

IAEA SAFETY STANDARDS SERIES

Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material

SAFETY GUIDE

No. TS-G-1.1 (ST-2)



INTERNATIONAL
ATOMIC ENERGY AGENCY
VIENNA



Documents

Document Sharing Hub

IAEA SAFETY RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

Under the terms of Article III of its Statute, the IAEA is authorized to establish standards of safety for protection against ionizing radiation and to provide for the application of these standards to peaceful nuclear activities.

The regulatory related publications by means of which the IAEA establishes safety standards and measures are issued in the **IAEA Safety Standards Series**. This series covers nuclear safety, radiation safety, transport safety and waste safety, and also general safety (that is, of relevance in two or more of the four areas), and the categories within it are **Safety Fundamentals**, **Safety Requirements** and **Safety Guides**.

Safety Fundamentals (blue lettering) present basic objectives, concepts and principles of safety and protection in the development and application of nuclear energy for peaceful purposes.

Safety Requirements (red lettering) establish the requirements that must be met to ensure safety. These requirements, which are expressed as 'shall' statements, are governed by the objectives and principles presented in the Safety Fundamentals.

Safety Guides (green lettering) recommend actions, conditions or procedures for meeting safety requirements. Recommendations in Safety Guides are expressed as 'should' statements, with the implication that it is necessary to take the measures recommended or equivalent alternative measures to comply with the requirements.

The IAEA's safety standards are not legally binding on Member States but may be adopted by them, at their own discretion, for use in national regulations in respect of their own activities. The standards are binding on the IAEA in relation to its own operations and on States in relation to operations assisted by the IAEA.

Information on the IAEA's safety standards programme (including editions in languages other than English) is available at the IAEA Internet site

www.iaea.org/ns/coordinet

or on request to the Safety Co-ordination Section, IAEA, P.O. Box 100, A-1400 Vienna, Austria.

OTHER SAFETY RELATED PUBLICATIONS

Under the terms of Articles III and VIII.C of its Statute, the IAEA makes available and fosters the exchange of information relating to peaceful nuclear activities and serves as an intermediary among its Member States for this purpose.

Reports on safety and protection in nuclear activities are issued in other series, in particular the **IAEA Safety Reports Series**, as informational publications. Safety Reports may describe good practices and give practical examples and detailed methods that can be used to meet safety requirements. They do not establish requirements or make recommendations.

Other IAEA series that include safety related sales publications are the **Technical Reports Series**, the **Radiological Assessment Reports Series** and the **INSAG Series**. The IAEA also issues reports on radiological accidents and other special sales publications. Unpriced safety related publications are issued in the **TECDOC Series**, the **Provisional Safety Standards Series**, the **Training Course Series**, the **IAEA Services Series** and the **Computer Manual Series**, and as **Practical Radiation Safety Manuals** and **Practical Radiation Technical Manuals**.

This publication has been superseded by SSG-26.

ADVISORY MATERIAL FOR THE
IAEA REGULATIONS FOR THE
SAFE TRANSPORT OF
RADIOACTIVE MATERIAL

The following States are Members of the International Atomic Energy Agency:

AFGHANISTAN	GHANA	PANAMA
ALBANIA	GREECE	PARAGUAY
ALGERIA	GUATEMALA	PERU
ANGOLA	HAITI	PHILIPPINES
ARGENTINA	HOLY SEE	POLAND
ARMENIA	HUNGARY	PORTUGAL
AUSTRALIA	ICELAND	QATAR
AUSTRIA	INDIA	REPUBLIC OF MOLDOVA
AZERBAIJAN	INDONESIA	ROMANIA
BANGLADESH	IRAN, ISLAMIC REPUBLIC OF	RUSSIAN FEDERATION
BELARUS	IRAQ	SAUDI ARABIA
BELGIUM	IRELAND	SENEGAL
BENIN	ISRAEL	SIERRA LEONE
BOLIVIA	ITALY	SINGAPORE
BOSNIA AND HERZEGOVINA	JAMAICA	SLOVAKIA
BOTSWANA	JAPAN	SLOVENIA
BRAZIL	JORDAN	SOUTH AFRICA
BULGARIA	KAZAKHSTAN	SPAIN
BURKINA FASO	KENYA	SRI LANKA
CAMBODIA	KOREA, REPUBLIC OF	SUDAN
CAMEROON	KUWAIT	SWEDEN
CANADA	LATVIA	SWITZERLAND
CENTRAL AFRICAN REPUBLIC	LEBANON	SYRIAN ARAB REPUBLIC
CHILE	LIBERIA	TAJIKISTAN
CHINA	LIBYAN ARAB JAMAHIRIYA	THAILAND
COLOMBIA	LIECHTENSTEIN	THE FORMER YUGOSLAV REPUBLIC OF MACEDONIA
COSTA RICA	LITHUANIA	TUNISIA
CÔTE D'IVOIRE	LUXEMBOURG	TURKEY
CROATIA	MADAGASCAR	UGANDA
CUBA	MALAYSIA	UKRAINE
CYPRUS	MALI	UNITED ARAB EMIRATES
CZECH REPUBLIC	MALTA	UNITED KINGDOM OF GREAT BRITAIN AND NORTHERN IRELAND
DEMOCRATIC REPUBLIC OF THE CONGO	MARSHALL ISLANDS	UNITED REPUBLIC OF TANZANIA
DENMARK	MAURITIUS	UNITED STATES OF AMERICA
DOMINICAN REPUBLIC	MEXICO	URUGUAY
ECUADOR	MONACO	UZBEKISTAN
EGYPT	MONGOLIA	VENEZUELA
EL SALVADOR	MOROCCO	VIET NAM
ESTONIA	MYANMAR	YEMEN
ETHIOPIA	NAMIBIA	YUGOSLAVIA, FEDERAL REPUBLIC OF
FINLAND	NETHERLANDS	ZAMBIA
FRANCE	NEW ZEALAND	ZIMBABWE
GABON	NICARAGUA	
GEORGIA	NIGER	
GERMANY	NIGERIA	
	NORWAY	
	PAKISTAN	

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".

© IAEA, 2002

Permission to reproduce or translate the information contained in this publication may be obtained by writing to the International Atomic Energy Agency, Wagramer Strasse 5, P.O. Box 100, A-1400 Vienna, Austria.

Printed by the IAEA in Austria
June 2002
STI/PUB/1109

This publication has been superseded by SSG-26.

SAFETY STANDARDS SERIES No. TS-G-1.1 (ST-2)

ADVISORY MATERIAL FOR THE IAEA REGULATIONS FOR THE SAFE TRANSPORT OF RADIOACTIVE MATERIAL

SAFETY GUIDE

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2002

VIC Library Cataloguing in Publication Data

Advisory material for the IAEA regulations for the safe transport of radioactive material : safety guide. — Vienna : International Atomic Energy Agency, 2002.

p. ; 24 cm. — (Safety standards series, ISSN 1020-525X;
no. TS-G-1.1 (ST-2))

STI/PUB/1109

ISBN 92-0-111802-3

Includes bibliographical references.

1. Radioactive substances — Transportation — Safety regulations.
I. International Atomic Energy Agency. II. Series.

VICL

02-00271

FOREWORD

by Mohamed ElBaradei
Director General

One of the statutory functions of the IAEA is to establish or adopt standards of safety for the protection of health, life and property in the development and application of nuclear energy for peaceful purposes, and to provide for the application of these standards to its own operations as well as to assisted operations and, at the request of the parties, to operations under any bilateral or multilateral arrangement, or, at the request of a State, to any of that State's activities in the field of nuclear energy.

The following bodies oversee the development of safety standards: the Commission for Safety Standards (CSS); the Nuclear Safety Standards Committee (NUSSC); the Radiation Safety Standards Committee (RASSC); the Transport Safety Standards Committee (TRANSSC); and the Waste Safety Standards Committee (WASSC). Member States are widely represented on these committees.

In order to ensure the broadest international consensus, safety standards are also submitted to all Member States for comment before approval by the IAEA Board of Governors (for Safety Fundamentals and Safety Requirements) or, on behalf of the Director General, by the Publications Committee (for Safety Guides).

The IAEA's safety standards are not legally binding on Member States but may be adopted by them, at their own discretion, for use in national regulations in respect of their own activities. The standards are binding on the IAEA in relation to its own operations and on States in relation to operations assisted by the IAEA. Any State wishing to enter into an agreement with the IAEA for its assistance in connection with the siting, design, construction, commissioning, operation or decommissioning of a nuclear facility or any other activities will be required to follow those parts of the safety standards that pertain to the activities to be covered by the agreement. However, it should be recalled that the final decisions and legal responsibilities in any licensing procedures rest with the States.

Although the safety standards establish an essential basis for safety, the incorporation of more detailed requirements, in accordance with national practice, may also be necessary. Moreover, there will generally be special aspects that need to be assessed on a case by case basis.

The physical protection of fissile and radioactive materials and of nuclear power plants as a whole is mentioned where appropriate but is not treated in detail; obligations of States in this respect should be addressed on the basis of the relevant instruments and publications developed under the auspices of the IAEA. Non-radiological aspects of industrial safety and environmental protection are also not explicitly considered; it is recognized that States should fulfil their international undertakings and obligations in relation to these.

The requirements and recommendations set forth in the IAEA safety standards might not be fully satisfied by some facilities built to earlier standards. Decisions on the way in which the safety standards are applied to such facilities will be taken by individual States.

The attention of States is drawn to the fact that the safety standards of the IAEA, while not legally binding, are developed with the aim of ensuring that the peaceful uses of nuclear energy and of radioactive materials are undertaken in a manner that enables States to meet their obligations under generally accepted principles of international law and rules such as those relating to environmental protection. According to one such general principle, the territory of a State must not be used in such a way as to cause damage in another State. States thus have an obligation of diligence and standard of care.

Civil nuclear activities conducted within the jurisdiction of States are, as any other activities, subject to obligations to which States may subscribe under international conventions, in addition to generally accepted principles of international law. States are expected to adopt within their national legal systems such legislation (including regulations) and other standards and measures as may be necessary to fulfil all of their international obligations effectively.

EDITORIAL NOTE

An appendix, when included, is considered to form an integral part of the standard and to have the same status as the main text. Annexes, footnotes and bibliographies, if included, are used to provide additional information or practical examples that might be helpful to the user.

The safety standards use the form 'shall' in making statements about requirements, responsibilities and obligations. Use of the form 'should' denotes recommendations of a desired option.

The English version of the text is the authoritative version.

