

This publication is no longer valid
Please see <http://www-ns.iaea.org/standards/>

safety series

safety series

No. 37

IAEA SAFETY GUIDES

**Advisory Material
for the IAEA Regulations
for the Safe Transport
of Radioactive Material
(1985 Edition)**

Third Edition



INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1987

CATEGORIES OF IAEA SAFETY SERIES

From 1978 onwards the various publications in the Safety Series are divided into four categories, as follows:

- (1) **IAEA Safety Standards.** Publications in this category comprise the Agency's safety standards as defined in "The Agency's Safety Standards and Measures", approved by the Agency's Board of Governors on 25 February 1976 and set forth in IAEA document INFCIRC/18/Rev.1. They are issued under the authority of the Board of Governors, and are mandatory for the Agency's own operations and for Agency-assisted operations. Such standards comprise the Agency's basic safety standards, the Agency's specialized regulations and the Agency's codes of practice. *The covers are distinguished by the wide red band on the lower half.*
- (2) **IAEA Safety Guides.** As stated in IAEA document INFCIRC/18/Rev.1, referred to above, IAEA Safety Guides supplement IAEA Safety Standards and recommend a procedure or procedures that might be followed in implementing them. They are issued under the authority of the Director General of the Agency. *The covers are distinguished by the wide green band on the lower half.*
- (3) **Recommendations.** Publications in this category, containing general recommendations on safety practices, are issued under the authority of the Director General of the Agency. *The covers are distinguished by the wide brown band on the lower half.*
- (4) **Procedures and Data.** Publications in this category contain information on procedures, techniques and criteria pertaining to safety matters. They are issued under the authority of the Director General of the Agency. *The covers are distinguished by the wide blue band on the lower half.*

Note: The covers of publications brought out within the framework of the NUSS (Nuclear Safety Standards) Programme are distinguished by the wide yellow band on the upper half.

This publication is no longer valid
Please see <http://www-ns.iaea.org/standards/>

**ADVISORY MATERIAL
FOR THE IAEA REGULATIONS
FOR THE SAFE TRANSPORT
OF RADIOACTIVE MATERIAL (1985 EDITION)**

Third Edition

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

AFGHANISTAN	GUATEMALA	PARAGUAY
ALBANIA	HAITI	PERU
ALGERIA	HOLY SEE	PHILIPPINES
ARGENTINA	HUNGARY	POLAND
AUSTRALIA	ICELAND	PORTUGAL
AUSTRIA	INDIA	QATAR
BANGLADESH	INDONESIA	ROMANIA
BELGIUM	IRAN, ISLAMIC REPUBLIC OF	SAUDI ARABIA
BOLIVIA	IRAQ	SENEGAL
BRAZIL	IRELAND	SIERRA LEONE
BULGARIA	ISRAEL	SINGAPORE
BURMA	ITALY	SOUTH AFRICA
BYELORUSSIAN SOVIET SOCIALIST REPUBLIC	JAMAICA	SPAIN
CAMEROON	JAPAN	SRI LANKA
CANADA	JORDAN	SUDAN
CHILE	KENYA	SWEDEN
CHINA	KOREA, REPUBLIC OF	SWITZERLAND
COLOMBIA	KUWAIT	SYRIAN ARAB REPUBLIC
COSTA RICA	LEBANON	THAILAND
COTE D'IVOIRE	LIBERIA	TUNESIA
CUBA	LIBYAN ARAB JAMAHIRIYA	TURKEY
CYPRUS	LIECHTENSTEIN	UGANDA
CZECHOSLOVAKIA	LUXEMBOURG	UKRAINIAN SOVIET SOCIALIST REPUBLIC
DEMOCRATIC KAMPUCHEA	MADAGASCAR	UNION OF SOVIET SOCIALIST REPUBLICS
DEMOCRATIC PEOPLE'S REPUBLIC OF KOREA	MALAYSIA	UNITED ARAB EMIRATES
DENMARK	MALI	UNITED KINGDOM OF GREAT BRITAIN AND NORTHERN IRELAND
DOMINICAN REPUBLIC	MAURITIUS	UNITED REPUBLIC OF TANZANIA
ECUADOR	MEXICO	UNITED STATES OF AMERICA
EGYPT	MONACO	URUGUAY
EL SALVADOR	MONGOLIA	VENEZUELA
ETHIOPIA	MOROCCO	VIET NAM
FINLAND	NAMIBIA	YUGOSLAVIA
FRANCE	NETHERLANDS	ZAIRE
GABON	NEW ZEALAND	ZAMBIA
GERMAN DEMOCRATIC REPUBLIC	NICARAGUA	ZIMBABWE
GERMANY, FEDERAL REPUBLIC OF	NIGER	
GHANA	NIGERIA	
GREECE	NORWAY	
	PAKISTAN	
	PANAMA	

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, 1987

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

Printed by the IAEA in Austria
June 1987

SAFETY SERIES No. 37

**ADVISORY MATERIAL
FOR THE IAEA REGULATIONS
FOR THE SAFE TRANSPORT
OF RADIOACTIVE MATERIAL (1985 EDITION)**

Third Edition

**INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 1987**

This publication is no longer valid
Please see <http://www-ns.iaea.org/standards/>

**THIS SAFETY SERIES IS ALSO PUBLISHED IN
FRENCH, RUSSIAN AND SPANISH**

**ADVISORY MATERIAL FOR THE IAEA REGULATIONS FOR
THE SAFE TRANSPORT OF RADIOACTIVE MATERIAL (1985 EDITION)
IAEA, VIENNA, 1987
STI/PUB/768
ISBN 92-0-123487-2**

FOREWORD

Since the early 1950s there has been an increasing use of products of the nuclear industry in medical and engineering fields, in scientific research and for the generation of electricity. Such uses require safe and effective transport arrangements for the varied radioactive and fissile materials involved.

It has always been recognized that transport arrangements must be such as to ensure the safety of persons, property and the environment. Accordingly, regulations have been developed to assure safe transport. Before 1959, the various national and international controls were, in the main, based on the United States Interstate Commerce Commission regulations, which essentially aimed to facilitate the movement of radioactive ores and concentrates and packages containing relatively small amounts of radionuclides for medical and industrial use. The rapid expansion of the nuclear industry required that these early regulations be further developed so as to facilitate the safe movement of all kinds and quantities of radioactive materials.

In July 1959, the United Nations Economic and Social Council expressed the desire "that the International Atomic Energy Agency (IAEA) be entrusted with the drafting of recommendations on the transport of radioactive substances". The IAEA Regulations for the Safe Transport of Radioactive Materials were first published in 1961.

In order for its Regulations to be acceptable for worldwide application in all modes of transport, the Agency recognized that, for any radioactive and fissile material, these Regulations must reduce the hazards to transport workers and the general public to an acceptably low level, i.e. be 'safe'. The following basic requirements must be met if safety is to be achieved:

- effective containment of the material;
- effective control of radiation emitted from the package;
- a subcritical condition for any fissile material, i.e. 'criticality' must be prevented; and
- adequate dissipation of any heat generated within the package.

The Regulations, although applying to the Agency's own operations, are, for administrative purposes, 'model regulations' recommended to Member States and appropriate international organizations. This means that to facilitate their conversion into the format and 'language' used by the various modes of transport, the Regulations must be practical concerning what the various parties involved, especially consignors and carriers, are required to do. Also, the Regulations must be clear and concise, stating 'what' has to be achieved, i.e. performance, rather than

'how' to achieve the desired degree of protection in terms of detailed design specifications, or the 'why' as to the basis or rationale behind any specific requirement. These two latter considerations are treated in the companion documents to the Regulations, IAEA Safety Series No. 37 and Safety Series No. 7, respectively.

The procedure used by the Agency for the preparation of its Regulations is largely responsible for their extensive national and international application. While it was expected that the basic principles underlying the Regulations would remain acceptable for a long period, it was also expected that a 'feedback' of experience in application would be facilitated by revision after some five years. This resulted in three comprehensive revisions being published in 1965, 1967 and 1973. Following minor amendments, the 1973 Edition of the Regulations was again published in 1979 as the 1973 Revised Edition as Amended. The Agency's procedures for the development of the Regulations have involved the convening of a panel, or panels, whose members represented not only an extensive coverage of Member States and international transport organizations, but also experience of the various administrative and technical problems requiring solution. On the basis of proposals co-ordinated by the Secretariat, these panels prepared draft Regulations for comment by all Member States and all international organizations concerned. Then, following panel consideration of such comments, a final draft was prepared for approval by the Agency's Board of Governors.

The Standing Advisory Group on the Safe Transport of Radioactive Materials (SAGSTRAM) was established in 1978 to advise on the Agency's transport programme and on the development and implementation of the Regulations. Recommendations concerning procedures and a schedule for a further comprehensive review of Safety Series No. 6 were made at the first meeting of SAGSTRAM in October 1978, and the review process was initiated early in 1979 by the Secretariat, which invited comments and proposals for changes from Member States and international organizations. The comments and proposals received were considered in September 1980 by an Advisory Group, which also took into account recommendations from SAGSTRAM, an Advisory Group on radiation protection and safety principles, and a Technical Committee on package test standards.

The revised drafts were sent for comment to Member States and to international organizations concerned with transport, while specific topics were examined by consultants and Technical Committees. During the period 1980-84, approximately 150 experts from 22 Member States and 12 international organizations attended meetings in connection with the review process. In February 1984, SAGSTRAM considered the fourth revised draft and, after making minor changes, endorsed the text for the 1985 Edition, which was subsequently approved by the IAEA Board of Governors in late 1984.

In order to provide an explanation of and advice on the Regulations, a second document, Safety Series No. 7, entitled Notes on Certain Aspects of the Regulations, was issued in 1961. This contained information developed in response to a general

consensus of the regulatory development panels that such a complementary document was needed to provide guidance to users as to the purpose of the Regulations and their implementation in practice. It thus provided both explanation and advice, i.e. it was both a 'how' and a 'why' document.

The recognition of the need for more current information of an advisory nature led the Agency to publish in 1973 a companion document to the 1973 Revised Edition of the Regulations, entitled Advisory Material for the Application of the IAEA Transport Regulations, IAEA Safety Series No. 37. An updated version was published in 1982, and the present third edition has been issued to reflect the 1985 Edition of Safety Series No. 6. It provides information about the technical requirements of the Regulations and about the methods and technology which may be employed to satisfy them, for the benefit of designers and manufacturers of packagings, consignors, carriers, competent authorities and others, i.e. it provides 'how' information. Whereas the earlier versions of Safety Series No. 37 also contained some explanatory information on the intent and rationale of the regulatory requirement, to the extent practicable such information has been omitted from this third edition. It is now presented in a third document, i.e. Safety Series No. 7, Explanatory Material for the IAEA Regulations for the Safe Transport of Radioactive Material.

Thus, the IAEA transport regulatory publications comprise four main documents, namely:

- (a) The regulatory document, prescribing 'what' is to be achieved (Safety Series No. 6);
- (b) The advisory document, providing examples of 'how' certain regulatory requirements can be met (Safety Series No. 37);
- (c) The explanatory document, giving the basis, i.e. the 'why' of certain regulatory requirements (Safety Series No. 7); and
- (d) The 'Schedules of Requirements for the Transport of Specified Types of Radioactive Material Consignments' (Safety Series No.80).

The purpose of Safety Series No. 37 is to provide advice on the application of the provisions of Safety Series No. 6 in order to help achieve compliance with the regulatory standards. As presented here, Safety Series No. 37 also reflects the corrections and changes implemented by the 1986 Supplement to the Regulations for the Safe Transport of Radioactive Material. Its intent is to describe methods, techniques and practices (citing any appropriate national and international standards) which can be considered a means of satisfying certain requirements. Safety Series No. 37 should always be read as offering 'a way' or 'ways' rather than 'the unique way' of achieving compliance. The information provided is to be considered purely advisory and never mandatory, except where a competent authority may require use of any part or parts of the text. All paragraphs from Safety Series No. 6 have been examined in the preparation of this document. In certain cases they were

considered self evident and therefore no advisory information has been provided on them.

The same basic system of numbering of paragraphs has been used in this document as in Safety Series Nos 6 and 7, so as to facilitate cross-referencing. For each paragraph of Safety Series No. 6 for which advisory text is provided, that text is identified by 'A' followed by the corresponding paragraph number from Safety Series No. 6 (e.g. A-117). If more than one advisory paragraph is associated with a specific paragraph from Safety Series No. 6, these are numbered sequentially (e.g. A-210.1, A-210.2, etc.). Reference to paragraphs in the 1985 Edition of Safety Series No. 6 is made for example by either 'para. 201' or 'paras 201 to 205'. Where references are made to paragraphs in Safety Series No. 7, these are indicated by 'E' before the paragraph number, e.g. E-210.1, etc.

To ensure that this Advisory Material will adequately reflect current developments and regulatory bases, it is planned that it will be revised, as needed, at periodic intervals by consultants or panels of experts. Such revisions will be carefully co-ordinated with the periodic reviews of the Regulations.

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 Regulations. Readers are encouraged and invited to send, through their competent authority, any comments they may wish to make, including proposals for modifications, additions, and deletions, to the Director, Division of Nuclear Safety, International Atomic Energy Agency, Wagramerstrasse 5, P.O. Box 100, A-1400, Vienna, Austria.

CONTENTS

SECTION I. INTRODUCTION.....	1
Purpose and scope.....	1
Definitions for the purpose of the Regulations.....	2
SECTION II. GENERAL PRINCIPLES AND PROVISIONS.....	12
General principles for radiation protection.....	12
Provision for avoiding radiation damage to film.....	13
General accident provisions.....	14
General provision for quality assurance programme.....	14
General provision for compliance assurance.....	15
General provision for special arrangement.....	20
SECTION III. ACTIVITY AND FISSILE MATERIAL LIMITS.....	21
Basic A_1/A_2 values.....	21
Determination of A_1 and A_2	21
Contents limits for packages.....	22
SECTION IV. PREPARATION, REQUIREMENTS AND CONTROLS FOR SHIPMENT AND FOR STORAGE IN TRANSIT....	23
Package inspection requirements.....	23
Transport of other goods.....	24
Other dangerous properties of contents.....	25
Requirements and controls for contamination and for leaking packages.....	27
Requirements and controls for transport of excepted packages.....	29
Requirements and controls for transport of LSA and SCO.....	31
Determination of transport index (TI).....	31
Limits on transport index and radiation level for packages and overpacks... Categories.....	32
Marking, labelling and placarding.....	34
Consignor's responsibilities.....	37
Transport.....	38
Storage in transit.....	45
Customs operations.....	46
Undeliverable packages.....	48

SECTION V. REQUIREMENTS FOR RADIOACTIVE MATERIALS AND FOR PACKAGINGS AND PACKAGES.....	49
Requirements for radioactive materials.....	49
General requirements for all packagings and packages.....	50
Additional requirements for packages transported by air.....	52
Requirements for industrial packages.....	52
Requirements for Type A packages.....	56
Requirements for Type B packages.....	59
Requirements for packages containing fissile material.....	70
SECTION VI. TEST PROCEDURES.....	74
Demonstration of compliance.....	74
Tests for special form radioactive material.....	79
Tests for packages.....	80
SECTION VII. APPROVAL AND ADMINISTRATIVE REQUIREMENTS.....	96
General.....	96
Approval of special form radioactive material.....	100
Approval of package designs.....	101
Notification and registration of serial numbers.....	102
Approval of shipments.....	102
Approval of shipment under special arrangement.....	103
Competent authority approval certificates.....	104
Contents of approval certificates.....	104
Validation of certificates.....	106
APPENDIX I. LIST OF REGULATORY DOCUMENTS OF INTERNATIONAL AND REGIONAL INTERNATIONAL ORGANIZATIONS.....	109
APPENDIX II. CONTAMINATION CONTROL.....	111
APPENDIX III. HALF-LIFE AND SPECIFIC ACTIVITY OF RADIONUCLIDES, AND SPECIFIC ACTIVITY OF URANIUM AND THORIUM.....	115
APPENDIX IV. QUALITY ASSURANCE IN THE SAFE TRANSPORT OF RADIOACTIVE MATERIAL.....	127
APPENDIX V. GUIDE FOR QUALITY ASSURANCE PROGRAMME.....	145

APPENDIX VI. EXAMPLE CALCULATIONS FOR ESTABLISHING MINIMUM SEGREGATION DISTANCE REQUIREMENTS.....	147
APPENDIX VII. ACCELERATION VALUES AND CALCULATION METHODS FOR PACKAGE TIE-DOWN FORCES.....	157
APPENDIX VIII. EXAMPLE OF A RADIATION PROTECTION PROGRAMME FOR EXCLUSIVE USE VESSELS.....	164
APPENDIX IX. INFLUENCE OF BRITTLE FRACTURE ON MATERIAL INTEGRITY.....	172
APPENDIX X. CRITICALITY SAFETY ASSESSMENTS.....	175
REFERENCES.....	183
BIBLIOGRAPHY.....	189
LIST OF MEETINGS RELATING TO THE THIRD EDITION OF SAFETY SERIES No. 37 AND OF PARTICIPANTS AT THOSE MEETINGS.....	193
INDEX.....	199

This publication is no longer valid
Please see <http://www-ns.iaea.org/standards/>

SECTION I

INTRODUCTION

PURPOSE AND SCOPE

A-101.1. The IAEA Regulations for the Safe Transport of Radioactive Material [1] provide standards for ensuring a high level of safety of people, property and the environment against radiation and criticality hazards as well as thermal effects associated with the transport of radioactive materials. The basic requirements to be met are:

- (a) Effective containment of radioactive material;
- (b) Effective control of radiation emitted from the package;
- (c) A subcritical condition for any fissile material; and
- (d) Adequate dissipation of any heat generated within the package.

A-101.2. For meeting the basic requirements and the declared purpose of the Regulations, effective quality assurance and compliance assurance programmes are required (see paras A-209 and A-210). Compliance means, for example, that:

- (a) Appropriate and sound packages are used;
- (b) The activity of radioactive material in each package does not exceed the regulatory activity limit for that material and that package type;
- (c) The radiation levels external to, and the contamination levels on, surfaces of packages do not exceed the appropriate limits;
- (d) Packages are properly marked and labelled and transport documents are completed;
- (e) The number of packages containing radioactive material in a conveyance is within the regulatory limits;
- (f) Packages of radioactive material are stowed in conveyances and are stored at a safe distance from persons and photosensitive materials;
- (g) Only those transport and lifting devices which have been tested are used in loading, conveying and unloading packages of radioactive material; and
- (h) Packages of radioactive material are properly secured for transport.

A-101.3. The control of the transport of radioactive materials may be necessary also for other reasons, e.g. safeguards control and physical protection of nuclear materials and control of a property. However, if any activities or controls are imposed for any of these other reasons, they need to be evaluated to demonstrate that they do not detract from the standards of safety of the Regulations.

A-105.1. For radioactive materials having other dangerous properties, the regulations of Member States, modal conventions and agreements, and other relevant

documents of international organizations need to be applied. Generally, the other dangerous properties are controlled as subsidiary hazards. However, according to the instructions of the International Civil Aviation Organization, for example, the other dangerous properties take precedence for excepted packages. Appendix I contains a list of the major documents of the international and regional organizations.

A-105.2. See also paras A-208 and A-407.1 to A-407.9.

A-106.1. Although the intent of the Regulations is to provide for the requisite safety in transport without the need for specified routing, the regulatory authorities in some Member States have imposed routing requirements. Routing restrictions are usually imposed on shipments of spent fuel and other high activity materials. 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 which contribute to the overall risk in transporting large amounts of radioactivity and not only on concerns for 'worst case' scenarios, i.e. 'low probability/high consequence' accidents.

A-106.2. Through a Co-ordinated Research Programme of the IAEA a computer based environmental impact code INTERTRAN [2] has been developed and is available for use by Member States. In spite of many uncertainties, including the use of generalized models and the difficulty of selecting adequate input values for accident conditions, this code is useful for calculating and understanding, at least on a qualitative basis, the factors which are significant in determining the radiological impact from the transport of radioactive materials. These factors are the important aspects which should be considered in any routing decision.

A-106.3. In some Member States the authorities at the state, provincial or even local government levels may be involved in routing decisions. In these cases it is often necessary either to provide them with evaluations to assess alternative routes or to provide them with very simple methodologies which they can use. 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. For example, the Government of the United States of America applies such an approach [3] and it has been found that both local and state governments can play a role in the route selection process.

DEFINITIONS FOR THE PURPOSE OF THE REGULATIONS

Competent authority

A-116.1. The competent authority is the one organization defined by legislative or executive authority to act on behalf of a nation, or an international authority, in

matters involving the transport of radioactive materials. 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 or water), or the package and radioactive material contents (Type A, Type B, special form, fissile) or on the hazard associated with the material (radioactive or other dangerous properties). A national competent authority may in some cases delegate the approval of package designs and certain types of shipments 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.

A-116.2. The competent authority should make the consignors, carriers, consignees and public aware of its identity and how it may be contacted. It is helpful to publish 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, telex numbers, and telephone numbers.

A-116.3. The primary source of competent authority identifications is the List of National Competent Authorities for transport, which is updated and published annually by the IAEA. 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.

A-116.4. Some competent authorities have formed an informal group, known as the 'Radioactive Transport Study Group', which meets periodically to discuss problems of mutual interest in the national and international administration of the IAEA Regulations. The membership in 1986 included representatives of Australia, Belgium, Canada, the Federal Republic of Germany, France, Italy, Japan, the Netherlands, Poland, Sweden, the United Kingdom and the United States of America. It should be emphasized that the group has no executive powers, but discusses problems freely with an interest in obtaining consensus solutions which will help assure compliance and, at the same time, simplify the implementation of the transport Regulations.

Compliance assurance

A-117. See paras A-210.1 to A-210.14.

Contamination

A-122. Any surface with levels of contamination lower than 0.4 Bq/cm^2 ($10^{-5} \mu\text{Ci/cm}^2$) for beta and gamma emitters, or 0.04 Bq/cm^2 ($10^{-6} \mu\text{Ci/cm}^2$) for

alpha emitters is considered as a non-contaminated surface in applying the Regulations. For instance, a non-radioactive solid object with levels of surface contamination lower than the above limits is out of the scope of the Regulations and no requirement is applicable for its transport.

A-123. Surface contamination may consist of both fixed and non-fixed radioactive material. In subsequent paragraphs within the Regulations, limits are given for non-fixed contamination, fixed contamination or both. Care is necessary to ensure the correct application of these limits. The limits for surface contamination on packages, where given in Bq/cm^2 (or $\mu\text{Ci/cm}^2$), all relate to non-fixed contamination. For advice on contamination and contamination measurement techniques, see paras A-408.1 to A-408.3 and Sections 3 and 4 of Appendix II. The fixed contamination on the surfaces of packages and on conveyances is limited by the radiation level limit (see para. 413).

A-124. See para. A-123.

Exclusive use

A-128. For exclusive use of a conveyance or of a 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 carrier 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

A-129.1. Most nuclides can be made to fission, but many can only be made to fission with difficulty using special equipment and controlled conditions. The distinguishing characteristic of the fissile nuclides named in the definition is that they are capable of initiating a self-sustaining neutron chain reaction by only the accumulation of sufficient mass. No other action, mechanism, or special condition is required. Plutonium-238 in the form in which it is encountered in transport does not have this property. Although plutonium-238 can be made to support a fast neutron chain reaction under stringent laboratory conditions, it cannot under any circumstances maintain a chain carried by thermal neutrons. Plutonium-238 is, therefore,

'fissionable' rather than 'fissile'. Since the former term is not recognized in the Regulations, it is classed as 'fissile'. Although other fissile nuclides are known, e.g. in the actinides [4], the definition in the Regulations is limited to those that are commonly used in the commerce of the Member States. If materials such as actinides are to be transported, consideration should be given to the fissile properties of those materials.

A-129.2. A fission chain is propagated by neutrons. Because a chain reaction is dependent on the behaviour of neutrons, fissile materials are packaged and shipped under requirements designed to control neutron behaviour in a manner to maintain subcriticality and, thus, provide criticality safety in transport.

A-129.3. There are three possible fates for a neutron in fissile material. It may encounter a fissile nuclide and induce fission, producing neutrons to continue the chain; it may be absorbed by a constituent of the materials or by a fissile nuclide without fissioning and thereby be removed from the chain; or it may escape the fissile material by leaking from the surface and be removed from the chain. Criticality is achieved when there is a balance between neutron production by fission and loss by neutron absorption in and leakage from the fissile material. The three possible outcomes for a neutron are influenced to provide criticality control. The fraction of neutrons leaking from fissile material is affected by the geometric configuration, both of each package involved and of the spacing of multiple packages. For example, dimensions and shapes that are favourable to subcriticality are so because the leakage fraction is too large for the chain reaction to be sustained. Neutrons leaking from a package containing fissile material may enter other similar packages and produce a fission event. Neutron interaction can be influenced by the package dimensions, which determine the spacing of the fissile material and can be adjusted to promote neutron leakage from an array of packages. Neutrons also may be removed from the system by the use of neutron absorbers (see para. A-401.6) which influence package designs, and hence the neutron leakage from an array of packages (see para. A-479 regarding the spacing of groups of packages). Good package design embodies a balance of many parameters influencing neutron behaviour and thus assures subcriticality.

Freight container

A-130.1. Freight containers are typically built in accordance with the standards of the International Organization for Standardization (ISO) [5]. They should be approved in accordance with the International Convention for Safe Containers (CSC) (see Appendix D) in order to facilitate international transport operations. If other freight containers are used, the competent authority (or authorities) should be consulted.

A-130.2. In addition, special rules may be specified by modal transport organizations. As an example, the International Maritime Dangerous Goods (IMDG) Code (see Appendix I) contains the provisions for the transport by sea of dangerous goods including radioactive materials.

Low specific activity material

A-131.1. According to para. 131(a) LSA-I materials cannot consist of:

- (i) concentrates of ores other than uranium or thorium concentrates (for example, radium ore concentrate cannot be LSA-I material);
- (ii) any fissile material.

A-131.2. 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. A-1, this external shielding material is not to be taken into account in determining the specific activity of LSA material.

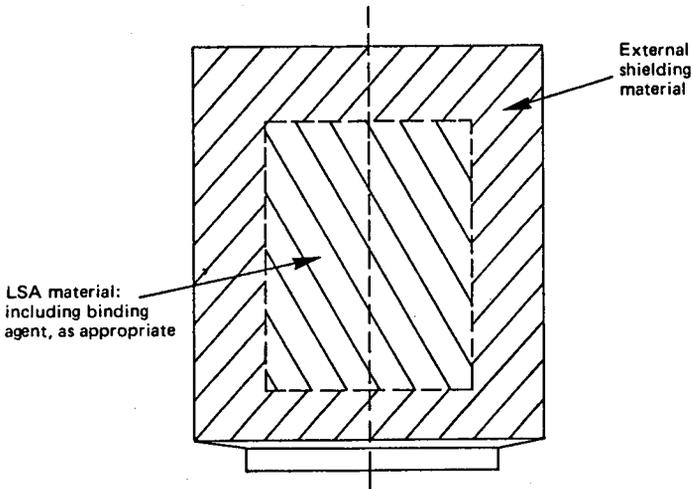


FIG. A-1. Low specific activity material surrounded by a cylindrical volume of non-radioactive shielding material.

A-131.3. For LSA-II solids, and for LSA-III materials not incorporated in 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, reasonable to recognize that the concept of limiting the estimated specific activity fails to be meaningful if in a large volume the activity distribution is clearly confined to a small percentage of that volume.

A-131.4. It is prudent to establish a method by which the significance of the estimated average activity, as determined, can be judged. There are clearly several methods that would be suitable for this particular purpose.

A-131.5. A simple method for assessing the average activity involves dividing the volume occupied by the LSA material into defined portions and then assessing and comparing the specific activity of each of these portions. It is suggested that specific activity differences between portions of less than a factor of ten would cause no concern.

A-131.6. Judgement needs to be exercised in selecting the size of the portions to be assessed. The method described in A-131.5 should not be used for volumes of material 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.

A-131.7. 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. Because 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.

A-131.8. As for the case discussed in para. A-131.3, there are several methods that would be suitable for assessing the activity of LSA-III materials, and the one presented here is just an example. This method is quite similar to the one outlined in paras A-131.5 and A-131.6 with the difference that the LSA material volume as such is not considered because the regulatory requirement puts emphasis on the radioactive material being essentially uniformly distributed in the binding agent. The method involves dividing the LSA material volume into a number of portions, the minimum number of which is suggested to be ten and the maximum volume of which is suggested to be 0.1 m^3 , whichever is more limiting in terms of portion volume. The specific activity of each volume should then be assessed (either 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. A-131.5 because the requirement of 'essentially

uniformly distributed' for LSA-III materials is intended to be more constraining than the requirement of 'distributed throughout' for LSA-II materials.

A-131.9. As a consequence of the definition of LSA material (materials for which limits of estimated average specific activity may apply, in addition to materials which by nature have a limited specific activity) 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. 422); and
- (b) the total activity of LSA material in any single conveyance (see para. 427 and Table VI).

Both requirements can be much more restrictive than the basic requirements for LSA material given in para. 131. 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 having 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 by 100 A_2 by other modes (see Table VI).

A-131.10. Objects which are both activated or otherwise radioactive and contaminated cannot be considered as SCO (see para. A-144.1). However, such objects may by definition qualify as LSA material since an object having activity throughout and also activity 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 in order for the objects to qualify as LSA material.

Maximum normal operating pressure

A-132.1. 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.

A-132.2. The environmental conditions to be applied to a package in determining the MNOP are the normal environmental conditions specified in paras 545 and 546, or in the case of air transport, para. 516. Other conditions to be applied in determining the MNOP are that the package is unattended for a one-year period and

that it is subject to the maximum internal heating. Being unattended means the absence of venting, external cooling by ancillary systems or other operational controls that could be used to limit internal pressure, i.e. it excludes any active corrective measures that could be applied to reduce the MNOP. Because 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 pressure of 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 encapsulation into the containment system. Some transient conditions may tend to reduce the MNOP (e.g. reduction with time of internal heat due to radioactive decay of the contents). These conditions may be taken into account if adequately justified.

Quality assurance

A-136. See paras A-209.1 to A-209.5.

Special arrangement

A-141. See paras A-211.1 to A-211.3.

Special form radioactive material

A-142.1. Unless the contents of a packaging are in special form, the quantity of radioactive material that can be carried in the packaging 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 $10^{-4} A_2$, 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 increase from A_2 to A_1 or appropriate multiples thereof. Depending on the isotope(s) involved, the A_1 values differ from the A_2 values by factors ranging from 1.0 to 1000 (see Table I). The capability to ship an increased quantity in a package if it is in special form only applies to Type A and excepted packages.

A-142.2. Special form radioactive material is defined as either the radioactive material itself in the form of an indispersible solid or a sealed capsule containing the radioactive material so the encapsulated material acts essentially as an indispersible solid. Special form material is in an indispersible, inert and insoluble form and is therefore unlikely to be dispersed in accident conditions. Those properties may be inherent in the radioactive material itself. Also, 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 A-503 and A-504.

Specific activity

A-143.1. The half-life and the specific activity for each individual radionuclide given in Table I of the Regulations are shown in Table AIII-1 of Appendix III. 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 radioisotope

$T_{1/2}$ is the half-life (in seconds) of the radioisotope

λ is the decay constant (seconds⁻¹) of the radioisotope = $\ln 2/T_{1/2}$.

A-143.2. The specific activity of any radioisotope not listed in Table AIII-1 of Appendix III can be calculated using the equation shown in para. A-143.1.

A-143.3. The specific activity of uranium, for various levels of enrichment, is shown in Table AIII-2 of Appendix III.

A-143.4. 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. 131) and in Table AIII-1 should be noted.

Surface contaminated object

A-144.1. Surface contaminated objects (SCO) 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 SCO. Such objects may, however, be regarded as LSA material insofar as the requirements specified in the LSA definition are complied with. See also para. A-131.10.

A-144.2. 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; and
- glove boxes with access ports blanked off for disposal.

A-144.3. Measurement techniques for fixed and non-fixed contamination of packages and conveyances are given in Sections 3 and 4 of Appendix II. These techniques are applicable to SCO materials. However, to properly apply these techniques a consignor needs to know the composition of the contamination.

Transport Index

A-146. See paras A-428.1 to A-429.

SECTION II

GENERAL PRINCIPLES AND PROVISIONS

GENERAL PRINCIPLES FOR RADIATION PROTECTION

A-201.1. The basic principles for radiation protection cover all activities involving ionizing radiation. The requirements for the transport of radioactive materials comply with the Basic Safety Standards for Radiation Protection, IAEA Safety Series No. 9 [6].

A-201.2. The term 'transport workers' usually covers all workers involved in all aspects of transport. The situation may vary from country to country and transport mode to transport mode.

A-201.3. In radiation protection, the system of dose limitation specified in Section IV of Safety Series No. 9 is summarized as follows:

— *Justification of practice*

No practice shall be adopted unless its introduction produces a positive net benefit;

— *Optimization of radiation protection*

All exposures shall be kept as low as reasonably achievable, economic and social factors being taken into account; and

— *Dose limits*

The dose to individuals shall not exceed the limits for the appropriate circumstances.

A-202.1. In the past, practical radiation protection was concerned mainly with ensuring compliance with 'dose limits' with less regard being paid to the other principles. In the currently recommended system of dose limitation more emphasis is placed on justification of practice and optimization of radiation protection. The dose limits are introduced as limiting conditions constraining the justification and optimization procedures, rather than permissible values to be used without optimization for the purposes of planning and design. For example, risks other than radiological may dominate the optimization procedure.

A-202.2. The IAEA, with the assistance of Member States and international organizations, has developed guidance on optimization of radiation protection in the transport of radioactive material [7]. Therefore, further guidance will not be provided here.

A-202.3. Workers involved with radioactive materials require training, especially at the beginning of the work. The extent of the radiation protection training depends

on the nature of transport operations. Reference [8] is an example of such guidance in handbook form.

A-203.1. Assessments of radiation exposure of workers and members of the public (including measurements where indicated) will, in addition to satisfying this requirement, provide necessary input to the competent authority and will help to achieve and maintain public confidence.

A-203.2. In order to comply with para. 203, information on the radiation doses to workers and to members of the public should be collected and reviewed at intervals of about 5 years. More frequent reviews should be made if circumstances warrant, e.g. if significant changes in transport patterns occur and when a new radioactive material related technology is introduced. The collection of relevant information may be achieved through a combination of radiation measurements and assessments.

A-204. Three categories for monitoring and assessing radiation doses are shown in para. 204. Most transport workers will be in the first category, where individual doses are most unlikely to exceed 5 mSv (500 mrem) per year. If higher doses are expected, then environmental monitoring and dose assessment are necessary. In many cases environmental monitoring may be satisfied 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 personal monitoring may also be required.

A-205.1. The Regulations state the principles of radiation protection which are to be applied in the determination through calculations 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 prepare these data in the form of segregation tables.

A-205.2. For developing procedures for properly segregating radioactive material from persons during transport and storage, the limiting dose values are, for workers, 5 mSv (500 mrem) per year and, for the public, 1 mSv (100 mrem) per year to members of the critical group (defined in para. 908 of Ref. [6]). These values are for segregation distance or dose rate calculation purposes only and are required to be used together with hypothetical but realistic parameters in order to obtain appropriate segregation distances and dose rates.

PROVISION FOR AVOIDING RADIATION DAMAGE TO FILM

A-206. 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 for film are being established for a specific transport mode, only a fraction of the limit prescribed in para. 206 should be committed to that mode.

GENERAL ACCIDENT PROVISIONS

A-207. To assist national authorities in preparing for and responding to transport accidents involving radioactive material, a Technical Document was published in 1982 [9] by the IAEA on Emergency Response Planning. It is intended to publish this material eventually as a Safety Series document.

A-208. In the event that the integrity of a package's containment system is violated in an accident, air and/or water may reach and, in some cases, chemically react with the contents. For some radioactive materials, these chemical reactions may produce caustic, acidic, toxic or poisonous substances which could be dangerous to people and the environment. Consideration should be given to this problem in the design of the package and in emergency response planning procedures to reduce the consequences of such reactions. In doing so, the quantities of materials involved, the potential reaction kinetics, the ameliorating effects of reaction products (self-extinguishing, self-plugging, insolubility, etc.), and the potential for concentration or dilution within the environment should all be considered. Such considerations may lead to restrictions on the package design, or its use, which go beyond considerations of the radioactive nature of the contents. Guidance on UF₆ transport is planned to be published in the IAEA-TECDOC Series.

GENERAL PROVISION FOR QUALITY ASSURANCE PROGRAMME

A-209.1 Organizations concerned with the transport of radioactive materials are required to develop and maintain an adequate quality assurance programme to ensure compliance with the Regulations and demonstrate that compliance to the competent authority. All aspects of transport mentioned in paras 103 and 209 should be covered by the programme. Appendix IV contains the foundation for such a programme and specifically covers the objectives and principles. The Appendix gives guidance to allow individual programmes to be developed flexibly to account for the wide variety of package designs and transport activities to be covered as well as the different approaches which may be required by the competent authorities of each Member State.

A-209.2. Each programme should be tailored to the specific organizational structure for which the programme is prepared, taking into account the specific transport activities of that organization.

A-209.3. The extent of the quality assurance programmes will depend on the type of transport activities being considered, ranging from minor requirements for the infrequent consignor of excepted packages, to extensive detailed requirements for regular consignors of packages subject to competent authority approval. Appendix V gives guidance in the form of a matrix on how to address each aspect of the quality assurance.

A-209.4. Each quality assurance programme should be available for review and audit by the respective competent authority or other applicable authority.

A-209.5. The following examples of non-conformity can easily be prevented by implementing an effective quality assurance programme:

- Failure to properly package a radioactive material
- Poor condition of packaging due to lack of maintenance, especially in the case of exposure devices and source changers used for radiography purposes
- Failure to properly prepare, label or document packages
- Understatement of Transport Index of packages — numbers specified on labels lower than true radiation level at an intermediate stage in journey (short-life nuclide) owing to the source being assessed at arrival radiation level or improper measurement performed prior to journey
- Failure to placard a vehicle correctly
- Improper removal of placards and labels from vehicles
- Packages insecurely stowed or improperly handled, resulting in lost and/or damaged packages
- Packages improperly stowed, inhibiting adequate heat rejection
- Improper application of exclusive use shipment controls
- Failure to comply with special arrangement conditions.

GENERAL PROVISION FOR COMPLIANCE ASSURANCE

A-210.1. An effective programme for compliance assurance by a competent authority should take account of all of the users of the Regulations, i.e. persons or organizations which at one time or another may be subject to the requirements of the Regulations such as

- consignors (shippers)
- carriers
- intermediaries (forwarders, brokers, warehousemen)
- packaging suppliers/manufacturers
- multiple regulatory organizations (i.e. shared responsibilities).

A-210.2. The competent authority which carries out the compliance assurance programme may vary from state to state, and within given states. For instance, in some states the consignor (shipper) may be regulated by the Ministry of Transport, the Nuclear Regulatory Authority, or both. Also, other organizations may be involved, e.g. in the area of emergency response. The carriage of radioactive materials may be regulated by the Ministry of Transport, Nuclear Regulatory Authority or local/provincial authority, and in many cases by regulatory authorities specific to the particular mode of transport.

A-210.3. A compliance assurance programme usually includes two major elements: on the one hand, the competent authority should review and approve certain activities in advance of the activity in question, and on the other hand, ensure through a regulatory inspection and enforcement programme that all the regulatory requirements are correctly followed in practice. The competent authority needs to be provided with adequate resources to perform these review, inspection and enforcement activities. In addition, the compliance assurance should also cover emergency response activities (see paras A-207 and A-208). General guidance on compliance assurance is being prepared by the IAEA in the form of a TECDOC.

A-210.4. A considerable effort in a compliance assurance programme is usually directed towards the user's application of quality assurance/quality control in the design, procurement, manufacture, use and maintenance of packages.

A-210.5. Periodically, the compliance assurance programme should be reviewed by the competent authority in the light of regulatory changes and the experience of user's performance since the programme was established. While the programme needs to be updated in a timely fashion as any specific change takes place, it is also desirable to periodically review the overall programme to ensure that it meets the goals it was designed to achieve. In some cases such reviews may be performed by review groups or ad hoc outside groups established solely to perform such reviews on an independent basis.

A-210.6. In order to assure safety of transport, the competent authority should review the design of a special form radioactive material and the design of certain types of packages as well as certain shipments and activities related to transport (see Section VII). In this work the competent authority needs many fields of expertise. The competent authority may also consider it necessary to perform independent assessments (analyses, tests) to verify the results presented in applications. Depending on the review results, the appropriate certificates will be given.

A-210.7. Through a compliance assurance programme, the competent authority should obtain assurance that all facets of the transport requirements are being met in practice by the users of the Regulations. The monitoring of the effectiveness of compliance by the user is most generally performed by routine, periodic inspections

(announced or unannounced) of the user's activities. For consignors these inspections are most generally an examination of the shipper's procedures before the transportation has actually taken place. For carriers, the inspections take place during transport itself. The frequency of inspections is established taking into account the scope and potential safety importance of the user's activity.

A-210.8. In inspections of the user's activity, the competent authority will usually direct its specific attention to assuring the following:

- (a) The user's management has provided the necessary personnel and resources to carry out an effective programme for compliance with the transport Regulations. This programme should clearly identify those persons who have the capability and who are responsible for the various specific requirements. Clear delegations of authority by management to those responsible persons are of extreme importance.
- (b) The user's management has provided the proper training of those persons who are responsible for carrying out the programme for compliance with the transport Regulations. Documentation of the training which has been provided should be submitted to the competent authority upon request.
- (c) The user is following established procedures for the design and fabrication or for the selection and procurement of packagings.
- (d) The consignor is using the proper packaging for the specific contents. Direct examination of packages being prepared for shipment should be made by the competent authority, when practicable.
- (e) The user has in his possession all of the required documentation, including the relevant competent authority certificates and any associated instructions for handling, loading, stowage, use and maintenance of the packaging. This is most often in the form of an instruction manual for the packaging.
- (f) The user follows established procedures for the preparation and use of the package in accordance with the approval certificate, instruction manual and related documents.
- (g) Procedures are established and followed to properly mark and label packages in accordance with the Regulations. This includes the proper determination and application of the correct transport index (TI). When practicable, the competent authority should directly observe these actions by the user.
- (h) Procedures are established and followed, and appropriate and properly calibrated instruments are provided, to monitor packages for both radiation and contamination.
- (i) Procedures are established and followed for preparing and controlling all relevant shipping documents correctly, for providing correct placarding of the carrier's vehicles, for providing all the required documentation to carriers and for providing any required notification to competent authorities of each country into which or through which the consignment is transported.

- (j) During transport, carriers are maintaining any required actions relating to placarding, stowage and separation of packages, etc., particularly any administrative controls relating to exclusive use shipments or supplementary operational controls specified in the competent authority certificate.

A-210.9. A highly important aspect of the competent authority's compliance assurance programme is the inspection effort which is directed to monitoring quality assurance. The main purpose is to ensure that the approved quality assurance programmes are implemented correctly. While all aspects of quality assurance should be audited, primary attention should be directed to the designers and manufacturers of packagings and should specifically include the inspection, maintenance and testing aspects of the quality assurance programme.

A-210.10. In verifying the effectiveness of the quality assurance procedures of a user, the competent authority will generally utilize a combination of inspection of records and physical inspections of users, especially designers' and manufacturers' facilities. The purpose is to ascertain that:

- the design of a package is unequivocally described by engineering drawings, material specifications, and methods of construction;
NOTE: For package designs requiring competent authority approval, this information is a required part of the application for the approval certificate. For other package designs, it is provided by the user upon request by the competent authority.
- the packagings are manufactured in complete accordance with the design;
NOTE: For package designs requiring competent authority approval, changes in the packaging construction methods, materials of construction, etc., are subject to the approval of the competent authority. For other package designs, such changes are documented and made available to the competent authority upon request. This applies equally to new package designs or to packagings in service.
- test and manufacturing equipment is properly controlled, qualified, calibrated, used and maintained in accordance with written procedures and schedules, with all results completely documented; and
- all the non-conformances are documented, reviewed and accepted or rejected correctly.

A-210.11. The user's management should be provided, upon completion of the inspection, with a summary of the results of the inspection findings, including non-compliances noted, affording an opportunity for management to present a reaction to such findings. Such a summary may be followed by a written letter from the competent authority, summarizing the findings, and asking for written response, if necessary.

