higher than those for the Type A package. In the field of safeguards implementation in Japan, the transport of plutonium bearing samples using as Type B package a DOT-6M packaging which is slightly modified has been applied only for importing NBS plutonium standard reference materials and plutonium samples for interlaboratory analysis from the USA by surface.

4. CONCLUSION

As described above, PNC has made great efforts, which have now become a considerable burden to the facility, to overcome the delays observed in the process of preparing and shipping plutonium bearing samples. It seems impossible to expect further improvement of these procedures through the operator's efforts. When larger plutonium handling facilities come into operation, it is quite certain that the number of samples to be verified will increase significantly and delays in evaluation would become more serious unless some new proposal is investigated. Consequently, the following three possibilities for on-site measurements may be proposed to improve this situation.

4.1. Application of high precision NDA to in-field measurement of plutonium bearing samples

An example is the compact K-edge densitometer which the IAEA is now considering for field application for the determination of plutonium concentration in solution samples taken at reprocessing and conversion plants in PNC. By this means, it can be expected to decrease the number of samples to be sent to the SAL-IAEA for DA.

4.2. Use of operator's measurement procedure and/or apparatus with appropriate authentication techniques

Examples are a K-edge densitometer and X-ray fluorescent analysis system installed at the reprocessing and the conversion plants, respectively. If the concentration of plutonium and of uranium in inspection samples could be determined by operator owned apparatus after application of appropriate authentication
techniques, this means is also promising for timely evaluation of measurement results.

4.3. Construction of an inspectorate laboratory at the site of future large facilities or use of operator’s space for on-site verification

Though this proposal leaves many problems to be solved such as expense, licensing and so on, it would be suggested as a drastic way of resolving the delay in evaluation due to pretreatment and transport of inspection samples when larger plutonium handling facilities start operation in the near future.
PRACTICAL PROBLEMS ENCOUNTERED IN THE TRANSPORT OF SAFEGUARDS SAMPLES.

Small quantities of fissile material are transported regularly from nuclear facilities to recognized laboratories. These shipments are required either by the IAEA and are bound for the Seibersdorf Laboratory in Austria, or by the Directorate for EURATOM Safeguards and bound for the Central Office for Nuclear Measurement in Belgium in connection with activities associated with EURATOM safeguards. In accordance with the agreements signed by Member States under NPT, the samples taken at nuclear facilities during inspections by IAEA and EURATOM personnel should reach the recognized laboratories within a shorter time than that required for detecting diversion of significant quantities of nuclear material that is to be used exclusively for peaceful purposes. In actual fact, difficulties associated with transport procedures and shipping authorizations have been identified as existing in various Member States and have led to delays in delivering the samples which are much greater than the time needed for detection. Over the past few years, action has been taken to find solutions to the problems mentioned above, some of which have been resolved gradually on a joint basis by the recognized radioactive material carriers and the competent authorities.

PROBLEMES PRATIQUES RESULTANT DU TRANSPORT D'ECHANTILLONS AUX FINS DES GARANTIES.

De petites quantités de matières fissiles sont transportées régulièrement à partir d'installations nucléaires vers des laboratoires agréés. Ces transports sont requis soit par l'AIEA, à destination principalement du Laboratoire de Seibersdorf en Autriche, soit par la Direction Contrôle de sécurité d'Euratom, à destination notamment du Bureau central des mesures nucléaires en Belgique en ce qui concerne les activités liées aux garanties exercées par Euratom. Conformément aux accords signés par les États Membres dans le cadre du TNP, les échantillons prélevés dans les installations nucléaires à la requête des inspecteurs de l'AIEA et d'Euratom doivent parvenir aux laboratoires qualifiés dans un délai inférieur à celui de détection du détournement de quantités significatives de matières nucléaires à utiliser exclusivement pour les activités pacifiques. En réalité, des difficultés en matière d'une part de procédures de transport et, d'autre part, d'approbations de transport ont été identifiées au niveau de divers États Membres, et ont conduit à des retards de livraison des échantillons dépassant largement les délais de détection. Depuis plusieurs années, de nombreuses actions ont été menées afin de trouver des solutions aux problèmes cités ci-dessus. Certains ont pu être résolus progressivement entre les compagnies agréées de transport de matières radioactives et leurs autorités respectives.
1. INTRODUCTION

Dans le cadre des activités menées par l'Agence internationale de l'énergie atomique (AIEA) ainsi que par la Communauté européenne de l'énergie atomique (Euratom), des procédures liées aux garanties ont été élaborées et mises en œuvre par ces organisations en vue du transfert de petites quantités de matières nucléaires à partir d'installations et à destination de laboratoires d'analyses.

Au cours de leurs inspections dans ces installations, les inspecteurs de ces organisations ont le pouvoir de faire prélever de petites quantités au sein de lots de matière, de les faire conditionner conformément aux règlements en vigueur et de faire procéder à leur expédition vers les laboratoires convenus pour analyse. Les résultats de ces analyses sont ensuite traités en fonction des conditions de non prolifération propres aux installations sous contrôle. Il apparaît également que le temps entre le prélèvement d'un échantillon et l'interprétation des mesures analytiques effectuées sur cet échantillon devrait être inférieur au temps de détection d'un détournement d'un type de matière nucléaire concernée.

2. LE TRANSPORTEUR

Le transporteur constitue la relation entre le client (l'AIEA ou Euratom) et une installation nucléaire (l'expéditeur) qui est soumise à un contrôle permanent par des représentants officiels du client.

3. LES MATIÈRES NUCLEAIRES

Les matières nucléaires, faisant l'objet d'une expédition, sont généralement d'un poids relativement faible, c'est-à-dire de l'ordre de quelques grammes pour des matières fissiles spéciales. Ce poids peut apparaître comme insignifiant. Toutefois, comme il s'agit de matières réglementées, les conditions suivantes doivent être impérativement remplies:

— l'emballage de ces matières doit répondre à des spécifications très strictes;
— les procédures administratives relatives à l'expédition sont d'application quel que soit le poids des matières;
— le transport doit être réalisé conformément aux normes en vigueur.

En conséquence, l'ensemble des opérations d'expédition constitue un travail tout aussi important pour une petite quantité que pour une quantité beaucoup plus importante.

Chaque section relative à l'expédition de ces échantillons doit donc être traitée avec diligence afin de respecter les impératifs de contrôle. Plusieurs aspects reliés au transport des échantillons sont traités ci-après.
TABLEAU I. LIMITES D'ACTIVITÉ PAR TYPE DE COLIS ET MODE DE TRANSPORT

<table>
<thead>
<tr>
<th>Type de colis</th>
<th>Mode de transport</th>
<th>Forme spéciale (en capsule)</th>
<th>Limites au contenu</th>
<th>Sous autre forme</th>
</tr>
</thead>
<tbody>
<tr>
<td>Exempté</td>
<td>Par voie postale&lt;sup&gt;a&lt;/sup&gt;</td>
<td>$10^{-4}A_1$</td>
<td>$10^{-4}A_2$</td>
<td>$10^{-5}A_2$</td>
</tr>
<tr>
<td></td>
<td>Autres</td>
<td>$10^{-3}A_1$</td>
<td>$10^{-3}A_2$</td>
<td>$10^{-4}A_2$</td>
</tr>
<tr>
<td>Type A</td>
<td>Autres</td>
<td>$\leq A_1$</td>
<td>$\leq A_2$</td>
<td>$\leq A_2$</td>
</tr>
<tr>
<td>Type B(U)</td>
<td>Autres</td>
<td>$&gt; A_1$</td>
<td>$&gt; A_2$</td>
<td>$&gt; A_2$</td>
</tr>
</tbody>
</table>

<sup>a</sup> Le transport par voie postale n’est pas autorisé dans tous les pays.

TABLEAU II. APPROBATION DES EXPÉDITIONS ($1 \text{ Ci} = 3.70 \times 10^{10} \text{ Bq}$)

<table>
<thead>
<tr>
<th>Pays</th>
<th>Approbation des expéditions</th>
<th>Limites d'exception pour les expéditions de Pu (par envoi)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Allemagne (RÉp. féd. d')</td>
<td>exigée</td>
<td>$&lt; 0,1 \mu \text{Ci}$</td>
</tr>
<tr>
<td>Belgique</td>
<td>exigée</td>
<td>$&lt; 0,1 \mu \text{Ci}$</td>
</tr>
<tr>
<td>Etats-Unis</td>
<td>non exigée</td>
<td>-</td>
</tr>
<tr>
<td>France</td>
<td>exigée</td>
<td>$&lt; 3 \text{ g}$</td>
</tr>
<tr>
<td>Italie</td>
<td>exigée</td>
<td>$&lt; 9 \text{ g}$</td>
</tr>
<tr>
<td>Japon</td>
<td>exigée</td>
<td>-</td>
</tr>
<tr>
<td>Pays-Bas</td>
<td>non exigée</td>
<td>-</td>
</tr>
<tr>
<td>Royaume-Uni</td>
<td>exigée</td>
<td>$&lt; 10 \text{ g}$</td>
</tr>
</tbody>
</table>

TABLEAU III. REFERENCES DE QUELQUES MODELES DE COLIS DE TYPE B(U) UTILISES POUR LE TRANSPORT D'ECHANTILLONS

<table>
<thead>
<tr>
<th>Dénomination</th>
<th>Origine</th>
<th>N° d'approbation</th>
<th>Masse brute</th>
<th>Masse fissile permise</th>
</tr>
</thead>
<tbody>
<tr>
<td>SAFPAK 2767B</td>
<td>Royaume-Uni</td>
<td>GB/2767/B(U)</td>
<td>11,8 kg</td>
<td>max. 15 g</td>
</tr>
<tr>
<td>PAT-2</td>
<td>Etats-Unis</td>
<td>USA/9150/B(U)</td>
<td>33 kg</td>
<td>max. 15 g</td>
</tr>
<tr>
<td>TNB 0145/2</td>
<td>Belgique</td>
<td>B/30/B(U)</td>
<td>50 kg</td>
<td>max. 15 g</td>
</tr>
<tr>
<td></td>
<td></td>
<td>B/30/B(U)F</td>
<td></td>
<td>max. 4500 g</td>
</tr>
</tbody>
</table>
4. LES CONDITIONS D’EMBALLAGE

En fonction de l’activité totale des isotopes présents dans les échantillons, ceux-ci doivent être placés dans des emballages du type soit exempté, soit A, soit B.