PREFACE

This Advisory Material is not a stand-alone text. It only has significance when used concurrently as a companion to the IAEA Safety Standards Series No. ST-1¹, *Regulations for the Safe Transport of Radioactive Material* (1996 edition), denoted henceforth as “the Regulations”. To facilitate cross-reference between it and the Regulations, each paragraph of the Advisory Material is numbered in correspondence with the paragraph of the Regulations to which it most directly relates. To distinguish paragraphs of the Advisory Material from those of the Regulations for reference purposes, Advisory Material paragraphs always have a numeral after the decimal point, even when there is only one subparagraph of text. Thus, for example, advice relating to para. 401 of the Regulations should be initially sought under para. 401.1 of the Advisory Material. Integral paragraph numbers which are cited in the text, either alone or accompanied by lower case letters in brackets, should be taken as identifying paragraphs of the Regulations.

Also, the publications listed under “References” are the versions which were used in the development of the 1996 edition of the Regulations and this Advisory Material. Some of the publications may have been superseded by later revisions. These may be consulted for the most recent information recognizing that the earlier editions are the basis for the discussions which follow.

Since the first edition in 1961, the Regulations for the Safe Transport of Radioactive Material of the International Atomic Energy Agency (IAEA Regulations) have served as the basis of safety for the transport of radioactive material worldwide. The provisions of the IAEA Regulations have been adopted in national regulations by most of the Member States of the Agency. The international regulatory bodies having responsibility for the various modes of transport have also implemented the IAEA Regulations. The safety record since the inception, and throughout several comprehensive revisions, of the Regulations has demonstrated the efficacy both of the regulatory provisions and of the arrangements for ensuring compliance with them.

In the discussions leading to the first edition of the IAEA Regulations, it was realized that there was need for a publication to supplement the Regulations which could give information on individual provisions as to their purpose, their scientific background and how to apply them in practice. The scientific basis of the classification of radioisotopes for transport purposes, then in use, and the factors that

¹ The *Regulations for the Safe Transport of Radioactive Material* were issued in 1996 as Safety Standards Series No. ST-1. In 2000 the Regulations were issued in English, with minor editorial corrections, as Safety Standards Series No. TS-R-1 (ST-1, Revised).

have to be taken into account by competent authorities in approving package designs, were examples adduced in support of this concept at the time. In response, the Agency published Safety Series No. 7, entitled, in its first edition in 1961, “Notes on Certain Aspects of the Regulations”.

As experience in applying the Regulations grew, it became increasingly evident that, while the provisions of the Regulations might be essentially clear and unambiguous, nevertheless they would often also be highly technical in nature and unavoidably complex. Moreover they intentionally state no more than ‘what’ must be achieved in relation to package characteristics and operational conditions in order to assure safety. They do not seek to prescribe ‘how’ the user should comply; indeed the freedom to innovate and to develop new ways to ensure compliance is recognized as intrinsically desirable in such a technically advanced field. An additional source of information on the Regulations, providing advice on ‘how’ to comply with them which could be augmented from time to time in the light of latest experience, was therefore provided by the Agency, initially in relation to the 1973 edition of the Regulations. This was entitled “Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material”. It was designated Safety Series No. 37.

Up to the time of publication of the previous edition of the IAEA Regulations, in 1985, Safety Series No. 37 had reached its third edition. Meanwhile, Safety Series No. 7, which embodied information on the scientific basis and rationale of the Regulations, had been retitled “Explanatory Material for the IAEA Regulations for the Safe Transport of Radioactive Material” and, embodying mainly information on the scientific basis and rationale of the Regulations, was in its second edition.

During the current regulatory revision, which culminated in 1996, the Agency’s senior advisory body for transport, the Transport Safety Standards Advisory Committee (TRANSSAC), in consultation with the Agency’s Publishing Section, agreed that it would be a useful simplification to combine the two Safety Guides previously known as Safety Series No. 7 and Safety Series No. 37 in a single publication, to be known as “Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material”. This would have the advantage of consolidating supporting information on the Regulations in one place, eliminating duplication. The advisory nature of the present publication has been made paramount. The inclusion of some explanatory material supports this function since a proper understanding of the background to the regulatory provisions helps users to interpret them correctly and to comply with them fully.

Thus the primary purpose of this publication (henceforth referred to as the Advisory Material) is to provide guidance to users on proven and acceptable ways of complying with and demonstrating compliance with the Regulations. It must be emphasized that the text is not to be construed as being uniquely prescriptive. It offers recommendations on ways of complying but it does not lay down ‘the only way’ to comply with any specific provision.

This publication has been superseded by SSG-26.

Member States and international organizations are invited to take note of this publication and to bring it to the attention of persons and organizations who make use of, or are subject to, the IAEA Regulations. Moreover, readers are encouraged to send, through their competent authority, any comments they may wish to make, including proposals for modifications, additions or deletions, to the International Atomic Energy Agency.

CONTENTS

SECTION I.	INTRODUCTION	1
	Background	1
	Objective	1
	Scope	1
	Reference to Section I	3
SECTION II.	DEFINITIONS	5
	References to Section II	28
SECTION III.	GENERAL PROVISIONS	29
	Radiation protection	29
	Emergency response	34
	Quality assurance	35
	Compliance assurance	36
	Special arrangement	38
	References to Section III	39
SECTION IV.	ACTIVITY LIMITS AND MATERIAL RESTRICTIONS	41
	Basic radionuclide values	41
	Determination of basic radionuclide values	43
	Contents limits for packages	44
	References to Section IV	48
SECTION V.	REQUIREMENTS AND CONTROLS FOR TRANSPORT	51
	Requirements before the first shipment	51
	Requirements before each shipment	53
	Transport of other goods	55
	Other dangerous properties of contents	56
	Requirements and controls for contamination and for leaking packages	57
	Requirements and controls for transport of excepted packages	63

Requirements and controls for transport of LSA material and SCO in industrial packages or unpackaged	67
Determination of transport index	68
Determination of criticality safety index	70
Limits on transport index, criticality safety index and radiation levels for packages and overpacks	71
Categories	71
Marking, labelling and placarding	73
Consignor's responsibilities	79
Transport and storage in transit	82
Customs operations	93
Undeliverable consignments	94
References to Section V	94
 SECTION VI. REQUIREMENTS FOR RADIOACTIVE MATERIALS AND FOR PACKAGINGS AND PACKAGES	
Requirements for radioactive materials	97
General requirements for all packagings and packages	101
Additional requirements for packages transported by air	104
Requirements for excepted packages	105
Requirements for industrial packages	105
Requirements for packages containing uranium hexafluoride	110
Requirements for Type A packages	113
Requirements for Type B(U) packages	119
Requirements for Type B(M) packages	134
Requirements for Type C packages	136
Requirements for packages containing fissile material	137
References to Section VI	148
 SECTION VII. TEST PROCEDURES	
Demonstration of compliance	153
Tests for special form radioactive material	160
Tests for low dispersible radioactive material	162
Tests for packages	163
References to Section VII	194

SECTION VIII. APPROVAL AND ADMINISTRATIVE REQUIREMENTS	199
General aspects	199
Approval of special form radioactive material and low dispersible radioactive material	200
Approval of package designs	200
Transitional arrangements	202
Notification and registration of serial numbers	206
Approval of shipments	207
Approval of shipments under special arrangement	208
Competent authority approval certificates	209
Contents of approval certificates	210
Validation of certificates	213
Reference to Section VIII	214
 APPENDIX I: THE Q SYSTEM FOR THE CALCULATION AND APPLICATION OF A_1 AND A_2 VALUES	 215
Introduction	215
Background	216
Basis of the Q system	217
Dosimetric models and assumptions	219
Special considerations	229
Applications	233
Tabulation of Q values	237
Decay chains used in the Q system	252
Conclusions	252
References to Appendix I	255
 APPENDIX II: HALF-LIFE AND SPECIFIC ACTIVITY OF RADIONUCLIDES, DOSE AND DOSE RATE COEFFICIENTS OF RADIONUCLIDES AND SPECIFIC ACTIVITY	 259
Reference to Appendix II	285
 APPENDIX III: EXAMPLE CALCULATIONS FOR ESTABLISHING MINIMUM SEGREGATION DISTANCE REQUIREMENTS	 287

Introduction	287
Below main deck stowage of one group of packages	
in passenger aircraft	289
Below main deck stowage of multiple groups of packages	
in passenger aircraft	292
Main deck stowage on combi or cargo aircraft	294
Segregation distances for undeveloped film	295
References to Appendix III	295
 APPENDIX IV: QUALITY ASSURANCE IN THE SAFE TRANSPORT OF RADIOACTIVE MATERIAL	 297
Introduction	297
Quality assurance programmes	301
Organization	302
Document control	303
Design control	304
Procurement control	306
Material control	307
Process control	307
Inspection and test control	308
Non-conformity control	310
Corrective actions	310
Records	310
Staff and training	311
Servicing	311
Audits	311
Definitions of terms used in Appendix IV	312
References to Appendix IV	313
 APPENDIX V: PACKAGE STOWAGE AND RETENTION DURING TRANSPORT	 315
Introduction	315
Types of retention system	315
Package acceleration factor considerations	317
Demonstrating compliance through testing	319
Examples of retention system designs and assessments	320
Definitions of terms used in Appendix V	326
References to Appendix V	327

APPENDIX VI: GUIDELINES FOR SAFE DESIGN OF SHIPPING PACKAGES AGAINST BRITTLE FRACTURE	329
Introduction	329
General consideration of evaluation methods	330
Considerations for fracture mechanics	334
Safety factors for Method 3	337
Evaluation procedure for Method 3	339
References to Appendix VI	344
APPENDIX VII: CRITICALITY SAFETY ASSESSMENTS	347
Introduction	347
Package description	347
Criticality safety analysis models	348
Method of analysis	350
Validation of calculational method	352
Calculations and results	358
Special issues	365
Design and operational issues	368
References to Appendix VII	370
CONTRIBUTORS TO DRAFTING AND REVIEW	375
BODIES FOR THE ENDORSEMENT OF SAFETY STANDARDS	380
INDEX	382

LIST OF TABLES

Table I	Correction factors for package and detector sizes	22
Table II	Sample segregation between classes of dangerous goods ...	85
Table III	Comparison of the four volumetric leak test methods recommended by Aston et al. [3]	98
Table IV	List of VRI codes by country	211
Table I.1	Dose coefficients for submersion	229
Table I.2	Type A package contents limits	238
Table I.3	Decay chains used in the Q system	253
Table II.1	Half-life and specific activity of radionuclides	259
Table II.2	Dose and dose rate coefficients of radionuclides	272

Table II.3	Specific activity values for uranium at various levels of enrichment	286
Table III.1	Transmission factors	290
Table III.2	Variation of segregation distance with transport index for a single group of packages stowed below main deck on a passenger aircraft	291
Table III.3	Variation of segregation distance with transport index for main deck stowage on a combi or cargo aircraft	294
Table IV.1	Basic elements of quality assurance programmes that should be considered and addressed in the safe transport of radioactive material	300
Table V.1	Acceleration factors for package retention system design	318
Table V.2	Acceleration factors for package retention system design for specific packages	319
Table V.3	Symbols used in calculation of a rectangular package with baseplate flange bolted to the conveyance	325

Section I

INTRODUCTION

BACKGROUND

103.1. When making national or international shipments it is necessary to consult the Regulations for the particular mode of transport to be used for the countries where the shipment will be made. While most of the major requirements are in agreement with the Regulations, there can be differences with respect to the assignment of responsibilities for carrying out specific actions. For air shipments, the International Civil Aviation Organization's (ICAO) Technical Instructions for the Safe Transport of Dangerous Goods by Air and the International Air Transport Association's (IATA) Dangerous Goods Regulations should be consulted, with particular regard to the State and operator variations. For sea shipments, the International Maritime Organization's (IMO) International Maritime Dangerous Goods (IMDG) Code should be consulted. Some countries have adopted the Regulations by reference while others have incorporated them into their national regulations with possibly some minor variations.