A-210.12. Any system for compliance assurance should include provisions for enforcement. In this context, enforcement means any formal actions taken by the competent authority against the user of the Regulations when cases of violations or non-compliance by that user are observed. These observations are most often derived from the routine inspection programmes of the competent authority. A range of types of enforcement actions may be applied depending on the safety implications of the circumstances of the non-compliance. The application of enforcement sanctions should be carried out in the appropriate manner under the legal framework of the individual state. These sanctions could include, for example, the following:

- (a) *Written notice.* A written notice from the competent authority to the user setting out the non-compliance which has been observed or reported. This would require the user to provide a written reply explaining the causes of the non-compliance and the corrective actions taken to prevent a recurrence. This may be the most common type of enforcement sanction.
- (b) *Suspension.* This could involve a written notice of non-compliance accompanied by a statement of intention to suspend or revoke or modify a user's authorization unless or until the user demonstrates good reason why the suspension should not be applied. The obligation would be on the user to demonstrate that the non-compliance had ceased or that steps to prevent recurrence had been taken. In applying this type of sanction the competent authority should take into account the safety significance of the non-compliance, the prior enforcement history of the user, and the financial impact of the suspension on the user.
- (c) *Prosecution.* In circumstances where a non-compliance has occurred and the foregoing are considered inappropriate or have failed to prevent the user from continuing with the non-compliance, the competent authority may wish to initiate legal action against the user as a higher form of sanction. Such a sanction may be appropriate where:
 - the user has refused to rectify a non-compliance
 - the user has failed to cease an unsafe practice
 - there is evidence of deliberate negligence, or
 - there is evidence of criminal action.

A-210.13. Guidelines for the use of these sanctions may be prepared by the appropriate authorities in individual states for ensuring a fair and uniform application of the sanctions.

A-210.14. For the implementation of a compliance assurance programme, the preparation and issuing of information and guidance is needed. This may take the form of immediate bulletins on important and timely safety related matters. It may also take the form of information notices and guides which are intended to assist the users in the application and interpretation of the Regulations. Finally, it may involve

the developing and sponsoring of seminars, conferences, short courses, etc. for regulatory, carrier and other personnel, to make clear the correct application of the Regulations.

GENERAL PROVISION FOR SPECIAL ARRANGEMENT

A-211.1. To satisfy the basic conditions for the approval of transport under special arrangement the applicant is required to demonstrate that the overall level of safety provided in the package design features and the operational controls during transport is at least equivalent to that which would be achieved if all the applicable requirements were met. Subject always to the discretion of the competent authorities concerned with the shipment in question, it is possible to establish this by means of safety arguments ranging from considered judgement based on relevant experience to full probabilistic risk analysis. The applicability of the latter will of course be limited by the availability of appropriate information on the probable level of risk based on the assumption of full compliance with the applicable regulatory provisions themselves.

A-211.2. Approval under special arrangement will be sought in respect of shipments where variations from standard package design features result in the need to apply compensatory safety measures in the form of more stringent operational controls.

A-211.3. Details of possible additional controls which can be used in practice for this purpose are included in para. A-721.

SECTION III

ACTIVITY AND FISSILE MATERIAL LIMITS

BASIC A_1/A_2 VALUES

A-301.1 In a number of provisions of the Regulations it is necessary to know the A_1 or A_2 values for a radionuclide or mixtures of radionuclides. For the nuclides listed in Table I the values in TBq should be used. The values listed in Table I in Ci are for information only and are always less, in absolute terms, than the TBq values — in some cases significantly so owing to the methods of rounding used to select the respective values. The curie values are not intended for calculations or documentation used to establish contents limits.

A-301.2. In cases where the A_1 or A_2 value is listed as 'Unlimited', unlimited quantities apply wherever the respective A_1 or A_2 value is used (e.g. quantity limits for excepted packages, Table IV).

A-301.3. For tritium in liquid form the more limiting of either the total activity limit or the concentration (the total activity/litre) limit should be used as the limit for the content of tritium in Type A packages. For application to LSA and excepted packages the total activity limit for tritium should be used.

DETERMINATION OF A_1 AND A_2

A-303. In the event that A_1 or A_2 values need to be calculated the methods outlined in Appendix I in Safety Series No. 7 should be used. Two situations are considered here. First, a decay chain including one or more radionuclides in equilibrium in which the half lives of all daughters are less than ten days and in which no daughter has a half-life more than the parent nuclide, and second, any other situation. In the former case only the chain parent need be considered because the contribution of the daughters was considered in developing the A_1/A_2 values (see Appendix I of Safety Series No. 7); whereas, in the latter case, all the nuclides should be considered separately and considered as a mixture of radionuclides in accordance with para. 304.

A-304. Reactor plutonium recovered from low enriched uranium spent fuel (less than 5% uranium-235) constitutes a typical example of a mixture of radionuclides with known identity and quantity for each constituent. Calculations according to para. 304 result in activity limits independent of the abundance of the plutonium isotopes and the burnup within the range 10 000 MW·d/t to 40 000 MW·d/t. The

following values for reactor plutonium can be used within the above range of burnup taking into account the Am-241 buildup, up to five years after recovery:

$$A_1 = 20 \text{ TBq}$$
$$A_2 = 3 \times 10^{-3} \text{ TBq}$$

It is emphasized that these values can be applied only to the case of plutonium separated from spent fuel from thermal reactors, where the original fuel comprised uranium enriched up to 5% in uranium-235, where the burnup was in the range not less than 10 000 MW·d/t to not more than 40 000 MW·d/t, and where the separation was carried out less than five years before completion of the transport operation. It will also be necessary to separately consider other contaminants in the plutonium.

A-305. For mixtures of radionuclides whose individual identities are known but individual activities are not known in detail it may be possible to allocate nuclides to groups, taking the lowest A_1 or A_2 value, as appropriate, for each group to apply to all radionuclides within the group. Thus, if the total activity in each group is known and the lowest A_1 or A_2 value for a member of that group is known, a composite A_1 or A_2 for the mixture can be determined. The composite A_1 or A_2 will be lower than the highest group A_1 or A_2 to allow for the contribution from other groups. This method is most appropriate if mixtures of alpha and beta/gamma radionuclides are present (e.g. mixed fission products associated with transuranics). In this case knowledge of the total alpha activity and total beta/gamma activity is required together with knowledge of the most restrictive A_1 or A_2 values for the alpha emitters and beta/gamma emitters present.

A-306. If the knowledge specified in the last sentence of para. A-305 is not available, then the A_1 and A_2 values from Table II may be used.

CONTENTS LIMITS FOR PACKAGES

Packagings containing fissile material

A-315. Changes in the contained fissile material, either in quantity, form, arrangement, physical or chemical state, or changes in the packaging itself could adversely affect the neutron multiplication factor and invalidate the Transport Index of the package. The requirements set forth in the certificate of approval for a package are those necessary to provide safety during transport. The criticality safety assessment performed on the package, as part of the approval process, is for a specific set or range of parameters. The full range of acceptable conditions on these parameters should be described in the application for approval. Those parameters significant to criticality safety should be identified.

SECTION IV

PREPARATION, REQUIREMENTS AND CONTROLS FOR SHIPMENT AND FOR STORAGE IN TRANSIT

PACKAGE INSPECTION REQUIREMENTS

Before the first shipment

A-401.1. In the design phase of the package, documents should be prepared to define how the requirements of para. 401 are fully complied with for each manufactured packaging. Each document required should be initialled by the persons directly responsible for each stage of manufacture. Specific values should be recorded, even when within tolerance. The completed documents should be retained on file in conformance with quality assurance requirements (see para. 209).

A-401.2. In the case of a containment system having a design pressure exceeding 35 kPa, as required in para. 401(a), it should be confirmed that the containment system in the as-fabricated state is sufficient. This may be accomplished, for instance, through a test. For packagings with fill/vent valves, these openings can be used to pressurize the containment system to its design pressure. If the containment system does not have such penetrations, the vessel and its closure may require separate testing using special fixtures. During these tests, seal integrity should be evaluated using the procedures established for normal use of the package.

A-401.3. In performing the tests and inspections on packagings following fabrication to assess the effectiveness of shielding, to satisfy para. 401(b), the shielding components may be checked by a radiation test of the completed assembly. The radiation source for this test need not be the material intended to be transported, but care should be taken such that shielding properties are properly evaluated relative to energy, energy spectrum and type of radiation. Particular attention should also be paid to the homogeneity of packaging materials and the possibility of increased localized radiation levels at joints. For methods of testing the integrity of a package's radiation shielding see Refs [10, 11] and para. A-542.4.

A-401.4. Containment integrity should be assessed using appropriate leakage rate tests (see paras A-548.1 to A-548.15).

A-401.5. Inspection of a packaging for heat transfer characteristics should, in addition to a dimensional check, include special attention to ventilation apertures, surface emissivity and absorptivity and continuity of conduction paths. Proof tests, which may normally be necessary only for a prototype package, may be conducted using electrical heaters in place of a radioactive source.

A-401.6. In cases where criticality safety is dependent on the presence of neutron absorbers it is preferred that the neutron absorber be a solid and an integral part of the packaging. Solutions of absorbers are not endorsed for this purpose because their continued presence cannot be assured.

A-401.7. For further information see Refs [10–13].

Before each shipment

A-402.1. The certificate of approval is the evidence that a package design of an individual package meets the regulatory requirements and that the package may be used for transport. The provisions of para. 402 are designed to ensure that the individual package continues to comply with these requirements. Each check should be documented and initialled by the person directly responsible for that operation. Specific values should be recorded, even when within tolerances, and compared with results of previous tests, so that any indication of deterioration may become apparent. The completed documents should be retained on file in conformance with quality assurance requirements (see para. 209).

A-402.2. Inspection and test procedures should be developed to ensure that the requirements of paras 402(a) and 402(b) are satisfied. Compliance should be documented as part of the quality assurance programme (see para. 209).

A-402.3. To be in compliance with para. 402(c), it is recommended that detailed procedures be developed and followed to ensure that steady state conditions have been reached by measuring the temperature and pressure over a defined period. In the performance of any test it should be ensured that the method selected does not degrade the integrity of the package and that it provides the required sensitivity. Non-conformance with the approved design requirements should be fully documented and also reported to the competent authority which approved the design.

A-402.4. Every Type B package should be tested, after loading and before transport, to ensure compliance with the required leaktightness standard (see para. 402(d)). Some national authorities may permit an assembly verification procedure followed by a less stringent leakage test as offering equivalent confidence in meeting the design conditions. It is recommended that the competent authority of the country concerned be consulted if such a procedure is envisaged.

A-402.5. For further information see Refs [12, 13].

TRANSPORT OF OTHER GOODS

A-406. The transport regulatory documents of international transport organizations and the provisions laid down in regulatory documents of individual states

include, in some cases, specific storage, stowage and segregation requirements. As these regulations and provisions are frequently amended, the current editions should be consulted in order to ascertain these requirements with respect to other goods. A list of international and regional regulatory documents is provided in Appendix I.

OTHER DANGEROUS PROPERTIES OF CONTENTS

A-407.1. The Regulations provide an acceptable level of control of the radiation and criticality hazards associated with the transport of radioactive material. The Regulations do not cover hazards which may exist due to the physical/chemical form in which radionuclides are transported. In some cases such subsidiary hazards may exceed the radiological hazards. Compliance with the provisions of the Regulations therefore does not absolve its users from the need to consider all of the other potential dangerous properties of the contents.

A-407.2. By convention in the United Nations Recommendations on the Transport of Dangerous Goods (see Appendix I), all radioactive materials are classified in Class 7, though the other dangerous properties for some materials may be considerably higher. The United Nations Recommendations on the Transport of Dangerous Goods classify such goods as follows:

- Class 1 — Explosives
- Class 2 — Gases; compressed, liquefied, dissolved under pressure or deeply refrigerated
- Class 3 — Flammable¹ liquids
- Class 4 — Flammable¹ solids; substances liable to spontaneous combustion; substances which, on contact with water, emit flammable¹ gases
- Class 5 — Oxidizing substances; organic peroxides
- Class 6 — Poisonous (toxic) and infectious substances
- Class 7 — Radioactive materials
- Class 8 — Corrosives
- Class 9 — Miscellaneous dangerous substances.

A-407.3. In addition to meeting the requirements of the Regulations for its radioactive properties, radioactive consignments are required to comply with the requirements specified by relevant international transport organizations (see Appendix I), and applicable provisions adopted by individual states, for each individual substance on account of its other hazardous properties. This may include additional package design requirements and approvals by appropriate authorities.

¹ The word 'inflammable' has the same meaning as 'flammable'.

A-407.4. As an example, contaminated toluene with a specific activity of $< 10^{-5}$ A₂/g, transported as LSA-II under exclusive use would, from the radiological standpoint require IP-2 packaging standard, which corresponds to the United Nations packing group III. Toluene, however, is classified in the United Nations Recommendations in Class 3 and packing group II, to which more stringent performance standards apply.

A-407.5. For radioactive material transported under pressure, or where internal pressure may develop during transport under the temperature conditions specified in the Regulations, or when the package is pressurized during filling or discharge, the package may fall under the scope of pressure vessel codes of the Member States of concern.

A-407.6. Uranium hexafluoride (UF₆) is a particular example of radioactive material having subsidiary hazards. Depending on the degree of enrichment and amount of fissile uranium, UF₆ may be transported, from the radiological standpoint, in excepted or industrial packages, in Type A packages or in Type B packages. Nevertheless, because of other dangerous properties of UF₆ and of the reaction products of UF₆ with a humid atmosphere or with water, these packagings have in addition to comply with the regulations concerning the transport of both corrosive and poisonous materials. Furthermore, because these packagings are pressurized during loading and unloading operations they have to comply with pressure vessel regulations, although they are not pressurized under normal transport conditions. As a worldwide accepted standard, the American National Standard ANSI N 14.1-1982 [14] describes relevant design criteria for packages of different sizes with volumes from 0.15 up to 4040 L. According to this standard, UF₆ cylinders need to have an internal design pressure of 1.4 MPa while being capable of withstanding a test pressure of 2.8 MPa. Also an ISO standard is under preparation on the packaging of uranium hexafluoride (UF₆) for transport (ISO/DIS 7195), and an IAEA document on recommendations for providing protection during the transport of uranium hexafluoride is planned.

A-407.7. Performance tests for packagings of goods with dangerous properties other than radioactivity are prescribed in Chapter 9 of the United Nations Recommendations on the Transport of Dangerous Goods (see Appendix I).

A-407.8. Additional labels denoting subsidiary hazards should be shown on packages; in this case, however, there should be no class number in the lower half of the subsidiary hazard labels. The additional labelling should comply with the requirements specified by international transport organizations (see Appendix I).

A-407.9. Since the regulations promulgated by the international transport organizations as well as by individual Member States are frequently amended, the current editions of them should be consulted to ascertain what additional provisions apply with respect to subsidiary hazards.

REQUIREMENTS AND CONTROLS FOR CONTAMINATION AND FOR LEAKING PACKAGES

A-408.1. In certain instances, such as in the case of fuel flasks which may have been immersed in cooling ponds whilst being loaded with irradiated fuel, the contamination may not only be adhering to the surface but may be adsorbed into painted or even into metal surfaces. The rigours of transport, especially weathering effects, may transform some of this fixed contamination to non-fixed contamination. Care is necessary prior to dispatch to utilize appropriate decontamination methods to reduce the level of contamination such that the limits of non-fixed contamination would not be expected to be exceeded during the journey. It should be recognized that on some occasions the non-fixed contamination limits may be exceeded at the end of the journey. However, this situation presents no significant hazard because of the pessimistic assumptions used in calculating the derived limits for non-fixed contaminations. In such situations the consignee should inform the consignor to enable him to determine the causes and to minimize such occurrences in the future.

A-408.2. The most effective way of keeping the levels of non-fixed contamination on the surfaces of packages low is to prevent them from becoming contaminated. Loading, unloading and handling methods should be kept under review to achieve this. In the case of fuel flasks the use of a 'skirt' to eliminate contact with contaminated water in cooling ponds can prevent contamination of surfaces of the flask. If this is not possible, the use of strippable paints, pre-wetting with clean water, minimizing the immersion time and initiating decontamination as soon as possible may significantly reduce contamination uptake. Particular attention should be taken to remove contamination from joints and seal areas. Surface soiling should also be avoided wherever possible. Wiping a dirty surface both removes dirt and abrades the underlying substrate, especially if the latter is relatively soft, e.g. paint or plastic. Thus soiling can contribute to non-fixed contamination either by becoming contaminated itself or by generating loose contamination from the underlying substrate. Paints and plastics weather on exposure to sunlight. Amongst other effects, ultraviolet light oxidizes paint or plastic surfaces, thus increasing cation exchange capacity. This renders surfaces exposed to the environment increasingly contaminable by some soluble contaminants.

A-408.3. Advice on contamination limits and methods of measuring contamination is given in Appendix II.

A-409. See paras A-408.1 to A-408.3.

A-410.1. The prime purpose of inspection by a qualified person is to assess whether leakage or loss of shielding integrity has occurred or could be expected to occur, and either give assurance that the package is safe and within the limits prescribed in the Regulations or, if this is not so, to assess the extent of the damage or leakage and

the radiological implications. On rare occasions it may be necessary to extend surveys and investigations back along the route, the conveyances and the handling facilities to identify and clean up any contaminated areas. Investigations may need to include the assessment of external dose and possible radioactive intake by transport workers and members of the public.

A-410.2. As an example, in the event that a contamination incident is discovered during the sea transport of an irradiated nuclear fuel flask, the Japanese have established the following priority procedures to be followed:

- (1) Measure radiation level
- (2) Estimate range of influence of radiation
- (3) Control radiation exposure of crew and rescue team members
- (4) Assess requirement for protective clothing
- (5) Set up and mark access restricted area
- (6) Set up crew clean area
- (7) Check for external damage of package
- (8) Measure package surface temperature
- (9) Measure hold internal temperature
- (10) Measure forced cooling water temperature of package (if it is provided)
- (11) Investigate extent of any loss of radiation shielding ability
- (12) Investigate amount of leakage and estimate of its trend
- (13) Attach drip tray or absorbent material to any leaking point of package
- (14) Transfer leaked water into the contaminated water tank
- (15) Prevent spread of contamination by leaked water
- (16) Implement forced cooling of package
- (17) Implement forced cooling of hold
- (18) Prevent diffusion of radioactivity out of the hold
- (19) Estimate inner pressure of package
- (20) Prevent contamination of sound packages
- (21) Allow sea water entry to holds if necessary by emergency flooding system
- (22) Decontaminate package and
- (23) Record, report and communicate accident situation.

A-412. A qualified person should determine the frequency with which radiation surveys should be made on conveyances used routinely for the transport of radioactive materials.

A-413.1. Conveyances may become contaminated during the carriage of radioactive material from the non-fixed contamination on the packages. As shown in Table III, the conveyances and other associated equipment may become contaminated to the lower level (i.e. 0.4 Bq/cm^2 for beta, gamma and low toxicity alpha emitters and 0.04 Bq/cm^2 for all other alpha emitters) when they are carrying only excepted

packages. If the conveyance has become contaminated above this level, decontamination to at least this level should be performed unless the conveyance is to be used immediately for carriage other than excepted packages (i.e. IP-1, IP-2, IP-3, Type A and Type B), in which case the contamination limits are ten times higher. Of course, in every case decontamination to lower levels should be accomplished, 'reasonably achievable, economic and social factors being taken into account'.

A-413.2. Advice on contamination limits and methods of measuring contamination is given in Appendix II.

REQUIREMENTS AND CONTROLS FOR TRANSPORT OF EXCEPTED PACKAGES

A-416. For measuring the radiation level, an appropriate instrument should be used, i.e. sensitive to and calibrated for the type of radiation to be measured. In most cases only penetrating radiation (gamma rays and neutrons) needs to be taken into account. For establishing the radiation level on the surface of a package, it is normally adequate to take the reading shown on the instrument when the instrument is held against the surface of the package (see also A-433.1 and Table A-IV-2). The instruments used should, where possible, be small compared with the size of the package. In view of the usually small dimension of excepted packages, instruments with a small detection chamber (Geiger-Müller tube, scintillation meter or ionization chamber) are most suited for the purpose. The instrument should be reliable, in good condition, properly maintained and calibrated, and possess characteristics acceptable in good radiation protection practice.

A-417.1. Facilities and handling methods should be devised to avoid, when practicable, contamination on the external surfaces of packages.

A-417.2. Advice on the measurement of non-fixed contamination is given in Appendix II.

A-418. Advice on the measurement of radiation levels is given in A-416.

A-420.1. Examples of articles manufactured from natural uranium, depleted uranium or natural thorium are aircraft counterweights made of depleted uranium and coated with an epoxy resin, and uranium encased in metal and used as a shield in packagings for X-ray and gamma ray radiography and medical treatment devices.

A-420.2. The inactive sheath should cover the outer surfaces of the uranium or thorium to protect them from abrasion, to absorb the alpha radiation emitted and to reduce the beta radiation level at the accessible surfaces of the article. The sheath may be used to control the oxidation of the uranium or thorium and the consequent buildup of non-fixed (i.e. loose) contamination on the outer surfaces of such articles.

In the case of a depleted uranium shield incorporated in a packaging it is recommended that the uranium be sheathed with steel and the continuity of the envelope be assured by careful seam welding. As an example, the national regulations in the United States of America stipulate that the steel sheath must be at least 3.2 mm thick and the outside of the packaging is labelled showing that it contains uranium, to prevent it from inadvertently being machined or disposed of as scrap.

Additional requirements and controls for transport of empty packagings

A-421.1. Empty packagings which once contained radioactive material present little hazard provided they are conscientiously cleaned to reduce the non-fixed contamination levels to the excepted package levels specified in Table III of the Regulations, have external surface radiation levels below $5 \mu\text{Sv/h}$ (see para. 416) and are in good condition so that they may be securely re-sealed (see para. 421(a)); under these conditions the empty packaging may be transported as an excepted package.

A-421.2. Besides the situation described in para. A-421.1 the following cases may occur:

- (a) An empty packaging which cannot be securely closed due to damage or other mechanical defects may be shipped by alternate means which are consistent with the provisions of the Regulations, for instance under special arrangement conditions;
- (b) An empty packaging containing residual radioactive material or internal contamination in excess of the non-fixed contamination limits as specified in para. 421(c) should only be shipped as a package category which is appropriate to the amount and form of the residual radioactivity and contamination; and
- (c) The Regulations do not apply to an empty packaging, cleaned to an extent that there exists no contamination above the limits of 0.4 Bq/cm^2 for beta and gamma emitters and 0.04 Bq/cm^2 for alpha emitters and containing no radioactive material with a specific activity higher than 70 kBq/kg . (Note: For tanks see para. 404.)

A-421.3. The determination of the residual internal radioactivity within the interior of an 'empty' radioactive material packaging (see para. 421(c)) can be a difficult task. In addition to direct smears (wipes), various methods or combinations of methods which may be used include:

- gross radioactivity measurement;
- direct measurement of radionuclide; and
- material accountability, e.g. by 'difference' calculations, from a knowledge of the activity or mass of the contents and the activity or mass removed in emptying the package.

Whichever method or combination of methods is used, care should be taken to prevent excessive and unnecessary exposure of personnel during the measuring process. Special attention should be paid to high radiation levels which could exist when the containment system of an empty packaging is open.

REQUIREMENTS AND CONTROLS FOR TRANSPORT OF LSA AND SCO

A-422. In the case of solid radioactive waste essentially uniformly distributed in a concrete matrix placed inside a thick wall concrete packaging, the shielding of the concrete wall should not be considered as satisfying the condition of para. 422. However, the radiation level at 3 m from the unshielded concrete matrix may be assessed by direct measurement outside the thick wall of the concrete packaging and then corrected to take into account the shielding effect of the concrete wall.

A-425. According to paras 144(a)(iii) and 425(c), SCO-I is allowed to have non-fixed contamination on inaccessible surfaces in excess of the values specified in para. 144(a)(i). Items such as pipes resulting from the decommissioning of a facility should be prepared for unpackaged transport in a way to ensure that there is no release of radioactive material into the conveyance. This can be done, for example, by using end caps or plugs at both ends of the pipes.

DETERMINATION OF TRANSPORT INDEX (TI)

A-428.1. The TI for radiation protection purposes is determined by scanning all surfaces of a package, including the bottom, at 1 m distance. The highest value measured is the value that determines the TI based on radiation levels. Similarly, the TI for a tank, a freight container and unpackaged LSA-I and SCO-I materials is determined by measuring at 1 m from the surfaces, but a multiplication factor according to the size of the load should be applied in order to define the TI. For over-packs this method is only permitted for the original consignor. Where there are protrusions on the exterior surface, the protrusion should be ignored in determining the 1 m distance except in the case of a finned package in which case the measurement should be made at 1 m distance from the external envelope of the package. For unpackaged LSA-I material the surfaces should be considered to be the walls of the conveyance or freight container.

A-428.2. It is relatively simple to make a direct measurement with a hand held monitoring instrument. In practice, only penetrating radiation is taken into account. In some cases consideration should be given to the possibility of radiation increase as a result of the buildup of daughter nuclides during transport. In such cases a proper correction should be applied so that the TI would represent the highest radiation level envisaged during the transport.

TABLE A-I. CONVERSION FACTOR (NEUTRON FLUENCE RATE ($\text{cm}^{-2} \cdot \text{s}^{-1}$) CORRESPONDING TO A DOSE EQUIVALENT INDEX RATE OF $1 \mu\text{Sv/h}$) [6]

Neutron energy (MeV)	Conversion factor ($\text{cm}^{-2} \cdot \text{s}^{-1}$) per ($\mu\text{Sv/h}$)
2.5×10^{-8} (thermal)	26.0
1×10^{-3}	27.0
2×10^{-2}	17.0
1×10^{-1}	4.8
5×10^{-1}	1.4
1	0.85
5	0.68
10	0.68

A-428.3. The radiation level due to neutrons may be measured with a neutron dose equivalent ratemeter [15] or calculated on the basis of Table A-I. 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 level being emitted by the radionuclide and that the calibration of the instrument is still valid. In performing both the initial measurement and a check measurement, it should be noted that the calibration accuracy of the instrument is approximately $\pm 15\%$. Differences as great as 30% are therefore possible from one reading to the other, with the readings still being acceptable.

A-429. The transport index assigned to a package becomes a characteristic of the package. Consignors, carriers, consignees, and emergency response personnel rely on the transport index for information on how the package should be treated and handled. Extreme conservatism in nuclear criticality safety evaluations is to be discouraged as an adequate margin of safety will be obtained through compliance with the requirements of para. 315 and paras 562 to 568.

LIMITS ON TRANSPORT INDEX AND RADIATION LEVEL FOR PACKAGES AND OVERPACKS

A-433.1. 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. Large instruments 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 5), the effect is negligible and can be ignored, otherwise the values in Table A-II 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.

A-433.2. When measuring surface radiation limits, the instrument should be held as close to the package as is practicable, but in the case of finned packages the measurement should be at the outside contour of the fins. It should be adequate, where other conditions are not specified, for measurements to be averaged over an area not larger than 10 cm² at a distance of 5 cm from the packaging (see Ref. [16], para. 70).

TABLE A-II. CORRECTION FACTORS FOR PACKAGE AND DETECTOR SIZES

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

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

A-433.3. In some cases consideration should be given to the possibility of an increase in the radiation level as a result of the buildup of daughter nuclides during transport. In such cases a proper correction should be applied in order to assure that the surface radiation limit is not exceeded.

A-434. Even though a package is permitted to have an external radiation level up to 10 mSv/h (1000 mrem/h) the requirements for a maximum dose limit of 2 mSv/h (200 mrem/h) on the surface of the conveyance or of 0.1 mSv/h (10 mrem/h) at any point 2 m from the surface of the conveyance (see para. 465) may be more limiting in certain instances. See also para. A-433.3.

CATEGORIES

A-435. All packages and overpacks other than those consisting entirely of excepted packages have to be assigned a category. This is a necessary prerequisite to labelling and placarding.

MARKING, LABELLING AND PLACARDING

Marking

A-436.1. Markings on packages should be boldly printed, of sufficient size and sensibly located to be legible, bearing in mind the likely handling means to be employed. A character height of 12.5 mm should be considered a suitable minimum for light weight packages (i.e. up to a few hundred kilograms) where close contact mechanical means, e.g. forklift trucks, are likely to be employed. Heavier packages will require more 'remote' handling methods and the character size should be increased accordingly to allow operators to read them at a distance. A size of 65 mm is considered to be sufficient for the largest packages of tens of tonnes to the hundred tonne range. To ensure legibility, a contrasting background should be applied prior to marking if the external finish of the package does not already provide a sufficient contrast. Black characters on a white background are suitable. Where packages have irregular outer surfaces (e.g. fins or corrugations), or surfaces unsuitable for direct application of the markings, it may be necessary to provide a flat board or plate on which to place the markings to enhance legibility.

A-436.2. Markings should be durable in the sense of being at least resistant to the rigours of routine transport, including the effects of open weather exposure and abrasion, without substantial reduction in effectiveness. Attention is drawn to the need to consult national and modal transport regulations which may contain stricter requirements. For example, the International Maritime Dangerous Goods (IMDG)

Code (see Appendix I) requires all permanent markings (and also labels) to remain identifiable on packages surviving at least three months immersion in the sea. When a board or plate is used to bear a marking, it should be fitted securely to the package in a manner which is consistent with the integrity standard of the package itself.

A-436.3. The means of marking will depend on the nature of the external surface of the packaging itself, ranging (in order of durability) from stencilling or soft stamping with indelible inks or paints (suitable for fibreboard or wooden packagings), through branding (for wooden packagings), painting with enamel or resin based paints (suitable for many surfaces, particularly metals), to hard stamping, embossing or 'cast-in' markings of metallic outer packagings.

A-436.4. The scheduled inspection and maintenance programme required for packagings should include provisions to inspect all permanent markings and to repair any damage or defects. Experience from such inspections will indicate whether durability has been achieved in practice.

A-436.5. Appropriate national and modal transport regulations should always be consulted to supplement the general advice in paras A-436.1 to A-436.4, as variations in detailed requirements may be considerable.

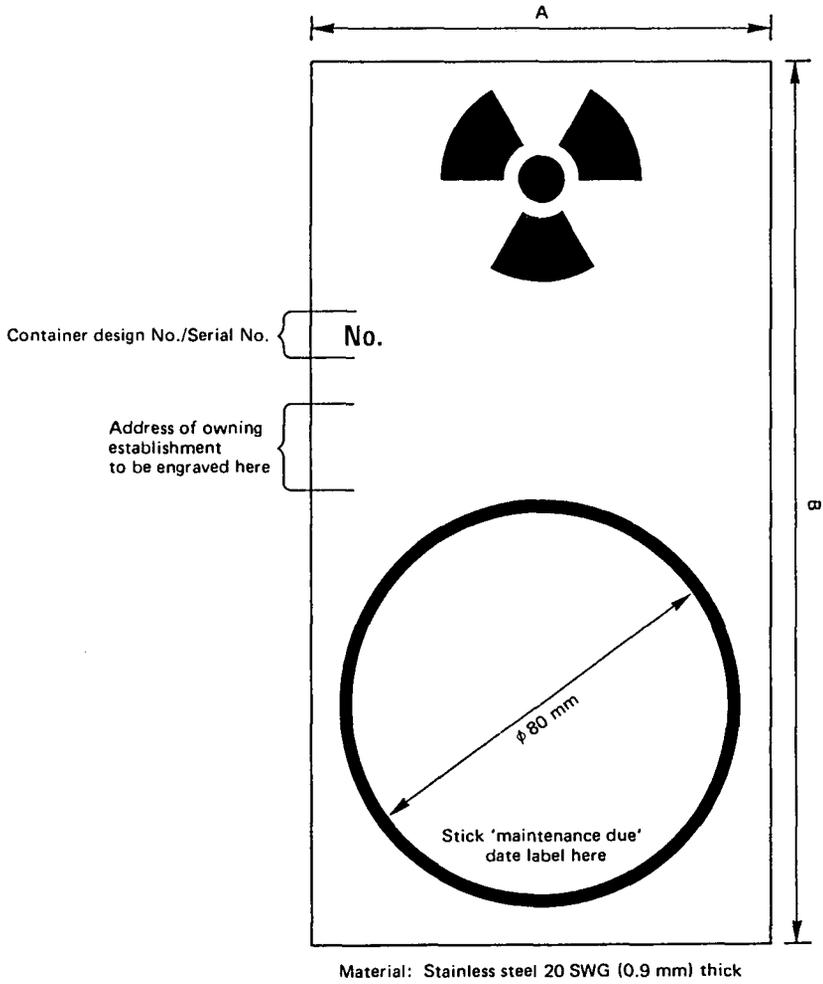
A-437. See paras A-436.1 to A-436.5.

A-438. General advice on legibility, durability of marking methods and inspection/maintenance of markings is given in paras A-436.2 to A-436.5 above. However, where possible the competent authority identification mark, serial number and Type B(U)/Type B(M) mark should not be rendered illegible, obliterated or removed even under accident conditions. It may be convenient to apply such markings adjacent to the trefoil symbol on the external surface of the package. For example an embossed metal plate as shown in Fig. A-2 may be used to combine these markings.

A-439. See para. A-438.

Labelling

A-442. In the identification of the most restrictive radionuclides for the purpose of identifying a mixture of radionuclides as the contents on a label, consideration should be given not only to the lowest A_2 values, but also to the relative quantities of radionuclides involved. Such considerations are used for example when determining the A_2 value of a mixture of radionuclides. In some cases it may not be necessary to determine the most restrictive radionuclides for each package containing a mixture because, for example, the material may come from the same source and have the same radionuclides as the most restrictive radionuclides for a wide range of mixture compositions. In some cases consideration should be given to the



All dimensions are in millimetres

A	B
95	180
180	250

FIG. A-2. Example of container identification plate [17].

possibility of activity increase as a result of the buildup of daughter nuclides during transport. In such cases a proper correction should be applied in order to determine the maximum activity during transport.

CONSIGNOR'S RESPONSIBILITIES

Particulars of consignment

A-447.1. The attention of the consignor is drawn to the particular requirement of para. 447(o) regarding consignments of packages in an overpack or freight container. Each package or collection of packages is required to have documents for the appropriate consignee. This is important in regard to the 'Consignor's declaration'. Nobody other than the consignor can make this declaration and so he is required to assure that appropriate documents are prepared for all parts of a mixed consignment so that they can continue their journey after being removed from an overpack or freight container.

A-447.2. Care should be exercised in the selection of the portion of the entry for United Nations No. 2910 (see Appendix of the Regulations) which constitutes the proper shipping name of the consignment. Portions of an entry not highlighted by capital letters are not considered part of the proper shipping name, e.g. the conjunction 'or'. Examples illustrating the selection of proper shipping names of the entry for UN No. 2910 are:

- UN No. 2910 RADIOACTIVE MATERIAL, EXCEPTED PACKAGE,
 - INSTRUMENTS or ARTICLES
 - LIMITED QUANTITY OF MATERIAL
 - ARTICLES MANUFACTURED FROM NATURAL URANIUM or DEPLETED URANIUM or NATURAL THORIUM
 - EMPTY PACKAGING

The proper shipping name is the applicable description from the following:

- UN No. 2910 RADIOACTIVE MATERIAL, EXCEPTED PACKAGE,
INSTRUMENTS
- UN No. 2910 RADIOACTIVE MATERIAL, EXCEPTED PACKAGE,
ARTICLES
- UN No. 2910 RADIOACTIVE MATERIAL, EXCEPTED PACKAGE,
LIMITED QUANTITY OF MATERIAL
- UN No. 2910 RADIOACTIVE MATERIAL, EXCEPTED PACKAGE,
ARTICLES MANUFACTURED FROM NATURAL
URANIUM

UN No. 2910 RADIOACTIVE MATERIAL, EXCEPTED PACKAGE,
ARTICLES MANUFACTURED FROM DEPLETED
URANIUM

UN No. 2910 RADIOACTIVE MATERIAL, EXCEPTED PACKAGE,
ARTICLES MANUFACTURED FROM
NATURAL THORIUM

UN No. 2910 RADIOACTIVE MATERIAL, EXCEPTED PACKAGE,
EMPTY PACKAGING

A-447.3 The last entry 'excepted package, empty packaging' is required to be used for empty packagings which have contained radioactive material. If there are residues or 'heels' in the packaging, e.g. in UF₆ packages, the packaging should not be called 'empty packaging' but should be shipped as a package. The quantity remaining would determine the package category.

A-447.4. The maximum activity of the contents during transport is required to be included in the transport documents (para. 447(i)). In some cases the activity may increase as a result of the buildup of daughter nuclides during transport. In such cases a proper correction should be applied in order to determine the maximum activity. In many cases, however, the activity will decrease during transport. In such cases, although not required by the Regulations, it may be desirable also to include in the transport documents, as additional information, the estimated activity of the contents at a later reference date.

TRANSPORT

Segregation during transport

A-460.1. In order to implement the principles for radiation protection contained in paras 201–205, simple procedures have been developed which will suitably limit radiation exposures to both persons and undeveloped film.

A-460.2. An effective way of limiting exposures to persons (transport workers, crews and members of the public) during transport is to require appropriate segregation distances between the radioactive material and the areas where people may be present. The Regulations provide the basis for the determination of segregation requirements but the actual determination and specification of these requirements is done at the modal level. Segregation distance requirements are prescribed by national regulatory bodies and international transport organizations such as the International Civil Aviation Organization (ICAO), the International Maritime Organization (IMO), etc. (see Appendices I and VI).

A-460.3. In the calculation of segregation distances it is usually sufficiently accurate to treat a package or a collection of packages as a point source of radiation. Radiation levels at various distances from the surface or surfaces can be calculated using the 'single point source inverse square' relationship and modelling the package as a sphere with the source at its centre. A package is unlikely to be spherical so it is conservative to take the radius of the sphere as half the minimum dimension of the package (which could be the length, width or height), while applying the maximum possible surface radiation level at all surfaces of the package. Under these conditions the following relationship holds:

$$\frac{RL_s}{RL_d} = \frac{(d + r)^2}{r^2}$$

where: RL_s is the maximum radiation level at any surface of the package (mSv/h)
 RL_d is the maximum possible radiation level at a distance d from all surfaces of the package (mSv/h)
 d is the distance of interest from all package surfaces (m)
 r is the radius (half the minimum dimension) of the package (m).

A-460.4. There are many transport mode specific considerations and conditions which should be factored into the models used to calculate segregation distances [18]. An example of the use of such a model is given in Appendix VI, i.e. calculations for determining minimum segregation and spacing distances for passenger and cargo aircraft.

A-460.5. If different classes of dangerous goods are being transported together, there is a possibility that the contents of leaking packages may affect adjacent cargo, e.g. a leak of corrosive material could reduce the effectiveness of the containment system for a package of radioactive material. Thus, in some cases it has been found necessary to restrict the classes of dangerous goods that may be transported near other classes. In some cases it may simply be stated which classes of dangerous goods must be segregated from others. In order to provide a complete and easy procedure for understanding the requirements it has been found that presentation of this information in a concise tabular form is useful. In such tables the various classes of dangerous goods are presented on both vertical and horizontal axes of a matrix. At an intersections of a column and a row it is possible to present information on the segregation requirement. For example, if the goods must not be transported together a number can be used; if no general restrictions need be applied, the entry can be an 'X'; and if specific conditions need to be met, footnotes and other symbols can be used. As an example of a segregation table, the one included in Section 15 of the General Introduction to the IMDG-Code (see Appendix I) is given in Table A-III.

TABLE A-III. SAMPLE SEGREGATION BETWEEN CLASSES

(Taken from the IMDG-Code, page 0112, Amendment 21-83, see Appendix I.)

The table below shows the general requirements for segregation between classes, but since the properties of substances or articles within each class may vary greatly, the individual schedules and relevant introduction to that class should always be consulted for individual requirements for stowage and segregation.

		1.1				2.2 3.1												
		1.5	1.2	1.3	1.4	2.1	2.3	3.2	3.3	4.1	4.2	4.3	5.1	5.2	6.1	6.2	7	8
Explosives	1.1	*	*	*	*	4	2	4	4	4	4	4	4	4	2	4	2	4
Explosives	1.5	*	*	*	*	4	2	4	4	4	4	4	4	4	2	4	2	4
Explosives	1.2	*	*	*	*	4	2	4	4	4	4	4	4	4	2	4	2	4
Explosives	1.3	*	*	*	*	4	2	4	4	3	3	4	4	4	2	4	2	2
Explosives	1.4	*	*	*	*	2	1	2	2	2	2	2	2	2	X	4	2	2
Flammable gases	2.1	4	4	4	2	X	X	2	2	1	2	1	2	4	X	4	2	1
Other than flammable gases	2.2	2	2	2	1	X	X	2	2	X	1	X	X	2	X	2	1	X
Flammable liquids	2.3	2	2	2	1	X	X	2	2	X	1	X	X	2	X	2	1	X
Flammable liquids	3.1	4	4	4	2	2	2	X	2	2	2	2	3	X	3	2	1	
Flammable liquids	3.2	4	4	4	2	2	2	X	2	2	2	2	3	X	3	2	1	
Flammable liquids	3.3	4	4	4	2	2	2	X	1	2	2	2	3	X	3	2	1	
Flammable solids	4.1	4	4	3	2	1	X	2	1	X	1	1	1	2	X	3	2	1
Spontaneously combustible substances	4.2	4	4	3	2	2	1	2	2	1	X	1	2	2	X	3	2	1
Substances which are dangerous when wet	4.3	4	4	4	2	1	X	2	2	1	1	X	2	2	X	2	2	1
Oxidizing substances	5.1	4	4	4	2	2	X	2	2	1	2	2	2	2	1	3	1	2
Organic peroxides	5.2	4	4	4	2	4	2	3	3	2	2	2	2	X	1	3	2	2
Poisons	6.1	2	2	2	X	X	X	X	X	X	X	X	1	1	X	1	X	X
Infectious substances	6.2	4	4	4	4	4	2	3	3	3	3	2	3	3	1	X	3	3
Radioactive materials	7	2	2	2	2	2	1	2	2	2	2	2	1	2	X	3	X	2
Corrosives	8	4	4	2	2	1	X	1	1	1	1	1	2	2	X	3	2	X
Miscellaneous dangerous substances	9	No general segregation recommended; individual schedules should be consulted.																

Numbers relate to the following terms as defined in subsection 15.8:

1. Away from
2. Separated from
3. Separated by a complete compartment or hold from
4. Separated longitudinally by an intervening complete compartment or hold from
- X No general segregation recommended; individual schedules should be consulted

* See subsection 5.4 of the introduction to class 1 for separation of goods of class 1 of different compatibility groups from one another.

Note: Poisonous gases which are also flammable should be segregated as class 2.1 gases.

A-460.6. Since mail bags often contain undeveloped film and will not be identified as such, it is prudent to protect mail bags in the same way as identified undeveloped film.

Stowing for transport

A-462.1. Within the context of the Regulations, 'stowage' means both the locating within a conveyance of a radioactive material package relative to other cargo (both

radioactive and non-radioactive) and the use of dunnage, braces, blocks or tie-downs to prevent movement within a conveyance during transport. Packages need to be stowed in conveyances to prevent loss or movement from inertial forces which may arise during normal conditions of transport. Such forces could be generated by uneven road or track, vibration, direction changes, acceleration/deceleration, rail shunting and motions in the air or seaway. Generally larger and heavier packages are secured to the conveyance by means of a tie-down system. Lightweight and small packages are normally carried in a closed conveyance and, provided that they are appropriately blocked and braced to prevent significant movement, do not pose a tie-down problem. Tie-down equipment and instructions will sometimes be provided by the consignor. Where no such equipment and instructions are provided it is the responsibility of the carrier to ensure that the package is secured within the conveyance, e.g. airlines normally provide means of securing packages in aircraft holds.

A-462.2. Maximum forces exerted by the tie-down system on the package under accident conditions cannot be reliably predicted unless a weak link of known strength is provided. This weak link could be intentionally fitted to some part of the tie-down system external to the package and in this case assurance of the tie-down under normal conditions of transport in the prescribed manner is necessary. Consequently the tie-down system or the package attachment lugs should be designed to withstand normal conditions of transport and fail under excessive load without impairing the ability of the package to meet the test requirements. Normally the designer will take this into account, thus avoiding reliance on the carrier to provide a weak link on the tie-down system. (See also paras A-527.1 to A-527.3 for additional information.)

A-462.3. Two basic schemes, the rigid or bolted tie-down and the tension member tie-down are in common use. The bolted tie-down, which combines the use of tie-down members and chocks (a chock is a means of preventing a package from sliding) is generally considered superior from a load carrying standpoint, particularly for heavy packages. However, unless the package is routinely transported on the same conveyance there may be practical disadvantages in fastening bolts to the conveyance. In such cases it may be desirable to secure the package with tension members such as wire ropes, chains or screwed rods which can be more readily adapted to conveyance platforms. A combination of chocks and tension members may also be considered. If modifications to the conveyance are not feasible, or not allowed, the tension member method may be the only possible means of tie-down.