La détermination de la classe d’emballage s’effectue par l’utilisation de valeurs de base dénommées $A_1$ et $A_2$ qui s’expriment en becquerels ou en curies. La liste de ces valeurs est donnée dans le Règlement de transport des matières radioactives (Collection Sécurité n° 6 de l’IAEA). Le tableau I donne les limites en activité pour l’expédition des matières radioactives en fonction du type de colis et suivant le mode de transport. La masse des isotopes fissiles par colis est limitée à 15 g.

5. APPROBATION DES MODELES DE COLIS

Seuls les modèles de colis renfermant des matières dont l’activité est supérieure aux valeurs $A_1$ et $A_2$ doivent être approuvés par les autorités compétentes des pays d’origine de ces colis. L’expérience a montré qu’une approbation supplémentaire pour ces modèles de colis est exigée par les autorités compétentes de certains pays tels que, par exemple, les États-Unis et le Japon.

D’autre part, il faut noter que, hormis les exceptions énoncées dans le règlement précité, tous les modèles de colis pouvant contenir plus de 15 g de matières fissiles doivent également être approuvés par les autorités compétentes concernées.

6. APPROBATION DES EXPEDITIONS

Le transport de matières radioactives en général et celui des matières fissiles en particulier ne peut être effectué, dans certains pays, que moyennant la délivrance d’autorisations de transport par les autorités compétentes des pays concernés. Le tableau II donne quelques exemples.

7. MODELES DE COLIS DE TYPE B(U)

Les modèles les plus utilisés pour l’expédition d’échantillons sont le SAFPAK 2767B, le PAT-2 et le TNB 0145. Quelques caractéristiques de ces modèles de colis sont données au tableau III.

8. EXPERIENCE EN MATIERE D’EXPEDITION DES ECHANTILLONS

TABLEAU IV. RELEVE DES TRANSPORTS EFFECTUES PAR TRANSNUBEL AU DEPART DE LA BELGIQUE VERS SEIBERSDORF (AUTRICHE) POUR LE COMPTE DE L'IAEA

<table>
<thead>
<tr>
<th>Année</th>
<th>Nombre d'expéditions</th>
<th>Nombre de colis</th>
<th>Expéditions</th>
<th>Délais pour chaque expédition</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>Belges</td>
<td>Autres</td>
</tr>
<tr>
<td>1981</td>
<td>7</td>
<td>13</td>
<td>6</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>1 semaine (6 exp.)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>1 mois (1 exp.)</td>
</tr>
<tr>
<td>1982</td>
<td>2</td>
<td>3</td>
<td>2</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>1 semaine (1 exp.)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>1 mois (1 exp.)</td>
</tr>
<tr>
<td>1983</td>
<td>4</td>
<td>5</td>
<td>4</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>1 à 2 semaines</td>
</tr>
<tr>
<td>1984</td>
<td>6</td>
<td>9</td>
<td>5</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>1 à 3 semaines</td>
</tr>
<tr>
<td>1985</td>
<td>5</td>
<td>15</td>
<td>4</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>1 à 3 semaines</td>
</tr>
</tbody>
</table>

TABLEAU V. RELEVE DES TRANSPORTS EFFECTUES PAR TRANSNUBEL VERS LE BUREAU CENTRAL DES MESURES NUCLEAIRES POUR LE COMPTE DE LA CCE A LUXEMBOURG

<table>
<thead>
<tr>
<th>Année</th>
<th>Nombre d'expéditions</th>
<th>Nombre de colis</th>
<th>Expéditeurs</th>
<th>Délais d'exécution</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>Bénélux</td>
<td>Autres</td>
</tr>
<tr>
<td>1981</td>
<td>2</td>
<td>21</td>
<td>5</td>
<td>14</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Bénélux et Danemark:</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>1 à 2 semaines</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>R.-U.: 1 à 2 mois</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>France: 8 mois</td>
</tr>
<tr>
<td>1982</td>
<td>14</td>
<td>15</td>
<td>5</td>
<td>9</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Bénélux: 15 semaines</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>France: 4 à 10 mois</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>R.-U.: 2 mois</td>
</tr>
<tr>
<td>1983</td>
<td>8</td>
<td>13</td>
<td>8</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Bénélux:</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>1 à 3 semaines</td>
</tr>
<tr>
<td>1984</td>
<td>3</td>
<td>3</td>
<td>3</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Bénélux:</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>1 à 3 semaines</td>
</tr>
<tr>
<td>1985</td>
<td>13</td>
<td>16</td>
<td>4</td>
<td>9</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Bénélux:</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>1 à 4 semaines</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Italie: idem</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>France: 4 à 6 mois</td>
</tr>
</tbody>
</table>
Dès qu'un colis est rendu disponible par l'installation d'expédition, celui-ci est directement acheminé vers l'aéroport de Bruxelles National et expédié vers Seibersdorf. Des approbations permanentes pour ces transports ont été délivrées par l'autorité compétente belge aux transporteurs agréés routier et aérien. L'accord sur les privilèges et immunités de l'Agence a été reconnu par la loi belge du 22 juillet 1985. Quelques problèmes douaniers se sont présentés du fait que l'Agence n'avait pas de représentation officielle en Belgique. Ces problèmes ont été résolus entre-temps.

Le tableau V concerne un relevé de transports d'échantillons effectués à destination du Bureau central des mesures nucléaires en Belgique pour le compte de la Commission des Communautés européennes à Luxembourg aux fins des garanties.

Les expéditeurs sont situés dans divers pays de la Communauté européenne dont la Belgique, la France, l'Italie, les Pays-Bas et le Royaume-Uni. Il apparaît que les délais entre la commande et l'exécution du transport sont de:

- 1 à 4 semaines pour les pays du Bénélux et l'Italie,
- 2 mois pour le Royaume-Uni,
- 4 à 8 mois pour la France.

Les délais de plusieurs mois, pour l'expédition d'échantillons notamment à partir de la France, résultent essentiellement des délais imposés pour obtenir les licences d'exportation. Cette situation a été toutefois corrigée en 1985 par la délivrance d'une licence globale d'exportation à la Transnucléaire (France) pour le compte de l'AIEA et valable pour un an. À notre connaissance, l'AIEA n'a pas demandé le renouvellement de cette licence pour 1986.

Pendant la même année 1985, la Transnuklear (RFA) a procédé à 17 expéditions vers Seibersdorf. D'après Transnuklear, les délais d'exécution proviennent essentiellement du fait que le transporteur ne reçoit pas en temps utile les informations d'expédition de la part soit des expéditeurs, soit des inspecteurs de l'AIEA. En particulier, si la date d'échantillonnage était donnée à l'avance, l'expédition pourrait être effectuée un jour après achèvement de cet échantillonnage. Ceci serait réaliste pour autant que les installations d'expédition procèdent immédiatement à l'emballage des échantillons.

9. CONCLUSIONS

Un effort permanent en vue d'accélérer l'acheminement des échantillons est réalisé à divers niveaux, en particulier par les transporteurs agréés par l'Agence. Les délais d'exécution pourraient être améliorés dans certains cas par une plus grande célérité de certains expéditeurs dans la transmission des informations aux transporteurs.
TRANSPORTATION OF SMALL QUANTITIES OF RADIOACTIVE MATERIALS IN ITALY

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Abstract

TRANSPORTATION OF SMALL QUANTITIES OF RADIOACTIVE MATERIALS IN ITALY.
Details of the transport safety requirements for samples and reference materials in Italy are presented. In compliance with these requirements, which are based upon the IAEA Regulations in Safety Series No. 6 and European Directives, controls are made in advance of the issue of the authorizations needed to perform the transport activity. In this way it is possible to ensure a good level of safety without delay to each single transport.

INTRODUCTION

In the nuclear industry, the shipment of samples and reference materials is necessary for a correct application of international safeguards regulations and for industrial and instrument operation purposes. The materials are always in the form of small quantities and do not entail high hazards or pose criticality or heat dissipation problems so that transports can be easily performed by all modes (road, railway, air and sea) in compliance with national regulations.

In Italy, safety in the transport of radioactive and fissile materials is based upon specific laws and regulations, which are not part of regulations concerning the transport of dangerous goods. Italian regulations make reference to European Community Directives for administrative procedures and radiation protection, while for technical aspects reference is made to the IAEA Regulations in Safety Series No. 6.

SAFETY REQUIREMENTS

European Community Directives require prior authorization for all activities entailing radiation risk, such as the production, handling and transport of radioactive and fissile materials. The nuclear plants, and other activities involving...
nuclear energy, are authorized by the Minister of Industry with the technical advice of ENEA-DISP. This procedure represents the first step for transport safety. In fact, package preparation and consignment play an important role, and the applicant is required to ensure compliance with the transport regulations and in particular:

(a) To define the radionuclides and quantities with a sufficient degree of confidence (otherwise pessimistic assumptions for radiotoxicity and quantity must be used);

(b) To define the physical state and concentration of the radionuclides together with other dangerous properties;

(c) To identify the packaging (excepted, industrial, Type A, Type B) required by regulations on the basis of the following characteristics: available volume, taking account of volume variations due to environmental temperature, corrosion, change of physical state; shielding required to comply with radiation limits; leakage from containment system;

(d) To provide maintenance and loading procedures for each type of packaging so that the safety of the package is the same as the certified one;

(e) To measure radiation levels and fill in the transport index if necessary on the label;

(f) To verify compliance of non-fixed contamination levels with the regulation limits;

(g) To choose the right marking and label for the packages.

These activities performed in nuclear plants are covered in many cases by a quality assurance programme so that a sufficient level of reliability is obtained.

Transport can be carried out by the nuclear plant owner or by other firms. In all cases, however, carriers must be authorized. The authorization is issued by the Minister of Industry, together with the Minister of Transport or the Minister of Mercantile Marine, with the technical advice of ENEA-DISP.