OBJECTIVE

104.1. In general the Regulations aim to provide a uniform and adequate level of safety that is commensurate with the inherent hazard presented by the radioactive material being transported. To the extent feasible, safety features are required to be built into the design of the package. By placing primary reliance on the package design and preparation, the need for any special actions during carriage, i.e. by the carrier, is minimized. Nevertheless, some operational controls are required for safety purposes.

SCOPE

106.1. Transport includes carriage by a common carrier or by the owner or employee where the carriage is incidental to the use of the radioactive materials, such as vehicles carrying radiography devices being driven to and from the operations site by the radiographer, vehicles carrying density measuring gauges being driven to and from the construction site, and oil well logging vehicles carrying measuring devices containing radioactive materials and radioactive materials used in oil well injections.

107.1. The Regulations are not intended to be applied to movements of radioactive material that forms an integral part of a means of transport, such as depleted uranium counterweights or tritium exit signs used in aircraft; or to radioactive material in persons or animals for medical or veterinary purposes, such as cardiac pacemakers or radioactive material introduced into humans or animals for diagnosis or treatment. The treating physician or veterinarian should give appropriate advice on radiological safety.

107.2. Consumer products are items available to the general public as the end user without further control or restriction. These may be devices such as smoke detectors, luminous dials or ion generating tubes that contain small amounts of radioactive substances. Consumer products are outside the scope of the Regulations only after sale to the end user. Any transport, including the use of conveyances between manufacturers, distributors and retailers, is within the scope of the Regulations to ensure that large quantities of individually exempted consumer products are not transported in an unregulated manner.

107.3. The principles of exemption and their application to the transport of radioactive material are dealt with in para. 401.

107.4. The scope of the Regulations includes consideration of those natural materials or ores which form part of the nuclear fuel cycle or which will be processed in order to use their radioactive properties. The Regulations do not apply to other ores which may contain naturally occurring radionuclides, but whose usefulness does not lie in the fissile, fertile or radioactive properties of those nuclides, provided that the activity concentration does not exceed 10 times the exempt activity concentration values. In addition, the Regulations do not apply to natural materials and ores containing naturally occurring radionuclides which have been processed (up to 10 times the exempt activity concentration values) where the physical and/or chemical processing was not for the purpose of extracting radionuclides, e.g. washed sands and tailings from alumina refining. Were this not the case, the Regulations would have to be applied to enormous quantities of material that present a very low hazard. However, there are ores in nature where the activity concentration is much higher than the exemption values. The regular transport of these ores may require consideration of radiation protection measures. Hence, a factor of 10 times the exemption values for activity concentration was chosen as providing an appropriate balance between the radiological protection concerns and the practical inconvenience of regulating large quantities of material with low activity concentrations of naturally occurring radionuclides.

108.1. Although these Regulations provide for the requisite safety in transport without the need for specified routing, the regulatory authorities in some Member

States have imposed routing requirements. In prescribing routes, normal and accident risks, both radiological and non-radiological, as well as demographic considerations should be taken into account. Policies embodied in the routing restrictions should be based upon all factors that contribute to the overall risk in transporting radioactive material and not only on concerns for 'worst case' scenarios, i.e. 'low probability/high consequence' accidents. Since the authorities at the State, provincial or even local levels may be involved in routing decisions, it may often be necessary to provide them with either evaluations to assess alternative routes or with very simple methods which they can use.

108.2. In assessing the radiological hazards and ensuring that the routing requirements do not detract from the standards of safety specified in the Regulations, analyses using appropriate risk assessment codes should be undertaken. One such code which may be used, INTERTRAN [1], was developed through a co-ordinated research programme of the IAEA. This computer based environmental impact code is available for use by Member States. In spite of many uncertainties stemming from the use of a generalized model and the difficulty of selecting appropriate input values for accident conditions, this code may be used to calculate and understand, at least on a qualitative basis, the factors significant in determining the radiological impact from routing alternatives involving the transport of radioactive material. These factors are the important aspects that should be considered in any routing decision. For routing decisions involving a single mode of transport, many simplifying assumptions can be made and common factors can be assigned which result in easy to use relative risk evaluation techniques.

108.3. The consignor may also be required to provide evidence that measures to meet the requirements for safeguards and physical protection associated with radioactive nuclear material shipments are being complied with.

109.1. See paras 506 and 507.

REFERENCE TO SECTION I

- [1] INTERNATIONAL ATOMIC ENERGY AGENCY, INTERTRAN: A System for Assessing the Impact from Transporting Radioactive Material, IAEA-TECDOC-287, IAEA, Vienna (1983).

Section II

DEFINITIONS

A₁ and A₂

201.1. See Appendix I.

Approval

204.1. The approval requirements in the Regulations have been graded according to the hazards posed by the radioactive material to be transported. Approval is intended to ensure that the design meets the relevant requirements and that the controls required for safety are adequate for the country and for the circumstances of the shipment. Since transport operations and conditions vary between countries, application of the ‘multilateral approval’ approach provides the opportunity for each competent authority to satisfy itself that the shipment is to be properly performed, with due account taken of any peculiar national conditions.

204.2. The concept of multilateral approval applies to transport as it is intended to occur. This means that only those competent authorities through whose jurisdiction the shipment is scheduled to be transported are involved in its approval. Unplanned deviations which occur during transport and which result in the shipment entering a country where the transport had not previously been approved would need to be handled individually. For this reason the definition of multilateral approval is limited to countries “through or into which the consignment is transported” and specifically excludes countries over which the shipment may be transported by aircraft. The countries that will be flown over are often not known until the aircraft is in the air and receives an air traffic control clearance. If an aircraft is scheduled to stop in a country, however, multilateral approval includes approval by the competent authority of that country.

204.3. Users of the Regulations should be aware that a Member State may require in its national regulations that an additional approval be given by its competent authority for any special form radioactive material, Type B(U) and Type C package which is to be used for domestic transport on its territory, even if the design has already been approved in another country.

205.1. For unilateral approval it is believed that the Regulations take into account the transport conditions which may be encountered in any country. Consequently,

only approval by the competent authority of the country of origin of the design is required.

Carrier

206.1. The term ‘person’ includes a body corporate as well as an individual (see also the Basic Safety Standards (BSS) [1], paras 2.10–2.14).

Competent authority

207.1. The competent authority is the organization defined by legislative or executive authority to act on behalf of a country, or an international authority, in matters involving the transport of radioactive material. The legal framework of a country determines how a national competent authority is designated and is given the responsibility to ensure application of the Regulations. In some instances, authority over different aspects of the Regulations is assigned to different agencies, depending on the transport mode (air, road, rail, sea or inland waterway) or the package and radioactive material type (excepted, industrial, Type A, Type B and Type C packages; special form radioactive material, low dispersible material; fissile material or uranium hexafluoride). A national competent authority may in some cases delegate the approval of package designs and certain types of shipment to another organization having the necessary technical competence. National competent authorities also constitute the competent authorities referred to in any conventions or agreements on the transport of radioactive material to which the country adheres.

207.2. The competent authority should make the consignors, carriers, consignees and public aware of its identity and how it may be contacted. This may be accomplished by publishing the organizational identity (department, administration, office, etc.), with a description of the duties and activities of the organization in question as well as detailed mailing address, telephone and facsimile numbers, email address, etc.

207.3. The primary source of competent authority identifications is the list of National Competent Authorities Responsible for Approvals and Authorizations in Respect of the Transport of Radioactive Material, which is published annually by the IAEA and is available on request. Each country should ensure that the listed information is current and accurate. The IAEA requests verification of this information annually, and prompt responses by Member States will ensure the continued value of this list.

Compliance assurance

208.1. See paras 311.1–311.9.

Confinement system

209.1. The confinement system should be that part of a package necessary to maintain the fissile material in the configuration that was assumed in the criticality safety assessment for an individual package (see para. 678). The confinement system could be (1) an inner receptacle with defined dimensions, (2) an inner structure maintaining the outer dimension of a fuel assembly and any interstitial fixed poisons, or (3) a complete package such as an irradiated nuclear fuel package with no inner container. The confinement system consists of specified packaging components and the package contents. Although the confinement system may have the same boundary as the containment system, this is not always the case since the confinement system maintains criticality control whereas the containment system prevents leakage of radioactive material. Each competent authority must concur that the confinement system defined in the criticality safety assessment is appropriate for the package design, for both damaged and undamaged configurations (see para. 678).

Containment system

213.1. The containment system can be the entire packaging but, more frequently, it makes up a portion of the packaging. For example, in a Type A package the containment system may be considered to be the vial containing the radioactive contents. The vial, its enclosing lead pot shielding and fibreboard box make up the packaging. The containment system does not necessarily include the shielding. In the case of special form radioactive material and low dispersible radioactive material, the radioactive material may be part of the containment system (see para. 640).

213.2. The leaktightness requirement for a containment system in a Type B(U), Type B(M) or Type C package depends on the radiotoxicity of the radioactive contents; for example, a Type B(U) or Type C package under accident conditions must have the release limited to a value of A_2 in the period of a week. This connection to the A_2 value means that for highly toxic radionuclides such as plutonium and americium the allowable volumetric leak rate will be much lower than for low enriched uranium. However, if fissile material is able to escape from the containment system under accident conditions, it must be demonstrated that the quantity that escapes is consistent with that assumed in the criticality safety assessment in applying para. 682(c).