A-462.4. The proper use of tension member tie-down systems presents many points that should be considered:

- (a) Chocks should be used, if at all possible, since the loads in the members of the tie-down system could be a factor of five less than the equivalent case without chocks. The means of securing the chocks to the conveyance platform need to be capable of withstanding the forces produced.

- (b) The angle formed by tie-down members with the conveyance platform when viewed from the side and above should be close to 45°. If the package is large in relation to the size of the conveyance platform the tie-down members should be crossed to achieve the nominally 45° angle. Rubbing of members on each other or on parts of the package or conveyance should be prevented.
- (c) Tie-down members should be pre-tensioned to avoid slackening during use. This tension should be checked and maintained throughout the journey. The loosening of members by vibration during transit should be prevented by the use of vibration resistant connections.
- (d) The tie-down should be anchored to the frame of the conveyance and not to the wooden or metal deck unless it can be demonstrated that these are capable of withstanding the expected forces.
- (e) Friction between the package and the conveyance platform should be ignored: it can be regarded as a bonus giving an additional unquantifiable margin of safety.

A-462.5. Information on acceleration values and on calculation methods for forces that can be imposed on tie-down members and the package when subjected to simultaneous accelerations in three directions is given in Appendix VII. National and international regulations and standards need to be consulted to obtain maximum acceleration values for particular modes of transport.

A-465. It should be noted that for the transport of a freight container there may be more than one entry in Table XI which is applicable. As an example, for a large freight container to be carried on a seagoing vessel the loading of packages into the freight container is controlled by the entry 'Freight Container — Large' in Column 1 of Table XI. Additionally, the loading of a large freight container aboard a vessel is controlled by the entries under 'Seagoing vessel' and both 'Hold, compartment or defined deck area' and 'Total vessel' in Column 1 of Table XI. It is also important to note that several requirements presented in the footnotes apply to certain shipments. These footnotes are requirements and are not just for information.

A-466. An overpack containing packages with a TI greater than 10 is required to be transported under exclusive use. The exclusive use requirement for transport of a package with a TI greater than 10 cannot be eliminated by the use of an overpack since overpacks with a TI greater than 10 are also required to be transported under exclusive use in the same way as packages under the same condition.

Additional requirements relating to transport by rail and by road

A-469.1. In most cases the radiation level at any point on the external surface of a package is limited to 2 mSv/h (200 mrem/h). For road and rail transport, when transported under exclusive use, packages and overpacks are allowed to exceed 2 mSv/h (200 mrem/h) if access to the enclosed areas in the vehicle is restricted.

Restricting of access to these areas may be achieved by using an enclosed vehicle that can be locked, or by bolting and locking a cage over the package with the surface of the cage meeting the 2 mSv/h (200 mrem/h) limit. In some cases the open top of a vehicle with side walls may be covered with a tarpaulin but this type of enclosure would generally not be considered adequate for preventing access.

A-469.2. During transit there should be no unloading or entering into the enclosed area of a vehicle. If the vehicle is being held in the carrier's compound for any period it should be parked in an area where access is controlled and where people are not likely to remain in close proximity for an extended period. If it is required to do maintenance work on the vehicle for an extended period, then arrangements should be made with the consignor or the consignee to ensure adequate radiation protection, e.g. by providing extra shielding and radiation monitoring.

A-469.3. It is essential to secure a package or overpack to prevent movement during transport which could cause the radiation level to exceed relevant limits or to increase the dose to the vehicle driver. For road transport a package or overpack should be secured for acceleration, braking, and turning as expected during normal conditions of transport. For rail transport, packages should also be secured to prevent movement during 'humping' of the rail car. See paras A-462.1 to A-462.5.

A-470.1. In the application of paras 204 and 470(b) consideration should be given to the probable exposure times of drivers and assistants. If the probable exposure time is greater than 750 hours per year, an annual dose in excess of 15 mSv (1.5 rem) could be received at the limiting dose rate of 0.02 mSv/h (2 mrem/h). The vehicle should be loaded in such a way that the radiation level in occupied positions is minimized. This may be achieved by placing packages with higher radiation levels furthest away from the occupied area and placing heavy packages with low radiation levels nearer to the occupied position.

A-470.2. The radiation levels in the normally occupied positions in a vehicle should be measured with portable instruments designed to measure penetrating radiation (i.e. gamma and neutrons), as described in paras A-433.1 and A-433.2.

Additional requirements relating to transport by vessels

A-471. Transport by sea of any package having a surface radiation level exceeding 2 mSv/h (200 mrem/h) is only allowed under special arrangement conditions, except when transported in or on a vehicle under exclusive use and when subject to the conditions of para. 469. However, if the latter situation occurs, it may be desirable for purposes of radiation protection that a specific area be allocated for that vehicle by the master of the ship or the competent authority concerned. This would be appropriate in particular for the transport of such vehicles aboard roll-on/roll-off ships or ferries. Further guidance will be found in the IMDG Code (see Appendix I).

A-472.1. Special use vessels which are employed for the transport by sea of radioactive material have been adapted and/or dedicated specifically for that purpose. The required radiation protection programme should be based upon preplanned stowage arrangements specific to the vessel in question and to the number and the nature of the packages to be carried. The radiation protection programme should take into account the nature and intensity of the external radiation likely to be emitted by packages. This information should be used to define stowage locations in relation to regularly occupied working spaces and living accommodation, in order to establish that radiation levels in such places will not exceed applicable limits. Occupancy factors based on the planned maximum duration of voyages should also be taken into account. The competent authority, normally the competent authority of the flag state of the vessel, may specify the maximum number of packages permitted, their identity and contents, the precise stowage arrangements to be observed and the maximum radiation levels permitted at key locations. The radiation protection programme would normally require that appropriate monitoring be carried out during and after completion of stowage as necessary to ensure that specified dose rates are not exceeded. Details of the results of such surveys, including any checks for contamination of packages and of cargo spaces may have to be provided to the competent authority.

A-472.2. For packages containing fissile material, the programme will also need to take appropriate account of the criticality contribution in an array and a maximum transport index allowable to ensure nuclear criticality control.

A-472.3. Although not directly part of a radiation protection programme, limitations on stowage associated with the heat output from each package may need to be considered. The means for heat removal, both naturally and mechanically, will need to be assessed for this purpose and heat outputs for individual packages need to be specified.

A-472.4. Records of measurements taken during each voyage may be required to be supplied to the competent authority on request. This is one method of assuring that the radiation protection programme and any other controls have functioned adequately.

A-472.5. "Persons qualified in the carriage of radioactive material" should be taken to mean persons who possess appropriate special knowledge of the handling of radioactive materials.

A-472.6. An example of a radiation protection programme of a special use vessel is shown in Appendix VIII.

Additional requirements relating to transport by air

A-473. This requirement relates to the presence of passengers on an aircraft rather than its capability to carry passengers. According to the definition, a passenger

aircraft (para. 112) is “an aircraft which is carrying any person other than a crew member, a carrier’s employee in an official capacity, an authorized representative of an appropriate national authority, or a person accompanying a consignment”. Thus an aircraft equipped to carry passengers, but which is carrying no passengers on that flight, then meets the definition of a cargo aircraft and so can be used for the transport of Type B(M) packages and of consignments under exclusive use.

A-475. Packages or overpacks that exceed 2 mSv/h (200 mrem/h) on the surface cannot be shipped under exclusive use by air unless they are transported under special arrangement. The special arrangement authorization should include consideration of handling, loading and in-flight arrangements in order to control the radiation doses to flight crew, ground support personnel and incidentally exposed persons. This may necessitate special instructions for crew members, notification to appropriate persons such as terminal staff at the destination and intermediate points and special consideration of transfer to other transport modes.

Additional requirements relating to transport by post

A-476. In establishing restrictions on the movement of radioactive materials in the domestic postal system, the postal authorities should concurrently consider the requirements for the international movement of such packages. In most cases these international requirements could be applied to domestic carriage.

A-477. Care will need to be taken in deciding which consignors may be authorized to dispatch radioactive materials by post. When authorization is given to an organization for the use of the postal service, it would be preferable that one suitably knowledgeable and responsible individual be appointed to ensure that the correct procedures and limitations are observed.

STORAGE IN TRANSIT

A-478.1. Workers in a transit area cannot always and continually be physically segregated from packages of radioactive materials and so control of radiation exposure is important. A limiting dose level of 5 mSv (500 mrem) per year is given in the Regulations (para. 204) for workers not subject to detailed radiological supervision. To establish that this is unlikely to be exceeded, estimates should be made for the areas where workers are in frequent contact with packages, or in the vicinity of such packages, or monitoring of these areas should be carried out (see also para. A-204).

A-478.2. For segregating members of the public from radioactive material packages, a limiting dose level of 1 mSv (100 mrem) per year to the critical group

is given in the Regulations (para. 205(b)). It is therefore necessary to establish adequate segregation distances for storage/transit areas for protection of the public. In calculating such distances the TI of the packages, and the maximum time of occupancy should be considered. An example of segregation distance calculations is given in Appendix VI. If there is any doubt regarding the effectiveness of the distance then a check may be carried out using appropriate instruments for the measurement of radiation levels. See also paras A-201.1 to A-205.2.

A-478.3. Undeveloped photographic film is sensitive to radiation and, therefore, is required to be stowed away from packages of radioactive material both in transit and in storage so as to avoid damage to the film. A limit of 0.1 mSv (10 mrem) per consignment of film is given in the Regulations. Road and air transport are the likely modes where there is a possibility that these two materials may come together and care will be necessary to prevent this. A driver of a vehicle carrying packages of radioactive materials should be instructed to ensure that, when he parks his vehicle, there is a clear space of at least 2 m all round the vehicle. Since vehicles carrying radioactive packages with significant external radiation levels will be placarded, other vehicles carrying film will be alerted to avoid parking in the clear space left (see also para. A-206).

A-479. The requirement to maintain a spacing of 6 m is necessary for nuclear criticality control. Where two storage areas are divided by a wall, or similar boundary, storage of the packages on opposite sides of the wall has still to meet the requirement for 6 m segregation.

CUSTOMS OPERATIONS

A-483.1. For the examination of packages containing radioactive materials by customs officials the following advice is given:

- (a) Clearance formalities should be carried out as quickly as possible;
- (b) Internal inspection of packages containing radioactive material should be waived if the nature and description of the goods can be established to the satisfaction of the customs authorities and if there is no suspicion of fraud; and
- (c) Any necessary internal inspection should be carried out at places where adequate facilities are available and radiation protection precautions can be implemented by qualified persons.

A-483.2. This advice is of importance, not only to limit radiation exposure to customs officers, but also to eliminate delays in customs clearance which may decrease the usefulness of valuable radioactive material. This applies in particular to material for medical purposes and to nuclear safeguards samples.

A-483.3. When it is noted that a package has been damaged, the customs official should immediately contact a qualified person, give him the necessary information and follow the instructions that he issues. No person should either remain near the package (a segregation distance of 3 m would generally be sufficient) or touch it unless absolutely necessary.

A-483.4. It is undesirable that inspection of packages of radioactive material beyond visual examination of the package should be made on the premises of the customs office. Opening of any package should only be performed by qualified persons after the consignee, consignor or competent authority has been contacted.

A-483.5. When necessary, packages should be placed for temporary storage in an isolated secure place. During such storage, the segregation distance between the packages and all persons should be as great as practicable. Warning signs should be posted around the package and storage area. Vehicles containing damaged packages which appear to be leaking, or appear to be severely dented or breached, should be detained and secured until they have been declared safe by a qualified person. See also paras A-410.1 and A-478.1 to A-479.

A-483.6. It is desirable that customs officials who regularly deal with packages that contain radioactive material receive some instruction on radiation hazards, radiation protection procedures and procedures to be taken in the event of an accident involving radioactive material. Information, possibly in the form of wall charts, can be made available in appropriate locations indicating the measures to be taken in the event of damage to packages, together with a list of the telephone numbers of qualified persons. Lectures, films and other training material also may be provided for customs officers. Training and instruction should be repeated at regular intervals.

A-483.7. Communications between consignors, consignees, carriers, competent authorities and customs authorities relating to the international transport of radioactive material should be maintained so that customs authorities are aware of:

- (a) Types and numbers of packages of radioactive material which are likely to be encountered in customs examination;
- (b) Qualified persons equipped with radiation measuring instruments who could be contacted in the event customs officers wish to open a package for inspection or in the event of the receipt of a damaged package which might present a radiological hazard; and
- (c) The national and international regulations for the handling, storage and transport of packages of radioactive material.

UNDELIVERABLE PACKAGES

A-484.1. Packages which cannot be delivered for any reason, e.g. because neither consignor nor consignee can be identified, should be placed in temporary storage in an isolated secure place. See also paras A-478.1 to A-479.

A-484.2. If the consignor, the consignee, the competent authority of the country in which the package is found, and the competent authority of the country from which the package originated, can be identified, they should be informed and instructions for further action requested.

SECTION V

REQUIREMENTS FOR RADIOACTIVE MATERIALS AND FOR PACKAGINGS AND PACKAGES

REQUIREMENTS FOR RADIOACTIVE MATERIALS

Requirements for special form radioactive material

A-503.1. The applicant should demonstrate that the solubility of the material evaluated in the leaching test is equal to or greater than that of the actual material to be transported. Results should also be extrapolated if material with reduced radioactivity is used; leaching values should be scaled up to values comparable to the total activity which will be transported. For material enclosed in a sealed capsule these extrapolation and scaling-up difficulties can be avoided by using suitable volumetric leakage assessment techniques such as vacuum bubble or helium leakage test methods (in these cases all test parameters which have an effect on sensitivity need to be thoroughly specified).

A-503.2. The Regulations allow alternative leakage assessment tests for sealed capsules. When by agreement with the competent authority concerned, the performance tests of a capsule design are not performed with radioactive contents the leakage assessment may be made by a volumetric leakage method. A rate of 10^{-5} Pa·m³/s for non-leachable solid contents and a rate of 10^{-7} Pa·m³/s for leachable solids², liquids and gases would in most cases be considered to be equivalent to the activity release of 2 kBq (50 nCi) prescribed in para. 503. Four volumetric leak test methods are recommended as being suitable for detecting leaks in sealed capsules and these are listed in Table A-IV together with their sensitivity.

A-504. Sealed sources which can be opened only by destructive techniques are generally assumed to be those of welded construction. They can be opened only by such methods as machining, sawing, drilling or flame cutting. Capsules with threaded end caps or plugs, for example, which may be opened without destroying the capsule, would not be acceptable.

² The definition of leachable and non-leachable is taken from Ref. [19], i.e. leachable — greater than 0.01% of the total activity soluble in 0.1 L of still water at 20°C in 48 hours.

TABLE A-IV. COMPARISON OF THE FOUR VOLUMETRIC LEAK TEST METHODS RECOMMENDED IN Ref. [20]

Leak test method	Sensitivity (Pa · m ³ /s)	Minimum void in capsule (mm ³)
Vacuum bubble		
(i) Glycol or isopropyl alcohol	10 ⁻⁶	10
(ii) Water	10 ⁻⁵	40
Pressurized bubble with isopropyl alcohol	10 ⁻⁸	10
Liquid nitrogen bubble	10 ⁻⁸	2
Helium pressurization	10 ⁻⁸	10

GENERAL REQUIREMENTS FOR ALL PACKAGINGS AND PACKAGES

A-506.1. In the selection of materials for lifting attachments, consideration should be given to the selection of materials which will not yield under the range of loads expected in normal handling, yet should yield under severe overloads without affecting the safety of the system. In addition, material should be suitable for withstanding friction forces to avoid undue scoring.

A-506.2. For the design of attachment points of packages lifted many times during their lifetime the fatigue behaviour should be taken into account in order to avoid failure cracks. Where fatigue failure may be assumed, the design should take into account the detectability of those cracks by non-destructive means and appropriate tests should be included in the maintenance programme of the package.

A-506.3. Snatch factors for lifting should be related to the expected lifting speeds of cranes and be clearly identified. It should be ensured, by an appropriate safety factor against yield, that there is no plastic deformation under these snatch loads in any part of the package.

A-506.4. Special attention should be given to lifting attachments of packages handled in nuclear facilities. In addition to damage of the package itself, the dropping of heavy mass packages onto sensitive areas could result in radioactive material releases from other sources within the facility or in a criticality or other event which could affect the safety of the facility. For these attachment points even higher safety margins may be required than for normal engineering practice [21-23].

A-508.1. In determining what is practicable as regards the design and finish of packaging, the primary consideration should be not to detract from the effectiveness

of any features which are necessary for compliance with other requirements of the Regulations. For example, features provided for safe handling, operation and stowage should be designed so that while they fulfil their essential functions under the appropriate provisions of the Regulations, any protrusions and potential difficulties of decontamination are minimized.

A-508.2. Cost is also a legitimate determinant of what is practicable. Measures to comply with para. 508 need not involve undue or unreasonable expense. For example, the choice of materials and methods of construction for any given packaging should be guided by commonly accepted good engineering practice for that type of packaging, always having due regard to para. 508, and need not invoke extravagantly expensive measures.

A-508.3. An exterior surface with a smooth finish having low porosity aids decontamination and is inherently less susceptible to absorption of contamination and subsequent leaching-out ('hide-out') than a rougher one.

A-509. For the purposes of compliance with para. 509 exactly similar considerations to those in paras A-508.1 to A-508.3 should be applied.

A-512.1. Chemical compatibility of contents with packaging materials and between different materials of the components of the packagings should take into account such effects as corrosion, embrittlement, accelerated ageing and dissolution of elastomers and elastics, contamination with dissolved material, initiation of polymerization, pyrolysis producing gases, and alterations of a chemical nature.

A-512.2. Compatibility consideration should include those materials which may be left from manufacturing, cleaning or maintaining the packaging, such as cleaning agents, grease, oil, etc., and also should include residuals of former contents of the package.

A-512.3. Consideration of physical compatibility should take into account thermal expansion of materials and contents over the temperature range of concern so as to cover the changes in dimensions, hardness, physical states of materials and contents.

A-512.4. One aspect of physical compatibility is observed in the case of liquid contents, where sufficient ullage must be provided in order to avoid hydraulic failure as a consequence of the different expansion rates of the contents and its containment systems within the admissible temperature range. Void volume values to provide sufficient ullage may be derived from regulations for the transport of other dangerous goods with comparable properties.

A-513. Locks are probably one of the best methods of preventing unauthorized operation of valves; they can be used directly to lock the valve closed or used on a lid or cover which prevents access to the valve. Whilst seals can be used to indicate that the valve has not been used they cannot be relied upon to prevent unauthorized operation.

ADDITIONAL REQUIREMENTS FOR PACKAGES TRANSPORTED BY AIR

A-515. This provision is meant to establish the heat rejection capability of the packaging such that at the given 38°C ambient temperature a differential surface temperature of 12°C is adequate for the internal heat load. If, during transport, the ambient temperature exceeds 38°C (e.g. see para. 516) the limit on accessible surface temperature no longer applies.

A-516. The extremes of ambient temperature that air transport authorities consider might be encountered during air transport are -40°C to +55°C. In designing the integrity of containment under these conditions, care may need to be taken when considering the effect that such extremes may have on resultant surface temperatures, contents, and thermal stresses and pressure variations to ensure adequate containment of the radioactive material.

A-517. For guidance on the no-leakage requirement see paras A-537.1 to A-537.5.

REQUIREMENTS FOR INDUSTRIAL PACKAGES

Requirements for industrial package Type 1 (IP-1)

A-518.1. The advisory information in paras A-506.1 to A-517 applies to all IP-1's; for an IP-1 intended to carry a liquid, the advisory information in paras A-518.2 to A-518.5 applies.

A-518.2. Neither the industrial package design requirements of the Regulations nor United Nations packing group III (see Appendix I) design requirements regard packages as pressure vessels.³ In the event that pressure vessels are used as industrial packages, the design principles of relevant pressure vessel codes should be taken into account for the selection of materials, design/calculation rules, and quality assurance requirements for the manufacturing and use of the package (e.g. pressure testing by independent inspectors). The comparably high wall thickness of pressure vessels is usually foreseen to provide safety with respect to internal service and/or test pressure. A design pressure higher than necessary to cover service conditions corresponding to the vapour pressure at the upper temperature limit may provide a margin of safety against mishaps or even accidents by necessitating a greater thickness of wall. In this case, it may not be necessary to prove safety

³ In this respect, packages can only be pressure vessels if they have a volume of less than 450 L in the case of liquid contents and of less than 1000 L in the case of gaseous contents. Pressure vessels with greater volumes are defined as tanks, for which paras 521 and 522 provide a comparable level of safety.

by drop- and stacking-performance tests, but rather the pressure test could suffice. However, the safety of associated service equipment (valves etc.) against mechanical loads needs to be assured, for example by the use of additional protective structures.

A-518.3. Pressure vessels with volumes less than 450 L for liquid contents and 1000 L for gaseous contents, and designed for a pressure of 265 kPa (see para. 521(b)), may provide an adequate level of safety and consequently may not need to be subjected to the IP-tests. It is understood that all precautions specified by the relevant pressure vessel codes for the use of pressure vessels are taken into consideration and applied as appropriate.

A-518.4. An example for this application is the pressure vessels used for the transport of uranium hexafluoride (UF₆). These cylinders are designed for a pressure much higher than occurs under normal transport and service conditions. Their service equipment therefore is inherently protected against mechanical loads.

A-518.5. The ullage requirement (see para. 538) is not specified as a requirement for the industrial packages. However, in the case of liquid contents, or solid contents such as UF₆ which may become liquid in the event of heating, it may be necessary to provide comparable precautions, as referred to in para. 538, in order to prevent rupture of the containment. Such rupture can occur in the case of insufficient ullage especially as a result of expansion of contents with temperature changes.

Additional requirements for industrial package Type 2 (IP-2)

A-519.1. For guidance on prevention of the loss or dispersal of contents and of the loss of shielding integrity, see paras A-537.1 to A-537.9.

A-519.2. For an IP-2 packaging intended to carry a liquid, see paras A-518.2 to A-518.5. For an IP-2 packaging intended to carry a gas, see paras A-518.2 to A-518.4.

A-519.3. In order to profit from the existence of a multitude of approved packaging designs according to the United Nations Recommendations on the transport of other dangerous goods (see Appendix I), which may also be suitable for the transport of some LSA/SCO materials, packagings of United Nations packing group III are allowed to be used without additional testing. This is because the United Nations Recommendations require general design requirements and performance tests, which in total provide the same level of safety as the Regulations. Leaktightness is one of the performance test criteria in the United Nations Recommendations, but shielding requirements need to be addressed separately since they are not adequate in United Nations package types.

A-519.4. As United Nations packing groups I and II require even more stringent performance test standards, IP-2 requirements are automatically complied with by

all of the United Nations packing groups I, II and III. This means that packagings marked with 'X', 'Y' or 'Z' according to the United Nations system are potentially suitable for the transport of LSA and SCO requiring an IP-2 package when no specific shielding is required. For these packages, determination of maximum relative density/gross mass and maximum total pressure/vapour pressure of contents is necessary.

A-519.5. United Nations packagings of packing group III, i.e. packagings which meet the specifications given in Chapter 9 of the United Nations Recommendations on the Transport of Dangerous Goods (see Appendix I), may be used for IP-2 packages provided the properties of the intended contents are consistent with those allowable in the particular packaging and that no specific shielding is required. The applicable restrictions can be determined from the United Nations marking which must appear on United Nations specification packagings. This marking consists of the following elements, which are required to be in the sequence shown:

- (a) The United Nations packaging symbol $\left(\begin{smallmatrix} u \\ n \end{smallmatrix}\right)$ (or UN for embossed metal packagings);
- (b) A packaging type code (e.g. '1A2' for a steel drum with a removable head);
- (c) 'X' for packing groups I, II and III
'Y' for packing groups II and III
'Z' for packing group III only;
- (d) (i) If intended to contain liquids — the relative density of liquid for which the design has been tested (can be omitted when not more than 1.2 g/cm³);
(ii) If intended to contain solids or inner packagings — the maximum allowable gross mass in kilograms;
- (e) If intended to contain liquids — the test pressure, in kilopascals, of the hydraulic pressure test;
- (f) The last two digits of the year of manufacture (plastic drums and jerricans are also marked with the month of manufacture);
- (g) The distinguishing motor vehicles sign used by the state which authorized the marking; and
- (h) The name or identification mark of the manufacturer.

Additional requirements for industrial package Type 3 (IP-3)

A-520. For an IP-3 intended to carry a liquid, see paras A-518.2 to A-518.5. For an IP-3 intended to carry a gas, see paras A-518.2 to A-518.4.

Alternative requirements for tanks and freight containers to qualify as IP-2 and IP-3

A-521.1. The general design criteria for tank containers with respect to safe handling, stacking and transport can be complied with if the structural equipment (frame) is designed in accordance with ISO 1496/3-1981 [24]. This standard prescribes a structural framework in which the tank is attached in such a manner that all static forces of handling, stowage and transport produce no undue stresses on the shell of the tank.

A-521.2. The dynamic forces under normal conditions of transport are considered in Appendix VII.

A-521.3. Tank containers approved according to The International Convention for Safe Containers (see Appendix I) are considered to be at least equivalent to those designed to ISO 1496/3-1981 [24], except in regard to any shielding requirement.

A-521.4. The shielding retention requirement (para. 521(c)) is complied with if after the tests the shielding material remains at its place, shows no significant cracks and prevents no more than a 20% increase in the radiation level as evaluated by calculation and/or measurements under the above mentioned conditions. In the case of tank containers with an ISO framework, the radiation level calculations/measurements may take the surfaces of the framework as the relevant surfaces.

A-523.1. ISO 1496/1-1978 [25] prescribes test criteria which provide the safety in handling, stacking and transport. The framework has to withstand forces in all three directions which are given as a function of the total mass. Other requirements provide the necessary protection of the floor and roof of the container against concentrated loads (e.g. forklift forces). (Note: the ISO Standard referred to in the Regulations has been updated and is now ISO 1496/1-1984 [26].)

A-523.2. Freight containers approved according to The International Convention for Safe Containers (see Appendix I) are considered to be at least equivalent to those designed to ISO 1496/1-1978 [25], except in regard to any shielding requirement. (See note in para. A-523.1 concerning the updated Standard.)

A-523.3. The dynamic forces under normal conditions of transport are considered in Appendix VII.

A-523.4. In the case of freight containers constructed of concrete the requirements should be complied with when the corner fittings are designed to withstand the forces specified in ISO 1496/1-1978 [25] and if, in addition, all concrete parts are designed and manufactured in accordance with the requirements for concrete buildings/constructions accepted in the country of origin and if the loss of shielding after the tests implies not more than 20% increase of the radiation level at any

external surface of the freight container. (See note in para. A-523.1 concerning the updated Standard.)

A-523.5. For interpretation of the shielding retention requirement, see para. A-521.4.

REQUIREMENTS FOR TYPE A PACKAGES

A-526.1. The type and mass of the package will, in the main, dictate the type of security seal to be used, but designers should ensure that the method chosen is such that it will not be impaired during normal handling of the package in transport.

A-526.2. There are many methods of sealing but the following are typical of those used on packages of radioactive materials.

- (a) When the packaging is a fibreboard carton, gummed or self-adhesive tape which cannot be re-used to seal the package may be used (the outer packaging and/or the tape will be effectively destroyed on being opened).
- (b) Crimped metal seals may be used on the closures of drums, lead and steel pots and small boxes. The seals are crimped onto the ends of a suitable lace or locking wire and are embossed with an identifying pattern. The method used to secure the closure itself should be independent of the security seal.
- (c) Padlocks may be used on timber boxes and also for steel or lead/steel packages. A feature such as a drilled pillar is incorporated into the box or packaging design so that when the padlock is fitted through the drilled hole it is not possible to gain entry into the package.

A-527.1. It is possible to design tie-down systems which do not impair the packaging for normal conditions of transport where acceleration values are known (see para. A-462). This may not apply for accident conditions where acceleration values are difficult to define. Tie-down systems should therefore be designed to fail before the failure of any structural component of the package which is essential to the containment or to the shielding of the content. Consideration can also be given to directionally preferred failures so the transport workers are protected in the event of head-on impacts while the package is protected against excessive side loads for side-on impacts [27].

A-527.2. With the exception of such Type A packages as freight containers or tanks, the tie-down to the vehicle will in general be a standard piece of equipment suitable for restraining packages which have a considerable variation in mass. It is therefore sensible to design the attachment of the tie-down to the package as the weak link; this can be done by designing the attachment point so that it will only accommodate a certain maximum size shackle pin, be held by pins that would shear, or bolts that would break, at the appropriate stress.

A-527.3. Lifting points may be used as tie-down attachments but if so used they should be designed specifically for both tasks. The lifting points and tie-down attachments should be clearly marked to indicate the specific purposes unless they can be so designed that alternative use is impossible, e.g. a hook type lifting point cannot normally be used for tie-down.

A-528. Type A package components should be designed for a temperature range from -40°C to $+70^{\circ}\text{C}$ corresponding to possible ambient temperatures within a vehicle or other enclosure or package temperatures when the package is exposed to direct sunlight. This range covers the conditions likely to be encountered in transport and storage operations. If a wider environmental temperature range is likely to be encountered during transport or handling or as a result of internal heat generation, then this may need to be allowed for in the design. Some of the items that may need consideration are:

- expansion/contraction of components relative to structural or sealing functions;
- decomposition or changes of state of component materials at extreme conditions;
- tensile/ductile properties and package strength; and
- shielding design.

A-530. Examples of positive fastening devices which may be suitable are:

- welded seams
- screw threads
- snap-fit lids
- crimping
- rolling
- peening
- heat shrunk materials, and
- adhesive tapes or glues.

Other methods may be appropriate depending on the package design.

A-533. Certain materials may react chemically or radiolytically with some of the substances intended to be carried in Type A packages. Tests may be required to determine the suitability of materials to ensure that the containment system is not susceptible to deterioration in use.

A-534. For guidance on the requirement on retention of radioactive contents, see paras A-537.1 to A-537.5.

A-535. Examples of enclosures which may be suitable are:

- (a) Blank caps on threaded valves using gaskets;
- (b) Blank flanges on flanged valves using gaskets; and

- (c) Specially designed valve covers or enclosures, using gaskets, designed to retain any leakage.

Other methods may be appropriate depending on the package design.

A-536. Examples of design features which may be suitable are:

- (a) Hinge operated interlock devices on covers;
- (b) Bolted, welded or padlocked frames surrounding the radiation shield; and
- (c) Threaded shielding plugs.

Other methods may be appropriate depending on the package design.

A-537.1. A maximum allowable leakage rate for normal transport of Type A packages has never been defined quantitatively in the Regulations but has always been required in a practical sense.

A-537.2 The intent of paras 517, 519, 537(a) and 540 is to ensure that under normal transport conditions the radioactive content of the package cannot escape in sufficient quantities to create a radiological or contamination hazard.

A-537.3. Practically, it is difficult to advise on a single test method that could satisfactorily incorporate the vast array of packagings and their contents that exist. A qualitative approach, dependent upon the packaging under consideration and its radioactive contents, may be employed.

A-537.4. For solid, granular and liquid contents, one way of satisfying the requirements for 'no loss or dispersal' would be to monitor the package (containing a non-active, control material) on completion of a vacuum test or other appropriate tests to determine visually whether any of the contents have escaped. For liquids, an absorbent material may be used as a test indicator. For liquids and powders it may be necessary for some package designs to use a pressure differential of at least 20 kPa in order to facilitate detection. Thereafter, a careful visual inspection of the package may confirm that its integrity is maintained and no leakage has occurred. Another method which may be suitable in some cases would be to weigh the package before and after a vacuum test to determine whether any leakage has occurred.

A-537.5. For gaseous contents, visual monitoring is unlikely to be satisfactory and a suction detection or pressurization method with a readily identifiable gas (or volatile liquid providing a gaseous presence) may be used. It may be necessary to apply a minimum differential pressure of 20 kPa to facilitate detection. Again, a careful visual inspection of the packaging may confirm that its integrity has been maintained and no escape paths exist. Another detection method would be a simple bubble test.

A-537.6. For packages exhibiting little external deformation and negligible internal movement of the radioactive contents or shielding, a careful visual examination may provide sufficient assurance that the surface radiation level is essentially unchanged.

A-537.7. If it is considered that a surface radiation level has probably increased, it is necessary to perform monitoring tests to ensure that the radiation level increase does not exceed 20%.

A-537.8. The method of evaluating the loss of shielding varies from one manufacturer to another. This could lead to discrepancies in evaluating a package's ability to satisfy the requirements of para. 537(b). One way of overcoming this problem may be to define the maximum surface area of the package over which the surface radiation level is assessed. Thus, for example, individual measurements may be taken over areas not greater than 10% of the total surface area of the package. The package surface may be marked to define the subdivisions to be considered and tests conducted by means of a test source suitable for the package (i.e. cobalt-60 or sodium-24 for general package use or specific nuclides for a certain package design). It may be necessary to consider the effect of increased localized radiation levels when evaluating shielding loss. Where other conditions are not specified, the procedures outlined in A-433.2 may be followed.

A-537.9. The loss of shielding should be evaluated on the basis of the measurements taken both before and after the tests specified in para. 537 and the resulting data should be compared to determine whether the package satisfies the requirement or not.

A-538. When designing ullage requirements it may be necessary to consider both extremes of package material temperature, -40°C and $+70^{\circ}\text{C}$ (see para. 528). At the lower temperature, pressure increases may occur as a result of expansion at transitional temperatures where the material changes its state from liquid to solid. At the higher temperature, pressure increases may occur as a result of expansion or vaporization of the liquid contents. Consideration may also be needed to ensure that excessive ullage is not provided as this may allow unacceptable dynamic surges within the package during transport. In addition, surging or lapping may occur during filling operations involving large liquid quantities and designers may need to consider this aspect for certain package designs.

A-540. For guidance on the requirement of no loss or dispersal of gaseous radioactive contents, see para. A-537.5.

REQUIREMENTS FOR TYPE B PACKAGES

A-542.1. In a package design, maximum radiation levels are established both at the surfaces (para. 434) and at 1 m from the surfaces of the package (as implied by para. 432 coupled with para. 428). After the tests for accident conditions have been performed, however, an increase in the radiation level is allowed provided the limit of $10\ \mu\text{Sv/h}$ ($1000\ \text{mrem/h}$) at 1 m from the surface is not exceeded when the package is loaded with its maximum allowed activity.

A-542.2. When shielding is required for a Type B package design, the shielding may consist of a variety of materials, some of which may be lost during the tests for accident conditions. This is acceptable provided the radioactive contents remain in the package and sufficient shielding is retained to ensure that the radiation level at 1 m from the 'new' (after test) external surface of the package does not exceed 10 mSv/h (1000 mrem/h).

A-542.3. The demonstration of compliance with this acceptance criterion of not more than 10 mSv/h (1000 mrem/h) at 1 m from the external surface of a Type B package after the applicable tests may be made by different means: calculations, tests on models, parts or components of the package, tests on prototypes, etc., or by a combination of them. In verifying compliance, attention should be paid to the potential for increased localized radiation levels emanating through cracks or gaps which could appear as a defect of design or manufacturing or could occur during the tests as a consequence of the mechanical or thermal stresses, particularly in drains, vents and lids.

A-542.4. When the verification of compliance is based on full scale testing, the evaluation of the loss of shielding may be made by putting a suitable radioactive source into the specimen and monitoring entirely the outside surface with an appropriate detector, for instance films, Geiger-Müller probes or scintillation probes. For thick shields a scintillation probe, e.g. NaI (thallium activated), of small diameter (about 50 mm) is usually employed because it allows the use of low activity sources, typically cobalt-60, and because its high sensitivity and small effective diameter permits an easy and effective detection of increased localized radiation levels. If measurements are made near the surface of the packaging, care must be taken to properly measure (see para. A-433.1) the radiation level and to average the results (see para. A-433.2). Calculations will then be needed to adjust the measured radiation level to 1 m from the external surface of the package. Finally, unless the radioactive contents for which the package is designed are used in the test, further calculations will be required to adjust the measured values to those which would have existed had the design contents been used.

A-542.5. The use of lead as a shielding material needs special care. It has a low melting temperature and high coefficient of expansion and, therefore, it should be protected from the effects of the thermal test. If it is contained in relatively thin steel cladding which could be breached in the impact test and if the lead melts in the fire it would escape from the package. Also, owing to its high coefficient of expansion the lead could burst the cladding in the thermal test and be lost. In both these cases the radiation level could be excessive after the thermal test. To overcome the expansion problem voids might be left to allow the lead to expand into them but it should be recognized that, when the lead cools, a void will exist whose position may be difficult to predict. A further problem is that uniform melting of the lead may not

necessarily occur, owing to non-uniformities in packaging structure and in the fire environment. In this event, localized expansion could result in the cladding being breached and the subsequent loss of lead, thus reducing the shielding capability of the package.

A-542.6. Additional guidance on testing the integrity of radiation shieldings may be found in Refs [28–33].

A-543.1. While para. 528 required that Type A packaging materials be evaluated for temperatures up to 70°C, para. 543 requires similar considerations be made for Type B packages for the temperature/insolation conditions specified in paras 545 and 546. In this case, the heat generated by the contents may be such that the maximum package temperature may be considerably in excess of 70°C.

A-543.2. See also paras A-528, A-545, A-546.1 to A-546.8 and A-556.2 and Appendix IX.

A-543.3. Practical tests may be used to determine the package's internal and external temperatures under normal conditions by simulating the heat source due to radioactive decay of the contents with electrical heaters. In this way, the heat source can be controlled and measured. Such tests should be performed in a uniform and steady thermal environment (i.e. fairly constant ambient temperature, still air, and minimum heat input from external sources such as sunlight). The package with its heat source should be held under test for sufficient time to allow the temperatures of interest to reach steady state. The test ambient temperature and internal heat source should be measured and used to adjust linearly all measured package temperatures to those corresponding to a 38°C ambient temperature.

A-543.4. For tests performed in uncontrolled environments (e.g. outside) ambient variations (e.g. diurnal) may make it impossible to achieve constant steady state temperatures. In such cases, the periodic quasi-steady state temperatures should be measured (both ambient and package) and from these correlations may be made between ambient and package average temperatures. These results, together with data on the internal heat source, can be used to predict package temperatures corresponding to a steady 38°C ambient temperature.

A-545. When analysis is used to determine temperatures within a package for transport of radioactive materials under normal conditions, a steady ambient temperature of 38°C is used. Heat is transferred from the package surface to the environment by convection and radiation.

A-546.1. The insolation data provided in Table XII of the Regulations are uniform heat fluxes. They are to be applied at the levels stated for 12 hours (daylight) followed by 12 hours of no insolation (nighttime). The cyclic step functions representing insolation should be applied until the temperatures of interest reach conditions of steady periodic behaviour.

A-546.2. A simple but conservative approach for evaluating the effects of insolation is to apply uniform heat flux continuously at the values stated in Table XII. Use of this approach avoids the need to perform transient thermal analysis; only a simple steady state analysis is performed.

A-546.3. For a more precise model, a time dependent sinusoidal heat flux may be used to represent insolation during daylight hours for flat surfaces or for curved surfaces. The integrated (total) heat input to a surface between sunrise and sunset is required to be equal to the appropriate value of total heat for the table values over 12 hours (i.e. multiply the table value by 12 hours to get total heat input in $W \cdot h/m^2$). The period between sunset and sunrise gives zero heat flux for this model. The cyclic insolation model should be applied until the temperatures of interest reach conditions of quasi-steady periodic behaviour.

A-546.4. Downward facing flat surfaces cannot receive any insolation, and the Table XII value of 'none' applies. For any upward facing horizontal surface, the Table XII value is applicable. Non-horizontal surfaces may include vertical or nearly vertical surfaces (i.e. up to 15° off the vertical); for these surfaces, the Table XII value for vertical surfaces applies. For upward tilted flat surfaces that are more than 15° off the vertical, the horizontal projection of the area may be used in conjunction with the insolation value for a flat upward facing horizontal surface. For downward tilted flat surfaces that are more than 15° off the vertical, the vertical projection of the area may be used in conjunction with the insolation value for a flat vertical surface.

A-546.5. The insolation value for curved surfaces given in Table XII should be applied to all curved surfaces of any orientation.

A-546.6. Components of the package that reduce insolation to any surface (i.e. provide solar shade to the surface of the package) may be taken into account in the thermal evaluation. Any such components assumed to reduce insolation should not be included in the thermal evaluation if their effectiveness would be reduced as a result of the package being subjected to the tests for normal conditions of transport.

A-546.7. Because radiation heat transfer depends on the emissivity and absorptivity at a surface, variations in these properties may be taken into account. These surface properties are wavelength dependent. Solar radiation corresponds to high temperature and short wavelength radiation while surface radiation from packages corresponds to relatively low temperature and longer wavelength radiation. In many cases, the absorptivity will be lower than the emissivity, so using the higher value for both will give a larger margin of safety when the objective is heat dissipation. In other cases, advantage might be taken of naturally occurring differences in these properties, or the surface could be treated to take advantage of such differences to reduce the effect of insolation. When differences in surface properties are used as

a means of thermal protection to reduce insolation effects, the performance of the thermal protection system should be demonstrated, and the system should be shown to remain intact under normal conditions of transport.

A-546.8. Evaluation of the package temperature for transport of radioactive material may be done by analysis or test. Tests if used should be performed on full scale models. If the radiation source is not sunlight, differences between solar wavelength and the source wavelength should be taken into account. The test should continue until thermal equilibrium is achieved (either constant steady state or periodic quasi-steady state depending on the source). Corrections should be made for ambient temperatures and internal heat, where necessary.

A-547.1. In general, coatings for thermal protection fall into two groups, those which undergo a chemical change in the presence of heat (e.g. ablative and intumescent) and those which provide a fixed insulation barrier (including ceramic materials).

A-547.2. Both groups are susceptible to mechanical damage. Materials of the ablative and intumescent type are soft and can be damaged by sliding against rough surfaces (such as concrete or gravel) or by the movement of hard objects against them. In contrast, ceramic materials are very hard, but are usually brittle and unable to absorb shock without cracking or fracturing.

A-547.3. Commonly occurring incidents, which could cause damage to thermal protection materials include: relative movement between package and contact surfaces of vehicle during transport; skidding across a road in whose surface gravel is embedded; sliding over a damaged rail track or against the edge of a metal member; lifting or lowering against bolt heads of adjacent structure or equipment; impact of other packages (not necessarily containing radioactive material) during stowage or transport; and many other situations which would not result from the tests required in paras 622 to 627. Packages that are tested by a simple drop test do not receive damage to the surface representative of the rolling and sliding action usually associated with a vehicle accident and packages subsequently thermally tested may have a coating which under practical accident conditions could be damaged.

A-547.4. The damage to a thermal protection coating may reduce the effectiveness of the coating, at least over part of the surface. The packaging designer should assess the effects of this kind of damage.

A-547.5. The effects of age and environmental conditions on the protective material need to also be taken into account. The properties of some materials change with time, and with temperature, humidity or other such conditions.

A-547.6. A coating may be protected by adding skids or buffers which would prevent sliding or rubbing against the material. A hard outer skin of metal or an over-pack would give good protection but could alter the thermal performance of the

package. In some cases, the external surface of the package could be designed so that thermal protection can be applied within recesses.

A-547.7. With the agreement of the competent authority, thermal tests with arbitrary damage to the thermal protection of a package may be made, to show the effectiveness of damaged thermal protection.

A-548.1. The Regulations require Type B packages to be designed to restrict loss of radioactive contents to an acceptable low level. This is specified as a permitted release of radioactivity expressed as a fraction of A_2 per unit time for normal and accident conditions of transport. These criteria have the advantage of expressing the desired containment performance in terms of the parameter of primary interest: the potential hazard of the particular radionuclide in the package. The disadvantage of this method is that direct measurement is generally impractical and it is required to be applied to each individual radionuclide in question in the physical and chemical form which is expected after the mechanical, thermal and water immersion tests. It is more practical to use well established leakage testing methods such as gas leakage tests. In general leakage tests measure material flow passing a containment boundary. The flow may contain a tracer material such as a gas, liquid, powder or the actual or surrogate contents. A means should therefore be determined to correlate the measured flow with the radioactive material leakage expected under the reference conditions. This radioactive material leakage can then be compared with the maximum radioactive material leakage rate that is permitted by the Regulations. If the tracer material is a gas, the leakage rate expressed as a mass flow rate can be determined. If the tracer material is a liquid, either the leakage rate, expressed as a volumetric flow rate, or the total leakage expressed as a volume can be determined. If the tracer material is a powder, the total leakage, expressed as a mass, can be determined. Finally if the tracer material is radioactive, the leakage expressed as an activity can be determined. Volumetric flow rates for liquids and mass flow rates for gases can be calculated by the use of established equations. If powder leakage is calculated by assuming that the powder behaves as a liquid or an aerosol, the result will be very conservative.