The authorization is given after verification of the procedures adopted to comply with the existing regulations. In particular the following issues are taken into account:

(1) Assessment is made of radiation doses to workers based on the magnitude and likelihood of the exposures. Individual radiation exposure monitoring and special health supervision is carried out if necessary. In any case, periodic environmental monitoring and assessments of radiation exposure levels in working areas should be carried out to verify that the workers are classified in the correct way;

(2) Workers involved in loading, transport and unloading shall be trained with reference to radiation risks and their minimization; workers shall be familiar with labels and marking of packages in order to be immediately informed about the radiation level existing under normal transport conditions. Worker
training has become more and more relevant for safe transport and this results in a more conscientious participation, preventing psychological unwillingness. Specific pre-job training courses and/or technical information can be given, taking into account the extent of the transport activity. In particular they must include:

(i) Procedures for package stowage in order to prevent damage under normal conditions and avoid the proximity of the general public (e.g. for railway transportation) or drivers (e.g. for road transportation); the measures may include the use of placards;

(ii) Procedures for the control of the packaging and of the vehicle performed by the carrier or a qualified expert with adequate instruments; such controls are needed to check vehicle contamination, the frequency being chosen on the basis of the use made of the vehicle;

(iii) Emergency procedures for accident conditions with reference to the type and quantity of transported packages; the importance of this is increased by the fact that in many circumstances the carrier is alone at the time of the accident, but is required to provide accident area delimitation and public evacuation, to contact a qualified expert and the fire brigade and provide them with the relevant documentation. Further actions can require the intervention of an organization other than the carrier.

(iv) Procedures to ensure the packages are delivered only to the appropriate person and location and then to remove placards from the vehicle to prevent undue alarm when the vehicle is empty.

It is requested, for road transportation, that an authorized vehicle provided with the following be used:

- fire extinguishers to cope with small fires in the driving cabin or luggage van;
- possibility of rapidly disconnecting the battery to prevent fire risks;
- loading compartment equipped with a quick stowage system and characterized by smooth surfaces to facilitate decontamination;
- signal equipment to keep the public away from the vehicle in the event of accident;

For a vehicle with a high cylinder displacement, a diesel engine is suggested.

On the basis of the above considerations, the Minister of Industry issues a permanent authorization for radioactive material transport valid for five years: during this period the carrier is requested only to supply ENEA-DISP with quarterly reports on the transport carried out.

Controls to verify compliance with safety procedures are performed by ENEA-DISP inspectors, who are independent judicial officers. Control results are reported in a formal minute in which prescriptions aimed at higher safety and protection standards may be indicated; in the event of violation the minute is
sent to the judicial authority. At present in our country there are about 150 authorized carriers, some for hire and some working on their own account, for the different transport modes.

Transport by post, even if it is regulated at an international level, has not yet been carried out. Our Minister of Post and Telecommunication is making preparations for this transport mode, although overseas experience shows evidence of problems among post personnel and there has been only limited use of this transport mode in the countries where it is accepted and regulated.

It has also to be remembered that international regulations for the safe transport of radioactive material (as set forth by the IAEA and accepted by IMO, ICAO, ADR, etc. and by many Member States) stress the relevance of intrinsic package safety as the major issue for safe transport. International regulations, as it is well known, demand increased containment capability and packaging resistance with increasing radioactivity content and require an approval certificate released by the Competent Authority for special form materials and for Type B packaging. Such certificates, which are granted in Italy by ENEA-DISP, entail a very high degree of safety in respect of the radiological risk and also in relation to possible contamination of the environment in the event of severe accidents. At present, approval certificates issued in another country must be endorsed by ENEA-DISP.

CONCLUSIONS

The transport of radioactive and fissile samples, which represents a small but relevant and specific activity in the nuclear field, is performed in compliance with the above mentioned standards by all modes. In the future, transport by post could be envisaged but as far as Italy is concerned a further effort in terms of organization is required for the administration involved.

The authorization requested in Italy is similar to that in other Member States, with the obvious differences arising from different laws, and the customs of local administrative and competent authorities. Generally, the authorization is issued with a validity of several years so that each single transport does not need specific administrative procedures in view of the small quantities involved in sample transport.

To enable easier transfer of samples from different countries to the IAEA Safeguards Analytical Laboratory, the Agency prepared a very detailed and specific document entitled "Arrangements for the transfer of safeguards samples". This has to be complemented by specific technical and administrative provisions requested by the procedures of specific nuclear centres. The completed document could then be approved by the competent authorities, as far as safety problems are concerned, and sent again to the nuclear centres and to the Agency. In this way each nuclear centre would have available technical guidelines which are in agreement with the requirements of the IAEA (which is charged with safeguards controls) and of the Competent Authorities (responsible for safe transport and for the radiation protection of workers and the general public).
IMPORT AND EXPORT OF SMALL QUANTITIES OF NUCLEAR MATERIAL.

The problems encountered in the import and export of small quantities of nuclear material do not differ essentially from those encountered in the international movements of larger quantities. Paradoxically, the insignificance of the quantities sometimes even creates additional difficulties. In France, for example, inspection procedures may seem complex, particularly in the case of small quantities which have very low or zero thresholds and are non-uniform. It is obvious, however, that inspections are essential even in the case of small quantities. Nevertheless, it might be desirable for national and international thresholds to be standardized and for a guide to be drawn up which is as general as possible and groups together in a single document the regulations that have to be met.

IMPORTATION ET EXPORTATION DE PETITES QUANTITES DE MATIERES NUCLEAIRES.

Les problèmes rencontrés dans l’importation et l’exportation de petites quantités de matières nucléaires ne sont pas fondamentalement différents de ceux rencontrés dans les mouvements internationaux de quantités plus importantes. Paradoxalement, il arrive même que la faiblesse des quantités vienne apporter des difficultés supplémentaires. En France, par exemple, les dispositions de contrôle peuvent paraître complexes, en particulier dans le cas des petites quantités aux seuils très bas ou nuls, et non uniformes. Il est évident cependant que le contrôle est indispensable, même dans le cas de petites quantités. On pourrait cependant souhaiter une harmonisation nationale et internationale des seuils et l’établissement d’un guide aussi général que possible qui regroupe dans un même document les dispositions à satisfaire.

Les problèmes rencontrés dans l’importation et l’exportation de petites quantités de matières nucléaires ne sont pas fondamentalement différents de ceux rencontrés dans les mouvements internationaux de quantités plus importantes. Paradoxalement, il arrive même que la faiblesse des quantités vienne apporter des difficultés supplémentaires.

On peut citer les problèmes suivants, parmi ceux qui ont été soulignés par un groupe d’experts réunis à Vienne en juin 1984:

- procédures administratives rigides;
- divergences dans l’interprétation et l’application de la réglementation nationale;
- volume de la documentation exigée;
— décisions des autorités locales en contradiction avec les règles nationales;
— manque de contacts et de coopération entre les autorités compétentes des pays intéressés;
— quantités faibles des matières nucléaires expédiées par rapport au volume global des marchandises exportées, d'où peu d'intérêt des intervenants.

Certains de ces problèmes deviennent particulièrement aigus dans le cas de petites quantités, en particulier le dernier cité dans l'énumeration ci-dessus si l'on compare les contraintes administratives à l'importance de l'envoi et à son faible ou nul impact économique.

1. CONDITIONS GENERALES DE L'IMPORTATION/EXPORTATION DES MATIERES NUCLEAIRES

Les dispositions concernant l'importation/exportation sont spécifiques de chaque Etat; cependant, on peut dire qu'il y a, en général, deux sortes de dispositions à satisfaire:
— celles qui sont particulières aux matières nucléaires,
— celles qui sont plus générales.

En ce qui concerne les premières, les informations sur les mesures détaillées prises dans le cadre national sont souvent tenues secrètes. Par ailleurs, les prescriptions sont en perpétuelle évolution, ce qui est source de difficultés accrues.

En ce qui concerne les secondes, il semble que les difficultés et les retards soient plus importants pour l'obtention de licences d'exportation que pour l'importation. Cependant, l'obtention des documents d'importation nécessaires pour la réexportation donnent lieu à des difficultés, en particulier dans le cas des expéditions de petites quantités (radioisotopes, échantillons, etc.).

Ces caractéristiques rendent difficile de s'étendre longuement, dans un tel cadre, sur les aspects généraux, les procédures de chacun des États étant particulières à chacun. C'est ailleurs que sont décrites les solutions apportées par chacun de ces États pour l'envoi des échantillons à l'AIEA ou à d'autres organisations, soit dans le cadre d'accords particuliers, soit dans celui de conventions générales. Cette question ne sera pas abordée ici.

Nous nous limiterons à la situation de la France dans le cas général des envois de petites quantités.

2. CAS DE LA FRANCE

Les principaux textes applicables aux exportations et importations de matières nucléaires sont évoqués ci-après.
2.1. La loi du 25 juillet 1980 sur la protection et le contrôle des matières nucléaires

Sont soumises à cette loi les matières fissiles, fusibles ou fertiles. Leur importation et leur exportation, faites en exécution de contrats conclus par les opérateurs français et étrangers, sont soumises à une autorisation et à un contrôle. L'exportateur est tenu de stipuler aux acquéreurs et sous-acquéreurs les conditions relatives à l'utilisation ultérieure des matières nucléaires, auxquelles peut être subordonnée la délivrance de l'autorisation de toute exportation. L'autorisation peut être soumise à des conditions concernant sa durée, les quantités et les formes des matières nucléaires concernées, les mesures à prendre pour en connaître la localisation, éviter leur vol, leur détournement ou leur perte.

Le contrôle porte sur l'exécution de ces mesures, qui est assortie de dispositions pénales.

2.2. Le décret du 12 mai 1981 précisant les modalités d'application de la loi

L'autorisation est délivrée par le ministre de l'industrie qui consulte à ce sujet les ministres de l'intérieur et des affaires étrangères. L'autorisation n'est pas requise si les quantités d'éléments importées ou exportées au cours d'une période de 12 mois ne dépassent, par exemple, à aucun moment, les seuils suivants:

- plutonium ou uranium 233: 3 g;
- uranium enrichi à 20% ou plus d'uranium 235: 15 g;
- tritium: 2 g.

Cependant, l'exportation ou l'importation de quantités se situant au-dessous de ces seuils doit faire l'objet d'une déclaration au ministère de l'intérieur lorsque ces quantités dépassent, par exemple:

- pour le plutonium, l'uranium enrichi, l'uranium 233: 1 g;
- pour le tritium: 0,01 g.