Contamination

214.1. Contamination includes two types of radioactive material on surfaces or embedded in surfaces, namely fixed contamination and non-fixed contamination. There is no definitive distinction between fixed and non-fixed contamination, and various terms have been used to describe the distinction. For practical purposes a distinction is made between contamination which, during routine conditions of transport, remains in situ (i.e. fixed contamination) and, therefore, cannot give rise to hazards from ingestion, inhalation or spreading, and non-fixed contamination which may contribute to these hazards. The only hazard from fixed contamination is that due to external radiation exposure, whereas the hazards from non-fixed contamination include the potential for internal exposure from inhalation and ingestion as well as external exposure due to contamination of the skin should it be released from the surface. Under accident conditions, and under certain use conditions such as weathering, fixed contamination may, however, become non-fixed contamination.

214.2. Contamination below levels of 0.4 Bq/cm^2 for beta and gamma emitters and for low toxicity alpha emitters, or 0.04 Bq/cm^2 for all other alpha emitters (see also para. 508), can give rise only to insignificant exposure through any of these pathways.

214.3. Any surface with levels of contamination lower than 0.4 Bq/cm^2 for beta and gamma emitters and low toxicity alpha emitters or 0.04 Bq/cm^2 for all other alpha emitters is considered a non-contaminated surface in applying the Regulations. For instance, a non-radioactive solid object with levels of surface contamination lower than the above levels is out of the scope of the Regulations, and no requirement is applicable to its transport.

215.1. See paras 214.1–214.3.

216.1. See paras 214.1–214.3.

Criticality safety index

218.1. The criticality safety index (CSI) is a new term defined for the first time in the 1996 edition of the Regulations. The 1973 and 1985 editions of the Regulations used the 'transport index' for both radiological control and control of criticality safety of packages containing fissile material. These editions of the Regulations defined the transport index (TI) so that a single number accommodated both radiological safety and criticality safety considerations. As the operational controls needed for radiological protection and for criticality safety are essentially independent, this edition of the Regulations has separated the CSI from the TI, which is now defined (see para. 243) for

radiological control only. This separation into two indices enables a clear recognition of the basis for operational control of a fissile package and eliminates potential unnecessary restrictions caused by the use of one index. However, with this new control, care should be taken not to confuse the ‘new TI’ and the ‘old TI’ used in the previous edition of the Regulations. Awareness of this change is necessary to ensure proper labelling for criticality safety (see paras 544 and 545) and criticality control for packages, overpacks and freight containers containing fissile material using the newly introduced CSI.

218.2. The CSI is a number used to control criticality safety for a shipment of fissile material and is obtained by dividing the number 50 by the value of N (see para. 528). The accumulation of packages containing fissile material is required to be controlled in individual consignments (see paras 529 and 530), in conveyances, freight containers and overpacks (see paras 566(d) and 567) and in-transit storage (see paras 568 and 569). To facilitate such control, the CSI is required to be displayed on a label (see paras 544 and 545) which is specifically designed to indicate the presence of fissile material in the case of packages, overpacks or freight containers where contents consist of fissile material not excepted under the provisions of para. 672.

Exclusive use

221.1. The special features of an ‘exclusive use’ shipment are, by definition, first, that a single consignor must make the shipment and must have, through arrangements with the carrier, sole use of the conveyance or large freight container; and, second, that all initial, intermediate and final loading and unloading of the consignment is carried out only in strict accordance with directions from the consignor or consignee.

221.2. Since ordinary in-transit handling of the consignment under exclusive use will not occur, some of the requirements which apply to normal shipments can be relaxed. In view of the additional control which is exercised over exclusive use consignments, specific provisions have been made for them which allow:

- Use of a lower integrity industrial package type for low specific activity (LSA) materials;
- Shipment of packages with radiation levels exceeding 2 mSv/h (but not more than 10 mSv/h) at the surface, or a TI exceeding 10;
- Increase by a factor of two in the total number of criticality safety indices for fissile material packages in a number of cases.

Many consignors find that it is advantageous to make the necessary arrangements with the carrier to provide transport under exclusive use so that the consignor can utilize one or more of the above provisions.

221.3. In the case of packaged LSA material, the Regulations take into account the controlled loading and unloading conditions which result from transport under exclusive use. The additional controls imposed under exclusive use are to be in accordance with instructions prepared by the consignor or consignee (both of whom have full information on the load and its potential hazards), allowing some reduction in packaging strength. Since uncontrolled handling of the packages does not occur under exclusive use, the conservatism which is embodied in the normal LSA packaging requirements regarding handling has been relaxed, but equivalent levels of safety are to be maintained.

221.4. Packages which may be handled during transport must necessarily have their allowable radiation levels limited to protect the workers handling them. The imposition of exclusive use conditions and the control of handling during transport help to ensure that proper radiation protection measures are taken. By imposing restrictions and placing a limit on the allowable radiation levels around the vehicle, the allowable radiation level of the package may be increased without significantly increasing the hazard.

221.5. Since exclusive use controls effectively prevent the unauthorized addition of radioactive materials to a consignment and provide a high level of control over the consignment by the consignor, allowances have been made in the Regulations to authorize more fissile material packages than for ordinary consignments.

221.6. For exclusive use of a conveyance or large freight container, the sole use requirement and the sole control requirement are the determining factors. Although a vehicle may be used to transport only radioactive material, this does not automatically qualify the consignment as exclusive use. In order to meet the definition of exclusive use, the entire consignment has to originate from or be controlled by a single consignor. This excludes the practice of a carrier collecting consignments from several consignors in a single vehicle. Even though the carrier is consolidating the multiple consignments onto one vehicle, it is not in exclusive use because more than one consignor is involved. However, this does not preclude a properly qualified carrier or consignee who is consolidating shipments from more than one source from taking on the responsibilities of the consignor for these shipments and from being so designated.

Fissile material

222.1. A fission chain is propagated by neutrons. Since a chain reaction depends on the behaviour of neutrons, fissile material is packaged and shipped under requirements designed to maintain subcriticality and, thus, provide criticality safety in transport. In the Regulations the term 'fissile material' is occasionally used to refer both to fissile

radionuclides and to material containing these radionuclides. Users of the Regulations should remain alert to the context in which the term 'fissile material' is used.

222.2. Most radionuclides can be made to fission, but many can only be made to fission with difficulty and with special equipment and controlled conditions. The distinguishing characteristic of the fissile nuclides implied by the definition is that they are capable of supporting a self-sustaining thermal neutron (neutron energies less than approximately 0.3 eV) chain reaction by only the accumulation of sufficient mass. No other action, mechanism or special condition is required. For example, Pu-238 is no longer listed in the definition because, although it can be made to support a fast neutron chain reaction under stringent laboratory conditions, in the form in which it is encountered in transport it does not have this property. Plutonium-238 cannot under any circumstances support a chain reaction carried by thermal neutrons. It is, therefore, 'fissionable' rather than 'fissile'.

222.3. As indicated in the above paragraph, the basis used to select the nuclides defined as fissile material for the purposes of the Regulations relies on the ease of accumulating sufficient mass for a potential criticality. Other actinides that have the potential for criticality are discussed in ANSI/ANS-8.15-1981 [2] and subcritical mass limits are provided for isolated units of Np-237, Pu-238, Pu-240, Pu-242, Am-241, Am-242m, Am-243, Cm-243, Cm-244, Cm-245, Cm-247, Cf-249 and Cf-251. The predicted subcritical mass limits for these materials range from a few grams (Cf-251) to tens of kilograms. However, the lack of critical experiment data, limited knowledge of the behaviour of these nuclides under different moderator and reflection conditions and the uncertainty in the cross-section data for many of these nuclides require that adequate attention (and associated subcritical margin) be provided to operations where sufficient quantities of these nuclides might be present (or produced by decay before or during transport). Advice of the competent authority should be sought on the need and means of performing a criticality safety assessment per the requirements of paras 671–682 whenever significant quantities of these materials may be transported.

Freight container

223.1. The methods and systems employed in the transshipment of goods have undergone a transformation since about 1965; the freight container has largely taken the place of parcelled freight or general cargo which was formerly loaded individually. Packaged as well as unpackaged goods are loaded by the consignor into freight containers and are transported to the consignee without intermediate handling. In this manner, the risk of damage to packages is reduced, unpackaged goods are consolidated into conveniently handled units and transport economies are realized. In the case of large articles such as

contaminated structural parts from nuclear power stations, the container may perform the function of the packaging as allowed under para. 627.

223.2. Freight containers are typically designed and tested in accordance with the standards of the International Organization for Standardization (ISO) [3]. They should be approved and maintained in accordance with the International Convention for Safe Containers (CSC) [4] in order to facilitate international transport operations. If other freight containers are used, the competent authority should be consulted. It should be noted, however, that the testing prescribed in CSC is not equivalent to that prescribed in ISO 1496/1. For this reason the Regulations require the design standard to be ISO.

223.3. In addition, special rules may be specified by modal transport organizations. As an example, the International Maritime Dangerous Goods (IMDG) Code [5] contains the provisions for the transport by sea of dangerous goods including radioactive material.

Low dispersible radioactive material

225.1. The concept of low dispersible radioactive material applies only to qualification for exemption from the requirements for Type C packages in the air transport mode.

225.2. Low dispersible radioactive material has properties such that it will not give rise to significant potential releases or exposures. Even when subjected to high velocity impact and thermal environments, only a limited fraction of the material will become airborne. Potential radiation exposure from inhalation of airborne material by persons in the vicinity of an accident would be very limited.

225.3. The low dispersible radioactive material criteria are derived in consistency with other safety criteria in the Regulations, as well as on the basis of established methods to demonstrate acceptable radiological consequences. The Regulations require that the performance of low dispersible material be demonstrated without taking any credit for the Type B packaging in which it is transported.

225.4. Low dispersible radioactive material may be the radioactive material itself, in the form of an indispersible solid, or a high integrity sealed capsule containing the radioactive material, in which the encapsulated material acts essentially as an indispersible solid. Powders and powder-like materials cannot qualify as low dispersible material.

Low specific activity material

226.1. The reason for the introduction of a category of LSA material into the IAEA Regulations was the existence of certain solid materials the specific activities of which are so low that it is highly unlikely that, under circumstances arising in transport, a sufficient mass of such materials could be taken into the body to give rise to a significant radiation hazard. Uranium and thorium ores and their physical or chemical concentrates are materials falling into this category. This concept was extended to include other solid materials, on the basis of a model which assumes that it is most unlikely that a person would remain in a dusty atmosphere long enough to inhale more than 10 mg of material. If the specific activity of the material is such that the mass intake is equivalent to the activity intake assumed to occur for a person involved in a median accident with a Type A package, namely $10^{-6} A_2$, then this material would not present a greater hazard during transport than that presented by a Type A package. This leads to a low specific activity material limit of $10^{-4} A_2/g$.