A-548.2. The basic calculative method therefore involves the knowledge of two parameters, the radioactive concentration of the contents of the package and the volumetric leak rate of the packaging.

A-548.3. The product of these two parameters should be less than the maximum permitted radioactivity leakage rate expressed as a fraction of A_2 per unit time.

A-548.4. For packages containing radioactive materials in liquid or gaseous form the concentration of the radioactivity is to be determined in order to convert Bq/h (activity leak rate) to m^3/s (volumetric leak rate) under equivalent transport conditions. When the contents include mixtures of radionuclides (R1, R2, R3 etc.) the 'unity rule' specified in para. 304 is used as follows:

$$\frac{\text{Potential release of R1}}{\text{Allowable release of R1}} + \frac{\text{Potential release of R2}}{\text{Allowable release of R2}} + \dots + \frac{\text{Potential release of Rn}}{\text{Allowable release of Rn}} \leq 1$$

A-548.5. From this, and assuming uniform leakage rates over the time intervals being considered, the radioactivity of the gas or liquid in the package and the volumetric leakage rate are required to fulfil the following:

For the conditions in para. 548(a)

$$\sum_i \frac{C_{(Ri)}}{A_{2(Ri)}} \leq \frac{10^{-6}}{3600L} = \frac{2.78 \times 10^{-10}}{L} \quad (1)$$

For the conditions in para. 548(b)

$$\sum_i \frac{C_{(Ri)}}{A_{2(Ri)}} \leq \frac{1}{7 \times 24 \times 3600L} = \frac{1.65 \times 10^{-6}}{L} \quad (2)$$

where $C_{(Ri)}$ is the concentration of each nuclide in TBq/m³ of liquid or gas at STP;

$A_{2(Ri)}$ is the limit specified in Table I in TBq for that nuclide;

L is the permitted leak rate in m³/s of liquid or gas at STP.

The quantity C can also be derived as follows:

$$C = GS \quad (3)$$

where G is the concentration of the nuclide in kg/m³ of liquid or gas at STP;

S is the specific activity of the nuclide in TBq/kg of the pure nuclide (see Appendix III)

$$\text{or } C = F\bar{G}S \quad (4)$$

where F is the fraction of the nuclide present in an element (percentage/100);

\bar{G} is the concentration of the element in kg/m³ of liquid or gas at STP.

A-548.6. Note that the allowable activity release after tests for normal conditions of transport is given in terms of TBq/h and after tests for accident conditions in terms of TBq/week. It is unlikely that any leakage after an accident will be at a uniform rate. The value of interest is the total leakage during the week and not the rate at any time during the week (i.e. the package may leak at a high rate for a short period of time following exposure to the accident environment and then release essentially

nothing for the remainder of the week as long as the total release does not exceed A_2 per week).

A-548.7. The calculated permitted radioactive liquid or gas leakage may then be converted to an equivalent test gas leakage under reference conditions, taking account of pressure, temperature and viscosity by means of the equations for laminar and/or molecular flow conditions, examples of which are given in American National Standard ANSI N14.5-1977 [13]. In particular cases where a high differential pressure may result in a high gas velocity, turbulent flow may be the more limiting and should be taken into account.

A-548.8. The test gas leakage determined by the above method may range from about $1 \text{ Pa m}^3/\text{s}$ to less than $10^{-10} \text{ Pa m}^3/\text{s}$, depending upon the A_2 values of the radionuclides and their concentration in the package. Generally in practice, a test need not be more sensitive than $10^{-8} \text{ Pa m}^3/\text{s}$ for a pressure difference of $1 \times 10^5 \text{ Pa}$ to qualify a package as being leaktight. Where the estimated allowable test leakage rate exceeds $10^{-2} \text{ Pa m}^3/\text{s}$, a limiting value of $10^{-2} \text{ Pa m}^3/\text{s}$ is recommended because it is readily achievable in practical cases.

A-548.9. When a package is designed to carry solid particulate material, test data on the transmission of solids through discrete leak paths or seals can be used to establish test gas conditions. This will generally give a higher allowed volumetric leak rate than by assuming the particulate material behaves as a liquid or an aerosol. In practice even the smallest particle size powder would not be expected to leak through a seal which has been tested with helium to better than $10^{-6} \text{ Pa m}^3/\text{s}$ with a pressure difference of $1 \times 10^5 \text{ Pa}$.

A-548.10. Packages designed for the transport of irradiated fuel pose a particular problem in that the radioactivity is concentrated as fission products in fuel pins which have been sealed prior to irradiation. Pins which were intact on loading into the package would generally be expected to retain this activity under normal conditions of transport.

A-548.11. Under accident conditions irradiated fuel pins may fail with subsequent release of activity into the package containment system. Data on the fuel fission product inventory, possible failure rate of pin cladding and the mechanism of activity transfer from the failed pin into the containment system are therefore required to enable the package leaktightness to be assessed.

A-548.12. The above methods of assessing the leaktightness requirements of packages are generally applied in two ways:

- (a) When the package is designed for a specific function, the radioactive contents are clearly defined and the standard of leaktightness can be established at the design stage.

- (b) When an existing package with a known standard of leaktightness is required to be used for a purpose other than that for which it was designed, the maximum allowable radioactive material contents has to be determined.

A-548.13. Another alternative is the method of calculating an 'effective' A_2 as appropriate for mixtures when considering leakage from Type B packages. In this case, $f(i)$ (the fraction of activity of nuclide i in the mixture as defined in para. 304) is the fraction in a form which can leak through the seals. Thus, it is not necessarily the fraction within the package itself since part of the contents may be in solid discrete pieces too large to pass through seal gaps. In general, for leakage of liquids and gases the fractional quantities relate to the gaseous or dissolved radionuclides. Care is necessary, however, to take account of finely divided suspended solid material.

A-548.14. If the package has elastomeric seals, permeation of gases or vapours may cause relatively high leakage rates. Permeation is the passage of a liquid or gas through a solid barrier (which has no direct leak paths) by an absorption-diffusion process. Where the radioactive material is gaseous (e.g. fission gas) the rate of permeation leakage is determined by the partial pressure of the gas and not by the pressure in the containment system. The tendency of elastomeric materials to absorb gases can also be taken into account.

A-548.15. It should be noted that, in the case of some large packages, very small leakage of activity over a long time period could result in contamination of the exterior surface. In these cases it may be necessary to reduce the leakage (para. 548(a)) to ensure that the surface contamination limit (paras 122, 408 and 409) is not exceeded.

Requirements for Type B(U) packages

A-550.1 The Regulations have been framed with the intention of ensuring that certain packages of irradiated fuel do not suffer significant damage from submersion at depths of up to 200 m to facilitate recovery work. Under such circumstances the gross retention of the containment together with the shielding of packages is necessary.

A-550.2. In many cases of B(U) package design, the need to meet other sections of the Regulations will result in a containment system which is unimpaired by immersion in 200 m of water.

A-550.3. In cases in which the containment efficiency is impaired, it is recognized that leakage into the package and subsequent leakage from the package is possible.

A-550.4. The aim under conditions of an impaired containment should be to ensure that only dissolved activity is released. Retention of solid radioactive material in the package reduces the problems in salvaging the package.

A-550.5. Degradation of the total containment system could occur with prolonged immersion and the recommendations made in the above paragraphs should be considered as being applicable for immersion periods of about 1 year.

A-553.1. Type B(U) packages are generally not pressure vessels and do not fit tidily within the various codes and regulations which cover such vessels. For the tests required to verify the ability of a Type B(U) package to withstand both normal and accident conditions of transport, assessment under the condition of maximum normal operating pressure (MNOP) is required. Under normal transport conditions the prime design considerations are to provide adequate shielding and to restrict the leakage of radioactivity under quite modest internal pressures. The accident situation represents a single extreme incident following which re-use is not considered as a design objective. Such an extreme incident is characterized by single short duration, high stress cycles during the mechanical tests at normal operating temperature, followed by a single, long duration stress cycle induced by the temperatures and pressures created during the thermal test. Neither of these stressing cycles fit the typical pattern of loading of pressure vessels, the design of which is concerned with time dependent degradation processes such as creep, fatigue, crack growth and corrosion. For that reason, specific reference to the allowable stress levels has not been included in the Regulations. Instead, strains in the containment system are restricted to values which will not affect its ability to meet the applicable requirements. Whilst other requirements might eventually assume importance, it is for the containment of radioactivity that the containment system exists. Before a fracture would occur it is likely that containment systems, particularly in re-usable packagings with mechanically sealed joints, will leak. It is of prime importance therefore to determine the extent to which the strains in the various components distort the containment system and impair its sealing integrity. Reduction of seal compression brought about, for example, by bolt extensions and local damage due to impact and by rotations of seal faces during thermal transients need to be assessed. One assessment technique is to predict the distortions on impact directly from drop tests on representative scale models and to combine these with the distortions calculated to arise during the thermal test using a recognized and validated computer code. The effects upon sealing integrity of the total distortion may then determined by experiments on representative sealed joints with appropriately reduced seal compressions.

A-553.2. The MNOP should be determined in accordance with the definition given in para. 132.

A-553.3. It is recommended that the strains in a containment system under normal conditions of transport at maximum normal operating pressure should be within the

elastic range. The strains under accident conditions of transport should not exceed the strains which would allow leakage rates greater than those stated in para. 548(b), nor increase the external surface radiation level beyond the requirements of para. 542.

A-553.4. When a package or package prototype model is tested, the test item should be pressurized to the MNOP prior to testing. If justifiable, pressures different from the MNOP may be used provided the results are adjusted to reflect the MNOP.

A-553.5. When analysis is used to evaluate package performance, the MNOP should be used as the basis for the calculations.

A-555.1. For a Type B(U) package transported under exclusive use, the maximum permitted temperature of readily accessible surfaces is 85°C. Readily accessible surface is not a precise description, but is interpreted here to mean those surfaces which could be casually contacted by a person who may not be associated with the transport operation. For example, the use of a ladder might make surfaces accessible, but this would not be cause for considering the surfaces as readily accessible. In the same sense, surfaces between closely spaced fins would not be regarded as readily accessible. If fins are widely spaced, say the width of a person's hand or more, then the surface between the fins could be regarded as readily accessible.

A-555.2. Barriers or screens may be used to give protection against higher surface temperatures and still retain the Type B(U) approval category. An example would be a closely finned package fitted with lifting trunnions where the use of the trunnions would require the fins to be cut away locally to the trunnions and thus expose the main body of the package as an accessible surface. Protection may be achieved by the use of a barrier, such as an expanded metal screen. The use of barriers or screens should not impair the ability of the package to meet heat transfer requirements nor reduce its safety. Such a screen or other device is not required to survive the regulatory tests for the package design to be approved. This provision permits approval of packages using such thermal barriers without the barriers having to be subjected to the tests which the package is required to withstand.

A-556.1. See paras A-528 and A-545 and Appendix IX.

A-556.2. The heating effect of the radioactive contents (which could prevent the package temperature falling to the ambient of -40°C) is usually ignored. Adequate performance of the package should not depend on some minimum internal heat load.

Requirements for Type B(M) packages

A-558.1. Intermittent venting is permitted in order to allow a package to be relieved of a buildup of pressure which might, under normal conditions of transport (see

paras 619–624) or when the package is subjected to the thermal test (see para. 628), cause it to fail to meet the Regulations. Activity release under normal conditions and under accident conditions, where no operational controls are used, is limited, however, by the provisions of para. 548.

A-558.2. Because there is no specified regulatory limit of activity release for intermittent venting, where operational controls are used the person responsible should be able to demonstrate to the competent authority, using a model which relates as closely as possible to the actual conditions of package venting, that transport workers and members of the public will not be exposed to doses in excess of those laid down by the relevant national authorities. When the intermittent venting operation is taking place under the control of a radiation protection adviser, the release may be varied on his advice taking account of measurements made during the operation to assure that transport workers and members of the public are adequately protected.

A-558.3. Factors taken into account in such an assessment will include:

- (a) Exposure due to normal activity leakage and external radiation from the package;
- (b) The location and orientation of the venting orifice in relation to the working position of the operator and the proximity of workers and members of the public;
- (c) Occupancy factors of transport workers and members of the public;
- (d) The physical and chemical nature of the material being vented, e.g. gaseous (halogen, inert gas, etc.), particulate, soluble/insoluble; and
- (e) Other dose commitments incurred by operators and the public.

A-558.4. In assessing the adequacy of the release operation, account should be taken of possible detriment from retaining and disposing of the released activity rather than allowing it to disperse.

REQUIREMENTS FOR PACKAGES CONTAINING FISSILE MATERIAL

A-561. Section IV of the Regulations has its relevance to fissile materials not only in delineating responsibilities and requirements, but also in prescribing the system for implementing criticality control in transport. It contains the method for assigning a transport index to a package containing fissile material (see paras 428 to 430 in conjunction with para. 567). The transport index establishes the character of the package, informing carriers how the package is to be handled, stowed, and stored in transit. A transport index of zero indicates that from the criticality safety standpoint there is no limit to the number of packages which may be grouped together. A transport index greater than zero designates packages which require separation in transport and in in-transit storage to provide not only criticality control but radiation

exposure control. The paragraphs of Section IV relevant to fissile material along with paras 559–568 are the bases for criticality control in transport.

A-562.1. Criticality safety assessments are described in Appendix X.

A-562.2. Subparas 562(a)–562(f) detail the contingencies to be examined. These are occurrences a package may experience while in transit. The form of the fissile material, containment and packaging materials, and the immediate environment of the package define both the neutron distribution and the neutron energy distribution in the system. The contingencies are examples of events which may influence the energy and distribution of neutrons. The effect could result in an increase or a decrease in the neutron multiplication factor of the system.

A-562.3. Water, commonly encountered in transport, influences criticality safety in several ways. When it is mixed with fissile material the resulting neutron moderation can significantly reduce the amount of material required to achieve criticality. As a reflector of neutrons, water also increases the neutron multiplication factor, though less dramatically. If the water reflector is located outside the containment system it is less effective, and less still outside the package. Thick layers of water (approx. 15 cm) between packages can reduce neutron interaction in arrays to an insignificant value.

A-562.4. Neutron absorbers are sometimes employed in the packaging to reduce the effect of moderation and the contribution to the neutron multiplication factor resulting from interaction among packages. A neutron absorber is more effective for neutrons of low energy; thus, it may be used in combination with a neutron moderator (subpara. 562(b)). The loss of effectiveness of neutron absorbers, e.g. by corrosion and redistribution, or as in the case of contained powders, by settling, can have a marked effect on the neutron multiplication factor.

A-562.5. Subparas 562(c) and 562(d) address contingencies which could result in an increase in the neutron multiplication factor of a package. For this reason, the rearrangement of the contents must be considered in establishing the margin of sub-criticality. A loss of the fissile material from the package, if credible, must be limited to a subcritical quantity under unspecified conditions. This usually will imply the minimum quantity required for criticality consistent with the type of fissile material. The reduction of spaces between packages, credible because of possible damage to the package in transport, will have a direct effect on the neutron interaction among packages; thus, it requires examination.

A-562.6. The effects of temperature changes (subpara. 562(f)) on the stability of fissile material form or on the neutron multiplication factor are to be examined. Temperature changes may also influence the package integrity. The temperatures which should be considered include those resulting from ambient condition requirements specified in paras 515, 545 and 556, as applicable; package material and

package surface temperature requirements specified in paras 515, 528, 544 and 555, as applicable; and temperatures resulting from the thermal test specified in para. 628 (as required by para. 564(b)).

A-563.1. These specific requirements for package design directly affect the criticality assessment of packages in normal transport. The prevention of entry of a 10 cm cube, subpara 563(a), originally was of concern when open, 'bird-cage' types of packages were permitted. This requirement now provides a criterion for evaluating the integrity of the outer container of the package; the 5% reduction in spacing limitation should be viewed in the same way. Water is again considered in subpara. 563(b) because of its effectiveness in altering the neutron energy spectrum and, hence, the neutron multiplication factor. This subparagraph provides guidance to the designer on the selection of the level of containment that may be needed.

A-563.2. Subpara. 563(c) provides guidance to the designer concerning the type of constraints required to prevent a significant change in the neutron multiplication factor, which may result from a rearrangement of the contents. The interpretation given to the word 'significantly' is ultimately a matter of judgement on the part of the analyst, as is the acceptability of a margin of subcriticality. Generally, the package loading is selected to result in an acceptable neutron multiplication factor. Should the package be subjected to the specified contingencies, the resulting increases in the multiplication factor, taken singly or in credible summation, will not exceed the accepted margin of subcriticality.

Individual packages in isolation

A-566.1. The neutron balance between capture, fission and leakage might be adversely affected by damage to the fissile material in an accident (para. 566(a)). For example, damage to fuel pins in an LWR fuel transport container might result in a limited redistribution of oxide material with a consequent increase in the neutron multiplication factor. The arrangement of fissile material in the damaged case is potentially complex and the use of sophisticated computer codes in the criticality safety evaluation may be necessary. These are capable of realistic representation of a package in its undamaged and damaged states (paras 566(a), 567). It may well be cost effective to employ such codes throughout the analysis in order to avoid undue restrictions, particularly if the package is in frequent and widespread use. The performance of several computer codes is described in two reports, one dealing with the criticality assessment of LWR fuel transport containers [34] and the other concerned with calculations of arrays of packaged and unpackaged material [35]. Derivation of the transport index might involve the calculation of large finite arrays for which, as is identified in the second report, there is an unfortunate lack of experimental data. In these circumstances, it was recommended that a 'specific supplementary allowance' be made in addition to other margins usually allowed for

random and systematic effects on calculated values of the neutron multiplication factor.

A-566.2. The variety of configurations possible for fissile material escaping from the containment system and the possibility of subsequent chemical or physical changes (para. 566(b)) requires that limitation be evaluated on the basis of the minimum critical mass. Allowance may be taken of material properties which will not be subject to change. 'Unchanged' means unaltered by exposure to the environments to which the package is subjected during the test sequence.

Subcriticality evaluation assumptions

A-568.1. The subcriticality evaluation assumptions deal with irradiated and unspecified fissile materials. Irradiation of fissile material depletes the fissile nuclide content and produces fission products which contribute to neutron absorption. As may be expected, the neutron multiplication factor for irradiated fissile material usually decreases with irradiation time. Subpara. 568(a) permits use of the reduced multiplication factor, provided the irradiation experience is known. When the experience is not known the condition producing the highest attainable neutron multiplication factor must be assumed (para. 568(b)). This is usually the unirradiated state. However, one method of extending the useful residence time of fissile material is to add a distributed burnable poison, allowing a larger fissile nuclide content than would otherwise be present. Since the burnable poison is depleted along with the fissile nuclide, it may be that the neutron multiplication factor will reach a maximum at an intermediate exposure.

A-568.2. Para. 568(c) permits those nuclear parameters that are known for the material to be used. Values of unknown parameters should be appropriately selected to produce the maximum neutron multiplication factor under credible conditions of transport. 'Credible conditions' is used here to imply that parameters significant to the criticality safety assessment are limited to ranges consistent with the conditions of the fissile material and packaging in transport. Typical unspecified fissile materials are generated in process operations, e.g. contaminated work clothes, gloves, or tools, residues of chemical analyses and operations, floor sweepings, etc.

SECTION VI

TEST PROCEDURES

DEMONSTRATION OF COMPLIANCE

A-601.1. It is generally not advisable to perform the tests required in Section VI of the Regulations using the radioactive material which is being evaluated to qualify as either LSA-III or special form radioactive material, or which would be the contents of special form radioactive material capsules or of packages. When determining whether the radioactive material or the intended radioactive contents are to be used in the tests, a radiological safety assessment should be made, consistent with Ref. [6].

A-601.2. In addition to full scale tests of a package, reference to previous demonstrations of a sufficiently similar nature, scale model tests, calculations, reasoned arguments, or combinations thereof may be used to demonstrate compliance. The decision concerning the method of demonstrating compliance should be based upon all factors involved, complexity of package design and phenomena requiring investigation, availability of facilities, and ability to accurately measure and/or scale responses. Simulated contents of packaging for full scale tests should be carefully chosen to represent the relevant physical characteristics of the actual contents.

A-601.3. When considering reference to previously satisfactory demonstrations of a similar nature, it is necessary to consider all the similarities and the differences between two packages. The areas of difference may require modification of the results of the demonstration. The ways and extent to which the differences and similarities will qualify the results from the previous demonstration depend upon their effects. In an extreme case, a packaging may be geometrically identical to an approved packaging but because of material changes in the new packaging, the reference to the previous demonstration would not be relevant and could not be used.

A-601.4. When scale models are used certain test parameters cannot be adjusted. For example, both time and gravitational acceleration are real, and therefore it will be necessary to adjust the results by use of scaling factors.

A-601.5. When scale models are used to determine damage, due consideration should be given to the mechanisms which cause the distortion. This is because damage due to elasticity, plasticity and instability may have different scale factors as a result of different parameters in the test being affected. Also, because the demonstration of compliance requires the combination of three tests (penetration, drop and thermal tests), conflicting requirements for the test parameters may require a compromise, which in turn would give results requiring scale factoring. In summary, the effect of scaling for all areas of difference should be considered.

A-601.6. Experience has shown that the testing of scale models may be very useful for demonstrating compliance with certain specific requirements of the Regulations, particularly the mechanical tests. Attempts to perform thermal tests using scale models are not recommended (see para. A-628.14). In mechanical tests, the conditions of similitude are relatively simple to create, provided the same materials and suitable methods of construction are used for the model as for the full sized package. Thus, in an economical manner, it is possible to study the relation of package orientation and the resulting damage, and the overall deformation of the package, and to obtain information concerning the deceleration of package parts. In addition, many design features can be optimized by model testing.

A-601.7. The details which should be included in the model are a matter of judgement and depend on the type of test for which the model is intended. For example, in the determination of the structural response from an end impact, the omission of lateral cooling fins from the scale model may result in more severe damage. This type of consideration may greatly simplify construction of the model without detracting from its validity. Only pertinent structural features which may influence the outcome of the test need be included. It is essential, however, that the materials of construction for the scale model and the full sized package are the same and that suitable construction and manufacturing techniques are used. In this sense, the construction and manufacturing techniques which will replicate the mechanical behaviour and structural response of the full sized package should be used, giving consideration to such processes as machining, welding, heat treatment and bonding methods. The stress-strain characteristics of the construction materials should not be strain rate dependent to a point which would invalidate the model results. This point needs to be made in view of the fact that strain rates in the model may be higher than in the full sized package.

A-601.8. In some cases it may not be practical to scale all components of the package precisely. For example, consider the thickness of an impact limiter compared to the overall length of the package. In the model, the ratio of the thickness to the overall length may differ from that of the actual package. When any appreciable geometrical discrepancy exists between the actual package and the model to be tested, the behaviour of both when subjected to the 9 m drop should be compared by computer code analyses to determine whether the effect of geometrical discrepancy is a significant consideration. The computer code employed should be a code which has been verified through appropriate bench mark tests. If the effects of the discrepancies are not significant, the model would be considered suitable for a scale model drop test. This applies to a scale ratio of 1:4 or greater.

A-601.9. The scale factor chosen for the model is another area where a judgement needs to be made since the choice of scale factor depends on the accuracy necessary to ensure an acceptable model representation. The greater the deviation from full scale, the greater the error that is introduced. Consequently, the reduction of scale

might be greater for a study of package deformation as a whole than for testing certain parts of the package and in some cases the scale factor chosen may be determined by the particular type of test being undertaken. In some tests, such as the penetration tests specified in the Regulations, it will also be necessary to scale the bar in order to produce accurate results. In other cases where the packaging may be protected by a significant thickness of deformable structure, the drop height may need to be scaled.

A-601.10. In general, the scale ratio M (the ratio model dimension:prototype dimension) should be not less than 1:4. For a model with a scale ratio of 1:4 or larger, the effect of strain rate dependence on the material mechanical properties will be negligibly small.

A-601.11. Scaling of drop tests is possible, taking into account the limitations given below, as a result of the following model laws, which are valid when the original drop height is maintained:

$$\begin{array}{ll} \text{Accelerations:} & a_{\text{model}} = 1/M \cdot a_{\text{original}} \\ \text{Forces:} & F_{\text{model}} = M^2 \cdot F_{\text{original}} \\ \text{Stresses:} & \sigma_{\text{model}} = \sigma_{\text{original}} \\ \text{Strains:} & \epsilon_{\text{model}} = \epsilon_{\text{original}} \end{array}$$

A-601.12. For lightweight models, the model attitude or velocity during drop testing could be affected by such things as the swing of an umbilical cord carrying wires for acceleration sensors or strain gauges, or by wind effects. Experience suggests that, for packages with mass up to 1000 kg, full scale models should be used for the test, or special guides should be used.

A-601.13. When acceleration sensors are used to evaluate impact behaviour of the package, the cut-off frequency should be considered. The cut-off frequency should be selected to suit the structure (shape and dimension) of the package. Experience suggests that, for a package with a mass of 100 Mg with impact limiter, the cut-off frequency should be 100 to 200 Hz, and that, for smaller packages with a mass of m [Mg], this cut-off frequency should be multiplied by a factor $(100/m)^{1/3}$.

A-601.14. When an application for approval of a package design is based to any extent on scale model testing, the application should include a demonstration of the validity of the scaling methods used. In particular, such a demonstration should include:

- definition of the scale factor;
- demonstration that the model constructed reproduces sufficiently accurately the details of the package or packaging parts to be tested;
- a list of parts or features not reproduced in the model;
- justification for deletion of parts or features in the model; and
- justification of the similitude criteria used.

A-601.15. Care has to be exercised when planning the instrumentation and analysis of either a scale model test or a full scale test. It is necessary to ensure that adequate and correctly calibrated instrumentation and test devices are provided so that the test results may be documented and evaluated in order to verify the test results. At the same time, it is necessary to ensure that the instrumentation, test devices, and electrical connections do not interfere with the model in a way that would invalidate the test results.

A-601.16. In the evaluation of the results of a scale model test, it is necessary to consider not only the damage sustained by the packaging, but, in some cases, the damage to the package contents. In particular, damage to the package contents should be considered when it involves a change in:

- release rate potential;
- parameters affecting criticality;
- shielding effectiveness;
- thermal behaviour.

A-601.17. It might be difficult to extrapolate the results of scale model testing involving seals and sealing surfaces to the responses expected in a full sized package. Although it is possible to acquire valuable information on the deformation and displacement of sealing surfaces with scale models, extrapolation of seal performance and leakage should be approached with caution (A-617.7 to A-617.10). When scale models are used for testing seals it is necessary to consider the possible effect of such factors as surface roughness, seal behaviour as a function of material thickness and type, and the problems associated with predicting leakage rates on the basis of scale model results.

A-601.18. Brittle fracture in testing is treated in Appendix IX.

A-601.19. Calculation is permitted in the demonstration of compliance (see para. 601(d)) when the procedures and parameters are reliable and conservative. This will require the examination of any engineering data used. Material properties in specifications are usually supplied to give a probability of not being under strength of between 95 % to 98 %. When tests are used for determining material property data, scatter in the data should be taken into account. It is usual to factor results where the number of tests are limited to give a limit of the mean plus twice the standard deviation on a normal (Gaussian) distribution (approximately 95 % probability). It is also necessary to consider scatter due to material and manufacturing tolerances unless all calculations are on the worst combination of possible dimensions.

A-601.20. Many calculations will require the use of commercially available computer codes. Prior to their use, two items should be considered as to their reliability. First, is the code applicable for the intended calculation? For example, for

mechanical assessments, can it accept impact calculations, is it suitable for calculating plastic as well as elastic deformations? Secondly, does the computer code adequately represent the packaging under review for the purpose of compliance? To satisfy these two requirements it will, in many cases, be necessary for the user to run 'bench mark' problems, which uses the code to model and calculate the parameters of a problem in which the results are known. For other uses, checks that the input and output balance in load or energy may be required. When the code used is not widely employed and known, proof of the theoretical correctness should also be given.

A-601.21. In the presentation of reasoned argument, care is required that argument is based on engineering experience. Where theory is used, due account should be made of design details which could modify the result of general theory, e.g. discontinuities, asymmetries, irregular geometry, inhomogeneities and variable material properties. The presentation of reasoned argument is a difficult method because of its subjective nature. It should always be based on scientific fact and be supported by engineering experience.

A-602.1. Any post-test assessment method used to assure compliance should incorporate the following techniques as appropriate to the type of package under examination:

- visual examination;
- assessment of distortion;
- seal gap measurements of all closures;
- non-destructive testing; and
- microscopic examination of damaged material.

A-602.2. In the evaluation of damage to a package after a drop test, all damage from secondary impacts should be considered as well. Secondary impact includes all additional impacts between the package and target, following initial impact. For evaluations which are based on numerical methods, it is also necessary to consider secondary impacts. Accordingly, the attitude of the package which produces maximum damage has to be determined with secondary as well as initial impacts taken into account. However, experience suggests that the effect of secondary impact is usually negligible except for a limited number of slender and rigid packages, including:

- a package with an aspect ratio (length to diameter) larger than 5;
- a rigid, large package when significant rebound is expected to occur following the 9 m drop; and
- a package in which the contents are rigid and slender and particularly vulnerable to lateral impacts.

A-602.3. See also the references given in Appendix IX.

TESTS FOR SPECIAL FORM RADIOACTIVE MATERIAL

General

A-604. The tests of a capsule design may be performed with simulated radioactive material. The term 'simulated' means a facsimile of a radioactive sealed source, the capsule of which has the same construction and is made with exactly the same materials as those of the sealed source that it represents but contains, in place of the radioactive material, a substance with mechanical, physical and chemical properties as close as possible to those of the radioactive material and containing radioactive material of tracer quantities only. The tracer should be in a form soluble in a solvent which does not attack the capsule. One procedure described in Ref. [19] utilizes either 2 MBq of strontium-90 and yttrium-90 as soluble salt, or 1 MBq of cobalt-60 as soluble salt. When possible, shorter lived nuclides should be used. However, if leaching assessment techniques are used, care needs to be taken when interpreting the results. The effects of scaling will have to be introduced, the importance of which will depend upon the maximum activity to be contained within the capsule, its physical form, and in particular its solubility compared with the tracer radio-nuclide. These problems can be avoided if volumetric leakage tests are used (see para. A-503.2).

A-606. Alternative volumetric leakage tests are fully discussed in para. A-503.2.

Test methods

A-608. Special attention should be paid to the percussion test conditions in order to get maximum damage.

A-611.1. The alternative test proposed in para. 611(a) is the ISO Impact Class 4 test which consists of the following: A hammer, with a mass of 2 kg, the flat striking surface having a diameter of 55 mm, with its edge rounded to a radius of 3 mm, is allowed to drop on the specimen from a height of 1 m; the specimen is placed on a steel anvil which shall have a mass of at least 20 kg. The anvil is required to be rigidly mounted and have a flat surface large enough to take the whole of the specimen. This test may be employed in place of both the impact test (para. 607) and the percussion test (para. 608).

A-611.2. The alternative test proposed in para. 611(b) is the ISO Temperature Class 6 test which consists of subjecting the specimen to a minimum temperature of -40°C for 20 minutes and heating over a period not exceeding 70 minutes from ambient to 800°C ; the specimen is then held at 800°C for 1 hour followed by thermal shock treatment in water at 20°C . This test is more severe than the test it replaces.

Leaching and volumetric leakage assessment methods

A-613. See para. A-503.2.

TESTS FOR PACKAGES

Preparation of a specimen for testing

A-615. Since, in certain cases, components forming a containment system may be assembled in different ways, it is essential for test purposes that the specimen and the method of assembly be clearly defined.

Testing the integrity of the containment system and shielding

A-617.1. In order to establish the performance of specimens which have been subjected to the tests specified in paras 619–633 it may be necessary to undertake an investigation programme involving both inspection and further subsidiary testing. Generally, the first step will be a visual examination of the specimen and recording by photography. In addition, other inspections may be necessary. If the tests were performed with specimens containing radioactive trace materials, wipe tests may give a measurement of the leakage. Leaktightness may be detected following the procedures outlined in paras A-537.1 to A-537.5 (IP, Type A, Type B). Likewise, the shielding integrity may be evaluated by the use of trace radiation materials placed inside the packaging. After examination of the outer integrity, the containment system should be disassembled to check the interior situation: integrity of capsules, glass, flasks, etc.; stability of geometrical compartments, particularly in the case where the intended contents are fissile material; distribution of absorbent material; stability of shielding; and function of mechanical parts. The investigatory programme should be aimed at examining three specific areas:

- integrity of the containment system;
- integrity of shielding;
- assurance, where applicable, that no rearrangement of the fissile contents or degree of moderation has adversely influenced the assumptions and predictions of the criticality assessment.

A-617.2. The integrity of the containment system can be evaluated in many ways. For example, the activity release from the containment system can be calculated on the basis of the volumetric (i.e. gaseous) release.

A-617.3. In the case of test specimens which are representative of full size containment systems, direct leakage measurements can be made on the test specimen.

A-617.4. The two following areas need attention:

- the performance of the normal closure system; and
- the leakage levels which may have occurred elsewhere in the containment system.

A-617.5. For small packages, Refs [36, 37] describe two examples of immersion test procedures, both of which relate to volumetric release. The first utilizes internal air pressure and has a practical sensitivity of about 10^{-4} Pa m³/s; the second utilizes a bath of ethylene glycol and a reduced external pressure, and has a practical sensitivity of about 10^{-6} Pa m³/s. Several minutes of observation may be required when the points of possible leakage are not fully exposed, since the escape of gas might not be visible immediately. Conversely, care needs to be taken since previously entrapped air external to the containment system could escape and be mistaken for leakage.

A-617.6. Containment, in accordance with the Regulations, involves so many variables that a single standard test procedure is not feasible. In the American National Standard N14.5-1977 [13], types of tests which are acceptable, listed in order of increasing sensitivity under usual conditions, include but are not limited to:

- gas pressure drop
- water immersion bubble or soap bubble
- gas pressure rise
- vacuum air bubble
- halogen detector
- helium mass spectrometer.

This standard

- relates the regulatory requirements for radioactive material containment to practical detectable mass flow leakage rates;
- defines the term 'leaktight' in terms of a volumetric flow rate;
- makes some simplifying, conservative assumptions so that many of the variables may be consolidated;
- describes a release test procedure; and
- describes specific volumetric leakage tests.

A-617.7. In the event that specimens less than full size have been used for test purposes, direct measurement of leakage past seals is not advisable as not all parameters associated with leakage past seals are readily scalable. In this instance, because loss of sealing is often associated with loss of seal compression resulting from, for example, permanent extension of the closure cover bolts, it is recommended that a detailed metrology survey be made to establish the extent to which bolt extension and distortion of the sealing faces has occurred on the test specimen following the mechanical tests. The data based on a detailed metrology survey can

be scaled and the equivalent distortion and bolt extension at full size determined. From tests with full size seals using the scaled metrology data the performance of the full size package can be determined.

A-617.8. For evaluating shielding integrity, attention is drawn to the fact that if a radioactive source is to be used to establish the post accident test condition, any damage or modification to the post-test package configuration caused by the insertion of the source might invalidate the results obtained (see Ref. [28]).

A-617.9. In the event that a full size specimen has been used for testing, one method of proving the integrity of the shielding is as follows:

- (a) With a suitable source inside the specimen, the entire surface of the specimen should be examined with an X-ray film or an appropriate instrument to determine whether there has been a loss of shielding.
- (b) If there is evidence of loss of shielding at any point on the surface of the specimen, the radiation level should be determined by actual measurement and calculation to ensure compliance with paras 537 and 542. For additional information, refer to paras A-537.6 to A-537.9 and paras A-542.3 to A-542.4.

A-617.10. Alternatively, a careful dimensional survey could be made of those parameters which contribute to shielding performance to ascertain that they have not been adversely affected, e.g. slumping or loss of lead from shields, giving rise to either a general increase in radiation or increased localized radiation levels.

A-617.11. A change in the geometry of the package contents or a change in the effective moderation may occur as a result of the tests, and this could change the validity of the considerations used in evaluating nuclear criticality safety. In this event, a re-evaluation of criticality may be necessary.

A-617.12. It should be noted that although the testing of the package at full or smaller scale can be carried out with simulated contents from which some data on the behaviour of any basket or skip used for positioning the contents can be obtained, the final geometry will in practice depend upon the interaction of the actual material (whose mechanical properties may be different from the simulated contents) with both the basket or skip and the other components of the packaging.

Target for drop test

A-618. One example of an unyielding target to meet the regulatory requirements is a steel plate as the upper surface of a concrete block. The combined mass of the steel and concrete should be at least 10 times that of the specimen to be dropped on it. The block should be set on firm soil and the steel plate should be at least 4.0 cm thick and floated onto the concrete while it is still wet. The plate should have protruding fixed steel structures on its lower surface to ensure tight contact with the

concrete. Since flexure of the target is to be avoided, especially in the vertical direction, it is recommended that the target should be close to cubic in form, with the depth of the target comparable to the width and length. Other targets which have been used are described in Refs [38, 39].

Tests for demonstrating ability to withstand normal conditions of transport

A-621. One method of performing the water spray test which is considered to satisfy the conditions prescribed in para. 621 is as follows:

- (a) The specimen is placed on or above a flat horizontal surface, in whichever orientation is likely to cause most damage to the package. A uniformly distributed spray is directed onto the surface of the package for a period of 15 minutes from each of four directions at right angles and changes in spray direction should be made as rapidly as possible.
- (b) The following additional test conditions are recommended for consideration:
 - (i) A spray cone apex angle sufficient to envelop the entire specimen at the distance employed in (ii);
 - (ii) A distance from the nozzle to the nearest point on the specimen of at least 3 m;
 - (iii) A water consumption equivalent to the specified rainfall rate of 5 cm/h, as averaged over the area of the spray cone at the point of impingement on the specimen and normal to the centre line of the spray cone;
 - (iv) Water draining away as quickly as delivered;
 - (v) The specimen supported so that it does not sit in a pool of water.
- (c) The requirement of para. 621 is intended to provide maximum surface wetting and this may be accomplished by directing the spray downwards at an angle of 45° from the horizontal:
 - (i) for rectangular specimens, the spray may be directed at each of the four corners;
 - (ii) for cylindrical specimens standing on one plane face, the spray may be applied from each of four directions at intervals of 90°.

A-622.1. As the number of possibilities of loading associated with normal transport is infinite, an enveloping drop test may be used to prove the strength of any package dropped. It is required to drop the packaging in such a way that would give the maximum damage for the feature being tested. It is usual to test all the various features of the package which are susceptible to impact damage.

A-622.2. The safety features to be tested depend on the type of the package to be tested. The safety features are those intended to meet the requirements of para. 519 for industrial packages, of para. 537 for Type A packages, and of paras 537, 542,

543 and 548 for Type B packages. However, since Type B packages are subjected to tests for accident conditions which give more severe impacts to the packages (para. 627), this paragraph is primarily intended to address safety features which pertain to the requirements of paras 519 and 537. Such safety features include structural components, materials and devices designed to prevent loss or dispersal of radioactive substances or loss of shielding materials, e.g. the entire containment system, such as lids, valves (and their seals), shielding and structural materials. For packages containing fissile materials, the safety features include, in addition to those mentioned above, components for maintaining subcriticality, such as a fuel holding frame and neutron absorbers.

A-622.3. The 'maximum damage' is the maximum impairment of the integrity of the package. To produce the 'maximum damage' for most packages, it is sufficient to drop the specimen in such a way that the impact acceleration and/or deformation of a safety feature under consideration is maximized. This means that the attitude to produce maximum damage should be determined. The geometric surface, the material and the structure need to be considered. A uniform spherical container could be dropped in any attitude, but most containers have some asymmetry giving different resistances to impact. In any investigation, sufficient structural elements should be considered to allow for the absorption of all the kinetic energy of the package. Arguments may be developed as to the damage in the various elements between the impact point and the main concentration of mass with regard to their performance in absorbing the energy, in developing internal loads, in distorting, collapsing or folding, and in the consequences of these behaviours.

A-622.4. For most packages, it is sufficient to drop the specimen onto the target in such attitude that the weakest component, in terms of structural strength, directly hits the target to generate the maximum impact acceleration and/or deformation. In this way, the safety feature being tested would sustain the maximum damage. In the selection of the attitude for directly hitting the weakest component, attitudes for reducing the performance of the structure when that structure is vital to the mechanical or subcriticality performance of the package should be selected.

A-622.5. Packages of low mass may be hand held above the target and dropped. As soon as the mass of the package is too great for a person to hold, test facilities need to be used. These may be simply a release mechanism suspended from an overhead structure, like a roof member or a crane, or a tower specially designed for drop tests. The design of dedicated drop facilities is as wide as the parameters to be tested, but all consist of four elements: the support, the release, the track guide (usually not used in direct drops), and the target which is defined in para. 618. Sufficient height is required to be provided in the support to allow for the release mechanism, the support cable or harness and the full depth of the test item and still make it possible to attain the correct attitude and dropping height between the bottom of the package

and the target. The release mechanism for a free drop test should allow easy setting and instantaneous release. Care has to be taken that the release mechanism does not give undesirable effects on the attitude of the specimen or produce any damage to it. Various types of mechanisms, such as mechanical, electromagnetic, or combinations of these, may be used. A number of test facilities are described in Ref. [40].

A-622.6. Scale model techniques may be useful in order to determine the most damaging drop attitude (see paras A-601.4 to A-601.17). Care should be taken in instrumentation since mounts and sensor frequencies may produce errors in the data obtained.

A-623.1. Any package whose normal top, i.e. the side opposite the one which it normally rests on, is parallel and flat, may be stacked. In addition, stacking may be achieved by adding feet, extension pads or frames to the package. In general, packages with convex surfaces cannot be stacked unless extension pads or feet are provided; however, stacking tests of these packages are required.

A-623.2. The specimen should be placed with the base down on an essentially flat surface such as a flat concrete floor or steel plate. If necessary, a flat plate, which has sufficient area to cover the upper surface of the specimen, should be placed on the upper surface of the specimen so that the load may be applied uniformly to it. The mass of the plate should be included in the total stacking mass being applied. If a number of packages of the same kind are stackable, a simple method is to build a stack of five packages on top of the test specimen. Alternatively, a steel plate or plates or other convenient materials with a mass five times that of the package may be placed on the package.

Tests for demonstrating ability to withstand accident conditions in transport

A-627.1. Paras A-627.2 to A-627.15 provide advice and suggest techniques for satisfying the mechanical test requirements. Additional information is provided in paras A-601.1 to A-601.21. Because of their physical characteristics, most packages will be subject to the 9 m drop (impact) test rather than the crush test. The crush test is only applicable to those packages that have a mass of 500 kg or less, have an overall density (that is to say the mass divided by the total volume of the package) of 1000 kg/m³ or less, contain more than 1000 A₂, and have radioactive contents which are not classified as special form.

A-627.2. The Regulations require that the attitudes of the package for both the impact (drop I), penetration (drop II) and crush (drop III) tests be such as to produce maximum damage taking into account the thermal test. In addition, the order in which the tests are carried out is that which will be most damaging. The assessment

of maximum damage should be made with concern for the containment of the radioactive material within the package, the retention of shielding to keep external radiation to the acceptable level and, in case of fissile materials, maintenance of subcriticality. Any damage which would give rise to increased radiation or loss of containment after the thermal test also needs to be considered. Damage which may render the package inappropriate for re-use but does not affect its ability to meet the safety requirements is not a reason for classifying the specimen as having failed.

A-627.3. Different modes of damage are possible as a result of the mechanical tests. It is necessary to consider the results of these modes for any analytical assessment to demonstrate compliance with the applicable requirements. The fracture of a critical component or the breach of the containment system may allow the escape of the radioactive material. Deformation may impair the function of radiation or thermal shields and it may alter the configuration of fissile material so that the assumptions and predictions in the criticality assessment are no longer valid. Local damage to shielding may, as a result of the subsequent thermal test, give rise to deterioration of both the thermal and radiation protection. Consequently, investigations should include stress, strain, instability and local effect for all attitudes of drop where symmetry does not prevail.

A-627.4. Owing to the possibility of damage during a test, multiple drops of a specimen are sometimes not feasible. The use of analysis or reasoned argument based on engineering data to predict the most damaging attitude may be acceptable. The geometry and construction of a package may define the most severe orientation.