2.3. L'avis aux importateurs et aux exportateurs relatif aux produits soumis au contrôle de la destination finale (Journal Officiel du 6 décembre 1985)

Ce texte stipule que toute opération internationale concernant un des produits soumis au contrôle de la destination finale et, à ce titre, figurant sur une liste annexée à l'avis, doit faire l'objet d'une autorisation préalable de l'administration, et en particulier d'une licence d'exportation. L'avis spécifie par ailleurs les modalités d'obtention et d'utilisation de la licence. Les produits soumis au contrôle de la destination finale doivent arriver dans le pays visé par la licence d'exportation, être livrés au destinataire prévu, dans la quantité prévue.

Ces modalités varient d'ailleurs avec les pays destinataires, classés en trois groupes.
Ces échanges donnent lieu, pour le premier de ces groupes, à l'établissement de documents dont les uns permettent un contrôle a priori de la destination (certificat international d'importation) et les autres un contrôle a posteriori (certificat de vérification de livraison); pour les autres pays, il existe des dispositions pratiquement équivalentes sous une forme un peu différente.

Ces modalités sont assorties d'un seuil en deçà duquel elles ne sont pas nécessaires.

Par exemple pour les isotopes du plutonium, et pour l'uranium 233 ce seuil est de 1 g.

2.4. L'avis aux exportateurs, relatif aux produits frappés de prohibition de sortie (Journal Officiel du 21 janvier 1986)

Ce texte, qui est destiné à éviter la prolifération des armes nucléaires, établit un contrôle renforcé sur l'exportation des produits figurant sur une liste annexée et dont font partie les produits spéciaux et autres produits fissiles, dont en particulier l'uranium enrichi en $^{235}\text{U}$, et/ou en $^{233}\text{U}$, et le plutonium sous toutes ses formes. Les procédures restent applicables même si les produits transformés sur le territoire français sont demeurés la propriété d'une personne physique ou morale établie à l'étranger.

L'accord préalable de l'administration est indispensable; il est accordé, comme dans le cas précédent, par l'administration générale des douanes, sur avis favorable des départements ministériels intéressés. Il n'est assorti d'aucun seuil inférieur pour les produits fissiles.

Une seule demande peut être déposée quand un produit est visé par ces deux derniers textes.

CONCLUSION

Ce rapide survol des principales dispositions montre que l'importation et l'exportation de petites quantités de matières nucléaires, dans le cas de la France, et en dehors de tout privilège et conditions particulières, n'échappe pas aux règles générales. Les seuils inférieurs qui permettent dans certains cas d'échapper à l'une de ces règles ne sont pas identiques, et pour l'une d'entre elles, le seuil est nul.

L'application de ces règles peut paraître complexe et elle explique en grande partie les délais quelquefois observés.

Il est cependant tout à fait évident que le contrôle est indispensable même dans le cas de petites quantités.

On pourrait cependant proposer, dans la mesure du possible, une harmonisation nationale et internationale des seuils et l'établissement d'un guide aussi général que possible qui tend à regrouper dans un même document les dispositions à prendre.
QUESTIONS OF LIABILITY
IN THE TRANSPORT OF SAMPLES*

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For the purposes of the paper 'samples' are defined as small quantities of radioactive material which are excepted from certain provisions of the IAEA Regulations for the Safe Transport of Radioactive Materials.

Examples of package limits for non-fissile material (see Table IV of the IAEA Regulations).

<table>
<thead>
<tr>
<th>Activity</th>
<th>Liability in the Fed. Rep. of Germany</th>
</tr>
</thead>
<tbody>
<tr>
<td>Special form solids</td>
<td>$10^{-3} A_1$ 2.0 GBq of $^{137}$Cs None</td>
</tr>
<tr>
<td>Other form solids</td>
<td>$10^{-3} A_2$ 0.5 GBq of $^{137}$Cs None</td>
</tr>
<tr>
<td>Liquids</td>
<td>$10^{-4} A_2$ 0.05 GBq of $^{137}$Cs None</td>
</tr>
</tbody>
</table>

Examples of package limits for fissile material (see para. 560 of the IAEA Regulations).

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>15 g $^{233}$U</td>
<td>$5.27 \times 10^{-3}$ TBq</td>
<td>$1 \times 10^{-3}$ TBq</td>
<td>1 million</td>
</tr>
<tr>
<td>15 g $^{235}$U</td>
<td>$1.11 \times 10^{-6}$ TBq</td>
<td>unlimited</td>
<td>None</td>
</tr>
<tr>
<td>15 g $^{238}$Pu</td>
<td>9.44 TBq</td>
<td>$2 \times 10^{-4}$ TBq</td>
<td>2.5 million</td>
</tr>
<tr>
<td>15 g $^{239}$Pu</td>
<td>$34.41 \times 10^{-3}$ TBq</td>
<td>$2 \times 10^{-4}$ TBq</td>
<td>2 million</td>
</tr>
<tr>
<td>15 g $^{241}$Pu</td>
<td>61.05 TBq</td>
<td>$1 \times 10^{-2}$ TBq</td>
<td>2 million</td>
</tr>
<tr>
<td>15 g $^{238}$U nat</td>
<td>$0.39 \times 10^{-6}$ TBq</td>
<td>unlimited</td>
<td>None</td>
</tr>
<tr>
<td>15 g U enriched &lt;5%</td>
<td>$1.50 \times 10^{-6}$ TBq</td>
<td>unlimited</td>
<td>None</td>
</tr>
<tr>
<td>15 g U enriched 93%</td>
<td>$3.88 \times 10^{-5}$ TBq</td>
<td>$1 \times 10^{-3}$ TBq</td>
<td>None</td>
</tr>
</tbody>
</table>

* Only a summary is published here.
According to the convention on Third Party Liability in the Field of Nuclear Energy in connection with the Additional Protocol to the Convention on Third Party Liability in the field of Nuclear Energy we have the situation that in the case of the carriage of nuclear substances — including storage in transit — the operator of a nuclear installation shall be liable for damage caused by a nuclear incident involving nuclear substances. As an example, the amounts of liability according to the Federal German 'Deckungsvorsorgeverordnung' are given in the last columns of the tables.
EXPERIENCE IN THE TRANSPORT OF SAMPLES CONTAINING NUCLEAR MATERIALS

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U. WENZEL
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Abstract

EXPERIENCE IN THE TRANSPORT OF SAMPLES CONTAINING NUCLEAR MATERIALS.

The Euratom Safeguards Directorate and the International Atomic Energy Agency routinely transport small amounts of nuclear materials destined for destructive and non-destructive assay to and from various locations. Both organizations frequently experience delays detrimental to the objectives of such transports. To reduce the delays, the authors present approaches that provide for exemption of the transports from export/import controls, a global nuclear risk insurance, standardized consignment composition and appropriate transport arrangements. A test conducted by the Agency resulted in transport times acceptable for the objectives related to such transports. Though elements of the approaches apply only to transports of the safeguards organizations, other elements may be of use in general for the transport of nuclear materials in small amounts.

1. INTRODUCTION

The International Atomic Energy Agency (Agency) and the Euratom Safeguards Directorate (Euratom) are responsible for the application of safeguards to nuclear materials in their Member States.

As an integral part of these activities, Euratom and the Agency carry out measurements on nuclear materials employing destructive and non-destructive assays. This entails frequent transports of nuclear materials in small amounts from a facility of a Member State to the laboratories of Euratom or the Agency and vice versa. The administration of such transports is an important parameter to judge the effectiveness of the safeguards systems.

Failures or undue delays in the transports adversely affect the safeguards activities because:

— Non-destructive measurements are carried out at the facility. For such measurements it is necessary to have working reference materials and often specific reference materials have to be provided for an individual inspection.

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Destructive assay involves the shipment of representative samples of the material to the analytical laboratories of Euratom or the Agency. Delay in the transport can be detrimental to the usefulness of this verification procedure, as changes of the sample affecting adversely its representativeness may occur and/or results of the analysis may not be available on time.

However, both the Agency and Euratom have experienced in the past frequent delays in the transport of safeguards samples. This paper deals with the sources of delay, a new approach for expediting such transports and the preliminary results of implementing this new approach. Practical aspects relating to the costs of transport are not covered by this paper.

2. PRESENT SITUATION

For practical purposes, samples collected for safeguards verification are categorized as follows:

- Uranium (U) samples including fresh fuel and U products from reprocessing and recycling plants;
- Plutonium (Pu) samples including Pu products and mixed oxide (MOX) samples;
- Spent fuel (input) samples.

All samples fall into the category of fissile exempt (less than 15 g fissile) material.

For verification of the flow and inventory of nuclear materials, inspectors of the Euratom Safeguards Directorate take approximately 1200 samples per year at different installations handling material in the nuclear fuel cycle. About 47% of the transfers involve transport across the borders between Member States.

In 1985, the Agency shipped 1350 samples corresponding to approximately 150 shipments from the Member States to Agency premises. The samples were collected at about 50 facilities situated in 40 countries. In the same period, the Agency distributed 176 of the incoming samples corresponding to 50 shipments to the Agency’s Network of Analytical Laboratories (NWAL). The laboratories are situated in six countries.

2.1. Delay statistics

The delays in transport differ for samples of different categories. This is demonstrated in Table I.

Euratom experiences somewhat higher average delays than the Agency. This is due to more complicated transport procedures from installations of two Member States.
TABLE I. DELAYS FOR THE SHIPMENT OF SAFEGUARDS SAMPLES

<table>
<thead>
<tr>
<th></th>
<th>IAEA shipments</th>
<th>Euratom shipments</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Total average transfer — 46 d/sample</td>
<td>Total average transfer — 68 d/sample</td>
</tr>
<tr>
<td></td>
<td>Shipment to IAEA</td>
<td>Shipment from IAEA</td>
</tr>
<tr>
<td>Uranium</td>
<td>38 d/sample</td>
<td>41</td>
</tr>
<tr>
<td>Plutonium</td>
<td>63 d/sample</td>
<td>19          (fresh and irradiated)</td>
</tr>
<tr>
<td>Spent fuel</td>
<td>50 d/sample</td>
<td>70</td>
</tr>
</tbody>
</table>

Several conclusions can be drawn from the values in Table I:

(1) Both organizations experience similar transport delays;
(2) Significant delays occur mainly for international transports;
(3) The delays for incoming Pu and spent fuel samples are too high to allow timely verification;
(4) The great discrepancy between the delays for incoming and outgoing shipments indicates that effective procedures can be established for expediting safeguards samples;
(5) The delays are far too long and seriously affect the timeliness of safeguards detection capability.