226.2. Consideration was given to the possibility of shipping solid objects without any packaging. The question arose for concrete blocks (with activity throughout the mass), for irradiated objects and for objects with fixed contamination. Under the condition that the specific activity is relatively low and remains in or fixed on the object's surface, the object can be dealt with as a package. For the sake of consistency and safety, the radiation limits at the surface of the unpackaged object should not exceed the limits for packaged material. Therefore, it was considered that above the limits of surface radiation levels for packages (2 mSv/h for non-exclusive use and 10 mSv/h for exclusive use) the object must be packaged in an industrial package which assures shielding retention in routine transport. Similar arguments were made for establishing surface contamination levels for unpackaged surface contaminated objects (SCOs).

226.3. The preamble to the LSA definition does not include the unshielded radiation level limit of 10 mSv/h at 3 m (see para. 521), because it is a property of the quantity of material placed in a single package rather than a property of the material itself (although in the case of solid objects which cannot be divided, it is a property of the solid object).

226.4. The preamble also does not include wording relative to the essentially uniform distribution of the radionuclides throughout the LSA material. However, it states clearly that the material should be in such a form that an average specific activity can be meaningfully assigned to it. In considering actual materials shipped as LSA, it was decided that the degree of uniformity of the distribution should vary depending upon

the LSA category. The degree of uniformity is thus specified, as necessary, for each LSA category (see, for example, para. 226(c)(i)).

226.5. LSA-I was introduced in the 1985 edition of the Regulations to describe very low specific activity materials. These materials may be shipped unpackaged, or they may be shipped in Industrial Packages Type 1 (Type IP-1) which are designed to minimal requirements (para. 621). According to para. 226(a)(i), LSA-I materials cannot consist of: concentrates of ores other than uranium or thorium concentrates (for example, radium ore concentrate cannot be LSA-I material), unless they meet para. 226(a)(iv). In the 1996 edition of the Regulations the LSA-I category was revised to take into account:

- the clarification of the scope of the Regulations concerning ores other than uranium and thorium ores according to para. 107(e);
- fissile materials in quantities excepted from the package requirements for fissile material according to para. 672;
- the introduction of new exemption levels according to para. 236.

The definition of LSA-I was consequently modified to:

- include only those ores containing naturally occurring radionuclides which are intended to be processed for the use of these radionuclides (para. 226(a)(i));
- exclude fissile material in quantities not excepted under para. 672 (para. 226(a)(iii)); and
- add radioactive material in which the activity is distributed throughout in concentrations up to 30 times the exemption level (para. 226(a)(iv)).

Materials containing radionuclides in concentrations above the exemption levels have to be regulated. It is reasonable that materials containing radionuclides up to 30 times the exemption level may be exempted from parts of the transport regulations and may be associated with the category of LSA-I materials. The factor of 30 has been selected to take account of the rounding procedure used in the derivation of the Basic Safety Standards [1] exemption levels and to give a reasonable assurance that the transport of such materials does not give rise to unacceptable doses.

226.6. Uranium enriched to 20% or less may be shipped as LSA-I material either in Type IP-1 packages or unpackaged in fissile excepted quantities. However, amounts exceeding fissile excepted quantities (see para. 672) will be subject to the requirements for packages containing fissile material, thus precluding transport of the material unpackaged, or in unapproved packages.

226.7. The materials expected to be transported as LSA-II could include nuclear reactor process wastes which are not solidified, such as lower activity resins and filter sludges, absorbed liquids and other similar materials from reactor operations, and similar materials from other fuel cycle operations. In addition, LSA-II could include many items of activated equipment from the decommissioning of nuclear plants. Since LSA-II materials could be available for human intake after an accident, the specific activity limit is based upon an assumed uptake by an individual of 10 mg. Since the LSA-II materials are recognized as being clearly not uniformly distributed (e.g. scintillation vials, hospital and biological wastes and decommissioning wastes), the allowed specific activity is significantly lower than that of LSA-III. The factor of 20 lower allowed specific activity as compared with the limit for LSA-III compensates for localized concentration effects of the non-uniformly distributed material.

226.8. While some of the materials considered to be appropriate for inclusion in the LSA-III category would be regarded as essentially uniformly distributed (such as concentrated liquids in a concrete matrix), other materials such as solidified resins and cartridge filters are distributed throughout the matrix but are uniformly distributed to a lesser degree. The solidification of these materials as a monolithic solid which is insoluble in water and non-flammable makes it highly unlikely that any significant portion of it will become available for intake into a human body. The recommended standard is intended to specify the lesser degree of activity distribution.

226.9. The provisions for LSA-III are intended principally to accommodate certain types of radioactive waste consignments with an average estimated specific activity exceeding the 10^{-4} A₂/g limit for LSA-II materials. The higher specific activity limit of 2×10^{-3} A₂/g for LSA-III materials is justified by:

- restricting such materials to solids which are in a non-readily dispersible form, therefore explicitly excluding powders as well as liquids or solutions;
- the need for a leaching test to demonstrate sufficient insolubility of the material when exposed to weather conditions like rainfall (see para. 601.2);
- the higher package standard Industrial Package Type 3 (Type IP-3) under non-exclusive use conditions, which is the same as Type A for solids; in the case of Industrial Package Type 2 (Type IP-2) (para. 524), the lack of the water spray test and the penetration test is compensated for by the leaching test and by operational controls under the exclusive use conditions, respectively.

226.10. The specific activity limit for LSA-II liquids of 10^{-5} A₂/g, which is a factor of 10 more restrictive than for solids, takes into account that the concentration of a liquid may increase during transport.

226.11. A solid compact binding agent, such as concrete, bitumen, etc., which is mixed with the LSA material, is not considered to be an external shielding material. In this case, the binding agent may decrease the surface radiation level and may be taken into account in determining the average specific activity. However, if radioactive material is surrounded by external shielding material, which itself is not radioactive, as illustrated in Fig. 1, this external shielding material is not to be taken into account in determining the specific activity of the LSA material.

226.12. For LSA-II solids, and for LSA-III materials not incorporated into a solid compact binding agent, the Regulations require that the activity be distributed throughout the material. This provision puts no requirement on how the activity is distributed throughout the material, i.e. the activity does not need to be uniformly distributed. It is, however, important to recognize that the concept of limiting the estimated specific activity fails to be meaningful if in a large volume the activity is clearly confined to a small percentage of that volume.

226.13. It is prudent to establish a method by which the significance of the estimated average activity, as determined, can be judged. There are several methods that would be suitable for this particular purpose.

226.14. A simple method for assessing the average activity is to divide the volume occupied by the LSA material into defined portions and then to assess and compare the specific activity of each of these portions. It is suggested that the differences in specific activity between portions of a factor of less than 10 would cause no concern.

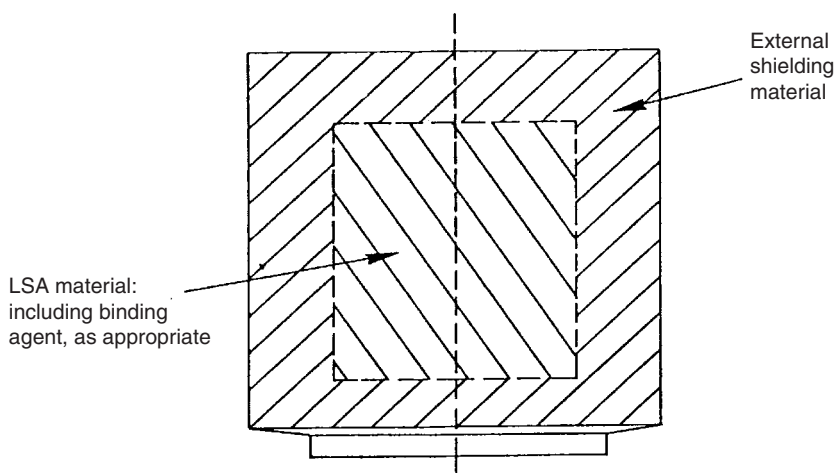


FIG. 1. Low specific activity material surrounded by a cylindrical volume of non-radioactive shielding material.

226.15. Judgement needs to be exercised in selecting the size of the portions to be assessed. The method described in para. 226.14 should not be used for volumes of material of less than 0.2 m^3 . For a volume between 0.2 m^3 and 1.0 m^3 , the volume should be divided into five, and for a volume greater than 1.0 m^3 into ten parts of approximately equivalent size.

226.16. For LSA-III materials consisting of radioactive material within a solid compact binding agent, the requirement is that they be essentially uniformly distributed in this agent. Since the requirement of ‘essentially uniformly distributed’ for LSA-III materials is qualitative, it is necessary to establish methods by which compliance with the requirement can be judged.

226.17. The following method is an example for LSA-III materials which are essentially uniformly distributed in a solid compact binding agent. The method is to divide the LSA material volume including the binding agent into a number of portions. At least ten portions should be selected, subject to the volume of each portion being no greater than 0.1 m^3 . The specific activity of each volume should then be assessed (through measurements, calculations or combinations thereof). It is suggested that specific activity differences between the portions of less than a factor of three would cause no concern. The factor of three in this procedure is more constraining than the suggested factor of ten in para. 226.14 because the ‘essentially uniformly distributed’ requirement is intended to be more constraining than the ‘distributed throughout’ requirement.

226.18. As a consequence of the definition of LSA material, additional requirements are specified for:

- (a) the quantity of LSA material in a single package with respect to the external radiation level of the unshielded material (see para. 521); and
- (b) the total activity of LSA material in any single conveyance (see para. 525 and Table V).

Both requirements can be much more restrictive than the basic requirements for LSA material given in para. 226. This can be seen from the following theoretical example: if it is assumed that a 200 L drum is filled with a solid combustible material with an estimated average specific activity of $2 \times 10^{-3} \text{ A}_2/\text{g}$, it would seem that this material could be transported as LSA-III. However, for example, if the density of the material is 1 g/cm^3 , the total activity in the drum will be 400 A_2 [$(2 \times 10^{-3} \text{ A}_2/\text{g}) (1 \text{ g/cm}^3) (2 \times 10^5 \text{ cm}^3) = 400 \text{ A}_2$] and transport as LSA-III would be precluded by the conveyance limit of 10 A_2 by inland waterway and of 100 A_2 by other modes (see Table V of the Regulations). See also para. 525.2.