A-627.5. The most severe attitudes for symmetric packagings that have either a cylindrical or cubic form may often be determined by the use of published information [40,41]. Asymmetries, especially where protrusions occur, are often sensitive when used as the impact point. Lifting and handling devices such as skids or attachment points will often have a different strength or stiffness relative to the adjacent parts of the package and should be considered as possible impact points.

A-627.6. Discontinuities such as lid or other penetration attachments often give a locally stiff element of structure of limited strength which may fail by either adjacent structural deformation or high loading (due to decelerations) on their retained masses.

A-627.7. Thin wall packages, such as drums, should be considered in terms of the possibility of plastic deformation either causing loss of the containment seal or distorting the lid attachment sufficiently to allow the loss of the lid.

A-627.8. Paras 562 and 564 require that, for fissile materials, criticality analyses be made with the damage resulting from the mechanical and thermal test included. Consideration is required of such aspects as efficiency of moderator, loss of neutron absorbers, rearrangement of package contents, geometric changes and temperature

effects (see paras A-562.1 to A-566.2). The assumptions made in the criticality analysis should be in conformity with the effects of the mechanical and thermal tests and all package orientations should be considered for the analysis.

A-627.9. It is intended that the drop of the package (drops I and II) or of the mass (drop III) be a free fall under gravity. If, however, some form of guiding is used, it is important that the impact velocity be at least equal to the impact velocity where the package or the mass is under free fall (approx. 13.4 m/s for drops I and III).

A-627.10. For drop II, the required minimum length of the penetrating bar is 20 cm. A greater bar length should be used when the distance between the outer surface of a package and any inner component of its containment system is greater than 20 cm. The material used in the construction of the bar is stated to be mild steel. It is suggested that the minimum yield stress of such material should not be less than 150 MPa nor more than 280 MPa. The yield to ultimate stress ratio should not be greater than 0.6.

A-627.11. For preliminary design of the outer shell of a steel-lead-steel packaging, the following equation may be used to estimate the shell thickness required to resist failure when the package is subjected to drop II:

$$t = 2148.5 \left(\frac{w}{s} \right)^{0.7}$$

where t is the outer shell thickness (cm)

w is the mass of the package (kg)

s is the tensile strength of the outer shell material (Pa).

This equation is based on tests employing annealed mild steel backed by chemical lead (see Ref. [41]). Packages using materials having physical properties different from the foregoing (different strength and impact resistance of steels, different alloys or backing materials, and packages with low radius of curvature of shell) may require different thicknesses of the outer steel shell to meet the requirements [41].

A-627.12. For the crush test (drop III) the packaging should rest on the target in such a way that it is stable. In order to achieve this it may be necessary to provide support, in which case the presence of the support should not influence the damage to the package. For further information on crush testing of small Type B packages see Ref. [42].

A-627.13. During mechanical testing, it is advised that the packaging does not contain radioactive material.

A-627.14. Instrumentation of test specimens and even of the target response to impact is often recommended for the following reasons:

- validation of assumptions in the safety analysis
- as a basis for design alterations
- as a basis for the design of comparable packages
- as a benchmark test for computer codes.

A-627.15. Examples of functions to be measured under impact/crushing conditions are: deceleration–time function and strain–time function (see para. A-601.13). Where electronic devices are used to acquire, record and store data, examination of any filtering, truncating or cropping should be made so that no data peaks of significance are lost. Most instruments will require cable connections to external devices (i.e. umbilicals). These connections should be such that they neither restrict the free fall of the package nor restrain the package in any way after impact (see para. A-601.12).

A-628.1. Paras A-628.3 to A-628.30 provide advice and suggest techniques pertaining to the thermal test. These are examples of what are generally acceptable to competent authorities; other methods or techniques may be used, but more justification might be expected in support of such an approach. It is important to note that the requirements of the thermal test may be met by a practical test, by a calculated assessment, or by a combination of both. The last approach may be necessary, if, for example, the initial conditions required for a practical test were not achieved or if all the package design features were not fully represented in the experiment. In many cases, the consequences of the thermal test need to be determined by calculation, which therefore becomes an integral part of the planning and execution of the practical test. The Regulations specify certain fire parameters which are essential input data for the calculation method but which are generally uncontrollable parameters in practical tests. Standardization of the practical test is therefore achieved by defining the fuel and test geometry for a pool fire and requiring other practical methods to provide the same or greater heat input.

A-628.2. With regard to the package design, some shielding materials have eutectics with melting temperatures which are lower than the 800°C environment of the thermal test. Therefore consideration should be given to the capability of any structural materials to retain them. Local shielding materials such as plastics, paraffin wax or water may vaporize, causing a pressure which may rupture a shell which may have been weakened by damage from the mechanical tests. A thermal analysis may be required to determine whether such pressures can be attained.

A-628.3. The bottom of the package to be tested is required to be positioned 1 m above the surface of the liquid fuel source, which should be contained with the minimum practical initial freeboard. Unless the fuel is replenished, or replaced by another liquid such as water, the level will fall during the test, probably by about

100 to 200 mm. The specimen package should be supported in such a way that the flow of heat and flames is perturbed by the minimum practical amount. For example, a larger number of small pillars is to be preferred to a single support covering a large area of the package. The transport vehicle, and any other ancillary equipment which might protect the package in practice, should be omitted from this test as the protection was taken into account in the test definition.

A-628.4. The pool size is required to extend between 1 and 3 m beyond the edges of the package. In general, the larger extension should be associated with the largest packages as flame thicknesses will vary more over the greater distances involved. The requirement for fully engulfing flames can be interpreted as a need for all parts of the package to remain invisible throughout the 30 minute test, or at least for a large proportion of the time. This is best achieved by designing for thick flame cover which can accommodate natural variations in thickness without becoming transparent. A low wind velocity (quiescent conditions) is also required for stable flame cover, although large fires might generate high local wind velocities. Wind screens or baffles can help to stabilize the flames, but care should be taken to avoid changing the character of the flames and to avoid reflected or direct radiation from external hot surfaces. This would enhance the heat input and therefore not detract from the acceptability of the test, but could make the test more stringent than necessary.

A-628.5. Wind speeds of less than about 2 m/s should not detract from the test and short duration gusts of higher speeds will not have a large effect on high heat capacity packages particularly if flame cover is maintained. Open air testing should only take place when rain, hail or snow will not occur before the end of the post-fire cool-down period. The package should be mounted with the shortest dimension vertical for the most uniform flame cover, unless a different orientation will lead to a higher heat input or greater damage, in which case such an arrangement should be chosen.

A-628.6. The fuel for a pool fire should comprise a distillate of petroleum with a distillation end point of 330°C maximum and an open cup flash point of 46°C minimum and with a gross heating value of between 46 and 49 MJ/kg. This covers most hydrocarbons derived from petroleum with a density of less than 820 kg/m³, e.g. kerosene and JP4-type fuels. A small amount of more volatile fuel may be used to ignite the pool as this will have an insignificant effect on the total heat input.

A-628.7. The choice of instrumentation will be dictated by the use to be made of a practical thermal test. Where a test involves a fully representative package, it could be performed without instrumentation, relying on post-test inspection. In this case the effects of insolation, ambient temperature and internal heat generation need to be represented or shown to be negligible. However, where a test provides data to be used in calculations to demonstrate compliance, some instrumentation is essential. The type and positioning of the instruments will depend on the data needed, e.g.

internal pressure and temperature measurements may be necessary and, where stress is considered important, strain gauges should be installed. In all cases, the cables carrying signals through the flames should be protected to avoid extraneous voltages created at high temperatures. As an alternative to continuous measurement, the package might be equipped in such a way that instruments could be connected soon after the fire and early enough to measure the peak pressure and temperature. A measurement of leakage can be achieved by pre-pressurization and re-measurement after the thermal test, where necessary making appropriate adjustments for temperature (see para. A-548.1 to A-548.15).

A-628.8. The duration of the test can be controlled by providing a measured supply of fuel calculated to ensure the required 30 minute duration, by removing the supply of fuel a predetermined time before the end of the test, by discharging the fuel from the pool at the end of the test or by carefully extinguishing the fire without affecting the package surfaces with the extinguishing agent. The duration of the test is the time between the achievement of good flame cover and the time at which such cover is lost.

A-628.9. Measurements should continue after the fire, at least until the internal temperatures and pressures are falling. If rain, or other precipitation, occurs during this period a temporary cover should be erected to protect the package and to prevent inadvertent extinguishing of combustion of the package materials, but care should be taken not to restrict heat loss from the package.

A-628.10. Where a fully representative package is subjected to the thermal test, an examination of the package after the test to determine compliance with appropriate requirements (leakage, shielding, etc.) may be sufficient.

A-628.11. Where the test supplies data for analytical evaluation of the package, measurements made during the test should be corrected for non-standard initial conditions of ambient temperature, insolation, internal heat load, pressure, etc. For packages containing liquids, the effects of partial loading, i.e. less than full contents, on the package heat capacity and heat transfer should be assessed.

A-628.12. A furnace test is often more convenient than an open pool fire test. Other possible test environments include pit fires. Any such test is acceptable provided it meets the requirements of para. 628.

A-628.13. Ensuring that the internal temperature increase is not less than that predicted for an 800°C fire ensures that the heat input is satisfactory, but the test should continue for at least 30 minutes, during which the time-averaged environment temperature should be at least 800°C. A high emissivity radiation source should be created by selecting a furnace either with an internal surface area very much larger than the envelope area of the package or with an inherently high emissivity internal surface (0.90 or higher). Many furnaces are unable to reproduce the convective heat

input of a pool fire, so an extension of the test duration might be necessary to compensate. Alternatively, a higher furnace temperature can be used but the test duration should be a minimum of 30 minutes. The furnace wall temperature should be measured at several places, sufficient to show that the average temperature is at least 800°C. The furnace can be pre-heated for a sufficient time to achieve thermal equilibrium, so avoiding a large temperature drop when the package is inserted. The 30 minute minimum duration should be such that the time-averaged environment temperature is at least 800°C.

A-628.14. The prediction of heat transfer into a full size package by radiation, convection and conduction and of phase changes in a thermal environment based on the extrapolation of the results from a test of a scale model is impossible without many different tests. A wide ranging programme simulating each process separately would require an extensive investigation using a theoretical model, so the technique has little inherent advantage over the normal analytical approach. Any scale testing, and the interpretation of the results, should be shown to be technically valid. However, the use of full scale models of parts of the package might be useful if calculation for a component (such as a finned surface) proves difficult. For example, the efficiency of a heat shield, or of a shock absorber acting in this role, could be most readily demonstrated by a test of this component with a relatively simple body beneath it. Component modelling is of importance for the validation of computer models. However, measurements of flame temperature and flame and surface emissivities are difficult and might not provide a sufficiently accurate specification for a validation calculation. Component size should be selected and appropriate insulation provided so that heat entering from the artificial boundaries (i.e. those representing the rest of the package) is not significant.

A-628.15. The most common method of package assessment for the thermal test is calculation. Many general purpose, heat transfer computer codes are available for such package modelling, although care should be taken to ensure that the provisions available in the code, in particular for representing radiation heat transfer from the environment to external surfaces, are adequate for the package geometry. Practical tests may ultimately be required for validation but arguments showing that the approximations or assumptions produce a more stringent test than required are often used. In general, code validation is accomplished by comparison with analytical solutions and comparison with other codes.

A-628.16. Generally, the normal conditions of transport will have been assessed by calculation, so detailed temperature and pressure distributions should be available. Alternatively, the package temperatures might have been measured experimentally, so, after correction to the appropriate ambient temperature, and for the effects of insolation and the heat load due to the contents, these provide the initial conditions for the calculated thermal test conditions. Ambient temperature corrections can be made in accordance with para. A-543.4.

A-628.17. The external boundary conditions of the fire should represent radiation, reflection and convection. The temperature is specified by the Regulations as an average of 800°C so, in general, a uniform average temperature of 800°C will be used for the radiation source and for convective heat transfer.

A-628.18. The flame emissivity is prescribed as 0.9. This can be used without ambiguity for plane surfaces but, for finned surfaces, the thin flames between the fins will have an emissivity much lower than that value. The dominant source of radiation to the finned surfaces will therefore be the flames outside the fins; radiation from flames within the fin cavity can be ignored. In all cases, appropriate geometric view factors should be used with the fin envelope radiation source, and reflected radiation should be taken into account. Care should be taken to avoid the inclusion of radiation 'reflected' from a surface representing flames as this is a non-typical situation.

A-628.19. The surface absorptivity is prescribed as 0.8 unless an alternative value can be established. In practice, demonstration of alternative values will be extremely difficult as surface conditions change in a fire, particularly as a result of sooting, and evidence obtained after a fire may not be relevant. The value of 0.8 is therefore most likely to be used in analytical assessments. It is important to take into account reflected radiation, particularly with complex finned surfaces, as multiple reflections increase the effective absorptivity to near unity. This complexity can be avoided by assuming unity for the surface absorptivity but, even in this case, surface to surface radiation should not be ignored, particularly during the cool-down period.

A-628.20. Convection coefficients during the fire need to be justified. Pool fire gas velocities are generally found to be in the range of 5 to 10 m/s [43]. Use of such velocities in forced convection, heat transfer correlations (e.g. the Colburn relation $Nu = 0.036 Pr^{1/3} Re^{0.8}$ quoted by McAdams [44]) results in convective heat transfer coefficients of about $10 W \cdot m^{-2} \cdot C^{-1}$ for large packages. Natural convection coefficients (about $5 W \cdot m^{-2} \cdot C^{-1}$) are not appropriate as this implies downward gas flow adjacent to the cool package walls, whereas, in practice, a general buoyant upward flow will dominate. The upper surface of a package is unlikely to experience such high gas velocities, in quiescent atmospheric conditions, as the region will include a stagnation area in the lee of the upward gas flow. The reduced convection there is adequately represented by the average coefficient as the averaging process includes this effect.

A-628.21. Convection coefficients for the post-test, cool-down period can be obtained from standard natural convection references, e.g. Ref. [44]. In this case coefficients appropriate for each surface can readily be applied. However, the turbulent natural convection equation

$$Nu = 0.13 (Pr Gr)^{1/3}$$

is probably sufficiently accurate for all surface orientations. Otherwise the boundary conditions used for the assessment of conditions under normal operation should be used. Changes to surface conditions and/or geometry resulting from the fire should be recognized in the post-fire assessment as these might affect both radiation and convection heat losses. Allowance should be made for continued heat input if package components would continue to burn following the thermal test exposure.

A-628.22. The calculations are likely to be performed using finite difference or finite element models of the package. These should have a sufficiently fine mesh or element distribution to represent internal conduction and external and internal boundary conditions, it being recognized that temperature gradients might be required for stress analysis. In particular, external features such as fins will require special attention as temperature gradients can be severe, perhaps requiring separate detailed calculations in order to determine the heat flux to the main body.

A-628.23. Consideration should be given to the choice of one, two or three dimensional models and to the decision whether the whole package or separate parts are to be evaluated. Such decisions require input from experienced heat transfer specialists familiar with the computer codes, or the use of acceptable technical information.

A-628.24. External surfaces of low thermal conductivity can lead to oscillations in computed temperatures. Special techniques (e.g. simplified boundary conditions) or assumptions (e.g. that time-averaged temperatures are sufficiently accurate) might be required to deal with this.

A-628.25. Generally, conduction and radiation can be modelled explicitly and external convection provides few problems for general purpose computer codes but experimental evidence may be required to support modelling assumptions and basic data used to represent internal convection and radiation. Radiation reflection will be important in gas filled packages and insufficient knowledge of thermal emissivities may restrict the final accuracy. A sensitivity study with different emissivities can be used to show that the assumptions are adequate or to provide conservative (i.e. maximum) limits on calculated temperatures.

A-628.26. Internal convection will be important for a water filled package and might be significant in a gas filled package. This process is difficult to predict unless there is experimental evidence to support modelling assumptions. Where water circulation routes are provided, internal heat dissipation will be rapid compared with other time constants and simplifying assumptions can be made (e.g. water can be modelled by an artificial material with a high conductivity). Care should be taken to consider areas not subject to circulation (stagnant regions) as high temperatures can occur there because of the inherently low thermal conductivity of water.

A-628.27. Gas gaps and contact resistances can vary with the differential expansion of components and it is not always clear whether an assumption will yield high or low temperatures. For example, a high resistance gas gap will prevent heat flow, minimizing temperatures inside but maximizing other temperatures because of the reduced effective heat capacity. In such cases calculations based on two extreme assumptions might provide evidence that both conditions are acceptable and, by implication, all variations in between are also acceptable. Seals are rarely represented explicitly, but local temperatures can be used as a close approximation to seal temperature.

A-628.28. The calculation of a thermal test transient should include representation of the initial conditions, 30 minutes with external conditions representing the fire and the cool-down period extended until all temperatures are decreasing with time. In addition, further calculation runs, perhaps with a different mesh distribution, should be performed to check the validity of the model and to assess the uncertainties associated with the modelling assumptions.

A-628.29. The results of the analysis will be used to confirm that the package has adequate strength and that leakage rates will be acceptable. The determination of pressures from calculated temperatures is thus an important step, particularly where the package contains a volatile liquid such as water. Items such as lead shields often may not be allowed to melt as the resulting condition cannot be accurately defined and thus shielding assessments may not be possible. Component temperatures should be examined to ensure that melting will not occur. The whole procedure should recognize the uncertainties in the model, the data (e.g. manufacturing tolerances) and the limitations of the computer codes, and allowances made for these uncertainties.

A-628.30. The post-exposure equilibrium temperatures and pressure might be affected by irreversible changes in the thermal test (perhaps due to protective measures such as the use of expanding coatings or the melting and subsequent relocation of lead within the package). These effects should be assessed.

A-629.1. The water immersion test may be satisfied by immersion of the package, a pressure test, a pressure test on critical components combined with calculations, or by calculations for the whole package.

A-629.2. The specimen is required to be immersed under a head of water for a period of not less than eight hours. All points of the package are required to be at least 15 m from the water surface.

A-629.3. For demonstration purposes, an external gauge pressure of at least 150 kPa is required to be considered to meet the immersion test conditions.

A-629.4. The entire package does not have to be subjected to a pressure test. Critical components such as the seal or lid area may be subjected to an external gauge

pressure of at least 150 kPa and the balance of the structure may be evaluated by calculation.

A-629.5. The package may be evaluated by calculations on the basis of a 150 kPa external pressure. Justification of model assumptions about the response of critical components should be included in the evaluation.

A-629.6. Transport packages may be designed with duplicate seal arrangements. Under these circumstances an inter-seal test at a pressure of at least 150 kPa for a period of eight hours may be used to demonstrate compliance with the 15 m immersion test.

Water immersion test for packages containing irradiated nuclear fuel

A-630.1. The water immersion test may be satisfied by the immersion of the package, a pressure test, a pressure test on critical components combined with calculations, or by calculations for the whole package.

A-630.2. If calculational techniques are adopted it should be noted that established methods are usually intended to define material, properties and geometries which will result in a design capable of withstanding the required pressure loading without any impairment. In the case of the 200 m immersion test requirement some degree of buckling or deformation is acceptable provided the final condition conforms with para. 550 (see also paras A-550.1 to A-550.5).

A-630.3. The entire package does not have to be subjected to a pressure test. Critical components such as the lid area may be subjected to an external gauge pressure of at least 2 MPa and the balance of the structure may be evaluated by calculation.

SECTION VII

APPROVAL AND ADMINISTRATIVE REQUIREMENTS

GENERAL

A-701.1. The Regulations distinguish between cases where the transport can be made without competent authority approval and cases where some kind of approval is required. In both cases the Regulations place primary responsibility for compliance on the consignor and the carrier. In the second case an independent assessment by the competent authority is required, as appropriate, in respect of special form radioactive material, packages containing fissile materials, Type B packages, special arrangements, certain shipments, radiation protection programmes for special use vessels and the calculation of unlisted A_1 and A_2 values.

A-701.2. Regarding the requirement that packages containing fissile material shall require competent authority approval it is noted that para. 560 excludes certain packages from those requirements that apply specifically to fissile material. However, all relevant requirements that apply with regard to the radioactive, non-fissile, properties of the package contents still apply.

A-701.3. The relationship between the competent authority and the applicant has to be clearly understood. It is the applicant's responsibility to 'make the case' to demonstrate compliance with the applicable requirements. The competent authority's responsibility is to judge whether or not the information submitted adequately demonstrates such compliance. It should be free to check statements, calculations and assessments made by the applicant, even, if necessary, by performance of independent calculations or tests. However, the competent authority should not 'make the case' for the applicant, because this would put the competent authority in the difficult position of being both 'advocate' and 'judge'. Nevertheless, this does not prohibit it from providing informal advice to the applicant, without commitment, as to what is likely to be an acceptable way of demonstrating compliance.

A-701.4. Further details of the role of the competent authority can be found in regulations issued nationally or by the international transport organizations (see Appendix I).

A-701.5. The applicant should contact the competent authority during the preliminary design stage to discuss the implementation of the relevant design principles and to establish both the approval procedure and the actions incumbent on him.

A-701.6. Experience has shown that many applicants make their first submission in terms of a specific and immediate need, which is rather narrow in scope, and then later make several requests for amendments to the approval certificate as they

attempt to expand its scope to use the packaging for other types of material and/or shipment. Whenever possible, applicants should be encouraged to make their first submission a general case, which will anticipate and cover their future needs. This will make the 'application–approval' system operate more efficiently and will result in much lower approval costs to the applicant. Additionally, in some cases, it is mutually advantageous for the prospective applicant and the competent authority to discuss a proposed application in outline before it is formally submitted in detail.

A-701.7. In some countries, it has been found to be advantageous to provide a guidance document, such as a check-list, to assist applicants for special form, package, and shipment approvals in submitting the necessary information in a convenient form. Such a guidance document is also of value to the competent authority in that it assists it in evaluating the completeness and accuracy of submissions. An example of such a guide is given below. The guide is not intended to be a substitute for the Regulations. To this end it is not necessary to print the text of a regulatory provision in the guide but to refer the applicant to the Regulations by quoting the relevant paragraph numbers, for example: "Show what the maximum pressure will be within each successive enclosure of the containment system

- (a) prior to shipment: (para. 402(c))
- (b) during normal transport: (paras 132, 554)
- (c) during and subsequent to accident conditions: (para. 553)."

A-701.8. In simple design cases, the guide may be used as a questionnaire, whereas in the more sophisticated designs, e.g. irradiated fuel packages, the submission needs to take the form of a more comprehensive safety report. In the latter case, however, the various sections and paragraphs should make reference to the question in the guide to which they relate. In both cases, the submission should be complete, with relevant extracts from any sources of reference which have been used in demonstrating compliance with the requirements.

A-701.9. It is desirable to divide the guide into parts, each of which refers to a particular aspect or type of approval. This ensures that the applicant uses only that part of the guide which is relevant to the type of approval required. Convenient divisions are:

- PART I — GENERAL INFORMATION
- II — ADMINISTRATIVE INFORMATION
- III — SPECIFICATION OF RADIOACTIVE CONTENTS
- IV — SPECIFICATION OF PACKAGING
- V — PACKAGE ANALYSES AND TESTS
- VI — SHIPMENT
- VII — SPECIAL ARRANGEMENT TRANSPORT OPERATION
- VIII — SPECIAL FORM RADIOACTIVE MATERIAL
- IX — QUALITY ASSURANCE

A-701.10. In general, Parts I, II and IX are always required to be provided by the applicant. In the case of an application for a special arrangement, the applicant should provide, in addition to that in Part VII, the information required in Parts III, IV and V (as applicable), since a necessary prerequisite of establishing compensatory measures is the identification of the package deficiencies.

A-701.11. Suggested items which could be included in the guide are:

Part I — General information

Instructions on the provision of approval information

Reference to related publications (Safety Series No. 6, Safety Series No.37, national standards, etc.)

List of transport regulations

Designs not requiring competent authority approval (excepted, industrial or Type A packages), and procedures to be followed for these designs

Information on the compliance assurance programme

Part II — Administrative information

Name, address, telephone number of applicant

Name, address, telephone number of designer

Name, address, telephone number of manufacturer

Type of approval required (e.g. Type B(U), Type B(M), packages containing fissile material)

Modes of transport

Competent authority identification mark, if previously allocated

Package title and general arrangement drawing number

Date of application

Date by which approval is requested

Part III — Specification of radioactive contents

General nature

Radionuclide

Physical state

Chemical state

Quantity in mass units and enrichment, where applicable (i.e. for packages containing fissile material)

Total activity/specific activity

Calculation of A_1/A_2 values, if not listed in Safety Series No. 6

Nature of emitted radiation

Information on irradiated fuel, e.g. rating, irradiation, initial enrichment and cooling time

Heat output

Hazards other than radioactivity

Part IV — Specification of packaging

Drawings (arrangement, assemblies, subassemblies and details)

Material specifications

Types of closures (e.g. welding)

Overall dimensions and mass

Handling facilities

Tie-down system

Radiation shielding

Neutron absorbers

Containment system

Quality control programme for manufacturing

Maintenance provisions

Actions before shipment

Actions during shipment

Restrictions (including modal restrictions)

Instructions for handling and stowage

Emergency instructions

Part V — Package analyses and tests

Radioactive material behaviour

Effects of radiolyses

Structural evaluation

Containment evaluation

Radiation shielding evaluation

Thermal evaluation, including surface heat flux

Criticality evaluation

Model tests

Prototype tests

Tests with real specimen

Part VI — Shipment

Mode of transport

Consignor

Carrier

Consignee

Consignment details

Exclusive use provisions

Operational controls

Storage in transit

Stowing, handling and lifting

Radiation protection programme

Transport instructions

Emergency instructions

Part VII — Special arrangement transport operation

Mode of transport

Consignor

Consignee

Consignment details

Reason for special arrangement

Proposed compensatory measures

Part VIII — Special form radioactive material

Drawings

Specifications of materials and closures

Overall dimensions and mass

Radionuclide

Physical and chemical state

Nature of emitted radiation

Heat output

Water concentration of contents

Demonstration of compliance with tests

Leakage tests and other quality control measures

Part IX — Quality assurance

Description of the organizations involved

Duties and responsibilities of different organizations

Quality assurance programmes (see Appendices IV and V)

APPROVAL OF SPECIAL FORM RADIOACTIVE MATERIAL

A-702.1. The design of special form radioactive material is required to receive unilateral competent authority approval prior to transport. Para. 702 specifies the minimum information to be included in an application for approval.

A-702.2. Further general advice on the information which may be required to be provided to the competent authority is given in paras A-701.9 to A-701.11. In particular for special form radioactive material all applicable information in para. A-701.11 Parts II, III, VIII and IX may be required.

A-703. Detailed advice on identification marks is given in para. A-724.

APPROVAL OF PACKAGE DESIGNS

Approval of Type B(U) package designs

A-705. General advice on the information which may be required to be provided to the competent authority is given in paras A-701.9 to A-701.11. In particular for Type B(U) package designs, all applicable information in para. A-701.11 Parts II, III, IV, V and IX may be required.

Approval of Type B(M) package designs

A-708.1. General advice on the information which may be required to be provided to the competent authority is given in paras A-701.9 to A-701.11. In particular for Type B(M) package designs, all applicable information in para. A-701.11 Parts II, III, IV, V and IX may be required.

A-708.2. Special attention should be given to stating which of the Type B(U) requirements of paras 545, 546 and 550–556 are not met by the package design. Proposed supplementary operational controls or restrictions (i.e. other than those already required by the Regulations) which are to be applied to compensate for failure to meet the above mentioned requirements should be fully identified, described, and justified. The maximum and minimum ambient conditions of temperature and solar insolation which are expected during transport should be identified and justified with reference to the regions or countries of use and appropriate meteorological data.

A-708.3. Where intermittent venting of Type B(M) packages is required a complete description of the procedures and controls should be submitted to the competent authority for approval. Further advice may be found in paras A-558.1 to A-558.4.

Approval of package designs for fissile material

A-710. A package design for fissile material is required to meet the requirements regarding both the radioactive and fissile properties of the package contents. Regarding the radioactive properties, a package is classified in accordance with the definition of package in para. 134. As applicable, a package design approval based on the radioactive, non-fissile properties of the package contents is required. In addition to such approval, a design approval is required relating to the fissile properties of the package contents. See para. 560 for exceptions regarding requirements on package design approval for fissile material.

A-711. The information provided to the competent authority with the application for approval is required to detail the demonstration of compliance with each require-

ment of paras 562 to 568. In particular, the information should include the items specifically quoted in the competent authority approval certificate as detailed in para. 729(m). The inclusion of appropriate information on any experiments, calculations or reasoned arguments used to demonstrate the subcriticality of the individual package or of arrays of packages is acceptable. Sufficient information should be submitted to permit the competent authority to verify compliance of the package with these regulations.

Approvals under the 1967, 1973 and the 1973 (As Amended) Editions of the Regulations

A-713. Any revision to the original package design, or increase in contained activity, or addition of other types of radioactive materials, which would significantly and detrimentally affect safety, as determined by the competent authority, will require the design to be reassessed according to the 1985 Regulations. This could include such things as an increase in the mass of the contents, changes to the closure, changes to any impact limiters, changes to the thermal protection and shielding and changes in the form of the contents.

A-714. See para. A-713.

NOTIFICATION AND REGISTRATION OF SERIAL NUMBERS

A-715. The packaging serial number should uniquely identify each packaging manufactured. For packagings manufactured to an approved Type B(U) or Type B(M) package design, or for fissile material package design, the appropriate competent authority is to be informed of the serial number. The term 'appropriate' has a broad interpretation and could pertain to any of the following:

- the country where the packaging design originated;
- the country where the packaging was manufactured; or
- the country or countries where the packaging is used.

In the case of packagings approved for continued use under paras 713 and 714, all competent authorities involved in the multilateral approval process should receive information on packaging serial numbers.

APPROVAL OF SHIPMENTS

A-716.1. Each requirement in para. 716 is intended to be applied separately. For example, a consignment of a vented Type B(M) package containing fissile material may need a shipment approval according to both paras 716(a) and 716(c).

A-716.2. The need to apply para. 716 is governed by the actual content of the package to be transported. For example, when a Type B(M) packaging, for which the package design approval certificate gives the permitted contents as ^{60}Co limited to 1600 TBq, is used for shipment of only 400 TBq ^{60}Co , no shipment approval is required since 400 TBq is less than $3000 A_2$ for ^{60}Co (which is $3000 \times 0.4 \text{ TBq} = 1200 \text{ TBq}$).

A-717. The Regulations conceptually differentiate between design approvals and shipment approvals. A shipment approval may be incorporated into the corresponding design approval certificate, and if this is done care should be exercised to clearly define the dual nature of the approval certificate and to apply the proper type of codes. For type codes see para. 724.

APPROVAL OF SHIPMENT UNDER SPECIAL ARRANGEMENT

A-720. Although an approval of a shipment under special arrangement will require consideration of both the shipment procedures and the packaging design, the approval is conceptually a shipment approval. The package design itself is not given any special approval.

A-721. The level of safety necessary in special arrangement shipments is normally achieved by imposing operational controls to compensate for any deficiencies in the packaging or the shipping procedures. Some of the operational controls which may be effectively employed are as follows:

- (a) Exclusive use of vehicle (see para. 128 and para. A-128).
- (b) Escort of shipment. The escort is normally a radiation protection specialist who is equipped with radiation monitoring instruments and is familiar with emergency procedures enabling him in the event of an accident or other abnormal event to identify quickly any radiation and contamination hazards present and to provide appropriate advice to the civil authorities. For road transport the escort, whenever possible, should travel in a separate vehicle so that he would not be incapacitated by the same accident. The escort should also be equipped with stakes, ropes, and signs to cordon off an accident area and with a fire extinguisher to control minor fires, and a communications system. If considered prudent, the radiation protection specialist could be accompanied by police and fire department escorts.
- (c) Routing of shipment may be controlled in order to select the safer routes and, if possible, to avoid areas of high population density and possible hazards, such as steep gradients and railway level crossings.
- (d) Timing of shipment may be controlled to avoid busy periods such as rush hours and weekend traffic peaks.

- (e) Shipments may be made directly, i.e. without stopover or trans-shipment, where possible.
- (f) Transport vehicle speeds may be limited, particularly if the impact resistance of the packaging is low and if the slower speed of the transport vehicle would not cause additional hazards (such as collisions involving faster moving vehicles).
- (g) Consideration should be given to notifying the civil authorities of the shipment beforehand, particularly police and fire departments and emergency plan participants, so that they may be prepared for any contingency.
- (h) Emergency procedures (either ad hoc or standing) should exist for any contingencies resulting from the shipment being involved in an accident.
- (i) Ancillary equipment such as package-to-vehicle tie-down or shock absorber systems and other protective devices or structures should be used, where necessary, to compensate for package deficiencies.

COMPETENT AUTHORITY APPROVAL CERTIFICATES

Competent authority identification marks

A-724. In applying and interpreting the type codes it is necessary to keep in mind that the code is based on the use of several indicators with the intent to quickly provide information on the type of package or shipment in question. The indicators provide information on package design characteristics (e.g. Type A, Type B(U) or Type B(M)), on the possible presence of fissile material in the package and on other specific aspects of the approval certificate (e.g. special arrangement, shipment or special form radioactive material). Specifically, the appearance of, for example, B(U)F in the code does not necessarily imply the presence of fissile material in a particular package, only the possibility that it might be present.

CONTENTS OF APPROVAL CERTIFICATES

A-726. The Regulations prescribe the basic information which must appear on certificates of approval and a competent authority identification mark system. Competent authorities are urged to follow these prescriptions as closely as possible to achieve international uniformity of certification. In addition to the applicable national regulations and the relevant international regulations (see Appendix I), each certificate should make reference to the appropriate edition of the Regulations, because this is the internationally recognized and known standard. The international vehicle registration (VRI) code, which is used in competent authority identification marks, is given in Table A-V.

TABLE A-V. LIST OF VRI CODES BY COUNTRY

Country Name	VRI Code	Country Name	VRI Code
Afghanistan	AFG	Iran, Islamic Rep. of	IR
Albania	AL	Iraq	IRQ
Algeria	DZ	Ireland	IRL
Argentina	RA	Israel	IL
Australia	AUS	Italy	I
Austria	A	Jamaica	JA
Bangladesh	BD	Japan	J
Belgium	B	Jordan	HKJ
Bolivia	BO ^a	Kenya	EAK
Brazil	BR	Korea, Republic of	ROK
Bulgaria	BG	Lebanon	RL
Burma	BUR	Luxembourg	L
Cameroon	CM ^a	Madagascar	RM
Canada	CDN	Malaysia	MAL
Chile	RCH	Mali	RMM
China	CN ^a	Mexico	MEX
Colombia	CO	Monaco	MC
Costa Rica	CR	Netherlands	NL
Cuba	C	New Zealand	NZ
Cyprus	CY	Niger	RN
Czechoslovakia	CS	Nigeria	WAN
Dem. P. R. of Korea	KP	Norway	N
Denmark	DK	Pakistan	PAK
Ecuador	EC	Paraguay	PY
Egypt	ET	Peru	PE
El Salvador	ES	Philippines	RP
Ethiopia	ETH	Poland	PL
Finland	SF	Portugal	P
France	F	Romania	R
Gabon	GA ^a	Saudi Arabia	SA
German Democratic Rep.	DDR	Sierra Leone	WAL
Germany, Fed. Rep. of	D	Singapore	SGP
Ghana	GH	South Africa	ZA
Greece	GR	Spain	E
Guatemala	GCA	Sri Lanka	CL
Hungary	H	Sudan	SUD
Iceland	IS	Sweden	S
India	IND	Switzerland	CH
Indonesia	RI	Syrian Arab Republic	SYR

TABLE A-V. (cont.)

Country Name	VRI Code	Country Name	VRI Code
Thailand	T	USSR	SU
Tunisia	TN	Venezuela	YV
Turkey	TR	Viet Nam	VN
United Kingdom	GB	Yugoslavia	YU
United States of America	USA	Zaire	ZRE
Uruguay	U	Zambia	Z

No VRI Code available for this country, code shown is ISO Code.

A-727. See para. A-726.

A-728. See para. A-726.

A-729. See para. A-726.

VALIDATION OF CERTIFICATES

A-730.1. The requirement for multilateral approval is equivalent to saying that a particular approval is required from two or more competent authorities, each needing to have demonstrated that an adequate level of safety is provided. There may, however, be differences as to how each competent authority will be satisfied.

A-730.2. The approval certificate of the competent authority of the country of origin is usually the first to be issued in the series of multilateral approval certificates. The other competent authorities have the option of either performing a separate safety assessment and evaluation or making use of the assessment already done by the original competent authority, thus limiting the scope and extent of their own assessment.

A-730.3. Subsequent approval certificates may take either of two forms. First, a competent authority in a subsequent country may endorse the original certificate, i.e. agree with and endorse the original certificate including any definition of controls incorporated in it. This is multilateral approval by validation of the original certificate. An approval by validation will not require any additional competent

authority's identification mark, either in terms of certificate identification or marking on packages. Second, a competent authority may issue an approval certificate which is associated with, but separate from, the original certificate in that this subsequent certificate would bear an identification mark other than that of the original identification mark. Furthermore, in this case packagings in use under such a multilateral approval have to be marked with the identification marks of both the original and the subsequent approval certificates.

This publication is no longer valid
Please see <http://www-ns.iaea.org/standards/>

Appendix I

LIST OF REGULATORY DOCUMENTS OF INTERNATIONAL AND REGIONAL INTERNATIONAL ORGANIZATIONS

INTERNATIONAL ORGANIZATIONS

Recommendations on the Transport of Dangerous Goods, the Committee of Experts on the Transport of Dangerous Goods, United Nations (UN), New York, United States of America.

Technical Instructions for the Safe Transport of Dangerous Goods by Air (TI), International Civil Aviation Organization (ICAO), 1000 Sherbrooke Street West, Suite 400, Montreal, Quebec, Canada H3A 2R2.

Dangerous Goods Regulations, International Air Transport Association (IATA), 2000 Peel Street, Montreal, Quebec, Canada H3A 2R4.

International Maritime Dangerous Goods (IMDG) Code, International Maritime Organization (IMO), 4 Albert Embankment, London SE1 7SR, United Kingdom.

Universal Postal Convention of Rio de Janeiro, Universal Postal Union, Case postale, CH-3000 Berne 15, Switzerland.

The International Convention for Safe Containers (CSC), as amended, International Maritime Organization (IMO), London, 4 Albert Embankment, London SE1 7SR, United Kingdom.

REGIONAL INTERNATIONAL ORGANIZATIONS

European Agreement Concerning the International Carriage of Dangerous Goods by Road (ADR) and Protocol of Signature, United Nations Economic Commission for Europe, Palais des Nations, CH-1211 Geneva 10, Switzerland.

Règlement international concernant le transport des marchandises dangereuses par chemins de fer RID, Convention relative aux transports internationaux ferroviaires (COTIF), Office central des transports internationaux par chemins de fer, Gryphenhübeliweg 30, CH-3006 Berne, Switzerland.

Règlement pour le transport des matières dangereuses sur le Rhin (ADNR), Central Commission for Navigation on the Rhine, Palais du Rhin, F-67 Strasbourg, France.

European Provisions Concerning the International Carriage of Dangerous Goods by Inland Waterway (ADN), Draft, United Nations Economic Commission for Europe, Palais des Nations, CH-1211 Geneva 10, Switzerland.

Regulations for the Transport of Radioactive Substances, Annex 4 to the Agreement on International Railroad Freight Traffic (SMGS), Railroad Co-operation Organization (OSZhD), Hoza Street 63/67, Warsaw, Poland.

Regulations for the Safe Transport of Spent Nuclear Fuel from Nuclear Power Plants of CMEA Member Countries — Transport by Rail, Council for Mutual Economic Assistance (CMEA), Kalinin Prospect 56, 12105 Moscow, Union of Soviet Socialist Republics.

Note: Most of the regulations and recommendations referred to above are updated periodically.

Appendix II

CONTAMINATION CONTROL

AII.1. FIXED AND NON-FIXED CONTAMINATION

Contamination, as defined in the Regulations, includes two types of radioactivity on surfaces or embedded in surfaces, namely fixed contamination and non-fixed contamination. The latter component has been and sometimes still is referred to as loose contamination. There is no definitive distinction between fixed and non-fixed and various terms have been used to describe the distinction. For practical purposes a distinction is made between contamination which remains in situ and, therefore, cannot provide hazards from ingestion, inhalation or spreading, and non-fixed contamination which can contribute to these hazards. The only hazard from fixed contamination is that due to external radiation exposure, whilst the primary hazard from non-fixed contamination is the potential for internal exposure due to intake or external exposure due to contamination of the skin should it be released from the surface.

In practice, however, contamination which appears fixed may become non-fixed as a result of the effects of weather, handling, etc. In most instances where small packages are slightly contaminated on the outer surfaces the contamination is almost entirely removable or non-fixed and the methods of measurement will reflect this. In some situations, however, such as in the case of irradiated fuel flasks, which are immersed in contaminated cooling pond water during loading and unloading, this is not necessarily so. Contaminants such as caesium-137 may strongly adhere onto or penetrate into steel surfaces. If these flasks remain in contaminated ponds for lengthy periods the degree of fixed contamination may be very high. Similarly, contamination may become ingrained in pores, and fine cracks, and in crevices, particularly in the vicinity of lid seals. Subsequent weathering, exposure to rain or even exposure to moist air conditions may cause some fixed contamination to be released or to become non-fixed. Thus a surface which is apparently free of non-fixed contamination at the beginning of a journey may subsequently be found to be contaminated at the end of its journey. Hence, considerable care in decontamination is necessary.

In all cases contamination levels on the external surface of packages should be kept as low as is reasonably achievable. The best way to ensure this is to prevent the surfaces from becoming contaminated in the first place. Thus loading and handling procedures should be devised and reviewed to prevent contamination whenever possible. In the particular case of fuel flasks mentioned above the pond immersion time should be minimized and effective decontamination techniques should be devised. Seal areas should be cleared by high pressure sprays, where possible, and

particular care taken to minimize the presence of contaminated water between the body and lid of the fuel flask. The problem can be alleviated by the provision of a 'skirt' around the flask during its immersion in the cooling pond. However, this may not be practicable at existing facilities and thus the potential problems will remain for some time.

AII.2. LIMITS OF CONTAMINATION

The Regulations prescribe limits on the levels of non-fixed contamination on the surfaces of packages. The limits correspond with values generally accepted for laboratories and plant working areas and are thus adequate in the context of transport where exposure time and handling time for transport workers are likely to be very much less than for laboratory or nuclear facility workers (e.g. reactor personnel). However, if all packages were contaminated close to the limits, the routine handling and storage of packages in transit stores, airport terminals, rail marshalling yards, etc. could lead to buildup of contamination in working areas. It may be necessary that checks be made to ensure that such buildup does not occur. Similarly, it is advisable to occasionally check gloves or other items of clothing of personnel regularly handling packages.

The Regulations set no specific limits for the levels of fixed contamination on packages, since the external radiation resulting therefrom will combine with the penetrating radiation from the contents, and the net radiation levels for packages are controlled by the specific requirements in, for example, paras 433 and 434, and indirectly by para. 432 when combined with para. 428(a). However, limits on fixed contamination are imposed for conveyances (see para. 413) to minimize the risk that it may become non-fixed as a result of abrasion, weathering, etc.

AII.3. MEASUREMENT OF NON-FIXED CONTAMINATION LEVELS

In a few cases a measurement of contamination may be made by direct reading contamination monitors. Such a measurement will include both fixed and non-fixed contamination. This will only be practicable where the level of background radiation from the installation in which the measurement is made or the radiation level from the contents does not interfere. In most cases the level of non-fixed contamination will have to be measured indirectly by wiping a known area with a smear and measuring the resultant radioactivity on the smear in an area not influenced by radiation background from other sources.

The derived limits for non-fixed contamination apply to the average level over an area of 300 m² or the total package if its total surface area is less than 300 cm². The level of non-fixed contamination may be determined by wiping an area of 300 cm² by hand with a filter paper or a wad of dry cotton wool or other material of similar nature. The number of smear samples taken on a larger package should be such as to be representative of the whole surface and should be chosen to include areas known or expected to be more contaminated than the remainder of the surface. For routine surveys on a very large package such as on an irradiated fuel flask, it is common practice to select a large number of fixed general positions to assist in identifying patterns and trends. Care should be taken that exactly the same position is not wiped on each occasion since this would leave large areas never checked, and would tend to 'clean' the areas checked.

The radioactivity on the smear sample may be measured either with a portable contamination monitor or in a standard counting castle. Care is necessary in converting the count rate to surface activity as a number of factors such as counting efficiency, geometrical efficiency, counter calibration and the fraction of activity removed from the surface to the smear sample will affect the final result.