2.2. Transport procedures for the shipment of safeguards samples

It is therefore worthwhile to discuss in detail the procedures routinely employed for shipments. Representative samples are collected from a batch of nuclear material. The relevant procedure is agreed upon between the organization concerned and the facility, carried out by the facility operator and witnessed by an inspector.
Some samples are subjected to treatment prior to shipment in order to ensure that:

— the requirements for safe transport are complied with
— the chemical integrity is maintained even if the transport is extraordinarily delayed.

Subsequently, the samples are loaded into a primary containment which is sealed by the inspector. At that point, the nuclear material involved is transferred into the custody of the Agency or Euratom and with that the organizations become fully responsible for the further handling of the samples.

Inspectors leave the plant before the transport arrangements are made and delegate the responsibility for initiating the transport either to the facility operator or to the relevant transportation units of their organization.

The further arrangements for the transport are made by a transport agent authorized by the Agency or Euratom. Upon receipt of the request for shipment, the agent applies for an export licence and, if necessary, for a shipment approval, arranges booking and routing of the transport, notifies the consignees and the Competent Authority for Transport Safety (CATS) of the transport and delivers the consignment to the carrier. In one Member State the Euratom Safeguards Directorate itself must apply for an export licence through a committee.

Agency samples are taken to the nearest airport and then by plane to Vienna. Delivery of the consignment to the Agency’s premises is carried out by the Agency’s Transportation Unit.

Euratom samples are transported either by air or in some cases by road.

3. SOURCES OF DELAY

Before one can attempt to reduce delays in the transfer of the samples one must take a more detailed look at the causes of the delays. These are:

— initiation of the transport
— import/export licences
— insurance cover for transport
— the approval of design of Type B(U) packages by the country of origin and, if requested, the validation of the certificate by all states from, through and into which the samples are transported
— approval for shipment and (in certain countries) authorization for carriers.

3.1. Initiation of the transport

The transport of the samples is most often initiated by the inspector, on return to Headquarters. This may cause a delay of several weeks.
3.2. Import/export licences

The Commission of the European Communities and the Agency become the owner of the samples taken by their inspectors. Both organizations have agreements with or legal rights in their Member States [1, 2] stipulating that articles in the property of the organizations are exempt from all customs duties, prohibitions and restrictions. Therefore, transport of samples from one Member State to another should not be restricted in any way and export licences for individual transports should not be necessary.

At present, however, some Member States require such export licences for safeguards samples, and Euratom and the Agency as the owners are required to follow the necessary administrative procedure in order to obtain authorization to move the sample. In some cases this is done through the operator or the carrier, in other cases through a special committee.

The situation is further complicated if the samples contain nuclear material which had been only temporarily imported for conversion, isotopic separation or reprocessing. Such samples must first of all be permanently imported into the country, an operation which is carried out (unwillingly) by the operator. Only then can an export licence be applied for.

These different administrative obstacles complicate the export and import of samples, and confuse the people involved in organizing them.

3.3. Insurance cover — civil liability for nuclear risks

Member States of the European Communities are party to the International Convention of Paris [3], which provides that for transport of plutonium, uranium and other radioactive materials, insurance is required covering third party liability for damage arising during transport. While many carriers and operators in the Communities already hold this insurance, certain national authorities may not recognize the validity of a policy issued in another country. This makes it necessary to obtain a second policy for the same shipment.

The Agency always requests its transport agents to take care of the insurance.

3.4. Authorization for carriers and approval of shipment

In general, national laws require that carriers of radioactive material are duly authorized by the competent authority. In some cases there are some serious difficulties in finding a carrier authorized in all the countries through which road transport must pass. Furthermore, except for very small quantities, a shipment of radioactive material may require special approval in compliance with national regulations implemented in addition to, and sometimes at variance with, the international regulations in force.

Frequently, additional information has to be provided on the samples being transported which may not be available until chemical analysis has been completed.
3.5. Validation of certificates

Several samples, calibration sources and standards qualify only for shipments in Type B(U) packages. According to the relevant international regulations, approval for this type of package is only required from the country of origin of the design. Unfortunately, some countries do not accept the original certificate of approval without further validation. Delays occur frequently in obtaining the validation or the extension of the expiry date. Certificates for the approval of radioactive material in special form may also require validation in certain countries.

4. ACTIONS TO SPEED TRANSPORTS OF SAFEGUARDS SAMPLES AND PRELIMINARY RESULTS

Actions to reduce the transport delays must take into account the legal status of Euratom (Community Legislation) and the Agency (Agreements with Member States). It is therefore evident that the authors came to conclusions differing in several points. The main points of each approach are presented below.

4.1. Euratom approach

The average transport delays now experienced for the shipment of Euratom samples must be reduced. Waiting times for the transport of samples containing Pu of six months or more are unacceptable for effective safeguards implementation. These times should be reduced so that they are shorter than the detection times, which are an important element of Euratom inspection goals. There are a number of short term and long term measures which could lead to a substantial reduction of the transport delays:

— *Global forward import/export licensing and transport authorization*

The total number of samples to be transported can be estimated from the inspectors’ sampling forecasts. It would then be possible to apply in advance for the necessary import/export licences and for transport authorization. If the competent authorities in the Member States concerned are prepared to consider and process such advance requests, this could lead to an overall reduction in transport delays.

— *Exemption of Euratom samples from import/export and transport licensing*

It is in the interest of all Member States that anomalies in the nuclear materials balance should be detected quickly and efficiently. They should therefore not impose any restrictions on the movement of Euratom samples. The best solution would be to implement the Protocol on Privileges and Immunities in such a way
that shipment of Euratom samples does not require import/export licences and that advance authorization can be granted for transport activities. As an alternative, the Commission of the European Communities could consider issuing a regulation on the subject, which would however be a lengthy process.

— Nuclear risk insurance

The Commission of the European Communities as the owner of the samples is liable for any damage resulting from their transport. At present this risk is covered by a special nuclear risk insurance taken out for the Commission by the carrier. Under Article 188 of the Treaty establishing the European Communities, any such policy is automatically valid in all Community countries.

— Package validation in the different Member States

This does not pose a major problem. There is a limited number of standard packages for the transport of small quantities of nuclear material and advance approval should be sought for packagings used for Euratom samples.

— Transport monitoring

It should be mentioned in passing that procedures have been issued and implemented to keep track of samples, to monitor delays and to ensure that the results are available for the various evaluations.

4.2. IAEA concept

The Agency developed a concept (STR-176) [4] which was recently distributed to the Member States for comments. This concept is based on:

— a standardized composition of an individual shipment
— national and international regulations for the safe transport of radioactive materials
— the application in its full scope of the privileges and immunities of the Agency to safeguards samples
— facility-specific transport procedures controlled by the Agency
— a system to keep track of safeguards samples during transport.

The concept was successfully tested at a facility at which parts of the concept could be already implemented.

4.2.1. Standardized shipment composition

The type and number of samples to be collected during an inspection depend on the inventory and the inventory flow of a facility at which safeguards are applied. These values are well known by the Agency.
The amount of nuclear material in a sample is prescribed by the sampling procedure, which is agreed on between the Agency and the facility operator and takes into account facility specific parameters [5].

The Agency has characterized all types of safeguards samples with respect to parameters other than nuclear materials but also relevant for transport safety.

With these data available, the Agency prescribes the composition of a consignment and the appropriate container type for the shipment (excepted, Type A, Type B(U) packagings) and prepares in advance the shipment documentation, based on maximum figures.

4.2.2. International and national regulations for the transport of radioactive materials

Shipments of Agency’s safeguards samples are carried out by airborne, border-crossing transports. Regulations governing such transports are based on the IAEA Regulations for the Safe Transport of Radioactive Material [6] and laid down in the ICAO Technical Instructions for the Safe Transport of Dangerous Goods by Air [7]. This document also contains variations imposed by several countries.

With the information contained in these documents, the Agency arranges in advance

— supply of approved, validated or otherwise accepted containers;
— general shipment approvals, if requested;
— notifications of the transport to the State Authorities involved.

The Agency requests its authorized transport agents to take care of the nuclear risk insurance and relevant security regulations. For shipments initiated at Agency premises, the Agency holds a global insurance policy.

4.2.3. Import/export of nuclear material

Samples collected for safeguards purposes are transferred at the facility into the custody of the Agency and are therefore exempt from customs duties and prohibitions and restrictions on import and export [2]. An instrument of acceptance of these privileges is deposited in every Member State where samples are collected for safeguards purposes.

In the past, the Agency has failed to inform the organizations involved in such transports of the provisions laid down in the agreement quoted above. With the distribution of STR-176 to the Member States, the Agency confirmed that the privileges should apply also to safeguards samples.
4.2.4. **Facility specific procedures for the transport of safeguards samples**

The actions described above should establish the prerequisites for a facility-specific transport procedure which allows expediting of safeguards samples.

The Agency and its authorized transport agent undertake preparatory activities:

- to supply transport equipment and documentation in advance
- to waive restrictions on export and import of nuclear material
- to obtain the necessary approvals and to maintain a global insurance policy.

Upon arrival at the facility, the inspector determines together with the operator the sampling schedule and notifies the transport agent about the expected end of sampling and the number of consignments.

Simultaneously with the sampling activities at the facility, the transport agent makes the necessary arrangements and is ready to pick up and forward the consignment at the expected date of the end of sampling.

The Agency, notified in advance, arranges the delivery of the consignment to the Safeguards Analytical Laboratory (SAL).

4.2.5. **Procedure for keeping track of safeguards samples**

The Agency has also set up a procedure to monitor the transport of the samples [8]. This procedure calls for a variety of notifications from the planning of the inspection to the receipt of the samples and allows the Agency to take appropriate actions whenever the sample flow is interrupted.

4.2.6. **Test of the concept**

In 1985, the Agency conducted an experiment at a Federal German reprocessing plant to test the components of the concept. In the period under consideration (9 months), 59 spent fuel solutions, 8 Pu product and 11 U product samples were shipped to the Agency in 14 consignments.

The mean time elapsed between end of sampling and receipt at SAL was:

- 7 days for spent fuel samples
- 6 days for plutonium samples
- 5 days for uranium samples.

A further reduction of these delays appears to be feasible, when this procedure is in routine use.

5. **CONCLUSIONS**

The proposals presented in this paper aim at expediting the transport of samples determined for chemical analysis and although some elements are applicable only to
materials in the custody of the organizations represented by the authors, they also contain other elements which may be of use for the transport of such samples in general.