226.19. Objects which are both activated or otherwise radioactive and contaminated cannot be considered as surface contaminated objects (SCOs) (see para. 241.5). However, such objects may qualify as LSA material since an object having activity throughout and also contamination distributed on its surfaces may be regarded as complying with the requirement that the activity be distributed throughout. For such objects to qualify as LSA material it is necessary to ascertain that the applicable limits on estimated average specific activity are complied with. In assessing the average specific activity, all radioactive material attributed to the object, i.e. both the distributed activity and the activity of the surface contaminations, needs to be included. As appropriate, additional requirements applicable to LSA material need to also be satisfied.

226.20. Compaction of material should not change the classification of the material. To ensure this, the mass of any container compacted with the material should not be taken into account in determining the average specific activity of the compacted material.

226.21. See also Appendix I.

Low toxicity alpha emitters

227.1. The identification of low toxicity alpha emitters is based on the specific activity of the radionuclide (or the radionuclide in its as-shipped state). For a nuclide with a very low specific activity, its intake cannot, because of its bulk, reasonably be expected to give rise to doses approaching the dose limit. The radionuclides U-235, U-238 and Th-232 have specific activities 4 to 8 orders of magnitude lower than Pu-238 or Pu-239 (4×10^3 to 8×10^4 Bq/g as compared with 2×10^9 to 6×10^{11} Bq/g). Although Th-228 and Th-230 have specific activities comparable with those of Pu-238 and Pu-239, they are only allowed as 'low toxicity alpha emitters' when contained in ores and physical and chemical concentrates, which inherently provides for the low activity concentration required.

Maximum normal operating pressure

228.1. The maximum normal operating pressure (MNOP) is the difference between the containment system maximum internal pressure and the mean sea-level atmospheric pressure for the conditions specified below.

228.2. The environmental conditions to be applied to a package in determining the MNOP are the normal environmental conditions specified in paras 653 and 654 or, in the case of air transport, in para. 618. Other conditions to be applied in determining

the MNOP are that the package is assumed to be unattended for a one year period and that it is subject to its maximum internal heat load.

228.3. A one year period exceeds the expected transit time for a package containing radioactive material; besides providing a substantial margin of safety in relation to routine conditions of transport, it also addresses the possibility of loss of a package in transit. The one year period is arbitrary but has been agreed upon as a reasonable upper limit for a package to remain unaccounted for in transit. Since the package is assumed to be unattended for one year, any physical or chemical changes to the packaging or its contents which are transient in nature and could contribute to increasing the pressure in the containment system need to be taken into account. The transient conditions that should be considered include: changes in heat dissipation capability, gas buildup due to radiolysis, corrosion, chemical reactions or release of gas from fuel pins or other encapsulations into the containment system. Some transient conditions may tend to reduce the MNOP, such as the reduction in pressure with time caused by a decrease in internal heat due to radioactive decay of the contents. These conditions may be taken into account if adequately justified.

Overpack

229.1. The carriage of a consignment from one consignor to one consignee may be facilitated by packing various packages or a single package, each of which fully complies with the requirements of the Regulations, into one overpack. Specific design, test or approval requirements for the overpack are not necessary since it is the packaging, not the overpack, which performs the protective function. However, the interaction between the overpack and the packages should be taken into account, especially concerning the thermal behaviour of the packages during routine and normal conditions of transport.

229.2. A rigid enclosure or consolidation of packages for ease of handling in such a way that package labels remain visible for all packages need not be considered as an overpack unless advantage is taken by the consignor of the determination of the TI of the overpack by direct measurement of the radiation level.

Package

230.1. The terms package and packaging are used to distinguish the assembly of components for containing the radioactive material (packaging) from this assembly of components plus the radioactive contents (package).

230.2. A package is the packaging plus its radioactive contents as presented for transport. For design and compliance assurance purposes, this may include any or all structural equipment required for handling or securing the package which is either permanently attached or assembled with the package.

230.3. In order to determine which structural components should be considered part of the package, it is necessary to examine the use and purpose of such equipment with respect to transport. If a package can only be transported with certain structural equipment, it is normal to consider that equipment part of the packaging. This does not mean that a trailer or transport vehicle should be considered part of the package in the case of dedicated transport.

230.4. Because the package may be transported either with or without certain structural equipment, it may be necessary to evaluate both situations in determining packaging suitability and compliance.

230.5. If certain equipment is attached during transport for handling purposes, it also may be necessary to consider its effect in normal and accident conditions of transport. In the case of Type B(U), Type B(M), Type C and packages designed to carry fissile material, the designer must reach agreement with the competent authority for certification.

230.6. A tank, freight container or intermediate bulk container with radioactive contents may be used as one of the types of package under these Regulations provided that it meets the prescribed design, test and any applicable approval requirements for that type of package. Alternatively, a tank, freight container or metal intermediate bulk container with radioactive contents may be used as an industrial package Type IP-2 or Type IP-3 if it meets the Type IP-1 requirements as well as other requirements which are specifically referenced in paras 625–628 of the Regulations.

Packaging

231.1. See paras 230.1 and 230.2.

Radiation level

233.1. One of the limiting quantities in radiological protection against exposure of people is the effective dose (the others being equivalent doses to the lens of the eye and to the skin (e.g. see Section II-8 of Ref. [1]). As this is not a directly measurable quantity, operational quantities had to be created which are measurable. These quantities are ‘ambient dose equivalent’ for strongly penetrating radiation and

‘directional dose equivalent’ for weakly penetrating radiation. The radiation level should be taken as the value of the operational quantity ‘ambient dose equivalent’ or ‘directional dose equivalent’, as appropriate.

233.2. In some cases consideration should be given to the possibility of an increase in radiation as a result of the buildup of daughter nuclides during transport. In such cases a correction should be applied to represent the highest radiation level envisaged during the transport.

233.3. In mixed gamma and neutron fields it may be necessary to make separate measurements. It should be ensured that the monitoring instrument being used is appropriate for the energy being emitted by the radionuclide and that the calibration of the instrument is still valid. In performing both the initial measurement and any check measurement, the uncertainties in calibration have to be taken into account.

233.4. For neutron dosimeters there is very often a significant dependence of the reading on the neutron energy. The spectral distribution of the neutrons used for calibration and the spectral distribution of the neutrons to be measured may affect the accuracy of dose determination considerably. If the energy dependence of the instrument reading and the spectral distribution of the neutrons to be measured are known, a corresponding correction factor may be used.

233.5. The Regulations require that, at the surfaces of packages and overpacks, specific radiation levels shall not be exceeded. In most cases a measurement made with a hand instrument held against the surface of the package indicates the reading at some distance away because of the physical size of the detector volume. The instrument used for the measurement of the radiation level should, where practicable, be small in relation to the dimensions of the package or overpack. Instruments which are large relative to the physical size of the package or overpack should not be used because they might underestimate the radiation level. Where the distance from the source to the instrument is large in relation to the size of the detector volume (e.g. a factor of five), the effect is negligible and can be ignored; otherwise the values in Table I should be used to correct the measurement. For radiographic devices where the source to surface distance is generally kept to a minimum, the effect is usually not negligible, and an allowance should be made for the size of the detector volume.

233.6. When monitoring finned flasks or other transport packages, care should be taken where narrow radiation beams may be encountered. A dose rate meter, with a detector area much larger than the cross-sectional area of the beam to be measured, will yield a proportionally reduced reading of dose rate because of averaging over the much larger detector area. An appropriate instrument should be chosen for the work.

Radioactive material

236.1. In previous editions of the Regulations, a single exemption value of 70 Bq/g was used to define radioactive material for transport purposes. Following publication of the BSS [1], it was recognized that this value had no radiological basis. The radiological protection criteria defined in the BSS were therefore used to establish radionuclide specific exemption values for transport purposes (see para. 401.3).

Shipment

237.1. In the context of the transport of radioactive material, the term ‘destination’ means the end point of a journey at which the package is, or is likely to be, opened, except during customs operations as described in para. 581.

Special arrangement

238.1. The use of the ‘special arrangement’ should not be taken lightly. This type of shipment is intended for those situations where the normal requirements of the Regulations cannot be met. For example, the disposal of old equipment containing radioactive material where there is no reasonable way to ship the radioactive material in an approved package. The hazard associated with repackaging and handling the radioactive material could outweigh the advantage of using an approved package,

TABLE I. CORRECTION FACTORS FOR PACKAGE AND DETECTOR SIZES

Distance between detector centre and package surface (cm)	Half linear dimension of package (cm)	Correction factor ^a
1	>10	1.0
2	10–20	1.4
	>20	1.0
5	10–20	2.3
	20–50	1.6
	>50	1.0
10	10–20	4.0
	20–50	2.3
	50–100	1.4
	>100	1.0

^a The reading should be multiplied by the correction factor to obtain the actual radiation level at the surface of the package.

assuming a suitable package is available. The special arrangement provisions should compensate for not meeting all the normal requirements of the Regulations by providing an equivalent level of safety. In keeping with the underlying philosophy of the transport regulations, reliance on administrative measures should be minimized in establishing the compensating measures.

Special form radioactive material

239.1. The Regulations are based on the premise that the potential hazard associated with the transport of non-fissile radioactive material depends on four important parameters:

- the dose per unit intake (by ingestion or inhalation) of the radionuclide;
- the total activity contained within the package;
- the physical form of the radionuclide;
- the potential external radiation levels.

239.2. The Regulations acknowledge that radioactive material in an indispersible form or sealed in a strong metallic capsule presents a minimal contamination hazard, although the direct radiation hazard still exists. Material protected in this way from the risk of dispersion during accident conditions is designated as ‘special form radioactive material’. Radioactive material which itself is dispersible may be adsorbed, absorbed or bonded to an inert solid in such a manner that it acts as an indispersible solid, e.g. metal foils. See paras 603.1–603.4, 604.1 and 604.2.

239.3. Unless the radioactive contents of a package are in special form, the quantity of radioactive material that can be carried in an excepted or Type A package will be limited to A_2 or multiples thereof. For example, a Type A package is limited to A_2 and the contents of excepted packages are limited to values ranging from A_2 to as low as $10^{-4} A_2$, or $10^{-5} A_2$ if transported by post, depending upon whether the material is solid, liquid or gas and whether or not it is incorporated into an instrument or article. However, if the material is in special form, the package limits change from A_2 to A_1 or appropriate multiples thereof. Depending on the radionuclide(s) involved, the A_1 values differ from the A_2 values by factors ranging from 1 to 10 000 (see Table I of the Regulations). The capability to ship an increased quantity in a package if it is in special form applies only to Type A and excepted packages.