To avoid underestimation, the beta energy of the calibration source used for a counter should not be greater than the beta energies of the contaminant being measured. The fraction of activity removed by the wipe test can, in practice, vary over a wide range and is dependent on the nature of the surface, the nature of the contaminant, the pressure used in wiping, the contact area of the material used for the test, the technique of rubbing (e.g. missing parts of the 300 cm² area or doubly wiping them) and the accuracy with which the operator estimates the area of 300 cm². It is common practice to assume that the fraction removed is 10% and this is usually conservative, so this results in overestimating the level of contamination. Other fractions may be used if determined experimentally.

It is recommended that users develop specific contamination measurement techniques relevant to their particular circumstances. Such techniques include the use of smears and appropriate survey instruments. The instruments and detectors selected should take into account the likely isotopes to be measured. Particular care is necessary in selecting instruments of appropriate energy dependence when low energy beta or alpha emitters are present. It should be recognized that the size of the smear and the size of the sensitive area of the detector are very important factors in determining overall efficiency.

It is further recommended that operators be adequately trained to achieve results as consistent as possible. Comparison between operators may be valuable in this respect. Attention is drawn to the difficulties which will occur if different organizations use techniques which are not fully compatible — especially in circumstances where it is not practical to maintain the levels of non-fixed contamination to near zero values.

AII.4. MEASUREMENT OF FIXED CONTAMINATION

If the non-fixed contamination on conveyances exceeds the limits in Table III of the Regulations, the conveyance should be decontaminated and following the decontamination a measurement should be made of the fixed contamination, for which a limit is specified in para. 413. The radiation level resulting from the fixed contamination on the surfaces may be measured using a portable instrument of an appropriate range held near to the surface of the conveyance. Such measurements should only be made before the conveyance is loaded.

Where packages having relatively high levels of fixed contamination are handled regularly by the same transport workers it may be necessary to consider not only the penetrating radiation but also the non-penetrating radiation from that contamination. An example of such a situation is the regular transport of irradiated fuel flasks by rail, when certain transport workers may regularly come in contact with the flasks. The (whole body) dose received by the workers from the penetrating radiation may be sufficiently low so that no special health surveillance or individual monitoring is necessary (see para. A-204). If it is known that the fixed contamination levels may be high, then it may be prudent to derive a working limit that prevents undesirable exposure of the workers hands.

As an example, for deriving a working limit for the hands one should assume that the limiting annual dose to the hands is 50 mSv (5 rem), which is 10% of the limit established by IAEA Safety Series No. 9 [6]. Assuming the contact time is 100 h per year and the effect of wearing gloves is ignored, the limiting contact dose rate would be 0.5 mSv/h (50 mrem/h).

In practice, measurements with portable instruments may underestimate (or overestimate) the true contact dose by a factor of two. Therefore, care should be taken in applying readings from instruments, and wherever possible the accuracy of the measuring device should be defined and considered in deriving the working limit.

Appendix III

HALF-LIFE AND SPECIFIC ACTIVITY OF RADIONUCLIDES, AND SPECIFIC ACTIVITY OF URANIUM AND THORIUM

Table AIII-1 provides a listing of the half-life and the specific activity of each radionuclide calculated using the equation shown in para. A-143.1 (Atomic Data and Nuclear Data Tables, Vol. 29, No. 2, Academic Press, New York (1983)).

Table AIII-2 provides the specific activity of uranium for various levels of enrichment. These figures for uranium include the activity of uranium-234 which is concentrated during the enrichment process.

TABLE AIII-1. HALF-LIFE AND SPECIFIC ACTIVITY OF RADIONUCLIDES

Symbol of radionuclide	Element and atomic number	Half-life		Specific activity ^a	
		T _{1/2} (s)	T _{1/2} (m,h,d,a)	Bq/g	Ci/g
Ac-225	Actinium (89)	8.640×10^5	1.000 × 10 d	2.147×10^{15}	5.803×10^4
Ac-227		6.865×10^8	2.177 × 10 a	2.678×10^{12}	7.239 × 10
Ac-228		2.207×10^4	6.130 h	8.296×10^{16}	2.242×10^6
Ag-105	Silver (47)	3.576×10^6	4.139 × 10 d	1.112×10^{15}	3.004×10^4
Ag-108m		4.005×10^9	1.270 × 10 ² a	9.650×10^{11}	2.608 × 10
Ag-110m		2.159×10^7	2.499 × 10 ² d	1.757×10^{14}	4.750×10^3
Ag-111		6.437×10^5	7.450 d	5.842×10^{15}	1.579×10^5
Al-26	Aluminium (13)	2.271×10^{13}	7.200 × 10 ⁵ a	7.071×10^8	1.911×10^{-2}
Am-241	Americium (95)	1.363×10^{10}	4.322 × 10 ² a	1.271×10^{11}	3.434
Am-242m		4.793×10^9	1.520 × 10 ² a	3.598×10^{11}	9.725
Am-243		2.324×10^{11}	7.370 × 10 ³ a	7.391×10^9	1.997×10^{-1}
Ar-37	Argon (18)	3.027×10^6	3.504 × 10 d	3.726×10^{15}	1.007×10^5
Ar-39		8.483×10^9	2.690 × 10 ² a	1.262×10^{12}	3.410 × 10
Ar-41		6.588×10^3	1.830 h	1.545×10^{18}	4.177×10^7
Ar-42		1.038×10^9	3.290 × 10 a	9.579×10^{12}	2.589×10^2
As-72	Arsenic (33)	9.360×10^4	2.600 × 10 h	6.194×10^{16}	1.674×10^6
As-73		6.938×10^6	8.030 × 10 d	8.242×10^{14}	2.227×10^4
As-74		1.536×10^6	1.778 × 10 d	3.672×10^{15}	9.924×10^4
As-76		9.475×10^4	2.632 × 10 h	5.796×10^{16}	1.567×10^6
As-77		1.398×10^5	3.883 × 10 h	3.878×10^{16}	1.048×10^6
At-211	Astatine (85)	2.596×10^4	7.210 h	7.622×10^{16}	2.060×10^6

TABLE AIII-1. (cont.)

Symbol of radionuclide	Element and atomic number	Half-life		Specific activity ^a	
		T _{1/2} (s)	T _{1/2} (m,h,d,a)	Bq/g	Ci/g
Au-193	Gold (79)	6.354 × 10 ⁴	1.765 × 10 h	3.404 × 10 ¹⁶	9.199 × 10 ⁵
Au-194		1.422 × 10 ⁵	3.950 × 10 h	1.513 × 10 ¹⁶	4.089 × 10 ⁵
Au-195		1.581 × 10 ⁷	1.830 × 10 ² d	1.354 × 10 ¹⁴	3.659 × 10 ³
Au-196		5.340 × 10 ⁵	6.180 d	3.988 × 10 ¹⁵	1.078 × 10 ⁵
Au-198		2.333 × 10 ⁵	2.700 d	9.037 × 10 ¹⁵	2.442 × 10 ⁵
Au-199		2.713 × 10 ⁵	3.140 d	7.732 × 10 ¹⁵	2.090 × 10 ⁵
Ba-131	Barium (56)	1.020 × 10 ⁶	1.180 × 10 d	3.125 × 10 ¹⁵	8.447 × 10 ⁴
Ba-133m		1.400 × 10 ⁵	3.890 × 10 h	2.241 × 10 ¹⁶	6.057 × 10 ⁵
Ba-133		3.324 × 10 ⁸	1.054 × 10 a	9.442 × 10 ¹²	2.552 × 10 ²
Ba-140		1.102 × 10 ⁶	1.275 × 10 d	2.707 × 10 ¹⁵	7.315 × 10 ⁴
Be-7	Beryllium (4)	4.605 × 10 ⁶	5.330 × 10 d	1.295 × 10 ¹⁶	3.500 × 10 ⁵
Be-10		5.046 × 10 ¹³	1.600 × 10 ⁶ a	8.273 × 10 ⁸	2.236 × 10 ⁻²
Bi-205	Bismuth (83)	1.323 × 10 ⁶	1.531 × 10 d	1.539 × 10 ¹⁵	4.160 × 10 ⁴
Bi-206		5.391 × 10 ⁵	6.240 d	3.758 × 10 ¹⁵	1.016 × 10 ⁵
Bi-207		1.053 × 10 ⁹	3.340 × 10 a	1.914 × 10 ¹²	5.174 × 10
Bi-210m		9.461 × 10 ¹³	3.000 × 10 ⁶ a	2.101 × 10 ⁷	5.678 × 10 ⁻⁴
Bi-210		4.329 × 10 ⁵	5.010 d	4.592 × 10 ¹⁵	1.241 × 10 ⁵
Bi-212		3.633 × 10 ³	6.055 × 10 m	5.420 × 10 ¹⁷	1.465 × 10 ⁷
Bk-247	Berkelium (97)	4.415 × 10 ¹⁰	1.400 × 10 ³ a	3.828 × 10 ¹⁰	1.035
Bk-249		2.765 × 10 ⁷	3.200 × 10 ² d	6.063 × 10 ¹³	1.639 × 10 ³
Br-76	Bromine (35)	5.832 × 10 ⁴	1.620 × 10 h	9.418 × 10 ¹⁶	2.545 × 10 ⁶
Br-77		2.056 × 10 ⁵	2.380 d	2.636 × 10 ¹⁶	7.125 × 10 ⁵
Br-82		1.271 × 10 ⁵	3.530 × 10 h	4.006 × 10 ¹⁶	1.083 × 10 ⁶
C-11	Carbon (6)	1.223 × 10 ³	2.038 × 10 m	3.103 × 10 ¹⁹	8.387 × 10 ⁸
C-14		1.807 × 10 ¹¹	5.730 × 10 ³ a	1.650 × 10 ¹¹	4.459
Ca-41	Calcium (20)	3.248 × 10 ¹²	1.030 × 10 ⁵ a	3.134 × 10 ⁹	8.471 × 10 ⁻²
Ca-45		1.408 × 10 ⁷	1.630 × 10 ² d	6.586 × 10 ¹⁴	1.780 × 10 ⁴
Ca-47		3.923 × 10 ⁵	4.540 d	2.264 × 10 ¹⁶	6.119 × 10 ⁵
Cd-109	Cadmium (48)	3.991 × 10 ⁷	4.619 × 10 ² d	9.596 × 10 ¹³	2.593 × 10 ³
Cd-113m		4.447 × 10 ⁸	1.410 × 10 a	8.307 × 10 ¹²	2.245 × 10 ²
Cd-115m		3.853 × 10 ⁶	4.460 × 10 d	9.419 × 10 ¹⁴	2.546 × 10 ⁴
Cd-115		1.927 × 10 ⁵	2.230 d	1.884 × 10 ¹⁶	5.092 × 10 ⁵
Ce-139	Cerium (58)	1.190 × 10 ⁷	1.377 × 10 ² d	2.524 × 10 ¹⁴	6.822 × 10 ³
Ce-141		2.808 × 10 ⁶	3.250 × 10 d	1.054 × 10 ¹⁵	2.849 × 10 ⁴
Ce-143		1.188 × 10 ⁵	3.300 × 10 h	2.457 × 10 ¹⁶	6.641 × 10 ⁵
Ce-144		2.462 × 10 ⁷	2.849 × 10 ² d	1.178 × 10 ¹⁴	3.183 × 10 ³

TABLE AIII-1. (cont.)

Symbol of radionuclide	Element and atomic number	Half-life		Specific activity ^a	
		T _{1/2} (s)	T _{1/2} (m,h,d,a)	Bq/g	Ci/g
Cf-248	Californium (98)	2.881 × 10 ⁷	3.335 × 10 ² d	5.841 × 10 ¹³	1.579 × 10 ³
Cf-249		1.107 × 10 ¹⁰	3.510 × 10 ² a	1.514 × 10 ¹¹	4.093
Cf-250		4.131 × 10 ⁸	1.310 × 10 a	4.042 × 10 ¹²	1.092 × 10 ²
Cf-251		2.832 × 10 ¹⁰	8.980 × 10 ² a	5.872 × 10 ¹⁰	1.587
Cf-252		8.326 × 10 ⁷	2.640 a	1.990 × 10 ¹³	5.377 × 10 ²
Cf-253		1.539 × 10 ⁶	1.781 × 10 d	1.072 × 10 ¹⁵	2.898 × 10 ⁴
Cf-254		5.227 × 10 ⁶	6.050 × 10 d	3.144 × 10 ¹⁴	8.497 × 10 ³
Cl-36	Chlorine (17)	9.492 × 10 ¹²	3.010 × 10 ⁵ a	1.221 × 10 ⁹	3.301 × 10 ⁻²
Cl-38		2.234 × 10 ³	3.724 × 10 m	4.916 × 10 ¹⁸	1.329 × 10 ⁸
Cm-240	Curium (96)	2.333 × 10 ⁶	2.700 × 10 d	7.456 × 10 ¹⁴	2.015 × 10 ⁴
Cm-241		2.834 × 10 ⁶	3.280 × 10 d	6.112 × 10 ¹⁴	1.652 × 10 ⁴
Cm-242		1.407 × 10 ⁷	1.628 × 10 ² d	1.226 × 10 ¹⁴	3.314 × 10 ³
Cm-243		8.988 × 10 ⁸	2.850 × 10 a	1.911 × 10 ¹²	5.165 × 10
Cm-244		5.708 × 10 ⁸	1.810 × 10 a	2.997 × 10 ¹²	8.100 × 10
Cm-245		2.681 × 10 ¹¹	8.500 × 10 ³ a	6.356 × 10 ⁹	1.718 × 10 ⁻¹
Cm-246		1.492 × 10 ¹¹	4.730 × 10 ³ a	1.138 × 10 ¹⁰	3.074 × 10 ⁻¹
Cm-247		4.920 × 10 ¹⁴	1.560 × 10 ⁷ a	3.435 × 10 ⁶	9.284 × 10 ⁻⁵
Cm-248		1.072 × 10 ¹³	3.400 × 10 ⁵ a	1.570 × 10 ⁸	4.243 × 10 ⁻³
Co-56	Cobalt (27)	6.808 × 10 ⁶	7.880 × 10 d	1.095 × 10 ¹⁵	2.959 × 10 ⁴
Co-57		2.348 × 10 ⁷	2.718 × 10 ² d	3.118 × 10 ¹⁴	8.428 × 10 ³
Co-58m		3.294 × 10 ⁴	9.150 h	2.185 × 10 ¹⁷	5.905 × 10 ⁶
Co-58		6.127 × 10 ⁶	7.092 × 10 d	1.175 × 10 ¹⁵	3.174 × 10 ⁴
Co-60		1.662 × 10 ⁸	5.270 a	4.186 × 10 ¹³	1.131 × 10 ³
Cr-51	Chromium (24)	2.393 × 10 ⁶	2.770 × 10 d	3.420 × 10 ¹⁵	9.243 × 10 ⁴
Cs-129	Caesium (55)	1.156 × 10 ⁵	3.210 × 10 h	2.800 × 10 ¹⁶	7.568 × 10 ⁵
Cs-131		8.372 × 10 ⁵	9.690 d	3.806 × 10 ¹⁵	1.029 × 10 ⁵
Cs-132		5.581 × 10 ⁵	6.460 d	5.666 × 10 ¹⁵	1.531 × 10 ⁵
Cs-134m		1.048 × 10 ⁴	2.910 h	2.973 × 10 ¹⁷	8.036 × 10 ³
Cs-134		6.496 × 10 ⁷	2.060 a	4.795 × 10 ¹³	1.296 × 10 ³
Cs-135		7.253 × 10 ¹³	2.300 × 10 ⁶ a	4.263 × 10 ⁷	1.152 × 10 ⁻³
Cs-136		1.137 × 10 ⁶	1.316 × 10 d	2.699 × 10 ¹⁵	7.296 × 10 ⁴
Cs-137		9.505 × 10 ⁸	3.014 × 10 a	3.206 × 10 ¹²	8.664 × 10
Cu-64	Copper (29)	4.572 × 10 ⁴	1.270 × 10 h	1.427 × 10 ¹⁷	3.855 × 10 ⁶
Cu-67		2.228 × 10 ⁵	6.190 × 10 h	2.796 × 10 ¹⁶	7.556 × 10 ⁵
Dy-159	Dysprosium (66)	1.248 × 10 ⁷	1.444 × 10 ² d	2.104 × 10 ¹⁴	5.678 × 10 ³
Dy-165		8.388 × 10 ³	2.330 h	3.016 × 10 ¹⁷	8.151 × 10 ⁶
Dy-166		2.938 × 10 ⁵	3.400 d	8.560 × 10 ¹⁵	2.313 × 10 ⁵

TABLE AIII-1. (cont.)

Symbol of radionuclide	Element and atomic number	Half-life		Specific activity ^a	
		T _{1/2} (s)	T _{1/2} (m,h,d,a)	Bq/g	Ci/g
Er-169	Erbium (68)	8.035 × 10 ⁵	9.300 d	3.074 × 10 ¹⁵	8.308 × 10 ⁴
Er-171		2.707 × 10 ⁴	7.520 h	9.017 × 10 ¹⁶	2.437 × 10 ⁶
Eu-147	Europium (63)	2.074 × 10 ⁶	2.400 × 10 d	1.369 × 10 ¹⁵	3.701 × 10 ⁴
Eu-148		4.709 × 10 ⁶	5.450 × 10 d	5.990 × 10 ¹⁴	1.619 × 10 ⁴
Eu-149		8.044 × 10 ⁶	9.310 × 10 d	3.483 × 10 ¹⁴	9.413 × 10 ³
Eu-150A		4.536 × 10 ⁴	1.260 × 10 h	6.135 × 10 ¹⁶	1.658 × 10 ⁶
Eu-150B		1.129 × 10 ⁹	3.580 × 10 a	2.465 × 10 ¹²	6.662 × 10
Eu-152m		3.348 × 10 ⁴	9.300 h	8.202 × 10 ¹⁶	2.217 × 10 ⁶
Eu-152		4.204 × 10 ⁸	1.333 × 10 a	6.533 × 10 ¹²	1.766 × 10 ²
Eu-154		2.775 × 10 ⁸	8.800 a	9.767 × 10 ¹²	2.640 × 10 ²
Eu-155		1.476 × 10 ⁸	4.680 a	1.825 × 10 ¹³	4.932 × 10 ²
Eu-156		1.313 × 10 ⁶	1.520 × 10 d	2.037 × 10 ¹⁵	5.507 × 10 ⁴
F-18	Fluorine (9)	6.588 × 10 ³	1.830 h	3.520 × 10 ¹⁸	9.513 × 10 ⁷
Fe-52	Iron (26)	2.977 × 10 ⁴	8.270 h	2.696 × 10 ¹⁷	7.287 × 10 ⁶
Fe-55		8.641 × 10 ⁷	2.740 a	8.783 × 10 ¹³	2.374 × 10 ³
Fe-59		3.845 × 10 ⁶	4.450 × 10 d	1.840 × 10 ¹⁵	2.973 × 10 ⁴
Fe-60		9.461 × 10 ¹²	3.000 × 10 ⁵ a	7.353 × 10 ⁸	1.987 × 10 ⁻²
Ga-67	Gallium (31)	2.817 × 10 ⁵	3.260 d	2.212 × 10 ¹⁶	5.978 × 10 ⁵
Ga-68		4.086 × 10 ³	6.810 × 10 m	1.502 × 10 ¹⁸	4.060 × 10 ⁷
Ga-72		5.076 × 10 ⁴	1.410 × 10 h	1.142 × 10 ¹⁷	3.087 × 10 ⁶
Gd-146	Gadolinium (64)	4.173 × 10 ⁶	4.830 × 10 d	6.851 × 10 ¹⁴	1.852 × 10 ⁴
Gd-153		2.091 × 10 ⁷	2.420 × 10 ² d	1.305 × 10 ¹⁴	3.527 × 10 ³
Gd-159		6.682 × 10 ⁴	1.856 × 10 h	3.929 × 10 ¹⁶	1.062 × 10 ⁶
Ge-68	Germanium (32)	2.340 × 10 ⁷	2.708 × 10 ² d	2.646 × 10 ¹⁴	7.091 × 10 ³
Ge-71		1.020 × 10 ⁶	1.180 × 10 d	5.767 × 10 ¹⁵	1.559 × 10 ⁵
Ge-77		4.068 × 10 ⁴	1.130 × 10 h	1.333 × 10 ¹⁷	3.602 × 10 ⁶
Hf-172	Hafnium (72)	5.897 × 10 ⁷	1.870 a	4.115 × 10 ¹³	1.112 × 10 ³
Hf-175		6.048 × 10 ⁶	7.000 × 10 d	3.944 × 10 ¹⁴	1.066 × 10 ⁴
Hf-181		3.663 × 10 ⁶	4.240 × 10 d	6.295 × 10 ¹⁴	1.701 × 10 ⁴
Hf-182		2.838 × 10 ¹⁴	9.000 × 10 ⁶ a	8.081 × 10 ⁶	2.184 × 10 ⁻⁴
Hg-194	Mercury (80)	1.640 × 10 ¹⁰	5.200 × 10 ² a	1.312 × 10 ¹¹	3.546
Hg-197m		8.568 × 10 ⁴	2.380 × 10 h	2.473 × 10 ¹⁶	6.684 × 10 ⁵
Hg-197		2.307 × 10 ⁵	2.670 d	9.185 × 10 ¹⁵	2.482 × 10 ⁵
Hg-203		4.025 × 10 ⁶	4.658 × 10 d	5.109 × 10 ¹⁴	1.381 × 10 ⁴
Ho-163	Holmium (72)	9.461 × 10 ⁸	3.000 × 10 a	2.707 × 10 ¹²	7.316 × 10
Ho-166m		3.784 × 10 ¹⁰	1.200 × 10 ³ a	6.645 × 10 ¹⁰	1.796
Ho-166		9.648 × 10 ⁴	2.680 × 10 h	2.606 × 10 ¹⁶	7.044 × 10 ⁵

TABLE AIII-1. (cont.)

Symbol of radionuclide	Element and atomic number	Half-life		Specific activity ^a		
		T _{1/2} (s)	T _{1/2} (m,h,d,a)	Bq/g	Ci/g	
I-123	Iodine (53)	4.752 × 10 ⁴	1.320 × 10 h	7.141 × 10 ¹⁶	1.930 × 10 ⁶	
I-124		3.612 × 10 ⁵	4.180 d	9.321 × 10 ¹⁵	2.519 × 10 ⁵	
I-125		5.196 × 10 ⁶	6.014 × 10 d	6.427 × 10 ¹⁴	1.737 × 10 ⁴	
I-126		1.125 × 10 ⁶	1.302 × 10 d	2.945 × 10 ¹⁵	7.959 × 10 ⁴	
I-129		4.951 × 10 ¹⁴	1.570 × 10 ⁷ a	6.535 × 10 ⁶	1.766 × 10 ⁻⁴	
I-131		6.926 × 10 ⁵	8.020 d	4.598 × 10 ¹⁵	1.243 × 10 ⁵	
I-132		8.280 × 10 ³	2.300 h	3.819 × 10 ¹⁷	1.032 × 10 ⁷	
I-133		7.488 × 10 ⁴	2.080 × 10 h	4.191 × 10 ¹⁶	1.133 × 10 ⁶	
I-134		3.156 × 10 ³	5.260 × 10 m	9.870 × 10 ¹⁷	2.668 × 10 ⁷	
I-135		2.380 × 10 ⁴	6.610 h	1.299 × 10 ¹⁷	3.512 × 10 ⁶	
In-111		Indium (49)	2.445 × 10 ⁵	2.830 d	1.538 × 10 ¹⁶	4.157 × 10 ⁵
In-113m			5.976 × 10 ³	1.660 h	6.181 × 10 ¹⁷	1.671 × 10 ⁷
In-114m			4.278 × 10 ⁶	4.951 × 10 d	8.560 × 10 ¹⁴	2.313 × 10 ⁴
In-115m			1.616 × 10 ⁴	4.490 h	2.246 × 10 ¹⁷	6.069 × 10 ⁶
Ir-189		Iridium (77)	1.140 × 10 ⁶	1.320 × 10 d	1.937 × 10 ¹⁵	5.234 × 10 ⁴
Ir-190	9.590 × 10 ⁵		1.110 × 10 d	2.291 × 10 ¹⁵	6.191 × 10 ⁴	
Ir-192	6.379 × 10 ⁶		7.383 × 10 d	3.408 × 10 ¹⁴	9.211 × 10 ³	
Ir-193m	9.158 × 10 ⁵		1.060 × 10 d	2.362 × 10 ¹⁵	6.382 × 10 ⁴	
Ir-194	6.894 × 10 ⁴		1.915 × 10 h	3.121 × 10 ¹⁶	8.435 × 10 ⁵	
K-42	Potassium (19)	4.450 × 10 ⁴	1.236 × 10 h	2.234 × 10 ¹⁷	6.037 × 10 ⁶	
K-43		8.028 × 10 ⁴	2.230 × 10 h	1.209 × 10 ¹⁷	3.268 × 10 ⁶	
Kr-81	Krypton (36)	6.623 × 10 ¹²	2.100 × 10 ⁵ a	7.781 × 10 ⁸	2.103 × 10 ⁻²	
Kr-85m		1.613 × 10 ⁴	4.480 h	3.045 × 10 ¹⁷	8.229 × 10 ⁶	
Kr-85		3.381 × 10 ⁸	1.072 × 10 a	1.453 × 10 ¹³	3.926 × 10 ²	
Kr-87		4.578 × 10 ³	7.630 × 10 m	1.048 × 10 ¹⁸	2.833 × 10 ⁷	
La-137	Lanthanum (57)	1.892 × 10 ¹²	6.000 × 10 ⁴ a	1.610 × 10 ⁹	4.352 × 10 ⁻²	
La-140		1.450 × 10 ⁵	4.028 × 10 h	2.056 × 10 ¹⁶	5.557 × 10 ⁵	
Lu-172	Lutetium (71)	5.789 × 10 ⁵	6.700 d	4.192 × 10 ¹⁵	1.133 × 10 ⁵	
Lu-173		4.320 × 10 ⁷	1.370 a	5.585 × 10 ¹³	1.509 × 10 ³	
Lu-174m		1.227 × 10 ⁷	1.420 × 10 ² d	1.955 × 10 ¹⁴	5.285 × 10 ³	
Lu-174		1.044 × 10 ⁸	3.310 a	2.298 × 10 ¹³	6.211 × 10 ²	
Lu-177		5.797 × 10 ⁵	6.710 d	4.068 × 10 ¹⁵	1.099 × 10 ⁵	
Mg-28	Magnesium (12)	7.524 × 10 ⁴	2.090 × 10 h	1.981 × 10 ¹⁷	5.355 × 10 ⁶	
Mn-52	Manganese (25)	4.916 × 10 ⁵	5.690 d	1.633 × 10 ¹⁶	4.413 × 10 ⁵	
Mn-53		1.167 × 10 ¹⁴	3.700 × 10 ⁶ a	6.750 × 10 ⁷	1.824 × 10 ⁻³	
Mn-54		2.700 × 10 ⁷	3.125 × 10 ² d	2.863 × 10 ¹⁴	7.738 × 10 ³	
Mn-56		9.288 × 10 ³	2.580 h	8.025 × 10 ¹⁷	2.169 × 10 ⁷	

TABLE AIII-1. (cont.)

Symbol of radionuclide	Element and atomic number	Half-life		Specific activity ^a	
		T _{1/2} (s)	T _{1/2} (m,h,d,a)	Bq/g	Ci/g
Mo-93	Molybdenum (42)	1.104 × 10 ¹¹	3.500 × 10 ³ a	4.066 × 10 ¹⁰	1.099
Mo-99		2.376 × 10 ⁵	2.750 d	1.775 × 10 ¹⁶	4.796 × 10 ⁵
N-13	Nitrogen (7)	5.976 × 10 ²	9.960 m	5.373 × 10 ¹⁹	1.452 × 10 ⁹
Na-22	Sodium (11)	8.199 × 10 ⁷	2.600 a	2.314 × 10 ¹⁴	6.254 × 10 ³
Na-24		5.407 × 10 ⁴	1.502 × 10 h	3.217 × 10 ¹⁷	8.693 × 10 ⁶
Nb-92m	Niobium (41)	8.770 × 10 ⁵	1.015 × 10 d	5.174 × 10 ¹⁵	1.398 × 10 ⁵
Nb-93m		5.077 × 10 ⁸	1.610 × 10 a	8.840 × 10 ¹²	2.389 × 10 ²
Nb-94		6.402 × 10 ¹¹	2.030 × 10 ⁴ a	6.936 × 10 ⁹	1.875 × 10 ⁻¹
Nb-95		3.022 × 10 ⁶	3.498 × 10 d	1.454 × 10 ¹⁵	3.929 × 10 ⁴
Nb-97		4.326 × 10 ³	7.210 × 10 m	9.947 × 10 ¹⁷	2.688 × 10 ⁷
Nd-147	Neodymium (60)	9.487 × 10 ⁵	1.098 × 10 d	2.993 × 10 ¹⁵	8.090 × 10 ⁴
Nd-149		6.228 × 10 ³	1.730 h	4.498 × 10 ¹⁷	1.216 × 10 ⁷
Ni-59	Nickel (28)	2.397 × 10 ¹²	7.600 × 10 ⁴ a	2.952 × 10 ⁹	7.978 × 10 ⁻²
Ni-63		3.157 × 10 ⁹	1.001 × 10 ² a	2.099 × 10 ¹²	5.673 × 10
Ni-65		9.072 × 10 ³	2.520 h	7.079 × 10 ¹⁷	1.913 × 10 ⁷
Np-235	Neptunium (93)	3.423 × 10 ⁷	3.962 × 10 ² d	5.189 × 10 ¹³	1.402 × 10 ³
Np-236		3.784 × 10 ¹²	1.200 × 10 ⁵ a	4.674 × 10 ⁸	1.263 × 10 ⁻²
Np-237		6.749 × 10 ¹³	2.140 × 10 ⁶ a	2.610 × 10 ⁷	7.053 × 10 ⁻⁴
Np-239		2.039 × 10 ⁵	2.360 d	8.565 × 10 ¹⁵	2.315 × 10 ⁵
Os-185	Osmium (76)	8.087 × 10 ⁶	9.360 × 10 d	2.790 × 10 ¹⁴	7.541 × 10 ³
Os-191m		4.716 × 10 ⁴	1.310 × 10 h	4.634 × 10 ¹⁶	1.252 × 10 ⁶
Os-191		1.331 × 10 ⁶	1.540 × 10 d	1.642 × 10 ¹⁵	4.439 × 10 ⁴
Os-193		1.098 × 10 ⁵	3.050 × 10 h	1.970 × 10 ¹⁶	5.324 × 10 ⁵
Os-194		1.892 × 10 ⁸	6.000 a	1.137 × 10 ¹³	3.073 × 10 ²
P-32	Phosphorus (15)	1.232 × 10 ⁶	1.426 × 10 d	1.059 × 10 ¹⁶	2.861 × 10 ⁵
P-33		2.189 × 10 ⁶	2.534 × 10 d	5.777 × 10 ¹⁵	1.561 × 10 ⁵
Pa-230	Protactinium (91)	1.503 × 10 ⁶	1.740 × 10 d	1.207 × 10 ¹⁵	3.263 × 10 ⁴
Pa-231		1.034 × 10 ¹²	3.280 × 10 ⁴ a	1.747 × 10 ⁹	4.721 × 10 ⁻²
Pa-233		2.333 × 10 ⁶	2.700 × 10 d	7.680 × 10 ¹⁴	2.076 × 10 ⁴
Pb-201	Lead (82)	3.359 × 10 ⁴	9.330 h	6.183 × 10 ¹⁶	1.671 × 10 ⁶
Pb-202		1.656 × 10 ¹³	5.250 × 10 ⁵ a	1.248 × 10 ⁸	3.373 × 10 ⁻³
Pb-203		1.866 × 10 ⁵	2.160 d	1.102 × 10 ¹⁶	2.978 × 10 ⁵
Pb-205		4.510 × 10 ¹⁴	1.430 × 10 ⁷ a	4.515 × 10 ⁶	1.220 × 10 ⁻⁴
Pb-210		7.033 × 10 ⁸	2.230 × 10 a	2.826 × 10 ¹²	7.639 × 10
Pb-212		3.830 × 10 ⁴	1.064 × 10 h	5.140 × 10 ¹⁶	1.389 × 10 ⁶

TABLE AIII-1. (cont.)

Symbol of radionuclide	Element and atomic number	Half-life		Specific activity ^a	
		T _{1/2} (s)	T _{1/2} (m,h,d,a)	Bq/g	Ci/g
Pd-103	Palladium (46)	1.465 × 10 ⁶	1.696 × 10 d	2.766 × 10 ¹⁵	7.474 × 10 ⁴
Pd-107		2.050 × 10 ¹⁴	6.500 × 10 ⁶ a	1.903 × 10 ⁷	5.144 × 10 ⁻⁴
Pd-109		4.846 × 10 ⁴	1.346 × 10 h	7.903 × 10 ¹⁶	2.136 × 10 ⁶
Pm-143	Promethium (61)	2.290 × 10 ⁷	2.650 × 10 ² d	1.275 × 10 ¹⁴	3.446 × 10 ³
Pm-144		3.136 × 10 ⁷	3.630 × 10 ² d	9.242 × 10 ¹³	2.498 × 10 ³
Pm-145		5.582 × 10 ⁸	1.770 × 10 a	5.157 × 10 ¹²	1.394 × 10 ²
Pm-147		8.262 × 10 ⁷	2.620 a	3.437 × 10 ¹³	9.288 × 10 ²
Pm-148m		3.568 × 10 ⁶	4.130 × 10 d	7.904 × 10 ¹⁴	2.136 × 10 ⁴
Pm-149		1.909 × 10 ⁵	2.210 d	1.467 × 10 ¹⁶	3.965 × 10 ⁵
Pm-151		1.022 × 10 ⁵	2.840 × 10 h	2.704 × 10 ¹⁶	7.307 × 10 ⁵
Po-208	Polonium (84)	9.145 × 10 ⁷	2.900 a	2.194 × 10 ¹³	5.931 × 10 ²
Po-209		3.217 × 10 ⁹	1.020 × 10 ² a	6.209 × 10 ¹¹	1.678 × 10
Po-210		1.196 × 10 ⁷	1.384 × 10 ² d	1.662 × 10 ¹⁴	4.493 × 10 ³
Pr-142	Praseodymium (59)	6.887 × 10 ⁴	1.913 × 10 h	4.268 × 10 ¹⁶	1.154 × 10 ⁶
Pr-143		1.173 × 10 ⁶	1.358 × 10 d	2.488 × 10 ¹⁵	6.724 × 10 ⁴
Pt-188	Platinum (78)	8.813 × 10 ⁵	1.020 × 10 d	2.519 × 10 ¹⁵	6.809 × 10 ⁴
Pt-191		2.506 × 10 ⁵	2.900 d	8.722 × 10 ¹⁵	2.357 × 10 ⁵
Pt-193m		3.741 × 10 ⁵	4.330 d	5.781 × 10 ¹⁵	1.562 × 10 ⁵
Pt-193		1.577 × 10 ⁹	5.000 × 10 a	1.372 × 10 ¹²	3.707 × 10
Pt-195m		3.473 × 10 ⁵	4.020 d	6.163 × 10 ¹⁵	1.666 × 10 ⁵
Pt-197m		5.652 × 10 ³	1.570 h	3.749 × 10 ¹⁷	1.013 × 10 ⁷
Pt-197		6.588 × 10 ⁴	1.830 × 10 h	3.216 × 10 ¹⁶	8.693 × 10 ⁵
Pu-236	Plutonium (94)	8.988 × 10 ⁷	2.850 a	1.968 × 10 ¹³	5.319 × 10 ²
Pu-237		3.898 × 10 ⁶	4.512 × 10 d	4.518 × 10 ¹⁴	1.221 × 10 ⁴
Pu-238		2.767 × 10 ⁹	8.774 × 10 a	6.338 × 10 ¹¹	1.713 × 10
Pu-239		7.600 × 10 ¹¹	2.410 × 10 ⁴ a	2.298 × 10 ⁹	6.211 × 10 ⁻²
Pu-240		2.069 × 10 ¹¹	6.560 × 10 ³ a	8.407 × 10 ⁹	2.272 × 10 ⁻¹
Pu-241		4.541 × 10 ⁸	1.440 × 10 a	3.814 × 10 ¹²	1.031 × 10 ²
Pu-242		1.186 × 10 ¹³	3.769 × 10 ⁵ a	1.455 × 10 ⁸	3.931 × 10 ⁻³
Pu-244		2.548 × 10 ¹⁵	8.080 × 10 ⁷ a	6.714 × 10 ⁵	1.815 × 10 ⁻⁵
Ra-223		Radium (88)	9.876 × 10 ⁵	1.143 × 10 d	1.895 × 10 ¹⁵
Ra-224	3.162 × 10 ⁵		3.660 d	5.893 × 10 ¹⁵	1.593 × 10 ⁵
Ra-225	1.279 × 10 ⁶		1.480 × 10 d	1.451 × 10 ¹⁵	3.921 × 10 ⁴
Ra-226	5.046 × 10 ¹⁰		1.600 × 10 ³ a	3.660 × 10 ¹⁰	9.893 × 10 ⁻¹
Ra-228	1.813 × 10 ⁸		5.750 a	1.010 × 10 ¹³	2.729 × 10 ²

TABLE AIII-1. (cont.)

Symbol of radionuclide	Element and atomic number	Half-life		Specific activity ^a	
		T _{1/2} (s)	T _{1/2} (m,h,d,a)	Bq/g	Ci/g
Rb-81	Rubidium (37)	1.649 × 10 ⁴	4.580 h	3.125 × 10 ¹⁷	8.447 × 10 ⁶
Rb-83		7.448 × 10 ⁶	8.620 × 10 d	6.753 × 10 ¹⁴	1.825 × 10 ⁴
Rb-84		2.840 × 10 ⁶	3.287 × 10 d	1.750 × 10 ¹⁵	4.729 × 10 ⁴
Rb-86		1.612 × 10 ⁶	1.866 × 10 d	3.011 × 10 ¹⁵	8.137 × 10 ⁴
Rb-87		1.514 × 10 ¹⁸	4.800 × 10 ¹⁰ a	3.170 × 10 ³	8.566 × 10 ⁻⁸
Rb (natural)					
Re-183	Rhenium (75)	6.048 × 10 ⁶	7.000 × 10 d	3.771 × 10 ¹⁴	1.019 × 10 ⁴
Re-184m		1.426 × 10 ⁷	1.650 × 10 ² d	1.591 × 10 ¹⁴	4.301 × 10 ³
Re-184		3.283 × 10 ⁶	3.800 × 10 d	6.910 × 10 ¹⁴	1.867 × 10 ⁴
Re-186		3.266 × 10 ⁵	3.780 d	6.871 × 10 ¹⁵	1.857 × 10 ⁵
Re-187		1.577 × 10 ¹⁸	5.000 × 10 ¹⁰ a	1.416 × 10 ³	3.826 × 10 ⁻⁸
Re-188		6.113 × 10 ⁴	1.698 × 10 h	3.632 × 10 ¹⁶	9.817 × 10 ⁵
Re-189		8.748 × 10 ⁴	2.430 × 10 h	2.525 × 10 ¹⁶	6.823 × 10 ⁵
Re (natural)					
Rh-99	Rhodium (45)	1.382 × 10 ⁶	1.600 × 10 d	3.050 × 10 ¹⁵	8.243 × 10 ⁴
Rh-101		1.009 × 10 ⁸	3.200 a	4.095 × 10 ¹³	1.107 × 10 ³
Rh-102m		1.788 × 10 ⁷	2.070 × 10 ² d	2.288 × 10 ¹⁴	6.184 × 10 ³
Rh-102		9.145 × 10 ⁷	2.900 a	4.475 × 10 ¹³	1.209 × 10 ³
Rh-103m		3.367 × 10 ³	5.612 × 10 m	1.204 × 10 ¹⁸	3.253 × 10 ⁷
Rh-105		1.273 × 10 ⁵	3.536 × 10 h	3.123 × 10 ¹⁶	8.440 × 10 ⁵
Rn-222	Radon (86)	3.300 × 10 ⁵	3.820 d	5.697 × 10 ¹⁵	1.540 × 10 ⁵
Ru-97	Ruthenium (44)	2.506 × 10 ⁵	2.900 d	1.717 × 10 ¹⁶	4.642 × 10 ⁵
Ru-103		3.391 × 10 ⁶	3.925 × 10 d	1.195 × 10 ¹⁵	3.230 × 10 ⁴
Ru-105		1.598 × 10 ⁴	4.440 h	2.487 × 10 ¹⁷	6.722 × 10 ⁶
Ru-106		3.211 × 10 ⁷	3.716 × 10 ² d	1.227 × 10 ¹⁴	3.315 × 10 ³
S-35	Sulphur (16)	7.534 × 10 ⁶	8.720 × 10 d	1.583 × 10 ¹⁵	4.278 × 10 ⁴
Sb-122	Antimony (51)	2.333 × 10 ⁵	2.700 d	1.467 × 10 ¹⁶	3.964 × 10 ⁵
Sb-124		5.201 × 10 ⁶	6.020 × 10 d	6.472 × 10 ¹⁴	1.749 × 10 ⁴
Sb-125		8.609 × 10 ⁷	2.730 a	3.879 × 10 ¹³	1.048 × 10 ³
Sb-126		1.071 × 10 ⁶	1.240 × 10 d	3.092 × 10 ¹⁵	8.357 × 10 ⁴
Sc-44	Scandium (21)	1.415 × 10 ⁴	3.930 h	6.705 × 10 ¹⁷	1.812 × 10 ⁷
Sc-46		7.243 × 10 ⁶	8.383 × 10 d	1.253 × 10 ¹⁵	3.386 × 10 ⁴
Sc-47		2.886 × 10 ⁵	3.340 d	3.078 × 10 ¹⁶	8.318 × 10 ⁵
Sc-48		1.573 × 10 ⁵	4.370 × 10 h	5.528 × 10 ¹⁶	1.494 × 10 ⁶
Se-75	Selenium (34)	1.035 × 10 ⁷	1.198 × 10 ² d	5.377 × 10 ¹⁴	1.453 × 10 ⁴
Se-79		2.050 × 10 ¹²	6.500 × 10 ⁴ a	2.578 × 10 ⁹	6.967 × 10 ⁻²

TABLE AIII-1. (cont.)