The major sources of delay are defined in this paper and the solutions recommended are as follows:

— **Export/import**

Most organizations dealing with shipments of nuclear materials in small amounts are not in the position to obtain exemption from export/import requirements. However, State Authorities approached indicated their willingness to issue licences for a total amount of nuclear material transported in a number of shipments over a defined period of time. In the case of Euratom a regulation on the subject might solve the problem.

— **Consignment composition**

The amount of nuclear materials in samples collected for chemical analysis is not exactly known. However, maximum figures can be derived with sufficient accuracy considering the sampling procedures and the characteristics of the items to be sampled. On this basis, the necessary documentation can be prepared in advance.

— **Transport containers**

Containers that qualify for the shipment of the samples can be determined on the basis of the consignment composition. Accepted, approved or validated containers are now available through the manufacturers or the transport agents.

— **Shipment approvals and global insurance**

Shipment approvals can be sought on a general basis or for a number of shipments from the Competent Authorities involved. Special care must be taken that the consignment is properly insured.

Effective implementation of IAEA safeguards in the States pursuant to the safeguards agreements and efficient application of Community law entail frequent transfers of radioactive samples, often involving more than one country.

Delays are detrimental to the value of any analytical programme, owing to chemical or physical changes in the sample, radiolytic decomposition, lateness of analysis results or other factors. The radioactive content of the samples is not known until the sample has been analysed after shipment. In some cases, this lack of information contributes significantly to the delays. We hope that the suggested new approaches will be welcomed by Member States and will progressively reduce the delays encountered at present.
REFERENCES


EMERGENCY RESPONSE
RESCUE CONTAINER SYSTEM FOR DEFECTIVE CYLINDERS LOADED WITH URANIUM HEXAFLUORIDE

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Urenco, Almelo,
Netherlands

Abstract

RESCUE CONTAINER SYSTEM FOR DEFECTIVE CYLINDERS LOADED WITH URANIUM HEXAFLUORIDE.

A system for rescuing defective UF₆ cylinders following an accident during transport or storage has been developed. It consists of a provisional repair method for the cylinders and a special rescue container licensed for transport. The system has been approved by the Federal German authorities. The container was used during the recovery activities after the 'Mont Louis' accident.

INTRODUCTION

Federal German licensing guidelines for the storage of UF₆ [1] require the availability of a rescue container for UF₆ to collect and ship leaking cylinders. Such a rescue container must be licensed for transport and it must be possible to unload its contents.

Unloading of the UF₆ is the problem with such a container. In Ref. [1] it is proposed to equip the rescue container with all the heating equipment, valves and instrumentation required for unloading. According to Ref. [2] such a container can be designed and built, but it is obvious that such a solution is extremely costly and complicated to operate in an emergency situation.

Transnuklear (TN) in co-operation with Urenco has worked out an improved solution which at the same time has increased the reliability of the system and drastically reduced the cost. The system for rescuing a defective UF₆ cylinder comprises the following features:

(1) Provisional repair of the defective cylinder by glass fibre reinforced polyester and epoxy resin compounds. It has been demonstrated that this repair is adequate to withstand the temperature and pressure necessary for emptying [3].
(2) Transport of the sealed cylinder in the TN rescue container (Figs 1–3). The container can be loaded/unloaded in a horizontal position. This simplifies handling operations.

(3) Contract with a company such as Urenco, Almelo, which has unloading facilities for UF$_6$ cylinders. Here the cylinder will be taken out of the rescue container and emptied by conventional methods.
FIG. 3. Rescue container with the top cover removed.

The system has been approved by the Federal German licensing authorities. The first rescue container was commissioned in July 1984. The characteristics of the container are as follows:

(a) The container is built in compliance with national/international regulations and codes for tank containers: ADR, RID, GGVS, GGVE, CSC (licence No. D/53300/TC issued by the Bundesanstalt für Materialprüfung, Berlin (West)).

(b) Handling attachments allow the use of conventional container transport and handling equipment.
(c) Technical data:

<table>
<thead>
<tr>
<th>Specification</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tare weight</td>
<td>4,980 kg</td>
</tr>
<tr>
<td>Maximum payload</td>
<td>19,020 kg</td>
</tr>
<tr>
<td>Maximum weight</td>
<td>24,000 kg</td>
</tr>
<tr>
<td>Total water capacity</td>
<td>9,800 L</td>
</tr>
<tr>
<td>Tank material</td>
<td>TSE.E 355</td>
</tr>
<tr>
<td>Shell thickness</td>
<td>8 mm</td>
</tr>
<tr>
<td>Maximum allowed working pressure</td>
<td>3 bar</td>
</tr>
<tr>
<td>Test pressure</td>
<td>4.5 bar (3 x 1.5)</td>
</tr>
<tr>
<td>Design pressure</td>
<td>13.0 bar</td>
</tr>
<tr>
<td>Rupture disk</td>
<td>3.75 bar</td>
</tr>
<tr>
<td>Design temperature</td>
<td>$-40^\circ$ C – $+65^\circ$ C</td>
</tr>
<tr>
<td>Dimensions</td>
<td>L = 6058 mm</td>
</tr>
<tr>
<td></td>
<td>H = 2200 mm</td>
</tr>
<tr>
<td></td>
<td>B = 2300 mm</td>
</tr>
</tbody>
</table>

REFERENCES

PROPOSED EMERGENCY RESPONSE SYSTEM FOR SEA TRANSPORT (ERSST)*

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Federal Republic of Germany

The Mont Louis accident in August 1984 highlighted a series of problems which could occur in the event of shipping accidents anywhere in the world. Accidents of this kind which produce severe damage (Amoco Cadiz) or cause no consequent damage (Mont Louis) require study of the many consequences from the viewpoint of transport legislation.

When an enterprise claims that it is specialized in the field of worldwide nuclear transport, it cannot wait for possible legal action but must demonstrate its competence by producing practicable, financially viable solutions using the means at its disposal.

International ocean going cargoes can produce problems for an international freight forwarder who, owing to the limited volume of freight, is not able to use special ships. Such problems include:

- seldom used routes and destinations
- use of shipping lines whose safety standards may be questionable
- an inability to influence the routes selected by shipping lines
- a lack of communication between shipowners, shipping agents and the freight forwarder in the event of unforeseen incidents
- no direct influence on the part of the freight forwarder in the event of unforeseen incidents
- a lack of supervision of the cargo while en route.

For such reasons, Transnuklear GmbH (TN) is developing its own programme to deal more effectively with possible problems through action on the following:

(1) Exclusive use of conference line carriers.
(2) Requesting routes which, as far as possible, bypass crisis areas and avoid transit ports.
(3) Agreement between the freight forwarder and the carrier that it is the carrier's duty to inform the freight forwarder in the case of untoward events.
(4) Constant reporting of the location of the cargo (not just the location of the transport) using satellite communication to the headquarters of the freight forwarder, especially for physical protection of shipments where definite and effective surveillance must be provided.

* Only a summary is published here.
(5) Constant supervision by a specialist international sea emergency service which has been supplied with the cargo movements by the freight forwarder, who is kept informed on a regular basis.

(6) Establishment of a permanent expert group which, in the case of emergency, forms an emergency team within a few hours. The team would consist of two qualified members of the freight forwarder and two to three members of a competent international salvage operator. Its task is to give technical and radiological advice to the captain, the shipowner and authorities in charge. It should travel as close as possible to the place of an event.

All of these aspects have been studied by TN. Some have been in operation for some years (e.g. use of conference lines, selected routes and ports of transit). Some are to be introduced shortly. A small number still need research and will be tested in the future.

However, TN is convinced that these measures (ERSST) are well within the capabilities of a freight forwarder. When coupled with changes in transport legislation, they will make a considerable contribution to the more effective organization and supervision of international ocean going transport of dangerous cargoes.
L'ACCIDENT DU MONT-LOUIS ET LA SECURITE NUCLEAIRE

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Abstract—Résumé

THE MONT-LOUIS ACCIDENT AND NUCLEAR SAFETY.

On 25 August 1984, the French cargo ship Mont-Louis, belonging to CGM, sank 15 km off the coast of Ostend, after having been involved in a collision with a ferry. It settled on a sand bank 15 metres deep at low tide. This commonplace type of accident, an outcome of the tough law of the sea, rapidly became an international issue on account of the cargo it carried. The cargo included 30 type 48Y containers of less than 1% enriched uranium hexafluoride intended for the USSR and weighing 350 t. In view of the nature of the cargo, particularly its value, it was decided to salvage the UF₆ containers as quickly as possible and to recover the material. This paper describes the salvage operations which, despite the difficulties associated with sea conditions normal at this time of year, ended with the recovery of all the 30 containers; they were all intact except one, which had a slight leak in the valve, thereby showing that the behaviour of the containers came up to expectations. Nevertheless, just as for any accident, there are lessons to be learnt from it. That is why French and international authorities as well as users have taken note of a number of interesting points covering both the regulatory and technical sides. Finally, some ideas are given on the application of the Paris Convention on Third Party Liability in the Field of Nuclear Energy and the protocol on intervention at sea in the event of pollution by radioactive materials.

L'ACCIDENT DU MONT-LOUIS ET LA SECURITE NUCLEAIRE.

Le 25 août 1984, à 15 km au large d’Ostende, le cargo français Mont-Louis, appartenant à la CGM, coulait après avoir été abordé par un transbordeur. Il gisait par 15 mètres de fond à marée basse sur un banc de sable. Cet accident, banal, conséquence de la dure loi de la mer, allait prendre rapidement une dimension internationale du fait de sa cargaison. En effet, il comprenait notamment 30 conteneurs de type 48Y remplis d'hexafluorure d'uranium d'enrichissement inférieur à 1%, destinés à l'URSS, d'un poids de 350 tonnes d'UF₆. Compte tenu de la nature de la cargaison, en particulier de sa valeur, la décision fut prise de procéder le plus rapidement possible à la récupération des conteneurs d'UF₆ afin de récupérer la matière. Ce mémoire décrit les opérations de récupération qui, malgré des difficultés inhérentes à l'état de la mer habituelles en cette période de l'année, se sont achevées par la récupération de l'ensemble des 30 conteneurs, tous intègres à l'exception d'un seul qui présentait une très petite fuite au niveau de la vanne, démontrant que la tenue de ces conteneurs a été conforme à ce qu'on pouvait en attendre. Néanmoins, comme pour tout accident, des enseignements doivent être tirés. C'est ainsi que les autorités françaises et internationales, ainsi que les utilisateurs, ont pris en considération un certain nombre de points intéressant tant la réglementation que l'aspect technique. Enfin, une réflexion est conduite au sujet de l'application de la Convention de Paris sur la responsabilité civile dans le domaine de l'énergie nucléaire et du protocole concernant l'intervention en haute mer en cas de pollution par des matières radioactives.
1. INTRODUCTION

Le jeudi 4 octobre 1984 à 5 h 30 du matin, le dernier conteneur d'hexafluorure d'uranium émergeait de la surface de la mer, retiré de l'épave du Mont-Louis. Ainsi s'achevait un événement qui, pendant plus de quarante jours, avait alimenté l'actualité française et internationale, démontrant une fois de plus que les questions nucléaires suscitent encore des réactions irrationnelles.