Specific activity

240.1. The definition of specific activity in practice covers two different situations. The first, the definition of the specific activity of a radionuclide, is similar to the

ICRU definition of specific activity of an element. The second, the definition of the specific activity of a material for the Regulations is more precisely a mass activity concentration. Thus, the definition of specific activity is given for both cases and depends upon its specific application in the requirements of the Regulations. The term 'activity concentration' is also used in some paragraphs of the Regulations (e.g. see para. 401 and the associated Table I of the Regulations).

240.2. The half-life and the specific activity for each individual radionuclide given in Table I of the Regulations are shown in Table II.1 of Appendix II. These values of specific activity were calculated using the following equation:

$$\begin{aligned}\text{Specific activity (Bq/g)} &= \frac{(\text{Avogadro's number}) \times \lambda}{(\text{atomic mass})} \\ &= \frac{4.18 \times 10^{23}}{A \times T_{1/2}}\end{aligned}$$

where

A is the atomic mass of the radionuclide,

$T_{1/2}$ is the half-life (s) of the radionuclide, and

λ is the decay constant (s^{-1}) of the radionuclide $= \ln 2/T_{1/2}$.

240.3. The specific activity of any radionuclide not listed in Table II.1 of Appendix II can be calculated using the equation shown in para. 240.2.

240.4. The specific activity of uranium, for various levels of enrichment, is shown in Table II.3 of Appendix II.

240.5. In determining the specific activity of a material in which radionuclides are distributed, the entire mass of that material or a subset thereof, i.e. the mass of radionuclides and the mass of any other material, needs to be included in the mass component. The different interpretations of specific activity in the definition of LSA material (para. 226) and in Table II.1 should be noted.

Surface contaminated object

241.1. A differentiation is made between two categories of surface contaminated objects (SCOs) in terms of their contamination level, and this defines the type of packaging to be used to transport these objects. The Regulations provide adequate flexibility for the unpackaged shipment of SCO-I objects or their shipment in an

Industrial package (Type IP-1). The higher level of non-fixed contamination permitted on objects classified as SCO-II requires the higher standard of containment afforded by Industrial package Type IP-2.

241.2. The SCO-I model used as justification for the limits for fixed and non-fixed contamination is based on the following scenario. Objects in the category of surface contaminated objects include those parts of nuclear reactors or other fuel cycle equipment that have come into contact with primary or secondary coolant or process waste, resulting in contamination of their surface with mixed fission products. On the basis of the allowable contamination levels for beta and gamma emitters, an object with a surface area of 10 m^2 could have fixed contamination up to 4 GBq and non-fixed contamination up to 0.4 MBq. During routine transport this object can be shipped unpackaged under exclusive use, but it is necessary to secure the object (para. 523(a)) to ensure that there is no release of radioactive material from the conveyance. The SCO-I object and other cargo is assumed to move in an accident such that 20% of the surface of the SCO-I object is scraped and 20% of the fixed contamination from the scraped surface is freed. In addition, all of the non-fixed contamination is considered to be released. The total activity of the release would thus be 160 MBq for fixed contamination and 0.4 MBq for non-fixed contamination. Using an A_2 value of 0.02 TBq for mixed beta and gamma emitting fission products, the activity of the release equates to $8 \times 10^{-3} A_2$. It is considered that such an accident would only occur outside so that, consistent with the basic assumption of the Q system developed for Type A packages (see Appendix I), an intake of 10^{-4} of the scraped radionuclides for a person in the vicinity of the accident is appropriate. This would result in a total intake of $0.8 \times 10^{-6} A_2$. Hence this provides a level of safety equivalent to that of Type A packages.

241.3. The model for an SCO-II object is similar to that for an SCO-I object, although there may be up to 20 times as much fixed contamination and 100 times as much non-fixed contamination. However, an Industrial package (IP-2) is required for the transport of SCO-II objects. The presence of this package will lead to a release fraction in an accident which approaches that for a Type A package. Using a release fraction of 10^{-2} results in a total release of beta and gamma emitting radionuclides of 32 MBq of fixed contamination and 8 MBq of non-fixed contamination, which equates to $2 \times 10^{-3} A_2$. Applying the same intake factor as in the previous paragraph leads to an intake of $0.2 \times 10^{-6} A_2$, thereby providing a level of safety equivalent to that of Type A packages.

241.4. If the total activity of an SCO is so low that the activity limits for excepted packages according to para. 408 are met, it can be transported as an excepted package, provided that all the applicable requirements and controls for transport of excepted packages (paras 515–519) are complied with.

241.5. Surface contaminated objects are by definition objects which are themselves not radioactive but have radioactive materials distributed on their surfaces. The implication of this definition is that objects that are radioactive themselves (e.g. activated objects) and are also contaminated cannot be classified as SCOs. Such objects may, however, be regarded as LSA material insofar as the requirements specified in the LSA definition are complied with. See also para. 226.19.

241.6. Examples of inaccessible surfaces are:

- inner surfaces of pipes the ends of which can be securely closed by simple methods;
- inner surfaces of maintenance equipment for nuclear facilities which are suitably blanked off or formally closed;
- glove boxes with access ports blanked off.

241.7. Measurement techniques for fixed and non-fixed contamination of packages and conveyances are given in paras 508.2 and 508.7–508.12. These techniques are applicable to SCOs. However, to apply these techniques properly, a consignor needs to know the composition of the contamination.

Tank

242.1. The lower capacity limit of 450 L (1000 L in the case of gases) is included to achieve harmonization with the United Nations Recommendations [6].

242.2. Paragraph 242 includes solid contents in tanks where such contents are placed in the tank in liquid or gaseous form and subsequently solidified prior to transport (for example, uranium hexafluoride, UF₆).

Transport index

243.1. The TI performs many functions in the Regulations, including providing the basis for the carrier to segregate radioactive materials from persons, undeveloped film and other radioactive material consignments and to limit the level of radiation exposure to members of the public and transport workers during transport and in-transit storage.

243.2. In the 1996 edition of the Regulations the TI no longer makes any contribution to the criticality safety control of packages containing fissile material. Control for criticality safety is now provided by a separate criticality safety index (CSI) (see paras 218.1 and 218.2). Although the previous approach of a single control value for

radiological protection and criticality safety provided for simple operational application, the current use of a separate TI and CSI removes significant limitations on segregation in the transport and storage in transit of packages not containing fissile material. The reason for retaining the designation of TI is that the vast majority of radioactive consignments are not carrying fissile material and, therefore, a new name for the 'radioactive only' TI could have created confusion because of the need to introduce and explain two new names. Care should be taken not to confuse the use of the TI value and to consider the CSI value as the only control for accumulation of packages for criticality safety.

243.3. See paras 526.1–526.4.

Unirradiated thorium

244.1. The term 'unirradiated thorium' in the definition of low specific activity material is intended to exclude any thorium which has been irradiated in a nuclear reactor so as to transform some of the Th-232 into U-233, a fissile material. The definition could have prohibited the presence of any U-233, but naturally occurring thorium may contain trace amounts of U-233. The limit of 10^{-7} g of U-233 per gram of Th-232 is intended to clearly prohibit any irradiated thorium while recognizing the presence of trace amounts of U-233 in natural thorium.

Unirradiated uranium

245.1. The term 'unirradiated uranium' is intended to exclude any uranium which has been irradiated in a nuclear reactor so as to transform some of the U-238 into Pu-239 and some of the U-235 into fission products. The definition could have prohibited the presence of any plutonium or fission products, but naturally occurring uranium may contain trace amounts of plutonium and fission products. In the 1985 edition of the Regulations, the limits of 10^{-6} g of plutonium per gram of U-235 and 9 MBq of fission products per gram of U-235 were intended to clearly prohibit any irradiated uranium while recognizing the presence of trace amounts of plutonium and fission products in natural uranium.

245.2. The presence of U-236 is a more satisfactory indication of exposure to a neutron flux. 5×10^{-3} g of U-236 per gram of U-235 has been chosen as representing the consensus view of ASTM Committee C-26 in specification C-996 for enriched commercial grade uranium. This value is incorporated into the 1996 edition of the Regulations and recognizes the possibility for trace contamination by irradiated uranium but ensures that the material may still be treated as unirradiated. This specification represents the composition with the maximum value for uranium radionuclides

for which the A_2 value for uranium hexafluoride can be demonstrated to be unlimited. The difference in A_2 for uranium dioxide is considered to be insignificant [7].

REFERENCES TO SECTION II

- [1] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANISATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, WORLD HEALTH ORGANIZATION, International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, Safety Series No. 115, IAEA, Vienna (1996).
- [2] AMERICAN NUCLEAR SOCIETY, American National Standard for Nuclear Criticality Control of Special Actinide Elements, ANSI/ANS-8.15-1981 (reaffirmed 1987), ANS, New York (1981).
- [3] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, Series 1 Freight Containers — Specifications and Testing — Part 1: General Cargo Containers, ISO 1496:1-1990(E), ISO, Geneva (1990).
- [4] INTERNATIONAL MARITIME ORGANIZATION, International Convention for Safe Containers, IMO, London (1984).
- [5] INTERNATIONAL MARITIME ORGANIZATION, International Maritime Dangerous Goods (IMDG) Code, 2000 edition including amendment 30-00, IMO, London (2001).
- [6] UNITED NATIONS, Recommendations on the Transport of Dangerous Goods, Ninth Revised Edition (ST/SG/AC.10/1/Rev.9), UN, New York and Geneva (1995).
- [7] INTERNATIONAL ATOMIC ENERGY AGENCY, Interim Guidance for the Safe Transport of Reprocessed Uranium, IAEA-TECDOC-750, IAEA, Vienna (1994).

Section III

GENERAL PROVISIONS

RADIATION PROTECTION²

301.1. The objectives of the Radiation Protection Programme (RPP) for the transport of radioactive material are:

- to provide for adequate consideration of radiation protection measures in transport;
- to ensure that the system of radiological protection is adequately applied;
- to enhance a safety culture in the transport of radioactive material; and
- to provide practical measures to meet these objectives.

The RPP should include, to the extent appropriate, the following elements:

- (a) scope of the programme (see paras 301.2–301.4);
- (b) roles and responsibilities for the implementation of the programme (see para. 301.5);
- (c) dose assessment (see para. 305);
- (d) surface contamination assessment (see paras 508, 513 and 514);
- (e) dose limits, dose constraints and optimization (see para. 302);
- (f) segregation distances (see paras 306–307);
- (g) emergency response (see paras 308–309);
- (h) training (see para. 303); and
- (i) quality assurance (see para. 310).