Symbol of radionuclide	Element and atomic number	Half-life		Specific activity ^a	
		T _{1/2} (s)	T _{1/2} (m,h,d,a)	Bq/g	Ci/g
Si-31	Silicon (14)	9.432 × 10 ³	2.620 h	1.428 × 10 ¹⁸	3.858 × 10 ⁷
Si-32		3.311 × 10 ⁹	1.050 × 10 ² a	3.939 × 10 ¹²	1.065 × 10 ²
Sm-145	Samarium (62)	2.938 × 10 ⁷	3.400 × 10 ² d	9.800 × 10 ¹³	2.649 × 10 ³
Sm-147		3.343 × 10 ¹⁸	1.060 × 10 ¹¹ a	8.494 × 10 ²	2.296 × 10 ⁻⁸
Sm-151		2.838 × 10 ⁹	9.000 × 10 a	9.740 × 10 ¹¹	2.632 × 10
Sm-153		1.681 × 10 ⁵	4.670 × 10 h	1.623 × 10 ¹⁶	4.386 × 10 ⁵
Sn-113	Tin (50)	9.945 × 10 ⁶	1.151 × 10 ² d	3.714 × 10 ¹⁴	1.004 × 10 ⁴
Sn-117m		1.176 × 10 ⁶	1.361 × 10 d	3.034 × 10 ¹⁵	8.200 × 10 ⁴
Sn-119m		2.532 × 10 ⁷	2.930 × 10 ² d	1.386 × 10 ¹⁴	3.745 × 10 ³
Sn-121m		1.734 × 10 ⁹	5.500 × 10 a	1.989 × 10 ¹²	5.375 × 10
Sn-123		1.116 × 10 ⁷	1.292 × 10 ² d	3.040 × 10 ¹⁴	8.216 × 10 ³
Sn-125		8.329 × 10 ⁵	9.640 d	4.009 × 10 ¹⁵	1.084 × 10 ⁵
Sn-126		3.154 × 10 ¹²	1.000 × 10 ⁵ a	1.050 × 10 ⁹	2.839 × 10 ⁻²
Sr-82	Strontium (38)	2.208 × 10 ⁶	2.555 × 10 d	2.306 × 10 ¹⁵	6.232 × 10 ⁴
Sr-85m		4.060 × 10 ³	6.766 × 10 m	1.210 × 10 ¹⁸	3.269 × 10 ⁷
Sr-85		5.602 × 10 ⁶	6.484 × 10 d	8.766 × 10 ¹⁴	2.369 × 10 ⁴
Sr-87m		1.008 × 10 ⁴	2.800 h	4.760 × 10 ¹⁷	1.286 × 10 ⁷
Sr-89		4.368 × 10 ⁶	5.055 × 10 d	1.074 × 10 ¹⁵	2.902 × 10 ⁴
Sr-90		9.019 × 10 ⁸	2.860 × 10 a	5.142 × 10 ¹²	1.390 × 10 ²
Sr-91		3.427 × 10 ⁴	9.520 h	1.338 × 10 ¹⁷	3.617 × 10 ⁶
Sr-92		9.756 × 10 ³	2.710 h	4.651 × 10 ¹⁷	1.257 × 10 ⁷
T all forms		Tritium (1)	3.885 × 10 ⁸	1.232 × 10 a	3.581 × 10 ¹⁴
Ta-178	Tantalum (73)	5.586 × 10 ²	9.310 m	4.198 × 10 ¹⁸	1.135 × 10 ⁸
Ta-179		5.740 × 10 ⁷	1.820 a	4.063 × 10 ¹³	1.098 × 10 ³
Ta-182		9.936 × 10 ⁶	1.150 × 10 ² d	2.308 × 10 ¹⁴	6.239 × 10 ³
Tb-157	Terbium (65)	4.730 × 10 ⁹	1.500 × 10 ² a	5.620 × 10 ¹¹	1.519 × 10
Tb-158		4.730 × 10 ⁹	1.500 × 10 ² a	5.585 × 10 ¹¹	1.509 × 10
Tb-160		6.247 × 10 ⁶	7.230 × 10 d	4.176 × 10 ¹⁴	1.129 × 10 ⁴
Tc-95m	Technetium (43)	5.270 × 10 ⁶	6.100 × 10 d	8.337 × 10 ¹⁴	2.253 × 10 ⁴
Tc-96m		3.090 × 10 ³	5.150 × 10 m	1.407 × 10 ¹⁸	3.803 × 10 ⁷
Tc-96		3.698 × 10 ⁵	4.280 d	1.176 × 10 ¹⁶	3.178 × 10 ⁵
Tc-97m		7.690 × 10 ⁶	8.900 × 10 d	5.596 × 10 ¹⁴	1.512 × 10 ⁴
Tc-97		8.199 × 10 ¹³	2.600 × 10 ⁶ a	5.248 × 10 ⁷	1.418 × 10 ⁻³
Tc-98		1.325 × 10 ¹⁴	4.200 × 10 ⁶ a	3.216 × 10 ⁷	8.691 × 10 ⁻⁴
Tc-99m		2.164 × 10 ⁴	6.010 h	1.949 × 10 ¹⁷	5.267 × 10 ⁶
Tc-99		6.717 × 10 ¹²	2.130 × 10 ⁵ a	6.277 × 10 ⁸	1.696 × 10 ⁻²

TABLE AIII-1. (cont.)

Symbol of radionuclide	Element and atomic number	Half-life		Specific activity ^a		
		T _{1/2} (s)	T _{1/2} (m,h,d,a)	Bq/g	Ci/g	
Te-118	Tellurium (52)	5.184 × 10 ⁵	6.000 d	6.824 × 10 ¹⁵	1.844 × 10 ⁵	
Te-121m		1.331 × 10 ⁷	1.540 × 10 ² d	2.593 × 10 ¹⁴	7.007 × 10 ³	
Te-121		1.452 × 10 ⁶	1.680 × 10 d	2.377 × 10 ¹⁵	6.423 × 10 ⁴	
Te-123m		1.034 × 10 ⁷	1.197 × 10 ² d	3.281 × 10 ¹⁴	8.869 × 10 ³	
Te-125m		5.011 × 10 ⁶	5.800 × 10 d	6.664 × 10 ¹⁴	1.801 × 10 ⁴	
Te-127m		9.418 × 10 ⁶	1.090 × 10 ² d	3.490 × 10 ¹⁴	9.432 × 10 ³	
Te-127		3.366 × 10 ⁴	9.350 h	9.764 × 10 ¹⁶	2.639 × 10 ⁶	
Te-129m		2.903 × 10 ⁶	3.360 × 10 d	1.115 × 10 ¹⁵	3.012 × 10 ⁴	
Te-129		4.176 × 10 ³	6.960 × 10 m	7.748 × 10 ¹⁷	2.094 × 10 ⁷	
Te-131		1.080 × 10 ⁵	3.000 × 10 h	2.950 × 10 ¹⁶	7.974 × 10 ⁵	
Te-132		2.817 × 10 ⁵	3.260 d	1.123 × 10 ¹⁶	3.034 × 10 ⁵	
Th-227		Thorium (90)	1.617 × 10 ⁶	1.872 × 10 d	1.137 × 10 ¹⁵	3.073 × 10 ⁴
Th-228			6.023 × 10 ⁷	1.910 a	3.039 × 10 ¹³	8.215 × 10 ²
Th-229	2.315 × 10 ¹¹		7.340 × 10 ³ a	7.875 × 10 ⁹	2.128 × 10 ⁻¹	
Th-230	2.378 × 10 ¹²		7.540 × 10 ⁴ a	7.632 × 10 ⁸	2.063 × 10 ⁻²	
Th-231	9.187 × 10 ⁴		2.552 × 10 h	1.967 × 10 ¹⁶	5.316 × 10 ⁵	
Th-232	4.447 × 10 ¹⁷		1.410 × 10 ¹⁰ a	4.046 × 10 ³	1.094 × 10 ⁻⁷	
Th-234	2.082 × 10 ⁶		2.410 × 10 d	8.567 × 10 ¹⁴	2.315 × 10 ⁴	
Ti-44	Titanium (22)	1.492 × 10 ⁹	4.730 × 10 a	6.360 × 10 ¹²	1.719 × 10 ²	
Tl-200	Thallium (81)	9.396 × 10 ⁴	2.610 × 10 h	2.221 × 10 ¹⁶	6.003 × 10 ⁵	
Tl-201		2.627 × 10 ⁵	3.040 d	7.906 × 10 ¹⁵	2.137 × 10 ⁵	
Tl-202		1.057 × 10 ⁶	1.223 × 10 d	1.956 × 10 ¹⁵	5.285 × 10 ⁴	
Tl-204		1.192 × 10 ⁸	3.780 a	1.716 × 10 ¹³	4.639 × 10 ²	
Tm-167	Thulium (69)	7.983 × 10 ⁵	9.240 d	3.131 × 10 ¹⁵	8.462 × 10 ⁴	
Tm-168		8.044 × 10 ⁶	9.310 × 10 d	3.089 × 10 ¹⁴	8.348 × 10 ³	
Tm-170		1.111 × 10 ⁷	1.286 × 10 ² d	2.210 × 10 ¹⁴	5.973 × 10 ³	
Tm-171		6.055 × 10 ⁷	1.920 a	4.031 × 10 ¹³	1.090 × 10 ³	
U-230	Uranium (92)	1.797 × 10 ⁶	2.080 × 10 d	1.010 × 10 ¹⁵	2.729 × 10 ⁴	
U-232		2.173 × 10 ⁹	6.890 × 10 a	8.280 × 10 ¹¹	2.238 × 10	
U-233		5.014 × 10 ¹²	1.590 × 10 ⁵ a	3.573 × 10 ⁸	9.656 × 10 ⁻³	
U-234		7.726 × 10 ¹²	2.450 × 10 ⁵ a	2.309 × 10 ⁸	6.240 × 10 ⁻³	
U-235		2.220 × 10 ¹⁶	7.040 × 10 ⁸ a	8.001 × 10 ⁴	2.162 × 10 ⁻⁶	
U-236		7.379 × 10 ¹⁴	2.340 × 10 ⁷ a	2.397 × 10 ⁶	6.478 × 10 ⁻⁵	
U-238		1.410 × 10 ¹⁷	4.470 × 10 ⁹ a	1.244 × 10 ⁴	3.363 × 10 ⁻⁷	
V-48		Vanadium (23)	1.380 × 10 ⁶	1.597 × 10 d	6.302 × 10 ¹⁵	1.703 × 10 ⁵
V-49	2.851 × 10 ⁷		3.300 × 10 ² d	2.988 × 10 ¹⁴	8.075 × 10 ³	

TABLE AIII-1. (cont.)

Symbol of radionuclide	Element and atomic number	Half-life		Specific activity ^a	
		T _{1/2} (s)	T _{1/2} (m,h,d,a)	Bq/g	Ci/g
W-178	Tungsten (74)	1.875×10^6	2.170×10 d	1.251×10^{15}	3.380×10^4
W-181		1.047×10^7	1.212×10^2 d	2.202×10^{14}	5.952×10^3
W-185		6.489×10^6	7.510×10 d	3.477×10^{14}	9.398×10^3
W-187		8.604×10^4	2.390×10 h	2.594×10^{16}	7.012×10^5
W-188		5.996×10^6	6.940×10 d	3.703×10^{14}	1.001×10^4
Xe-127	Xenon (54)	3.146×10^6	3.641×10 d	1.045×10^{15}	2.824×10^4
Xe-131m		1.028×10^6	1.190×10 d	3.099×10^{15}	8.376×10^4
Xe-133		4.571×10^5	5.290 d	6.867×10^{15}	1.856×10^5
Xe-135		3.269×10^4	9.080 h	9.459×10^{16}	2.556×10^6
Y-87	Yttrium (39)	2.894×10^5	3.350 d	1.658×10^{16}	4.480×10^5
Y-88		9.210×10^6	1.066×10^2 d	5.150×10^{14}	1.392×10^4
Y-90		2.307×10^5	2.670 d	2.010×10^{16}	5.434×10^5
Y-91m		2.983×10^3	4.971×10 m	1.538×10^{18}	4.157×10^7
Y91		5.055×10^6	5.851×10 d	9.074×10^{14}	2.452×10^4
Y-92		1.274×10^4	3.540 h	3.560×10^{17}	9.622×10^6
Y-93		3.636×10^4	1.010×10 h	1.234×10^{17}	3.336×10^6
Yb-169	Ytterbium (70)	2.767×10^6	3.202×10 d	8.928×10^{14}	2.413×10^4
Yb-175		3.620×10^5	4.190 d	6.589×10^{15}	1.781×10^5
Zn-65	Zinc (30)	2.109×10^7	2.441×10^2 d	3.045×10^{14}	8.229×10^3
Zn-69m		4.954×10^4	1.376×10 h	1.221×10^{17}	3.301×10^6
Zn-69		3.336×10^3	5.560×10 m	1.813×10^{18}	4.901×10^7
Zr-88	Zirconium (40)	7.206×10^6	8.340×10 d	6.583×10^{14}	1.779×10^4
Zr-93		4.825×10^{13}	1.530×10^6 a	9.302×10^7	2.514×10^{-3}
Zr-95		5.532×10^6	6.403×10 d	7.942×10^{14}	2.147×10^4
Zr-97		6.120×10^4	1.700×10 h	7.031×10^{16}	1.900×10^6

^a The values of the specific activity do not include any daughter product contribution.

TABLE AIII-2. SPECIFIC ACTIVITY VALUES FOR URANIUM AT VARIOUS LEVELS OF ENRICHMENT

Mass per cent of uranium-235 present in uranium mixture	Specific activity ^a	
	Bq/g	Ci/g
0.45	1.8×10^4	5.0×10^{-7}
0.72 (natural)	2.6×10^4	7.06×10^{-7}
1.0	2.8×10^4	7.6×10^{-7}
1.5	3.7×10^4	1.0×10^{-6}
5.0	1.0×10^5	2.7×10^{-6}
10.0	1.8×10^5	4.8×10^{-6}
20.0	3.7×10^5	1.0×10^{-5}
35.0	7.4×10^5	2.0×10^{-5}
50.0	9.3×10^5	2.5×10^{-5}
90.0	2.2×10^6	5.8×10^{-5}
93.0	2.6×10^6	7.0×10^{-5}
95.0	3.4×10^6	9.1×10^{-5}

^a The values of the specific activity include the activity of uranium-234 which is concentrated during the enrichment process, and do not include any daughter product contribution. The values are for the material originating from natural uranium enriched by a gaseous diffusion method. If the origin of the material is not known the specific activity should be either measured or calculated using isotopic ratio data.

Appendix IV

QUALITY ASSURANCE IN THE SAFE TRANSPORT OF RADIOACTIVE MATERIAL

AIV.1. INTRODUCTION

AIV.1.1 General

It is the aim of the Regulations to achieve, through the application of effective quality assurance and compliance assurance programmes, the safety of the public and workers in the transport of radioactive material.

This appendix is based essentially upon Safety Series No. 50-C-QA [45]. However, since this reference relates to nuclear power plants and this appendix relates to the transport of radioactive materials, two additional main topics were added to accommodate the needs specific to transport. Subtopic areas have also been modified accordingly. Table AIV-1 compares the structure of this appendix with that of Ref. [45]. Two main topics which have been added for transport are: (10) Control of use and care of packages, and (14) Staff and training; the latter is subsection 3.3 of Ref. [45]. Also, the subsection headings in (2) Quality assurance programmes, (3) Organization, (7) Material control, (11) Non-conformity control, (13) Records, and (15) Audits are different from those in Ref. [45].

Where organizations have quality assurance programmes based upon the structure of Ref. [45], consideration should be given to changing the programme for transport activities to the structure shown in this appendix, since it provides the principles and objectives to be adopted when establishing both a satisfactory overall quality assurance programme for transport of radioactive materials and also separate quality assurance programmes for each of the constituent areas of activity. The principles in each case for each type of programme are the same and are to ensure that all requirements applicable to the package and shipment are properly met and that this can be demonstrated to any competent authority at any time during the useful life of a package.

The quality assurance principles described in this appendix may in many cases be implemented by more than one organization. Whether they are carried out by one or more than one organization is not of prime importance because their manner of implementation may vary from country to country and organization to organization. These variations will be due to such considerations as regulatory requirements, the general organization of industry, and the degree of sophistication and experience of the technical organizations involved in transporting radioactive materials. In any event, the basic intent of the principles should be kept in mind at all times and the detailed implementational procedures should be arranged accordingly.

TABLE AIV-1. COMPARISON OF CONTENTS OF APPENDIX IV WITH
SAFETY SERIES NO. 50-C-QA

Quality Assurance in the Safe Transport of Radioactive Material (Appendix IV)	Quality Assurance for Safety in Nuclear Power Plants (Safety Series No. 50-C-QA [45])
1. INTRODUCTION	1. INTRODUCTION
1.1. General	1.1. General
1.2. Scope	1.2. Scope
1.3. Responsibility	1.3. Responsibility
2. QUALITY ASSURANCE PROGRAMMES	2. QUALITY ASSURANCE PROGRAMMES
2.1. Introduction	2.1. General
2.2. Graded approach to quality assurance	2.2. Procedures, instructions and drawings
2.3. Relationship of grading to package type	2.3. Management review
2.4. User's guide	
3. ORGANIZATION	3. ORGANIZATION
3.1. Responsibility, authority and communication	3.1. Responsibility, authority and communications
3.2. Organizational interfaces	3.2. Organizational interfaces
	3.3. Staffing and training
4. DOCUMENT CONTROL	4. DOCUMENT CONTROL
4.1. Document preparation, review and approval	4.1. Document preparation, review and approval
4.2. Document release and distribution	4.2. Document release and distribution
4.3. Document change control	4.3. Document change control
5. DESIGN CONTROL	5. DESIGN CONTROL
5.1. General	5.1. General
5.2. Design interface control	5.2. Design interface control
5.3. Design verification	5.3. Design verification
5.4. Design changes	5.4. Design changes

TABLE AIV-1. (cont.)

Quality Assurance in the Safe Transport of Radioactive Material (Appendix IV)	Quality Assurance for Safety in Nuclear Power Plants (Safety Series No. 50-C-QA [45])
6. PROCUREMENT CONTROL	6. PROCUREMENT CONTROL
6.1. General	6.1. General
6.2. Supplier evaluation and selection	6.2. Supplier evaluation and selection
6.3. Control of purchased items and services	6.3. Control of purchased items and services
7. MATERIAL CONTROL	7. MATERIAL CONTROL
7.1. Identification and control of materials, parts and components	7.1. Identification and control of materials, parts and components
7.2. Handling, storage and shipping of materials, parts and components	7.2. Handling, storage and shipping
8. PROCESS CONTROL	8. PROCESS CONTROL
9. INSPECTION AND TEST CONTROL	9. INSPECTION AND TEST CONTROL
9.1. Programme of inspection	9.1. Programme of inspection
9.2. Test programme	9.2. Test programme
9.3. Calibration and control of measuring and test equipment	9.3. Calibration and control of measuring and test equipment
9.4. Indication of inspection, test and operating status	9.4. Indication of inspection, test and operating status
10. CONTROL OF USE AND CARE OF PACKAGES	
11. NON-CONFORMITY CONTROL	10. NON-CONFORMANCE CONTROL
	10.1. General
	10.2. Non-conformance review and disposition
12. CORRECTIVE ACTIONS	11. CORRECTIVE ACTIONS

TABLE AIV-1. (cont.)

Quality Assurance in the Safe Transport of Radioactive Material (Appendix IV)	Quality Assurance for Safety in Nuclear Power Plants (Safety Series No. 50-C-QA [45])
13.RECORDS	12.RECORDS
13.1. Collection, storage and preservation of quality assurance	12.1. Preparation of quality assurance records
13.2. Package records and logbook	12.2. Collection, storage and preservation of quality assurance records
14.STAFF AND TRAINING	
15.AUDITS	13.AUDITS
	13.1. General
	13.2. Scheduling
DEFINITIONS	DEFINITIONS

Quality assurance programmes are subject to review and audit by competent authorities. Responsible organizations should give all reasonable assistance to competent authorities and their agents in this work.

AIV.1.2. Scope

Quality assurance programmes should be established for the design, manufacture, testing, documentation, use, maintenance and inspection of all packages, and for transport, storage in transit, operations and safety assessment to ensure compliance with the relevant provisions of the Regulations irrespective of whether competent authority approval of the design or shipment is required. All activities such as cleaning, assembly, testing, commissioning, inspecting, maintaining, repairing, loading, unloading, modifying and decontamination should be covered.

The principles and objectives are applicable to all those responsible for the transport of radioactive materials, and other organizations participating in activities affecting quality.

AIV.1.3. Responsibility

The overall responsibility for the establishment and implementation of quality assurance programmes rests with the licensee or applicant for competent authority approval. In cases where an approval is not needed the responsible party for a shipment activity is also responsible for overall quality assurance. Some duties may be delegated to other organizations or persons within the responsibility of the above mentioned parties.

If it is not possible according to individual national practices to clearly identify one responsible party or organization then the constituent parts and interfaces of an overall quality assurance programme must be clearly understood and agreed by all parties and the competent authorities.

In the case of a purchased packaging, the purchaser should be made aware that the packaging has been designed, manufactured and tested according to a quality assurance programme complying with the requirements of this appendix.

In its responsibility for ensuring health and safety, the national government should decide whether, in addition to the transport Regulations, a general legal framework should be established, requiring an effective overall quality assurance programme for the transport of radioactive material.

AIV.2. QUALITY ASSURANCE PROGRAMMES

AIV.2.1. Introduction

An overall quality assurance programme should be established consistent with the requirements of this appendix and covering the various aspects of the safe transport of radioactive materials, e.g. packaging, packing, handling, storage and training of personnel. The arrangements should be subject to regular review by management as to the status and adequacy of activities for which they have responsibility. Measures should be included to remedy any deficiencies discovered or to introduce any improvements recommended.

The programme should be commensurate with the complexity of packaging or its components and the degree of hazard associated with the contents that may be carried by means of a graded system of measures. Items, activities and processes to which quality assurance programmes apply should be identified and appropriate methods or levels of control and verification assigned consistent with their importance for safety.

All programmes should ensure that the activities affecting quality are accomplished in accordance with written procedures, instructions or drawings of a type appropriate to the circumstances and that they include appropriate quantitative and/or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Procedures for implementing the quality assurance programmes on a planned and systematic basis for different phases of packaging and transport of radioactive materials should be developed and documented by the organization performing the constituent activities.

All measures established (see Sections AIV.2 to AIV.15 of this appendix) should be adequately documented and steps taken to ensure that persons performing the quality assurance function have an adequate knowledge of the language in which the programme is written. Translations of the documentation into other languages should be verified by competent persons referring to the original.

Irrespective of whether competent authority approval is required the manufacturer, consignor or user of any package should be prepared to provide facilities for competent authority inspection of the packaging during construction and use and to demonstrate to any cognizant competent authority that:

- (a) The construction methods and materials used for the construction of the packaging are in accordance with the design specifications;
- (b) All re-use packagings built to a design are periodically inspected and, as necessary, repaired and maintained in good condition so that they continue to comply with all relevant requirements and specifications, even after repeated use; and
- (c) All transport activities are performed in conformity with relevant requirements.

Where marking of items is specified it should be clear, unambiguous and durable and applied in such a manner as not to affect the function of the item. The marks should not be hidden by surface treatment or coatings unless other means of identification are provided.

AIV.2.2. Graded approach to quality assurance

Quality assurance applies to all packages, particularly packages as presented for transport, and also to shipping controls including pre-dispatch checks and maintenance of packages in a state of continued compliance with requirements.

The design and operational requirements of packages are interrelated and it is essential to identify the safety features of the package and items which collectively comprise a package to ensure that appropriate quality assurance procedures are applied and hence to minimize the probability of failure or malfunction of these items or components.

A logical system of grading packages or components of packages is related to their safety significance. GRADE 1 items are those essential to safety, GRADE 2 items are those with a significant impact on safety and GRADE 3 items are those with minor or no impact on safety. For example:

- GRADE 1 items would be those directly affecting package leaktightness or shielding, or for packages of fissile material, those directly affecting geometry and so affecting criticality control, etc.
- GRADE 2 items would be structures or components or systems whose failure could indirectly affect safety whereby the unsafe condition could only exist in combination with a secondary event or failure.
- GRADE 3 items would be those affecting structures, components or systems whose malfunction would not affect the packaging effectiveness and so would be unlikely to affect safety.

The following examples of detailed requirements illustrate the application of a graded approach to quality assurance.

GRADE 1 items

- (a) The design should be based on applicable industrial standards or codes, and design verification should be accomplished by design review, by prototype testing or by the use of calculations or computer codes.
- (b) The procurement documentation for materials or services should specify that only approved suppliers (see Section AIV.6.2) are used.
- (c) The manufacturing planning should specify traceability of raw materials and the use of certified welders and processes.
- (d) Test and inspection should require the use of qualified test methods and qualified inspectors to verify conformity with specified standards and codes.
- (e) Audits should only be carried out by qualified and nominated personnel.
- (f) Acceptance after manufacture and authorization for use of such items should only be carried out by the consignor or his nominated representative.

GRADE 2 items

- (a) The design should be based on applicable industrial codes and standards, and design verification may be through the use of calculations or computer codes.
- (b) Specified processes need to be carried out by certified personnel.
- (c) Tests and inspections should require the use of inspectors qualified to verify conformity with appropriate codes, standards or industrial specifications.
- (d) The lead auditors need to be properly qualified and nominated personnel.

GRADE 3 items

- (a) In general, the design needs to follow accepted engineering or industrial practice in which items would be standard ('off the shelf' or proprietary). All items would be subject to inspection to confirm acceptability for use.

AIV.2.3. Relationship of grading to package type

The level of quality assurance applied to a package can be commensurate with the hazard inherent to the radioactive contents.

The following guidance applicable to each category of package listed should be regarded as a guide and is not intended to rigorously cover all situations. However, it gives a general indication of the level of quality assurance to be aimed at. Clearly, higher grades than those suggested may be used, and should be considered especially for those packages designed for radioactive materials having other significant dangerous properties (subsidiary hazards), such as uranium hexafluoride.

AIV.2.3.1. Excepted packages and industrial package Type 1 (IP-1)

In determining the radioactive contents and package radiation level, instrumentation and processes used should be subject to quality assurance at Grade I. In all other aspects, such as design, manufacture, etc., Grade 3 should be applied.

AIV.2.3.2. Non-fissile Type A packages and industrial package Types 2 and 3

Matters affecting shielding integrity and containment should be subjected to quality assurance at Grade 1. All other matters should be subjected to Grade 2 except where there is a minimal effect on safety, in which case Grade 3 is appropriate.

AIV.2.3.3. Special form radioactive material

In all matters affecting compliance with the special form radioactive material requirements quality assurance at Grade 1 is appropriate.

AIV.2.3.4. Fissile packages (other than Type B packages)

In the case of criticality assessment and other factors affecting the assumptions in the criticality assessment, quality assurance at Grade 1 is appropriate. All other aspects should be treated as in Section AIV.2.3.2.

AIV.2.3.5. Type B packages (non fissile and fissile)

In all aspects contributing to the integrity of shielding and containment together with criticality safety (where applicable), quality assurance at Grade 1 is appropriate. All other aspects should be treated as in Section AIV.2.3.2.

AIV.2.3.6. Shipments and special arrangements

Quality assurance should be applied to shipments and special arrangements according to the individual features of each case.

AIV.2.4. User's guide

Guidance is given in Appendix V on identifying the various elements of quality assurance programmes which are of principal interest to the various parties involved in the transport of radioactive material.

AIV.3. ORGANIZATION

AIV.3.1. Responsibility, authority and communication

A documented organizational structure, with clearly defined functional responsibilities, levels of authority and lines of internal and external communication for management, direction and execution of the quality assurance programme should be established. The organizational structure and functional assignments should recognize that execution of a quality assurance programme established for each field of activities, for example, manufacturing and use, involves both performers and verifiers and is not the sole domain of a single group. The organization's structure and the functional assignments should be such that:

- (a) Attainment of quality objectives is accomplished by those who have been assigned responsibility for performing the work; this may include examination, checks and inspections of the work by the individuals performing the work; and
- (b) When verification of conformity to established requirements is necessary it is carried out by those who do not have direct responsibility for performing the work.

The persons and organizations ensuring that an appropriate quality assurance programme is established and effectively executed should have sufficient authority and organizational freedom to identify quality problems, to review all pertinent information, to initiate, recommend or provide solutions and, where necessary, to

initiate actions to control further processing, delivery or installation of an item which is non-conforming, deficient or unsatisfactory until proper compliance has been achieved. They should be sufficiently independent of cost and schedule considerations. The responsibility for the grading of quality assurance measures lies with the designer of a package or the person responsible for planning transport operations. Care should be taken to ensure that the grading of packages, components or operational controls is performed in accordance with the recommendations of Section AIV.2 and is acceptable to the competent authority.

AIV.3.2. Organizational interfaces

Where multiple organizational arrangements exist, the responsibility of each organization should be clearly established and interfaces and co-ordination among organizations ensured by appropriate measures with provision made for regular review and amendment if appropriate.

AIV.4. DOCUMENT CONTROL

AIV.4.1. Document preparation, review and approval

The preparation, review, approval and issue of documents essential to the performance and verification of the work, such as instructions, procedures and drawings, concerned with all activities affecting quality relating to design, manufacture, use, etc. of the packaging and transport operations, should be subject to control. Instructions, procedures and drawings should include appropriate qualitative and quantitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Responsible individuals and organizations should be clearly identified and have authority as specified in Section AIV.3.1.

AIV.4.2. Document release and distribution

A document release and distribution system should be established to make the documents readily available by means of up to date distribution lists. Measures should be provided for ensuring that those participating in an activity are aware of and use appropriate and up to date documents for performing the activity.

AIV.4.3. Document change control

Changes to documents should be recorded and be subject to review and approval in accordance with documented procedures by designated persons or organizations having relevant background information, and knowledge and under-

standing of the original document. Written information on document revision and status should be prompt and timely in its distribution. Care should be taken to ensure that out of date, redundant documents are destroyed, although an original document file should be maintained for establishing the history and assuring traceability.

AIV.5. DESIGN CONTROL

AIV.5.1. General

Design control measures consistent with Sections AIV.2 and AIV.3 should be established and documented to ensure that applicable specified design requirements, such as regulatory requirements, design bases, codes and standards are correctly translated into specifications, drawings, procedures or instructions. They should include provisions to ensure that applicable quality standards are specified and stated in design documents, including quantitative and qualitative acceptance criteria.

Changes and deviations, including concessions, from specified design requirements and quality standards should be justified, controlled and documented in a manner that is consistent with the quality assurance programme.

Measures should also be established for the selection, and for the review for suitability of application, of any materials, parts, equipment and processes that are essential to the function of the packaging, subassembly, systems or components relative to their operating environments.

Design activities should be documented to permit adequate evaluation by technical personnel other than those performing the original design and by the competent authority or its agents.

AIV.5.2. Design interface control

Methods consistent with Section AIV.3.2 should be established for communicating design information, including changes, in a controlled and documented manner across interfaces.

AIV.5.3. Design verification

Design control measures consistent with the requirements of Sections AIV.2 and AIV.3 should be established for verifying the adequacy of design, for example by the performance of design reviews or by the use of alternative calculational methods or by the performance of a suitable testing programme in accordance with the requirements of the Regulations. They should also include reviews of basic input data employed.

AIV.5.4. Design changes

Documented procedures should be established for effecting design changes, including in-service changes or modification in a manner consistent with the design control measures for the original design and Sections AIV.2 and AIV.3 of this appendix.

The full impact of changes should be carefully considered and the need, justifications and required actions documented.

Written information concerning the changes should be sent to all affected persons and organizations in a manner consistent with Section AIV.4.

AIV.6. PROCUREMENT CONTROL

AIV.6.1. General

Procurement control measures should be established and documented to ensure that applicable regulatory requirements, design bases, standards, specifications and other requirements necessary to assure adequate quality are included or referenced in the documents for procurement of items and services.

Procurement requirements for assuring quality should include those items specified in this appendix that are deemed applicable, with particular attention paid to the graded approach specified in Section AIV.2.

The procurer and the competent authority and their representatives should have access to plant facilities, items, materials and records for inspection and audit and have appropriate records forwarded when required for review or approval. These records should be retained for an appropriate time.

The provisions for quality should extend, as appropriate, to lower tier subcontractors and suppliers.

AIV.6.2. Supplier evaluation and selection

The selection of suppliers should be based on their evaluated and documented capability to provide items or services in accordance with the requirements of the procurement documents and this quality assurance appendix.

AIV.6.3. Control of purchased items and services

Purchased items and services should be subject to control to assure the conformity to procurement documents. Documentary evidence should be made available to the manufacturer, consignor or user, and the competent authority if required.

AIV.7. MATERIAL CONTROL

AIV.7.1. Identification and control of materials, parts and components

Material control measures should be established and documented for the identification and control of package structures, associated subassemblies, materials and components, including partially fabricated items, as required throughout fabrication, assembly and use.

AIV.7.2. Handling, storage and shipping of materials, parts and components

Material control measures should be established and documented to control handling, storage and shipping of all materials and components through the entire production process in order to prevent damage, deterioration or loss.

AIV.8. PROCESS CONTROL

Processes affecting quality, e.g. welding or heat treatment, should be controlled in accordance with this appendix and other documented requirements, e.g. codes, standards, specifications, criteria or special requirements. Where specified, measures should be taken to ensure that these processes are accomplished by qualified personnel using qualified procedures and equipment.

AIV.9. INSPECTION AND TEST CONTROL

AIV.9.1. Programme of inspection

A programme for inspection of items and services and the activities affecting their quality during manufacture, maintenance, repair and use should be established and executed by or for the organization performing these activities. This should include in-service inspections of packaging at designated intervals, training and equipment of personnel, condition and equipment of conveyances, and emergency procedures. Such inspections should be performed by qualified individuals other than those performing the activities, and the inspections should be of an adequate extent to assure quality and to verify conformity to documented instructions, procedures and drawings. Procedures should be established to ensure that packages found, during in-service inspection or otherwise, not to conform with all necessary requirements are removed from use until the deficiency is remedied.

Work should not proceed beyond any designated inspection point without prior specific documented approval.

AIV.9.2. Test programme

A programme should be established to ensure the adequacy of the specification, performance and documentation of all testing required to demonstrate that the package and its parts will perform satisfactorily in continued service.

Testing should be performed in accordance with written test procedures which incorporate the requirements and acceptance criteria specified in design documents, by trained personnel using properly calibrated instrumentation.

The test results should be recorded and evaluated to ensure that the requirements have been satisfied.

Tests should include those for materials, manufacture, compliance with the requirements for normal and accident conditions of transport, in-service inspection and maintenance.

AIV.9.3. Calibration and control of measuring and test equipment

Measures should be taken to ensure that tools, gauges, instruments, and other inspection, measuring and test equipment and devices used in determining conformity to acceptance criteria are of the proper range, type, accuracy and precision and are controlled, calibrated and adjusted at specified intervals to maintain accuracy. Suitable records should be maintained. Calibration should be traceable back to national or international standards, as appropriate.

When deviations beyond prescribed limits are detected, an evaluation should be made of the validity of previous measurements and tests, and acceptance of tested items reassessed.

Controls should be established to ensure proper handling, storing and use of calibrated instruments.

AIV.9.4. Indication of inspection, test and operating status

Test and inspection status of packagings or their parts should be identified by the use of markings, stamps, tags, labels, routing cards, inspection records, security seals or other means in accordance with Section AIV.2 that can indicate the acceptability or non-conformity of items with regard to tests and inspections performed. The identification of the inspection and test status should be maintained as necessary throughout manufacturing, maintenance, repair and use of the item, to ensure that only items that have passed the required inspection and test are used.

AIV.10. CONTROL OF USE AND CARE OF PACKAGES

Measures should be established and documented to control all aspects of transport operations including handling, labelling, dispatch, receipt, carriage and storage.

These measures should include, when applicable, identification and control of contents, cleaning, packing, preserving, and any special process controls, including monitoring of leaktightness, radiation and contamination levels relating to package material. These measures should minimize any safety hazards, prevent damage, deterioration or loss of contents, and enable compliance with the relevant regulations for packages or consignments to be confirmed.

AIV.11. NON-CONFORMITY CONTROL

Measures should be established to control items, processes and package contents which do not conform to the requirements, in order to prevent their inadvertent use or dispatch for transport. Non-conforming items should be identified by marking, tagging and/or by physical segregation, where practical, and in order to control further processing, delivery or assembly they should be handled in accordance with documented procedures. Such items should be reviewed and rejected, modified, repaired, reworked or accepted without modification. The responsibility for review and authority for disposal or acceptance of non-conforming items should be defined.

AIV.12. CORRECTIVE ACTIONS

The quality assurance programme should provide actions, consistent with Section AIV.2, to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective or incorrect material and equipment, and any other non-conformity are identified.

The quality assurance programme should provide that the cause of all such conditions is determined and corrective action is taken to prevent recurrence. The identification and cause of the condition, and the corrective actions taken should be documented and reported to appropriate levels of management.

AIV.13. RECORDS

AIV.13.1. Collection, storage and preservation of quality assurance records

A record system for the management of the quality assurance programme should be established and executed in accordance with written procedures and instructions and Section AIV.2 of this appendix. Records, including the results of reviews, inspections, tests, audits, monitoring of work performance, materials analyses as well as closely related data, such as qualifications of personnel, procedures and equipment, repairs required and other related documents, should be

maintained to furnish proper evidence of activities affecting quality. All records should be legible, complete and easily identifiable with respect to the item involved.

Accepted non-conforming items should be reported to the purchaser and, when necessary, to the competent authority. The description of changes, waivers or deviations that have been accepted should be documented to denote the 'as built' condition.

In the case of non-conformity affecting the package contents or part of the process, the reason for non-conformity should be established and should be reported and recorded as required, and appropriate measures taken to remedy the deficiency, including, if appropriate, approval of a concession.

The system should provide for identification, collection, indexing, filing, storing, maintenance and disposal of the records. They should be stored in such a way that they are readily retrievable and maintained in a suitable environment to minimize deterioration or damage or to prevent loss. Retention and disposal should be in accordance with written procedures which make provision for the maintenance of the record of the 'as built' condition by or for the responsible organization for the useful life of the item.

AIV.13.2. Package records and logbook

The owner or user of a package should establish and maintain service and maintenance records for each package. The records of quality assurance measures for each individual package should be maintained and may be referred to in a logbook which should be available for inspection. The logbook should contain appropriate references to the following information and records (in parentheses):

- (1) Competent authority approval of the package design and the individual package serial number (package design No. and package serial No.)
- (2) Operating and maintenance instructions (instruction ref. No.)
- (3) Certificate of conformity or commissioning certificate, including a summary of the applied test procedures (certificate No.)
- (4) Test procedure for re-inspection tests (procedure No.)
- (5) Certificates of re-inspection tests (certificate No.)
- (6) Movement or transport record of the package (actual record)
- (7) Authorized modifications to the package (modification Nos. or certificates)
- (8) Record of significant damage and subsequent repairs (damage/repair certificate Nos.)

When a package is to be serviced or maintained at a point remote from the location of the above detailed records then the owner or user should make available such information as may be required to satisfactorily accomplish the servicing or maintenance tasks.

AIV.14. STAFF AND TRAINING

All personnel responsible for performing activities affecting quality should be qualified on the basis of general education, experience and proficiency required for performing the specific assigned tasks.

In support of quality assurance programmes suitable training programmes and procedures, including maintenance of records, should be established to ensure proficiency is achieved and maintained.

AIV.15. AUDITS

A system should be established for internal and external audits to be carried out in accordance with written procedures on a regular basis to verify compliance with all aspects of the quality assurance programme and to determine the effectiveness of the programme.

Audits should be carried out by qualified persons selected for their independence from the activity under audit.

The documented audit and results should be reviewed by those responsible for the activity and action taken to rectify any deficiency identified or to introduce any improvement thought desirable.

Additional audits should be instituted when thought desirable, for example when significant organizational changes have occurred or in order to verify that corrective actions have been taken to remedy identified or suspected deficiencies.

DEFINITIONS

Audit

A documented activity performed to determine by investigation, examination and evaluation of objective evidence the adequacy of, and adherence to, established procedures, instructions, specifications, codes, standards, administrative or operational programmes and other applicable documents, and the effectiveness of implementation.

Competent Authority

Any national or international authority designated or otherwise recognized as such for any purpose in connection with the IAEA Regulations.

Compliance Assurance

A systematic programme of measures applied by a competent authority which is aimed at ensuring that the provisions of the IAEA Regulations are met in practice.

Documentation

Recorded or pictorial information describing, defining, specifying, reporting or certifying activities, requirements, procedures or results related to quality assurance.

Examination

An element of inspection, such as non-destructive testing and the investigation of materials, components, supplies or services to determine conformity with the specified requirements.

Inspection

A quality control action which by means of examination, observation or measurement determines the conformity of materials, parts, components, systems, structures, as well as processes and procedures, with predetermined quality requirements.

Logbook

A document which contains references to the history and status of packagings.

Qualified Person

A person who, having complied with specific requirements and met certain conditions, has been designated to discharge specified duties and responsibilities.

Quality Assurance

A systematic programme of controls and inspections applied by any organization or body involved in the transport of radioactive material which is aimed at providing adequate confidence that the standard of safety prescribed in the IAEA Regulations is achieved in practice.

Records

Documents which furnish objective evidence of the quality of items and/or activities affecting quality.

Responsible Organization/Party/Person

The organization/party/person having overall responsibility for one or more areas of transport (e.g. approval, manufacturing, shipment, in-transit storage).

Specification

A written statement, drawing or check-list of requirements to be satisfied by a product, a material or a process.

Appendix V

GUIDE FOR QUALITY ASSURANCE PROGRAMME

The chart below shows the main elements of a quality assurance programme (see Appendix IV) indicating those elements which are of principal interest to designers, manufacturers, users, carriers and safety assessors. It should be stressed that the table is not intended to exclude quality assurance provisions if no X appears in the appropriate box. For example, carriers may have no direct interest in quality assurance for package design but will clearly have an interest in quality assurance relating to vehicle design, as relevant.

Quality assurance element ^a	Designers ^b	Manufacturers ^c	Users ^d	Carriers ^e	Safety assessors ^f
1. Quality assurance programmes	X	X	X	X	X
2. Organization	X	X	X	X	X
3. Document control	X	X	X	X	X
4. Design control	X				X
5. Procurement control	X	X	X		X
6. Material control		X	X		X
7. Process control		X	X		X
8. Inspection and test control	X	X	X	X	X
9. Controls of use and care of packages	X		X	X	X
10. Non-conformity control	X	X	X	X	X

Quality assurance element ^a	Designers ^b	Manufacturers ^c	Users ^d	Carriers ^e	Safety assessors ^f
11. Corrective actions	X	X	X	X	X
12. Records	X	X	X	X	X
13. Staffing and training	X	X	X	X	X
14. Audits	X	X	X	X	X

^a The grade of quality assurance may be determined from Appendix IV for different types of package or package component.

^b It is assumed that the design function is independent of the manufacturers' function. However if manufacturing is controlled by a design office then additional X's would be required for Material control and Process control.

^c As above, it is assumed that design and manufacture are independent.

^d 'User' usually means 'consignor'.

^e It is assumed that the carrier is independent of all other groups.

^f This column applies to organizations in which an independent safety assessment (either internal or external) is carried out by safety assessors. This assessment may in fact cover the design, manufacture, maintenance, preparation and handling of packages and transport operations.

Appendix VI

EXAMPLE CALCULATIONS FOR ESTABLISHING MINIMUM SEGREGATION DISTANCE REQUIREMENTS

AVI.1. INTRODUCTION

Segregation is used in the Regulations for transport and storage in transit in three ways:

- (1) To separate radioactive material packages from places regularly occupied by people for providing adequate radiation protection (paras 205 and 460(a));
- (2) To separate radioactive material packages from packages of undeveloped photographic film for providing protection of the film from 'fogging' (paras 206 and 460(a)); and
- (3) To separate radioactive material packages from packages of other dangerous goods (paras 406 and 460(b)).

This appendix provides guidance on one way of developing criteria for segregating radioactive material packages from areas regularly occupied by transport workers and members of the public. A similar procedure can be used for developing criteria for protection of undeveloped film. A method for segregating radioactive material packages from other dangerous goods is briefly summarized in para. A-460.5.

Generally, modal transport authorities accomplish segregation for radiation protection by establishing tables of minimum segregation distances which are based upon the limiting values for dose required by para. 205 of the Regulations. Since these are limiting values, the resulting tables should be considered as minimum requirements, and greater segregation distances may be necessary to be consistent with the need for implementing operational requirements 'for keeping radiation exposures as low as reasonably achievable' (see paras 202 and 203).

The procedure outlined below is conservative in many ways. For example, the limiting values for dose from para. 205 are applied at the boundary to a regularly occupied area. Since persons will move around within the occupied area during the period when radioactive material packages are present, their resultant exposure will be less than the limiting values. Also, the limiting values are for whole body exposure. Since the centre of gravity of a person and accordingly the sensitive organs will always be located away from the boundary, the exposure will again be less than the limiting value. In addition, the limiting values for individual organs and tissues have not been considered in para. 205. If they had been specified, the values for individual organs and tissues would have been higher than those for the whole body (see, for example, paras 411 and 418 of Ref. [6]). Thus, individual organs and

tissues are well protected by the procedure of establishing segregation criteria based on whole body limits. Finally, the radiation levels used in the procedure are based on the transport index (TI) of a package or on the summation of the TIs in an array of packages. Thus, for arrays of packages, self shielding within the array is not considered and actual radiation levels will be lower than those upon which the calculations are based.

In order to establish minimum segregation distance requirements which will implement the radiation protection principles of the Regulations it is first necessary to develop a model of transport conditions for a given mode of transport. Numerous variables need to be considered in the development of a model. These considerations have been documented for previous calculations made for air transport [18, 46] and for sea transport [47]. Important parameters in such a model will include:

- (a) The maximum annual travel periods (MATP) for crew and for the critical groups of members of the public;
- (b) The radioactive traffic factors (RTF), defined as the ratio of the annual number of journeys made in company with Category II-YELLOW and Category III-YELLOW packages of radioactive materials⁴ to the annual total of all journeys;
- (c) The maximum annual exposure times (MAET), for both crew and public, which are the relevant MATP multiplied by the appropriate RTF, i.e.

$$\text{MAET (h/a)} = \text{MATP (h/a)} \times \text{RTF}; \quad (\text{VI.1})$$

- (d) The applicable dose equivalent limits (DEL) from para. 205 for crew⁵ and members of the public; and,
- (e) The reference dose equivalent rates (RDER) for crew and members of the public, which are used as the basis for establishing the minimum segregation distances and are derived by dividing the applicable dose equivalent limit by the applicable maximum annual exposure time, i.e.

$$\text{RDER (mSv/h)} = \text{DEL (mSv/a)}/\text{MAET (h/a)}. \quad (\text{VI.2})$$

The following provides an example of how segregation distances may be determined for the situations of passenger carrying and cargo-only aircraft. This example

⁴ Category I-WHITE packages are excluded from this because they present no essential radiation exposure hazard.