2. LES CIRCONSTANCES DE L'ACCIDENT ET LA CARGAISON

Le samedi 25 août 1984, par temps bouché, à 14 h 23 et à 18 km au large d'Ostende, le cargo français Mont-Louis de la CGM jaugeant 4000 tonnes, qui faisait route vers le port soviétique de Riga, était éperonné par le car-ferry Olen-Bratia transportant un millier de passagers vers la Grande-Bretagne. Quelques heures plus tard, le Mont-Louis coulait malgré les tentatives de la société de sauvetage belge URS, appelée immédiatement au secours pour le remorquer vers des hauts fonds sur lesquels il aurait pu s'échouer. L'épave gisait alors au large de l'embouchure de l'Escaut par 15 mètres de fond sur un banc de sable, couchée sur le côté, la partie supérieure étant découverte à marée basse.

Au journal télévisé de 20 heures, la télévision française relatait laconiquement cet événement. Un accident grave, puisque le navire avait coulé, mais non dramatique était ainsi annoncé. Le Mont-Louis s'ajoutait à la longue liste des navires qui coulent chaque année dans le monde, conséquence inéluctable de la dure loi de la mer.

Le lendemain 26 août, l'accident lui-même cédait la vedette à la nature de la cargaison du Mont-Louis. L'organisation écologiste Greenpeace annonçait que le Mont-Louis transportait une cargaison de produits radioactifs. L'information était confirmée peu après par la CGM, armateur du navire: il s'agissait en effet d'hexafluorure d'uranium. Dès le lundi 27 août, le directeur de l'Institut de protection et de sûreté nucléaire (IPSN) du CEA exposait la juste mesure des risques et, en particulier, l'insignifiance du risque nucléaire. Cependant, les jours suivants, certains médias, et ceci pendant plus d'un mois, allaient s'emparer de cette affaire, alimentant les rumeurs les plus alarmistes.

Des parlementaires indignés interpellaient le gouvernement français, parlant de «désastre sans précédent qui aurait pu se produire», tandis qu'au Parlement européen, une résolution était votée le 13 septembre demandant de prendre au niveau européen les mesures destinées à éviter à l'avenir «toute catastrophe de ce type».

Un comité de crise était mis en place par le gouvernement belge qui se préoccupait en premier lieu des produits pétroliers qui s'échappaient de l'épave.

De leur côté, les autorités françaises mettaient en place, sous l'autorité du secrétariat d'Etat à la Mer, un dispositif interministériel de coordination visant, d'une part, à évaluer plus précisément les risques éventuels, et d'autre part, à
récupérer la cargaison d'uranium dans les meilleures conditions de sécurité. La cargaison comprenait, outre du matériel divers, 30 conteneurs de type 48Y remplis d'hexafluorure d'uranium et répartis en 3 lots: 18 conteneurs d'UF$_6$ appauvri à 0,67% expédiés par la Cogéma, 9 conteneurs d'UF$_6$ naturel expédiés par la Comurhex et 3 conteneurs d'UF$_6$ enrichi à 0,88% expédiés par la Cogéma, soit un total de 350 tonnes d'UF$_6$ (236 tonnes d'uranium).

3. CONCEPTION DES EMBALLAGES 48Y

Les emballages 48Y, conçus il y a vingt ans aux États-Unis, sont utilisés universellement pour le transport de l'UF$_6$ enrichi à moins de 1%. Ce sont des cylindres (figure 1) à fonds bombés soudés, protégés par des jupes contre les chocs, d'une contenance maximale de 12 500 kg et d'un poids total en charge de 14 860 kg. Ils comportent une vanne de remplissage et de vidange vissée et étamée sur un des fonds (située en partie supérieure pendant le transport) et une bouteille en aluminium fixée sur le même fond que la vanne et à proximité de cette dernière. La vanne est protégée elle-même contre des chocs éventuels par un capot de protection.

Le transport d'hexafluorure d'uranium est soumis à la réglementation du transport des matières radioactives. Compte tenu du risque extrêmement faible présenté par la radioactivité de l'uranium naturel (activité: 0,7 mCi/kg U), la réglementation n'impose pas une tenue à des conditions accidentelles.

Du point de vue de la radioactivité, il n'y a pas de différence significative entre les trois lots transportés en raison des spécifications qui imposent une purification poussée de l'uranium à la sortie des usines de retraitement.

En fait, les emballages 48Y sont capables de résister à des conditions accidentelles sévères. Ceci est la conséquence, d'une part de la prise en compte par les concepteurs du risque additionnel chimique, et d'autre part, du fait qu'en raison de leur mode de remplissage et de vidange (en phase liquide sous une pression de 5 bar), ces conteneurs doivent répondre à la réglementation des appareils à pression. Leur paroi en acier doux de 16 mm d'épaisseur a été définie pour une pression de service de 14 bar. Leur fabrication et leur exploitation répondent à des spécifications très sévères. Ils sont soumis à des inspections périodiques tous les cinq ans, comprenant en particulier une épreuve sous 28 bar de pression interne et le changement systématique des vannes. Ces conteneurs sont utilisés depuis 20 ans et certains sont encore en exploitation depuis.

Alors que la réglementation du transport des matières radioactives n'exige pas pour ce type d'emballage de garantir leur intégrité à des conditions accidentelles, des évaluations ont été faites qui permettent de garantir leur tenue dans des conditions voisines de celles imposées aux emballages de type B et à une pression externe de 20 bar, soit 10 fois supérieure à celle à laquelle ont été soumis les conteneurs suite à l'accident du Mont-Louis. Compte tenu de ces performances et des conditions de l'accident du Mont-Louis, les experts consultés dès l'annonce
de l'accident ont ainsi pu garantir que les emballages avaient gardé leur intégrité, le seul problème pouvant résulter d'une rupture éventuelle au niveau de la vanne.

Néanmoins, des évaluations de risques ont été faites dans l'hypothèse hautement improbable d'une rupture d'un conteneur dans la mer. Des essais réalisés à l'intention des médias, quelques jours plus tard (bouteille contenant 1,9 kg d'UF₆ ouverte dans un bac de 100 litres d'eau de mer), confirmaient ces évaluations et montraient que le risque était limité. L'évaluation du risque part de l'hypothèse majorante où l'eau de mer entrait en réaction avec la totalité de l'hexafluorure contenu dans un conteneur. La réaction n'aurait aucun caractère explosif, l'acide fluorhydrique gazeux se dissoudrait très rapidement dans l'eau ou se dégagerait dans l'atmosphère; quant au fluorure d'uranyle, il se dissoudrait plus lentement avant de se déposer sur le fond. Le risque est essentiellement de nature chimique et il est dominé par le dégagement d'acide fluorhydrique. C'est sous forme gazeuse, par inhalation, qu'il est le plus important (le seuil létal s'établit à une concentration de 50 mg/m³ pendant 1 heure d'inhalation). Ceci explique que, pour les opérations de récupération, des masques à cartouche filtrante et des moyens de colmatage rapide de fuites ont été mis à la disposition des équipes chargées de la récupération au cas où une fuite serait découverte à la sortie d'un conteneur de la mer, ce qui a été le cas pour un des conteneurs.

Dans l'eau, la dilution de l'acide fluorhydrique intervient très rapidement. On peut considérer qu'aucun effet toxique ne serait décelable en dessous d'une
concentration de l'ordre du milligramme par litre. Dans ces conditions, avec l'hypothèse majorante d'une réaction instantanée entre tout l'hexafluorure contenu dans un conteneur et l'eau de mer, ce qui libérerait environ 3 tonnes d'acide fluorhydrique, on évalue le pic de concentration pendant la première demi-journée à 20 mg/L à 500 m. L'effet des courants et des marées devait permettre un retour à la normale dans des temps de l'ordre de quinze jours.

Le risque radioactif est peu significatif, l'atteinte à l'homme ne pouvant se faire que par reconcentration dans les poissons et les coquillages. L'impact maximal a été évalué à moins de 10% de la limite autorisée pour la population.

4. LA RECUPERATION

Compte tenu de la valeur marchande importante représentée par une telle cargaison et des risques présentés pour l'environnement (risque radioactif négligeable, mais risque toxique), la décision a été prise de repêcher les conteneurs et, dès le 26 août, un patrouilleur de la Marine nationale était dépêché sur les lieux pour procéder à des prélèvements d'eau pour analyses. Le navire océanographique belge Belgica procédait aussi toutes les six heures à des prélèvements en surface et en eau profonde. Aucune radioactivité et aucune teneur en fluor anormales n'étaient décelées, confirmant l'avis des experts sur l'état des conteneurs. Des prélèvements effectués par la suite pendant toute la durée de l'opération, qui a duré plus d'un mois, ont toujours été négatifs (plus de 700 analyses ont été ainsi effectuées).

Le lundi 27 août, les dispositions étaient prises pour procéder au repêchage dans les plus brefs délais des conteneurs. Cette opération était confiée par la CGM aux sociétés belge URS et hollandaise Smit Tak.

Un ponton grue de 120 mètres de long, équipé de deux grues de 40 tonnes, était sur le site dès le vendredi 31 août. Pendant toute l'opération, des techniciens du groupe CEA et de la société Transnucleaire ont assisté, à la demande de la CGM, les entreprises de récupération à titre de conseil pour la protection sanitaire et les mesures à mettre en œuvre au cas où il y aurait des fuites au niveau des conteneurs. Des prélèvements dans la cale, au milieu des conteneurs, à trois niveaux, ont été faits le 1er septembre afin de s'assurer qu'aucune fuite n'existaît sur le chargement. Les opérations pouvaient donc commencer une semaine après l'accident, ce qui constitue un délai remarquablement court.