301.2. The scope of the RPP should include all the aspects of transport as defined in para. 106 of the Regulations. However, it is recognized that in some cases certain aspects of the RPP may be covered in RPPs at the consigning, receiving or storage-in-transit sites. Since the magnitude and extent of measures to be employed in the RPPs will depend on the magnitude and likelihood of exposures, a graded approach should be followed.

² After the text of this publication had been prepared, the IAEA issued Safety Standards Series No. RS-G-1.1, Occupational Radiation Protection, IAEA, Vienna (1999). This Safety Guide may provide additional guidance on the development and implementation of radiation protection programmes and the monitoring and assessing of radiation doses.

301.3. Both the package type and the package category need to be considered. For routine transport the external radiation is important and the package category provides a classification for this; under accident conditions, however, it is the package type (Excepted, Industrial, Type A, Type B or Type C) that is important. Excepted, Industrial and Type A packages are not required to withstand accidents. For those aspects of the RPP related to accident conditions of transport, the possibility of leakage from these package types as the result of transport or handling accidents will need to be considered. In contrast, Type B and Type C packages can be expected to withstand all but the most severe accidents.

301.4. The external radiation levels from excepted packages and Category I-WHITE label packages are sufficiently low so as to be safe to handle without restriction, and a dose assessment is therefore unnecessary. Consideration of radiation protection requirements can be limited to keeping handling times as low as reasonably achievable, and segregation can be met by avoiding prolonged direct contact of packages with persons and other goods during transport. A dose assessment will, however, be needed for Category II- and III-YELLOW label packages, and segregation, dose limits, constraints and optimization will need to be considered in its light.

301.5. The RPP will best be established through the co-operative effort of consignors, carriers and consignees engaged in the transport of radioactive material. Consignors and consignees should normally have an appropriate RPP as part of fixed facility operations. The role and responsibilities of the different parties and individuals involved in the implementation of the RPP should be clearly identified and described. Overlapping of responsibilities should be avoided. Depending on the magnitude and likelihood of radiation exposures, the overall responsibility for establishment and implementation of the RPP may be assigned to a health physics or safety officer recognized through certification by appropriate boards or societies, or other appropriate means (e.g. by the relevant competent authorities), as a 'qualified expert' [1].

302.1. Optimization of protection and safety requires that both normal and potential exposures be taken into account. Normal exposures are exposures that are expected to be received under routine and normal transport conditions as defined in para. 106 of the Regulations. Potential exposures are exposures that are not expected to be delivered with certainty but that may result from an accident or owing to an event or sequence of events of a probabilistic nature, including equipment failures and operating errors. In the case of normal exposures, optimization requires that the expected magnitude of individual doses and the number of people exposed be taken into account; in addition, in the case of potential exposures, the likelihood of the occurrence of accidents or events or sequences of events is also taken into account. Optimization should be documented in the RPPs. See also Ref. [2].

302.2. The Basic Safety Standards [1] define radiological protection requirements for practices (activities that increase the overall exposure to radiation) and for intervention (activities that decrease the overall exposure by influencing the existing causes of exposure). The system of radiological protection for practices as set out in the Basic Safety Standards (Section 2, Principal Requirements) is summarized as follows:

- No practice is to be adopted unless it produces a positive net benefit (justification of a practice).
- All exposures are to be kept as low as reasonably achievable, economic and social factors being taken into account (optimization of protection).
- Total individual exposure is to be subject to dose limits or, in the case of potential exposures, to the control of risk (individual dose and risk limits).

302.3. In practical radiological protection there has in the past existed, and there continues to exist, a need to establish standards associated with quantities other than the basic dose limits. Standards of this type are normally known as secondary or derived limits. When such limits are related to the primary limits of dose by a defined model, they are referred to as derived limits. Derived limits have been used in the Regulations.

302.4. Examples of derived limits in the Regulations include the maximum activity limits A_1 and A_2 , maximum levels for non-fixed contamination, radiation levels at the surfaces of packages and in their proximity, and segregation distances associated with the transport index. The Regulations require assessment and measurement to ensure that standards are being complied with.

302.5. It should be a task of the competent authority to ensure that all transport activities are conducted under a general framework of optimization of protection and safety.

303.1. The provision of information and training is an integral part of any system of radiological protection. The level of instruction provided should be appropriate to the nature and type of work undertaken. Workers involved in the transport of radioactive material require training concerning the radiological risks in their work and how they can minimize these risks in all circumstances.

303.2. Training should relate to specific jobs and duties, to specific protective measures to be undertaken in the event of an accident or to the use of specific equipment. It should include general information relating to the nature of radiological risk, knowledge of the nature of ionizing radiation, the effects of ionizing radiation and its

measurement, as appropriate. Training should be seen as a continuous commitment throughout employment and involves initial training and refresher courses at appropriate intervals. The effectiveness of the training should be periodically checked.

303.3. Information on specific training requirements has been published [3, 4].

304.1. The competent authority assessments may be used to evaluate the effectiveness of the Regulations, including those for RPPs, and may be part of the compliance assurance activities detailed in Ref. [5] (see also paras 311.1–311.8). Of particular importance is the assessment of whether there is effective optimization of radiation protection and safety. This may also help in achieving and maintaining public confidence.

304.2. In order to comply with para. 304 of the Regulations, information on the radiation doses to workers and to members of the public should be collected and reviewed as appropriate. Reviews should be made if circumstances warrant, e.g. if significant changes in transport patterns occur or when a new technology related to radioactive material is introduced. The collection of relevant information may be achieved through a combination of radiation measurements and assessments. Reviews of accident conditions of transport are necessary in addition to those of routine and normal conditions.

305.1. The Basic Safety Standards [1] set a limit on effective dose for the members of the public of 1 mSv/a, and for workers of 20 mSv, averaged over five consecutive years and not exceeding 50 mSv in a single year. Dose limits in special circumstances, dose limits in terms of equivalent dose for the lens of the eye, extremities (hand and feet) and skin, and dose limits for apprentices and pregnant women are also set out in the Basic Safety Standards and should be considered in the context of the requirements of para. 305. These limits apply to exposures attributable to all practices, with the exception of medical exposures and of exposures from certain natural sources.

305.2. Three categories for monitoring and assessing radiation doses are shown in para. 305. The first category establishes a dose range where little action needs to be taken for evaluating and controlling doses. The upper value of this range is 1 mSv in a year, which was chosen to coincide with the dose limit for a member of the public. The second category has an upper value of 6 mSv/a, which is 3/10 of the limit on effective dose for workers (averaged over 5 years). This level represents a reasonable dividing line between conditions where dose limits are unlikely to be approached and conditions where dose limits could be approached. The third category is for any situation where the occupational exposure is expected to exceed the 6 mSv/a upper value in the second category.

305.3. Many transport workers will be in the first category, and no specific measures concerning monitoring or control of exposure are required. In the second category, a dose assessment programme will be necessary. This may be based upon either individual monitoring or monitoring of the workplace. In the latter case, workplace monitoring may often be achieved by radiation level measurements in occupied areas at the start and end of a particular stage of a journey. In some cases, however, air monitoring, surface contamination checks and individual monitoring may also be required. In the third category, individual monitoring is mandatory. In most cases this will be accomplished by the use of personal dosimetry such as film badges, thermoluminescent dosimeters and, where necessary, neutron dosimeters (see also footnote 2).

305.4. Some studies of particular operations have shown a correlation between dose received by workers and the number of transport indexes handled [6]. It is unlikely that carriers handling less than 300 TI per year will exceed doses of 1 mSv/a and such carriers would not therefore require detailed monitoring, dose assessment or individual records.

305.5. Given that relatively high radiation levels are permitted during carriage under exclusive use, additional care should be taken to ensure that the requirements of para. 305 are met, since it would be relatively easy to exceed the 1 mSv level, and consequently specific measures regarding monitoring or control of exposures should be taken. In the assessment of the correct exposure category, exposures received during the carriage phase of transport should be considered together with those received elsewhere, particularly during loading and unloading.

306.1. The dose level of 5 mSv/a for occupationally exposed workers and of 1 mSv/a to the critical group [1] for members of the public are specifically defined values to be used for the purposes of calculating segregation distances or dose rates for regularly occupied areas. The distances and dose rates are, for convenience, often presented in segregation tables. The dose values given in para. 306 are for segregation distance or calculation purposes only and are required to be used together with hypothetical but realistic parameters in order to obtain appropriate segregation distances. Using the given values provides reasonable assurance that actual doses from the transport of radioactive materials will be well below the appropriate average annual dose limits.

306.2. These values together with simple, robust modelling have been used for a number of years to derive segregation tables for different modes of transport. Assessments of radiation exposures arising indicate that continued use of these values is acceptable. In particular, surveys of exposure occurring in air and sea

transport [7, 8] have shown that segregation distances derived from them have resulted in doses to the public below the relevant annual dose limits and that doses to workers not involved in direct handling are also less than 1 mSv/a. The use of segregation distances does not in itself remove the requirement for undertaking the evaluation required in para. 305 of the Regulations.

306.3. The Regulations state the requirements for radiation protection which are to be fulfilled in the determination of segregation distances (i.e. minimum distances between radioactive material packages and regularly occupied areas of a conveyance) and of dose rates in regularly occupied areas. For practical purposes it may be helpful to provide this information in the form of segregation tables.

307.1. Although not a radiation protection issue, an evaluation of the effect of radiation on fast X ray films in 1947 [9] determined that they may show slight fogging after development when exposed to doses exceeding 0.15 mSv of gamma radiation. This could interfere with the proper use of the film and provide incorrect diagnostic interpretation. Other types of film are also susceptible to fogging although the doses required are much higher. Since it would be impracticable to introduce segregation procedures which vary with the type of film, the provisions of the Regulations are designed to restrict the exposure of undeveloped films of all kinds to a level of not more than 0.1 mSv during any journey from consignor to consignee.

307.2. The different time durations involved for sea transport (in terms of days or weeks) and air or land transport (in terms of hours or days) mean that different tables of segregation distances are used, so that the total film exposure during transit is the same for each mode. More than one mode of transport and more than one shipment may be involved in the distribution and ultimate use of photographic film. Thus, when segregation distance tables are being established for a specific transport mode, only a fraction of the limit prescribed in para. 307 should be committed to that mode.

307.3. In road transport a driver may ensure sufficient segregation from photographic film carried in other vehicles by leaving a clear space of at least 2 m all around the vehicle when parking.

EMERGENCY RESPONSE

308.1. The requirements established in the Regulations, when complied with by the package designer, consignor, carrier and consignee, ensure a high level of safety for the transport of radioactive material. However, accidents involving such packages may happen. Paragraph 308 of the Regulations recognizes that advance planning and