Un conteneur vide était dépêché de Pierrelatte au port d'Ostende afin que les plongeurs se familiarisent avec les emballages et des protège-vannes adaptables sous l'eau étaient réalisés de manière à remplacer les protège-vannes dans l'hypothèse où ces derniers auraient été détériorés ou arrachés après la collision. Cette mesure s'est avérée fondée. Un conteneur enveloppe a été approvisionné en République fédérale d'Allemagne et installé sur la barge pour le cas où une grave détérioration serait constatée sur un emballage. Un contrôle détaillé de l'état de chaque
emballage récupéré a été fait sur la barge par des spécialistes habilités. Des procédures d’inspection, d’arrimage sur la barge, de contrôle de contamination et de débit de dose, d’étiquetage ainsi que de manutention et de réparation éventuelle (colmatage sur place des fuites avec un produit polyester) ont été rédigées à cette fin.

Afin de faciliter la récupération, une brèche a été pratiquée sur la partie avant de la coque du navire.

Les opérations qui s'effectuaient de jour ou de nuit à marée basse ont été entravées par l'état de la mer: tempêtes multiples (vent de force 7 à 8), courantes en cette période de l'année, et opacité de l'eau (suspension de grains de sable sur ces fonds) ont rendu très difficile la localisation des conteneurs, lesquels s'étaient désarrimés. Les «mafis» sur lesquels ils reposaient par groupes de trois n'ont en effet pas résisté lors du renversement du bateau et se sont vrillés en libérant les conteneurs. Lors de la tempête du 10 septembre, le navire s'est cassé en deux au niveau de la brèche qui avait été pratiquée. Une nouvelle brèche a été effectuée les 17 et 18 septembre à l'aide d'explosifs pour faciliter l'accès aux conteneurs.

Le premier conteneur 48Y était récupéré le jeudi 13 septembre à midi (figure 2), et le dernier conteneur était récupéré le jeudi 4 octobre à 5 h 30 du matin.
5. ETAT DES CONTENEURS

Les conteneurs ont été récupérés intégrès du point de vue de l'étanchéité, à l'exception d'un seul qui a présenté une légère fuite au niveau de la vanne. Ils avaient tous subi des déformations quelquefois importantes des jupes et des viroles. Les jupes de protection ont joué efficacement leur rôle. Par contre, les capots de protection des vannes ont été insuffisants et même nuisibles. Ils ont mal supporté les chocs latéraux et sont à l'origine de la détérioration des vannes qui ont subi des déformations, allant jusqu'à des inclinaisons de 60°, ce qui n'a cependant pas altéré leur étanchéité, sauf pour une vanne, démontrant ainsi leur grande flexibilité (figure 3).

Dès réception sur la barge, les conteneurs étaient arrimés et inspectés, une couche de vernis protectrice était déposée systématiquement sur la vanne pour garantir l'étanchéité et un protège-vanne neuf était posé.

FIG. 3. Vanne très déformée d'un conteneur récupéré.
Le conteneur qui a présenté une légère fuite a été mis par mesure de précaution dans l'enveloppe de protection après avoir eu sa fuite colmatée, et il a été transporté immédiatement à Pierrelatte. Un contrôle à l'arrivée a fait apparaître que le conteneur était resté étanche pendant le transport. Une mesure de poids a mis en évidence une prise de poids de l'ordre de 50 kg, ce qui a confirmé l'existence d'une fuite au niveau de la vanne avec une entrée d'eau de mer de l'ordre de cette quantité.

Les autres conteneurs ont été transportés par la barge au port de Dunkerque. L'ensemble de l'opération de récupération était achevé le 4 octobre.

Après contrôle général sur le quai, les conteneurs ont été placés sur des wagons et acheminés par train spécial à Pierrelatte. Les contrôles effectués lors du déchargement ont montré qu'aucune fuite ne s'était produite pendant ce transport depuis le port de Dunkerque.

L'expertise finale à Pierrelatte de chacun des conteneurs, par mesure de poids et de la pression interne, a mis en évidence que seulement deux conteneurs avaient subi une perte d'étanchéité du fait de leur immersion, correspondant respectivement à des entrées d'eau de 7 kg et 50 kg.

6. **ENSEIGNEMENTS**

L'accident du Mont-Louis ne peut pas être considéré comme un événement imprévu compte tenu de la probabilité relativement élevée d'accident maritime et de perte de navire qui en découle. Dans le déroulement et les suites de l'accident, la tenue des conteneurs a été conforme à ce qu'on pouvait attendre. L'évaluation des risques a confirmé qu'il s'agissait de matières relativement peu dangereuses en cas d'immersion en mer. Si le naufrage avait eu lieu en eaux plus profondes, au-delà de 200 m, on peut imaginer qu'ils auraient perdu leur étanchéité, mais la dilution aurait été plus efficace et l'impact sans réelle signification. L'accident n'a eu aucun impact sur l'environnement et ne conduit donc pas à remettre en cause la réglementation du transport des matières radioactives. Une réflexion a été néanmoins entreprise, tant en France qu'à l'étranger, pour en tirer les enseignements. Cette réflexion porte sur différents points, en particulier le problème du transport des matières présentant plusieurs dangers, les conditions du transport maritime des matières dangereuses, les mesures à prendre dans le but d'assurer les meilleures conditions possibles d'intervention et l'organisation de l'information du public en cas d'accident.

6.1. **Prise en compte de tous les dangers**

L'ensemble des dangers présentés par une matière radioactive doit être pris en compte au niveau réglementaire. C'est le cas de l'UF₆ qui, outre le danger radioactif, présente un danger chimique d'ailleurs prépondérant pour l'UF₆ natu-
6.1. Les conditions du transport maritime et l'organisation de l'intervention

Lors de l'accident du Mont-Louis, les dispositifs d'arrimage des conteneurs ont cédé, rendant plus difficiles encore les opérations de récupération. Les pratiques usuelles de fixation des colis à bord des navires nous paraissent devoir faire l'objet d'une réflexion en vue d'aboutir à un code de bonne pratique qui prenne en compte les accidents et les conditions de récupération éventuelles des colis.
A la suite de l'accident du Mont-Louis, certains ont proposé que le transport maritime des matières radioactives soit interdit ou soumis à des règles particulièrement contraignantes.

C'est oublier que le faible trafic des matières radioactives, associé au danger potentiel relativement peu élevé, au risque résiduel négligeable, ceci résultant de la réglementation internationale, conduit à un risque très faible comparé à celui présenté par d'autres matières dangereuses et, en particulier, les matières chimiques.

Les dispositions mises en place apparaissent donc globalement satisfaisantes. Une réflexion générale sur les risques présentés par le transport maritime de toutes les matières dangereuses serait par contre utile. Parmi les dispositions existantes, on peut citer, pour notre pays, le système SURNAV d'obligation d'information préalable des autorités responsables de tout mouvement du navire ayant à bord des matières nucléaires dans les eaux territoriales, permettant un suivi et une intervention en cas d'accident. Par ailleurs, le plan NUCMAR d'intervention en cas d'accident définit l'organisation des pouvoirs publics et les dispositions à mettre en place pour intervenir en cas d'accident dans les eaux territoriales ou au-delà de ces limites lorsque le littoral et les intérêts connexes sont menacés (Convention de Bruxelles du 29 novembre 1969). Ce plan prévoit en particulier la récupération éventuelle des colis. Des moyens doivent être développés pour effectuer ces opérations, permettant de relever des charges pouvant aller jusqu'à 100 t (colis de combustibles irradiés) et à des profondeurs de l'ordre de 200 m.

De manière à faciliter leur récupération, les colis devraient être conçus pour résister à une immersion de 200 m comme d'ailleurs la réglementation de l'AIEA le prévoit pour les colis d'éléments combustibles irradiés. Nous pensons qu'il faudrait étendre cette exigence à d'autres colis pour lesquels on estime qu'en cas d'immersion la récupération serait décidée; c'est le cas, par exemple, des colis de plutonium et d'UF₆. Ces exigences ne sont d'ailleurs pas exorbitantes car, pour d'autres raisons, la conception de ces colis leur confère une tenue à la pression externe généralement très élevée.

Pour terminer sur le problème de l'intervention, il ne faut pas oublier le contexte juridique qui peut amener des litiges, par exemple dans le cas où celle-ci est imposée par un Etat pour prévenir un risque ou ce qu'on appelle, au sens de la Convention de Paris, un accident nucléaire, si l'expéditeur, seul responsable légal, estime que cette intervention ne se justifie pas.

6.3. L'information du public

L'accident du Mont-Louis a mis en évidence, une fois de plus, la difficulté de l'information du public qui peut être soumis à un «matraquage» dramatisant l'événement de la part de certains médias. En cas d'accident mettant en cause la sécurité nucléaire, l'information au sens le plus large revêt une importance capitale. Il ne s'agit pas, en effet, d'organiser uniquement la circulation d'informations entre les différentes autorités publiques et les divers responsables des services ou
établissements concernés, mais également d’apporter à la population, aux représentants élus et au public, de façon générale, les informations que ceux-ci sont en droit d’obtenir. En l’absence d’informations ou sur la base de déclarations officielles insuffisamment coordonnées ou non cohérentes, les médias peuvent transformer un accident de type industriel, dans la plupart des cas sans gravité, en un événement susceptible de revêtir dans l’opinion publique une dimension sans commune mesure avec ses conséquences réelles. Pour ces raisons, il importe d’abord de réagir le plus rapidement possible à l’événement par une information immédiate (cette information peut et doit être donnée par les acteurs du transport, qui par leur responsabilité, connaissent tous les éléments de l’expédition et notamment les risques présentés par l’expédition concernée: il s’agit de l’expéditeur, du transporteur et du destinataire), puis de donner des informations coordonnées, cohérentes, au plan local et national si nécessaire. Une directive interministérielle vient d’être à cet effet, édictée dans notre pays (Directive SGSN en date du 13 mars 1986) qui confie cette coordination, au niveau local, au préfet maritime concerné et, au niveau national, au ministère chargé des transports.
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