⁵ Members of the crew are defined as transport workers and therefore their dose equivalent limit is defined by para. 205(a).

is based upon a particular set of assumptions and calculational techniques. Assumptions used in such calculations need to be verified before they are used as a basis for regulatory control. Other calculational techniques are also possible.

Three possible configurations are considered as follows:

- (a) Below main deck stowage in a passenger aircraft of radioactive material packages in a single group;
- (b) Below main deck stowage in a passenger aircraft of radioactive material packages in multiple groups with prescribed spacing distances between groups; and
- (c) Main deck stowage on either a combined cargo/passenger aircraft (known in the airline industry as a 'Combi' aircraft) or a cargo aircraft.

In the following calculations, all packages and groups of packages are treated as single point sources whose radiation levels can be described by the inverse square relationship. Experience has shown this to be a reasonable working assumption for small size packages usually transported by air. Consideration of the details of package dimensions and of the stowage configurations will generally lead to a small decrease in the segregation distance required; thus, treating all groups of packages as single point sources is conservative.

AVI.2. BELOW MAIN DECK STOWAGE OF ONE GROUP IN PASSENGER AIRCRAFT

In a typical passenger carrying aircraft packages are loaded in a cargo compartment directly below the passenger compartment. The reference dose equivalent rate is assumed to occur at seat height. This is conservative since the occupant's centre of gravity and accordingly individual sensitive organs (gonads, other reproductive organs, major body organs) will be above this level and they would also have higher DEL values if individually considered (paras 411 and 418 of Ref. [6]), and the tissues of the lower limbs, although below seat height, also are much less sensitive to radiation exposure and would have higher DEL values if individually considered. The highest radiation level would be experienced by a passenger located in a seat directly above a package or group of packages of radioactive material. All other passengers would be exposed at lower levels. This situation is depicted in Fig. AVI-1.

The actual minimum distance (AMD) of segregation needed between a source within a package (or group of packages) from the point of interest (representing a passenger) on a typical aircraft will be the sum of the required segregation distances (S , in m) between the package and the passenger compartment boundary, the height of the seat (although the actual seat height in most aircraft would be approximately

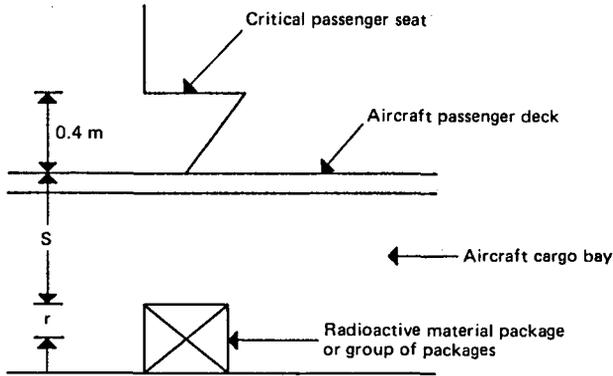


FIG. AVI-1. Typical configuration of passenger and cargo in passenger aircraft, used for determining the segregation distance S .

0.5 m, it is conservatively assumed to be 0.4 m here) and the radius of the package (r , in m):

$$AMD = S + 0.4 + r \tag{VI.3}$$

The transport index provides an accurate measure of the maximum radiation level at 1 m from the package surface. For fissile material packages, the radiation level could be less than the value calculated using the TI. In order to use the SI radiological units of measurements, the TI needs to be divided by 100. Hence, the inverse square law gives:

$$RDER = (TI/100) (TF_f) (1.0 + r)^2 / (AMD)^2 \tag{VI.4}$$

where:

- RDER is the radiation level at seat height (mSv/h)
- TI is the transport index which, when divided by 100, is an expression of the radiation level at 1 m from the package surface (mSv/h)
- TF_f is the transmission factor of the passenger compartment floor, the fraction of radiation which passes through the aircraft structures between the source and the dose point (dimensionless)
- r is the radius of a package or a collection of packages (half of the minimum dimension) (m)
- AMD is the actual minimum distance to dose point (m).

Substitution of Eq. (VI.3) into Eq. (VI.4) yields:

$$RDER = (TI/100) (TF_p)^2 / (S + 0.4 + r)^2 \quad (VI.5)$$

Solving for S, we get:

$$S = [(TI \times TF_p)/(100 \times RDER)]^{1/2} (1 + r) - (r + 0.4) \quad (VI.6)$$

The transmission factor (TF_p) varies with the energy of the radiation emitted from the package and the aircraft floor construction. Typical transmission factors range from 0.7 to 1.0. The combinations of transport index, transmission factor, and package size shown in Table AVI-1 were selected as conservative but realistic models.

TABLE AVI-1. TRANSMISSION FACTORS

Transport index (TI)	Transmission factor (TF_p)	Package radius (r)
0-1.0	1.0	0.05 m
1.1-2.0	0.8	0.1 m
2.1-50	0.7	0.4 m

The reference dose equivalent rate (RDER) is determined from Eqs (VI.1) and (VI.2). In a study by the United States Atomic Energy Commission [46], it was found that the radioactive traffic factor (RTF) for 20 major US airports in 1973 was approximately 1 in 30. To be conservative, and to account for changes which have occurred in passenger traffic patterns since 1973, it is assumed that RTF is 1 in 10. Data need to be developed to establish an internationally applicable value of RTF for the development of sound segregation tables. It is estimated that regular commuters such as sales persons may fly 500 hours each year, hence the MATP for the critical group is assumed to equal 500 h/a.

Thus, from Eq. (VI.1) we get

$$MAET = (500 \text{ h/a}) \times (0.1) = 50 \text{ h/a.}$$

The applicable DEL, from para. 205(b) of the Regulations is 1.0 mSv/a; and thus the applicable RDER, from Eq. (VI.2) is

$$RDER = (1 \text{ mSv/a})/(50 \text{ h/a}) = 0.02 \text{ mSv/h.}$$

TABLE AVI-2. VARIATION OF SEGREGATION DISTANCE WITH TRANSPORT INDEX FOR A SINGLE GROUP OF PACKAGES STOWED BELOW MAIN DECK ON A PASSENGER AIRCRAFT

Total of TIs for packages in the group	Vertical segregation distance (top of group of packages to floor of main deck (m))	
	Calculated here ^a	In 1986 ICAO Technical Instructions ^b
1.0	0.29	0.30
2.0	0.48	0.50
3.0	0.63	0.70
4.0	0.86	0.85
5.0	1.05	1.00
6.0	1.23	1.15
7.0	1.39	1.30
8.0	1.54	1.45
9.0	1.68	1.55
10.0	1.82	1.65

^a Calculated using Eq. (6) and assumptions outlined in this appendix.

^b ICAO Technical Instructions for the Safe Transport of Dangerous Goods by Air (see Ref. [18]).

For below main deck stowage on passenger aircraft the exposure to pilots should be minimal because of the location of the cockpit relative to the cargo areas.

With these assumptions, Eq. (VI.6) is used to calculate the segregation distances shown in column two of Table AVI-2. Also shown for comparison are the segregation values used in the 1986 edition of the International Civil Aviation Organization's Technical Instructions [48]. For use in international transport organization regulations, values such as these are often rounded for convenience.

To demonstrate the conservatism provided by this method for tissue and individual organs, tissue at deck level can be considered. For the lower limbs, consistent with the philosophy of para. 418 of IAEA Safety Series No. 9 [6] (which requires that no single organ should be exposed to 10 times the whole body dose), the DEL would be increased to 10 mSv/a, resulting in an RDER of 0.2 mSv/h. If this value is substituted into Eq. (VI.6) and the distance value of '0.4' is deleted, it can be seen that the segregation distances specified in Table AVI-2 are more

than adequate. For example, for a TI of 10 the segregation distance for tissue at floor level is only 0.09 m whereas the value in Table AVI-2 is a factor of 20 greater, i.e. 1.82 m.

For children in the critical group, two factors should be considered, the expected MATP and the possible presence on the floor of the aircraft during flight of these individuals. It would be expected that the MATP for these individuals would be significantly less than 500 h/a (although accurate data are not available for this), and the MATP for their possible presence on the floor of the aircraft would be much lower still. Therefore the segregation distances shown in Table AVI-2 are adequate also for children.

AVI.3. BELOW MAIN DECK STOWAGE OF MULTIPLE GROUPS IN PASSENGER AIRCRAFT

It should be noted that the calculated vertical segregation distance of 1.05 m for a single package or group of packages with a TI of 5 can be obtained in most aircraft, but that for many aircraft it would be impossible to obtain a vertical segregation distance above 1.6 m. This would limit the total TI in one group of packages which could be placed on a passenger aircraft. To increase the total TI which can be carried on a passenger aircraft, it would be necessary to space the packages or groups of packages within the belly cargo compartments of the aircraft. A configuration of five groups of packages, each having a different total TI value, with equal spacing distance S' between groups is depicted in Fig. AVI-2. The highest radiation level for passengers would be at the seat directly above the centre group of packages.

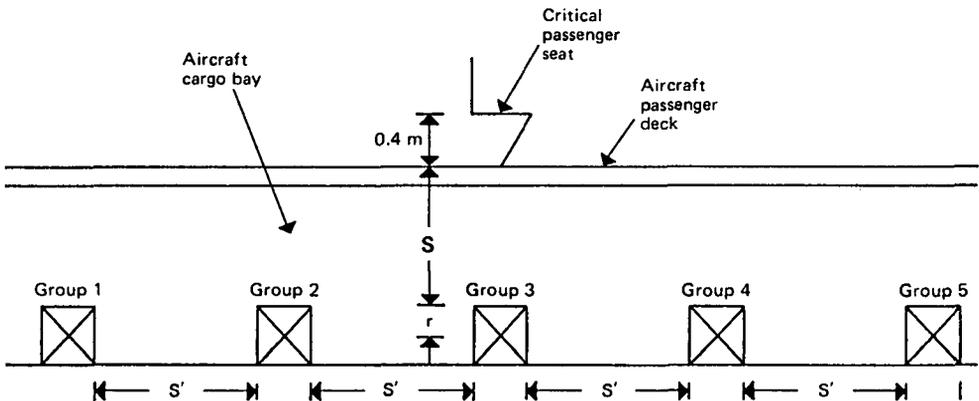


FIG. AVI-2. Typical configuration of passenger and special cargo in passenger aircraft, used for determining the segregation distance S and spacing distance S' .

For a configuration such as that shown in Fig. AVI-2, the inverse square law gives:

$$\text{RDER} = \text{TF}_f \sum_{i=1}^5 (\text{TI}_i/100) (1.0 + r_i)^2 / (\text{AD}_i)^2 \quad (\text{VI.7})$$

If it is assumed that

$$\text{TI}_i = 4, i = 1 \text{ to } 5$$

$$r_i = 0.4 \text{ m}, i = 1 \text{ to } 5$$

$$\text{TF}_f = 0.7$$

$$\text{then RDER} = 0.02 \text{ mSv/h}$$

It is noted that

$$\text{AMD}_1 = \text{AMD}_5 = \sqrt{(r + S + 0.4)^2 + (4r + 2S')^2} \quad (\text{VI.8})$$

$$\text{AMD}_2 = \text{AMD}_4 = \sqrt{(r + S + 0.4)^2 + (2r + S')^2}$$

$$\text{AMD}_3 = r + S + 0.4$$

Equations (VI.7) and (VI.8) combine to give one equation with two unknowns, S and S' . Various combinations of S and S' would allow a consignment of packages having a total TI of 20 to be carried with a segregation distance S less than 2.9 m. For example, placing the five groups, each with a total TI of 4, as shown in Fig. AVI-2, a segregation distance S of 1.6 m with a spacing distance S' of 2.11 m would give a maximum radiation level at seat height of 0.02 mSv/h. Thus various combinations of segregation and spacing will safely control the radiation exposure of passengers for large TI consignments.

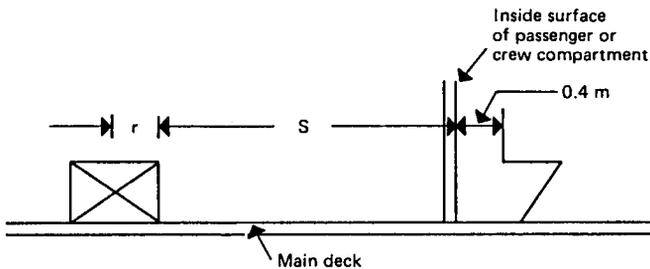


FIG. AVI-3. Typical configuration of main deck stowage on a Combi or cargo aircraft.

TABLE AVI-3. VARIATION OF SEGREGATION DISTANCE WITH TRANSPORT INDEX FOR MAIN-DECK STOWAGE ON A COMBI OR CARGO AIRCRAFT

Total of TIs for packages in the group	Horizontal segregation distance (forward face of group of packages to inside wall of occupied compartment (m))
1.0	0.29
2.0	0.48
5.0	1.18
10.0	2.00
20.0	3.16
30.0	4.05
40.0	4.80
50.0	5.46
100.0	8.05
150.0	10.04
200.0	11.72

AVI.4. MAIN DECK STOWAGE ON COMBI OR CARGO AIRCRAFT

For this condition, all parameters previously assumed are used, except TF_w (transmission factor for the wall of an occupied compartment) is assumed (without verification) to be greater than or equal to 0.8.

For the crew, the following assumptions⁶ are made:

$$MATP = 1000 \text{ h/a}$$

$$RTF = 1/4$$

$$MAET = (1000 \text{ h/a}) \times (1/4) = 250 \text{ h/a}$$

$$DEL = 5.0 \text{ mSv/a (from para. 205(a) of the Regulations)}$$

$$RDER = (5.0 \text{ mSv/a}) / (250 \text{ h/a}) = 0.02 \text{ mSv/h}$$

⁶ The values of MATP and RTF assumed here for crew members have not been verified for actual flight situations.

The MATP and MAET values used before for passengers in passenger aircraft are used here also. With these assumptions, the calculations for passengers in a Combi and for crew in a cargo aircraft will result in the same segregation distances.

The situation for Combi or cargo aircraft is depicted in Fig. AVI-3. The minimum horizontal distance between the seat back of a seated person and the inside wall of the occupied compartment is also assumed to be 0.4 m. This is probably a conservative value because, if the cargo is forward, the passenger's feet will be against the partition; and if the cargo is aft, there will usually be instruments, a galley, toilets or at least luggage or seat-reclining space between the partition and the rear seat. For this situation Eq. (VI.3) applies for AMD, and

$$S = [(TI \times TF_w)/(100 \times RDER)]^{1/2} (1 + r) - (r + 0.4) \quad (VI.9)$$

The calculated segregation distances for Combi and cargo aircraft are shown in Table AVI-3.

AVI.5. SEGREGATION DISTANCES FOR UNDEVELOPED FILM

An approach similar to that described above may be used for determining segregation distance requirements for packages marked as containing undeveloped film. However, instead of modelling the time of exposure for repetitive trips a single trip is considered. For this single trip a maximum allowed dose of 0.1 mSv (10 mrem), see para. 206, is normally used to calculate the segregation distance (S) for given transit times.

Appendix VII

ACCELERATION VALUES AND CALCULATION METHODS FOR PACKAGE TIE-DOWN FORCES

AVII.1. ACCELERATIONS

Each mode of transport will be subjected to certain accelerations which could be expected to occur under normal conditions of transport. For example, in starting from rest a forward acceleration will occur, braking would give a rearward acceleration and, because of the centrifugal effect, turning while in motion would produce a sideways acceleration. Vertical acceleration would be caused by bumps and waves or in the case of aircraft in climbing or descending. Below certain expected acceleration values which could be considered as normal, the package should remain secure on the conveyance. Above this acceleration the package is allowed to separate from the conveyance by the breakage of its restraint, but it still needs to meet the test requirements.

In considering normal conditions, there are some factors of transport which are undesirable, but sufficiently common to be called mishaps rather than accidents. These would produce moderate forces on the tie-downs. Examples of these are: for road, skids, the impact into soft surfaces (such as snow drifts or sand patches) and the impact into hard structures at very low velocity; for rail, the impacts and changes of direction experienced in marshalling, humping and shunting; and for aircraft, high velocity gusts and emergency landing conditions.

National and international modal standards and regulations need to be consulted to obtain the mandatory or recommended acceleration and load factors, together with any special conditions for transport. Many packages, however, are designed to be able to be used in many countries and with several transport modes. Table AVII-1 gives an indication of the magnitude of the load factors (or accelerations) which might be used for the design of attachments for a package. The values given for each mode would be in accordance with most national and international regulations. The final row of the table ('Radioactive material packages in the USA') is shown because a larger lateral load factor is required than for the other standards examined. Specific examples of acceleration values are given in Refs [48-50].

AVII.2. EXAMPLE CALCULATION OF PACKAGE TIE-DOWN REQUIREMENTS

Precise calculations of the tensions in the tie-down members arising from accelerations assumed to act simultaneously in different directions are complex. A generalized method which gives acceptable results for asymmetric packages and tie-downs is given in Ref. [48]. In practice most packages are symmetrical and can be

TABLE AVII-1. LOAD FACTORS FOR ENVELOPE DESIGN^a

Mode	Longitudinal	Lateral	Vertical ^b
Road	+ 2g	± 1g	3g down, 2g up
Rail	± 10g	± 2g	± 4g
Sea ^c	± 2g	± 2g	± 2g
Air	9g, -1.5g aft	± 1.5g	6g down ^d , 2g up
Radioactive material packages in the USA ^e	10g	5g	2g

^a Load factors are non-dimensional (i.e. multiples of the acceleration due to gravity, g) but may be considered as accelerations when multiplied by the gravitational constant g (9.81 m/s²). The direction of the load (and load factor) will be in the opposite sense to the acceleration causing it.

^b Vertical accelerations include gravity and are actual absolute factors experienced by the package.

^c These values are for general sea transport. For domestic barge transport, Ref. [51] recommends for dynamic loadings: 1.5g acceleration longitudinally and 0.5g laterally; and for wave motion (not to be superimposed on the dynamic accelerations), two cases are cited:

- for head seas, 1.4g acceleration longitudinally and 2g dynamic and 1g static vertically; and
- for beam seas, 1.6g acceleration in the transverse direction and similarly 2g dynamic and 1g static vertically.

^d The vertical factor in an aircraft depends on the pitch acceleration of the type of aircraft when subjected to the maximum gust condition and the position of the cargo relative to the aircraft's centre of gravity. The values shown are maximum for most modern transport aircraft.

^e These values are required for US design of packages, which are supplied with structural attachments for tie-down (see 10CFR part 71.45, Federal Register, August 5, 1983, Vol. 48, No. 152).

generally considered to be of cubic or cylindrical shape and the computational methods for these particular cases is given below. The effect of not chocking a package of cubic shape is also considered.

AVII.2.1. Cubic package — chocked

The package mass is M. All dimensions, X, Y and Z are equal and the centre of gravity is at the point X/2, Y/2, Z/2. The angles ϕ are equal and in the plane of the tie-down member. The angles α are equal.

The package is restrained symmetrically by four tie-down members 1, 2, 3 and 4, as shown in Fig. AVII-1. The tensions in the ties are respectively P₁, P₂, P₃ and P₄. The vehicle accelerations are a_x, a_y and a_z.

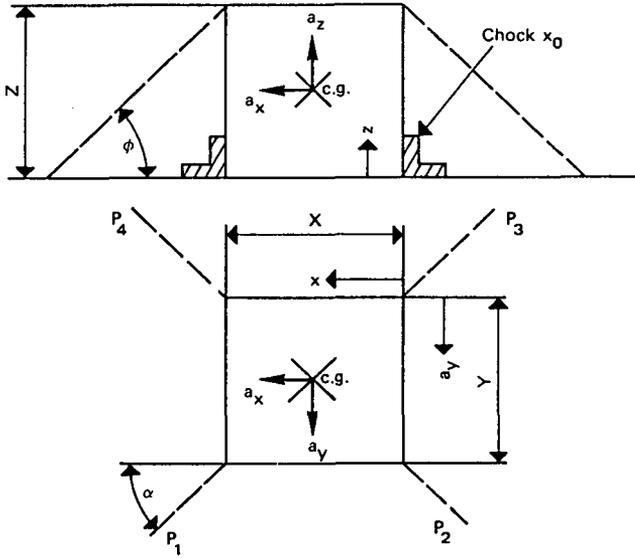


FIG. AVII-1. Cubic package, chocked (from Ref. [48]).

Forces on tie-down members

Tie-down members 1 and 4 resist acceleration a_x , thus the tension $P_{1x} = P_{4x}$. Tie-down members 1 and 2 resist acceleration a_y , thus the tension $P_{1y} = P_{2y}$. The maximum tension in tie-down 1: $P_1 = P_{1x} + P_{1y}$.

Consider the acceleration a_x . Taking moments about the tipping edge ($x = 0$), we get

$$Ma_x \frac{Z}{2} = 2Z P_{1x} \cos \phi \cos \alpha + 2Z P_{1x} \sin \phi + Mg \frac{X}{2} - Ma_z \frac{X}{2}$$

$$P_{1x} = \frac{M(a_x + a_z - g)}{4(\cos \phi \cos \alpha + \sin \phi)}$$

Similarly, for the acceleration a_y we have

$$P_{1y} = \frac{M(a_y + a_z - g)}{4(\cos \phi \sin \alpha + \sin \phi)}$$

Consider the following example. Assuming that

$$a_x = a_y = 2g \text{ and } a_z = 1g$$

the angles $\phi = 45^\circ$ and $\alpha = 45^\circ$.

The load in the tie-down

$$\begin{aligned} P &= 0.414 Mg + 0.414 Mg \\ &= 0.828 Mg \end{aligned}$$

Forces on chocks

We neglect the effect of friction, which is indeterminate and would be small if $a_z = 1g$. We consider the horizontal forces acting on the chocks due to a vehicle acceleration a_x . The package is restrained by tie-downs 1 and 4 and chock X. Since the package and tie-downs are symmetrical, $P_1 = P_4$, and

$$Ma_x = 2P_1 \cos \phi \cos \alpha + F_c$$

where F_c is the force on the chock. We assume that $a_x = 2g$ and $a_z = 1g$ as above. Thus, the tension in tie-down 1 (considering only accelerations a_x and a_z) is

$$P_1 = P_{1x} = 0.414 Mg$$

and $F_c = Ma_x - 2P_1 \cos \phi \cos \alpha$

$$F_c = 1.586 Mg$$

AVII.2.2. Cubic package — not chocked

Figure AVII-1 applies but without chocks. The assumptions and nomenclature used for Fig. AVII-1 also apply.

Tie-down members 1 and 4 resist acceleration a_x ; therefore $P_{1x} = P_{4x}$. Tie-down members 1 and 2 resist acceleration a_y ; therefore $P_{1y} = P_{2y}$. All tie-down members resist acceleration a_z ; therefore $P_{1z} = P_{2z} = P_{3z} = P_{4z}$. The maximum tension in tie-down 1: $P_1 = P_{1x} + P_{1y} + P_{1z}$.

It should be noted that the vertical components of the tensions in the tie-downs which arise from the vehicle accelerations will add to the friction load between the package and the conveyance platform.

Consider the vehicle acceleration a_x . The coefficient of friction between package and platform is μ . With angles ϕ all the same and α all the same, the tension will be equally shared by tie-downs 1 and 4. The vertical load imparted to the platform is $2P_{1x} \sin \phi + M(g - a_z)$. The horizontal loads:

$$Ma_x = 2P_{1x} \cos \phi \cos \alpha + \mu[2P_{1x} \sin \phi + M(g - a_z)]$$

$$P_{1x} = \frac{M[a_x - \mu(g - a_z)]}{2(\cos \phi \cos \alpha + \mu \sin \phi)}$$

Similarly, for the acceleration a_y we have

$$P_{1y} = \frac{M[a_y - \mu(g - a_z)]}{2(\cos \phi \sin \alpha + \mu \sin \phi)}$$

Considering the acceleration a_z

$$M(a_z - g) = 4P_{1z} \sin \phi$$

$$P_{1z} = \frac{M(a_z - g)}{4 \sin \phi}$$

Consider the following example. Assume that

$$a_x = a_y = 2g$$

$$a_z = 1g$$

The angles $\alpha = 45^\circ$ and $\phi = 45^\circ$ and

$$P_{1x} = \frac{2 Mg}{2(0.5 + 0.707\mu)}$$

$$P_{1y} = \frac{2 Mg}{2(0.5 + 0.707\mu)}$$

$$P_{1z} = 0$$

Assume that $\mu = 0.4$, then the tension in tie-down 1:

$$P_1 = 1.28 Mg + 1.28 Mg + 0 = 2.56 Mg$$

If friction is ignored, $\mu = 0$ and the tension in tie-down 1:

$$P_1 = 2 Mg + 2 Mg + 0 = 4 Mg$$

The tension is almost a factor of 5 greater than the equivalent chocked case considered in Section AVII.2.1.

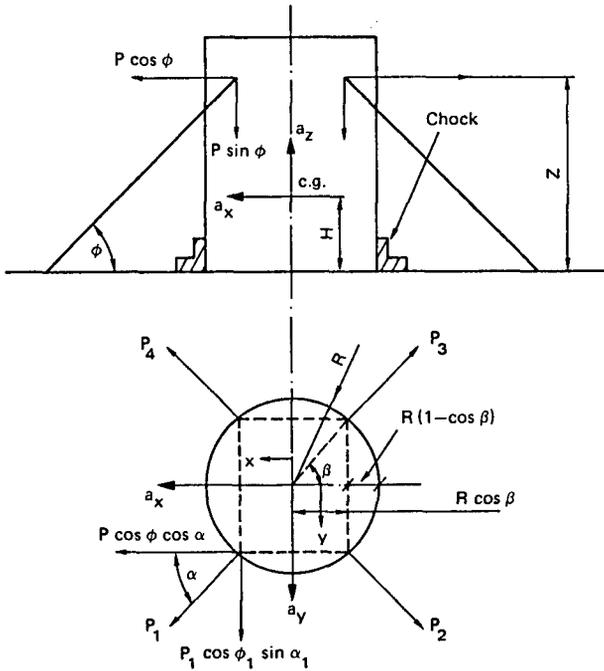


FIG. AVII-2. Cylindrical package, chocked [48].

AVII.2.3. Cylindrical package — chocked

We will calculate the maximum tension in tie-down 1 arising from vehicle accelerations a_x , a_y and a_z . From the method for a cubic package in Section AVII.2.1 above, it follows that

$$P_1 = P_{1x} + P_{1y}$$

$$P_{1x} = \frac{M (a_x H + a_z R - gR)}{2 [Z \cos \phi \cos \alpha + R \sin \phi (1 + \cos \beta)]}$$

$$P_{1y} = \frac{M (a_y H + a_z R - gR)}{2 [Z \cos \phi \sin \alpha + R \sin \phi (1 + \cos \beta)]}$$

The force on the chock $F_c = Ma_x - 2P_1 \cos \phi \cos \alpha$

Consider the following example. Assume that

$$H = R = \frac{Z}{2}$$

$$a_x = a_y = 2g \text{ and } a_z = 1g$$

Then the angles $\phi = \alpha = \beta = 45^\circ$, and

$$P_1 = P_{1x} + P_{1y} = 0.454 \text{ Mg} + 0.454 \text{ Mg} = 0.908 \text{ Mg}$$

The force on the chock is $2 \text{ Mg} - 0.454 \text{ Mg} = 1.546 \text{ Mg}$.

Appendix VIII

EXAMPLE OF A RADIATION PROTECTION PROGRAMME FOR EXCLUSIVE USE VESSELS

The following provisions are reproduced from the Manual of Radiation Control of a Vessel Exclusively Carrying Spent Nuclear Fuel, prepared in Japan. It should be noted that this programme complies in all respects with the requirements of the Regulations but also includes specific additional measures required by the authorities in Japan.

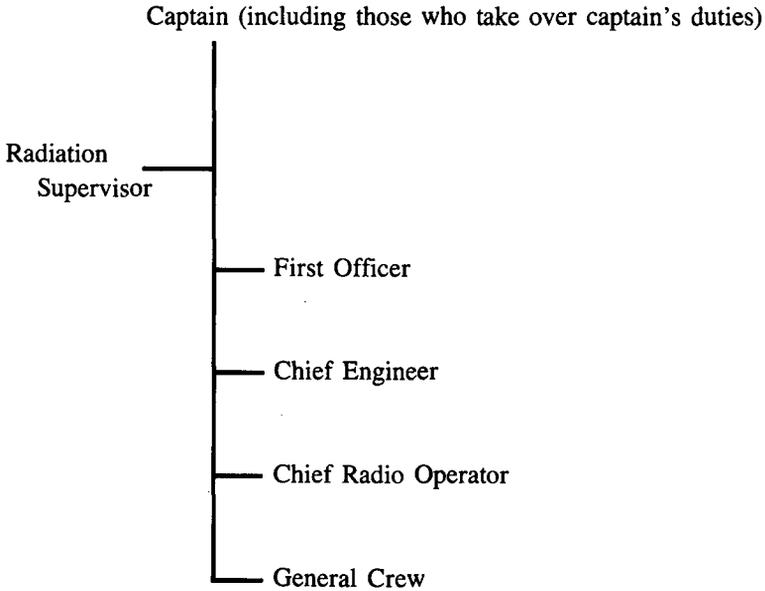
(General)

Art. 1. The object of this manual on operating vessels exclusively carrying spent nuclear fuel is to ensure the safety of the vessel operation and the maintenance of safety and health of crew, by prescribing the items necessary for vessel operation and for radiation control during loading and unloading of radioactive packages, in accordance with Art. 16-2 of 'Regulations for Carriage and Storage of Dangerous Goods in Ships'.

(System of Radiation Control)

- Art. 2.**
1. In connection with vessel operation, a qualified radiation supervisor is to be on board the vessel.
 2. The duties of the radiation supervisor are to conduct the radiation control services and aid the captain with regard to the radiation control by performing the following practices:
 - (1) with regard to the vessel operation and loading/unloading operation, he shall propose positive measures to the captain, when necessary, for radiation control, in compliance with the request of the captain, or on his own judgement. In this case, the captain shall respect his opinions.
 - (2) with regard to the radiation control duties with which he is entrusted by the captain, he shall give instruction and guidance directly to the crew;
 - (3) he shall perform the measurements of external radiation dose rates in the vessel, the access control of the access-restricted area of the vessel and the exposure control of the crew;
 - (4) he shall maintain and manage the radiation measuring instruments and protective clothing in the vessel; and
 - (5) he shall participate in planning radiation control and, if necessary, he shall execute the plan of his own accord.

TABLE AVIII-1. SAFETY CONTROL ORGANIZATION



- Captain:** responsible for the radiation control.
- Radiation Supervisor:** assists captain with regard to radiation control, aids captain in executing radiation monitoring practices, gives instruction and guidance directly to crew with regard to matters entrusted by the captain, manages the radiation exposure records of crew.
- First Officer:** assists captain with regard to radiation control.
- Chief Engineer:** performs services in connection with operation and maintenance of cask cooling, hold ventilation, fire fighting and emergency power generation devices.
- Chief Radio Operator:** performs services in connection with emergency communication.
- General Crew:** when necessary, assist in radiation control.

TABLE AVIII-2. FREQUENCY OF RADIATION MEASUREMENT

Measurement	Frequency	Number of measuring points	Instrument	Remarks
Area gamma radiation	Continuous	5	Area monitor	3 in hold 1 in accommodation space 1 in refrigerator room
	Regularly	Once per day	10	Ion chamber type survey meter 2 on hatch cover 3 on upper deck 1 in engine room 1 in captain's room Scintillation type survey meter 1 in accommodation room 1 in administration room 1 in refrigerator room
	Temporarily	When necessary	Where necessary	ditto When handling cask, etc.

Neutron	Regularly	Once per day	5	Neutron survey meter	2 on upper deck 1 in accommodation room 1 in administration room 1 in refrigerator room
	Temporarily	When necessary	Where necessary	ditto	When handling cask, etc.
Air concentration of radioactive particles	Regularly	Once per day	1	Dust sample	1 in accommodation room
	Temporarily	When necessary	Where necessary	ditto	When handling cask, etc.
Surface contamination density	Regularly	Once per day	10	Smear method (GM survey meter)	2 on upper deck 1 in engine room 2 in captain's room 3 in accommodation rooms 1 in administration room 1 in refrigerator room
	Temporarily	When necessary	Where necessary	ditto	When handling cask, etc.

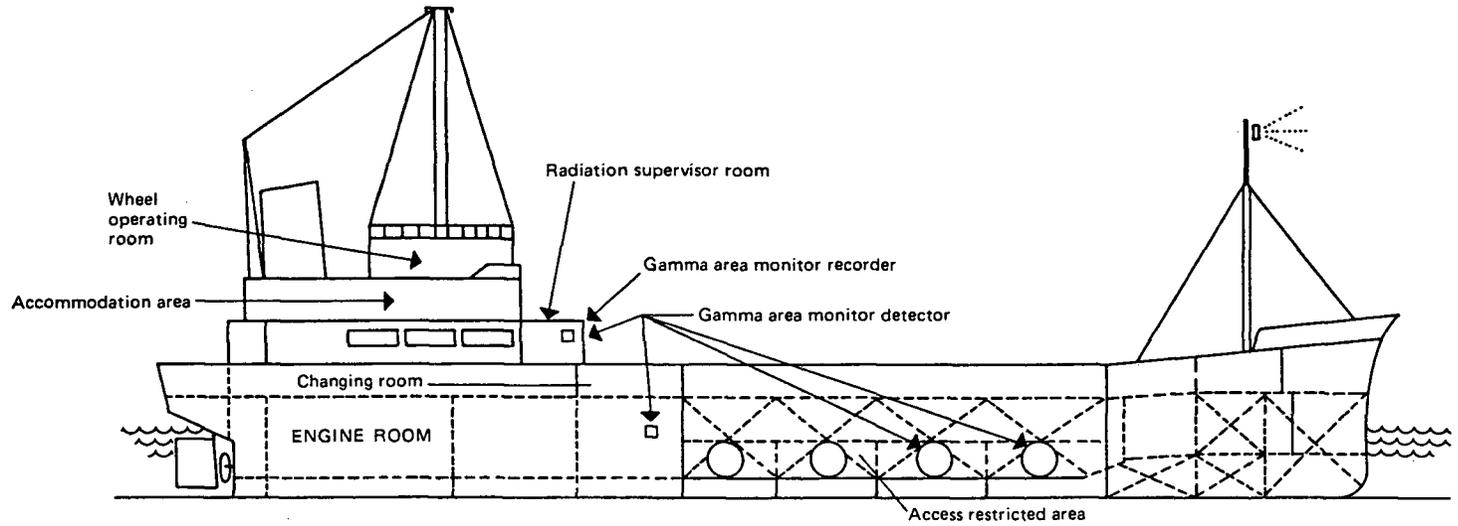


FIG. AVIII-1. Access-restricted area.

(Safety Control Organization)

Art. 3. The safety organization of the vessel is as shown in Table AVIII-1 — Safety Control Organization.

(Measurement of Radiation)

Art. 4. The radiation supervisor shall perform, when his ship is carrying radioactive packages on board, the measurement of radiation doses, etc., in the vessel, in principle in accordance with Table AVIII-2 — Frequency of Radiation Measurement.

(Measures Taken for Packages)

Art. 5. The captain shall, with the co-operation of the radiation supervisor, take measures such as cooling the holds where necessary to keep package surface temperatures below 85°C and hold atmospheric temperatures below 38°C.

(Wearing of Personnel Dosimeter)

Art. 6. 1. The crew shall always wear a film badge on board.
2. Those who enter the access-restricted area of the vessel shall wear a pocket dosimeter as well.

(Estimation of External Exposure Dose)

Art. 7. 1. The radiation supervisor shall estimate the external exposure doses received by the crew, as follows:
(1) estimation of doses by film badges every month;
(2) estimation of doses received by those who entered the access-restricted area by pocket dosimeter after each entry.
2. The estimation of the external exposure doses received by those other than the crew is to be done by pocket dosimeter only after an entry is made to the access-restricted area.

(Access-Restricted Area)

Art. 8. 1. The access-restricted area of the vessel is the area shown in Fig. AVIII-1 — Access-Restricted Area, and prescribed markings are to be posted on the area.
2. Those who enter this area shall notify the radiation supervisor and change their clothes to “clothes, shoes and gloves, etc. exclusively used in this area” in the changing room.
3. Upon leaving the area, they shall take off the clothes described above and be inspected with regard to radioactive contamination in the

changing room. Should any abnormality of contamination be detected, they shall immediately contact the radiation supervisor and request his instruction.

(Working Area)

Art. 9. The area, other than the access-restricted area, necessary for loading/unloading operation of packages is to be set up as the working area and partitioned by boundaries such as panels and posted markings in order to restrict the entrance of personnel not involved in the working operation.

(Exposure Dose)

Art. 10. The exposure doses during the normal operation are to be less than the values indicated below:

- (1) for the radiation supervisor..... 15 mSv
- (2) for the crew other than the radiation supervisor..... 5 mSv

(Management of Tools and the Like)

Art. 11. When carrying articles such as tools from the access-restricted area, the following are to be observed:

- (1) determining the surface contamination levels of the articles concerned,
- (2) performing decontamination in accordance with the instructions of the radiation supervisor when surface contamination levels exceed the following values:
 - (a) for contamination with beta and gamma emitters
and low toxicity alpha emitters 4 Bq/cm²
 - (b) for contamination with radioisotopes
which emit alpha particles 0.4 Bq/cm²

(Inspection of Contamination and Performance of Decontamination)

- Art. 12.
- 1. The captain and the radiation supervisor shall confirm that there is no significant radioactive contamination on the surfaces of packages before unloading.
 - 2. The captain and the radiation supervisor shall confirm that there is no significant radioactive contamination on the areas concerned after the completion of the unloading operation.
 - 3. When the measured values of the above two inspections exceed those described in Art. 11(2), the radiation supervisor shall take the measures necessary for decontamination.

(Management of Contaminated Articles)

- Art. 13. 1. Radioactively contaminated articles are to be delivered to the nuclear power station concerned, after temporary storage on board the vessel.
2. The waste water, including hot shower water, stored in the bilge tanks, is to be treated in the nuclear power station concerned when the radioactive concentration of the water exceeds $10^{-7} \mu\text{Ci}/\text{cm}^3$ *

* The value with natural background subtracted.

Appendix IX

INFLUENCE OF BRITTLE FRACTURE ON MATERIAL INTEGRITY

It is widely known that ferritic steels are less ductile at low temperatures and at high rates of loading. The material is susceptible to brittle fracture if unstable crack propagation occurs, even when nominal stresses in the material are less than the yield stress. Small crack-like defects in the material are often sufficient to initiate this behaviour. Such behaviour can be exhibited in many materials other than ferritic steels, such as ductile cast iron, some aluminium alloys, and many plastics and elastomers.

The condition is characterized by a considerable lowering of the energy required to cause failure under impact as the temperature is reduced. The concept of a ductile to brittle transition temperature describing the change in behaviour is attractive and several definitions of such a temperature have been proposed.

Various transition temperatures have been defined; the best known of these are:

- (a) *Nil ductility transition temperature* (NDTT). The temperature at which the energy curve first begins to rise.
- (b) *Fracture appearance transition temperature* (FATT). The temperature at which the fracture appearance is 50% crystalline and 50% fibrous.

Knott [52] states:

“The information obtained from notched impact tests cannot be applied directly to assess the resistance to fast crack propagation of a piece in service, because neither the fracture appearance nor the amount of energy absorbed can be related in a quantitative manner to the applied design stress, even if the geometry and strain rate associated with the impact test could be said to produce effects identical to those produced by service conditions. Impact-test information should really be used only to correlate with known performance in service. Provided that such correlations are made, the impact test provides comparative measurements of the toughness of different batches of steels of the same nominal composition, i.e. it may be used to give figures for quality control. Extensive correlations of this type have, however, been made for only a few specific applications.”

“One example is given by the tests carried out on steel plates from ‘Liberty Ship’ hulls, which fractured by brittle cleavage fracture during or after the Second World War. Here, it was found that notched impact testpieces cut from plates which had failed in a brittle manner absorbed less than 20 J (15 ft lb) at 10°C, whereas those cut from ‘tough’ plates absorbed more than 20 J. The figure of 20 J energy absorption at 10°C was therefore suggested

as a critical value for ensuring that ship steel is not susceptible to brittle fracture. Much more extensive analysis of failures suggested that a criterion of greater than 47 J (35 ft lb) energy absorption and less than 70% crystallinity at 0°C was necessary to prevent brittle fractures. Present specifications relate the required impact properties both to the strength level of the steel and to the thicknesses of plate used in service.”

“An alternative treatment applied to impact results is the measurement and specification of the transition temperature. This is primarily a feature of the impact test itself, but can again be correlated, for a given material and application, with a transition temperature which is characteristic of the service conditions. Often, an intermediary large-scale test is used to simulate the anticipated extremes of service environment in the testing laboratory. The transition curve for the large test then represents (hopefully) the worst case that could occur and the ‘worst case’ transition temperature is compared with the transition temperature in notched impact. Typically, a specification for material quality would then be that its notched impact transition temperature should be greater than, say, 30°C below the operating temperature in service.”

Reference [53] makes recommendations for carbon and carbon manganese steel pressure vessels required to operate at low temperatures. It states the following:

“*Design minimum temperature.* The design minimum temperature should be the lowest of the following temperatures:

- (a) The lowest metal temperature occurring coincidentally with loadings resulting in a calculated membrane stress equal to or exceeding $2f/3$ [where f is the nominal design strength value as defined in Ref. [53]];
- (b) 10°C higher than the lowest metal temperature occurring coincidentally with loadings resulting in a calculated membrane stress equal to or exceeding 50 N/mm^2 but not exceeding $2f/3$;
- (c) 50°C higher than the lowest metal temperature occurring coincidentally with loadings resulting in a calculated membrane stress not exceeding 50 N/mm^2 . In this case the membrane stresses should take account of internal and external pressure, static head and self weight.

In the case of components thermally insulated externally, the lowest metal temperature for the determination of design minimum temperature should correspond to the minimum temperature of the contents of the vessel at the appropriate loading condition. In the case of components not thermally insulated, the minimum temperature of the contents may also be used; however, attention is drawn to the effect that temperature may have upon the temperature of the components under operating conditions and the method used for assessing the lowest metal temperature should be subject to agreement.”

A more thorough discussion of brittle fracture can be found in Ref. [52], Chapters 7 and 8, and those listed in Refs [54–65]. The Pellini Izod and Charpy tests are described in Refs [66–69].

The underlying purpose of all that has so far been described is to ensure that materials are highly stressed only whilst in a ductile state. More quantitative descriptions of the behaviour of components in the presence of defects, real or postulated, may be obtained by using the techniques of fracture mechanics. When the material is in the brittle state, linear elastic fracture mechanics are generally sufficient [70, 71], using standardized fracture toughness values [72, 73]. For more general applications, including ductile behaviour, general yielding fracture mechanics methods are necessary. Typical procedures are described in Ref. [74]. Specialist guidance is recommended for the application of fracture mechanics methods.

A special circumstance which applies to transport packages is that the worst stresses would generally be induced by the mechanical test for accident conditions. Under these conditions the stress distributions and magnitudes are not well defined and consequently there is often little to be gained by applying the more refined fracture mechanics techniques.

Although model tests may be used to define stress levels, and hence confirm the avoidance of brittle fracture through the use of fracture mechanics, they may not be used as a direct demonstration of the avoidance of brittle fracture. Low temperature testing of representative full sized components could provide such a demonstration though this is not always practicable. In conducting such tests the objective should be to show that components under the high rates of strain produced by the mechanical test for accident conditions do not fail in a brittle fashion or exhibit a brittle response. Naturally, adequate standards of production need to be applied in order to ensure that all packages comply with the design intent. The identification of suitable material defect standards is important, and, in this regard, fracture mechanics techniques may give useful guidance. Since sections thicker than 10 cm may not exhibit the same brittle fracture characteristics as thinner sections, thick walled vessels may not be as susceptible to low temperature failures and may not be subject to as high a stress level as thinner vessels. Such considerations should be included in evaluating designs using 'brittle' construction materials. It should also be remembered that the occurrence and consequences of brittle fracture may be minimized by special design techniques such as lamination of thick sections, composite materials, etc., although these do preclude in-service inspections. Austenitic stainless steels (including the high nitrogen and warm worked varieties) and some aluminium alloys are not susceptible to low stress brittle fracture and no special requirements are necessary for their use at temperatures well below -40°C (i.e. down to -196°C) [53].

Appendix X

CRITICALITY SAFETY ASSESSMENTS

AX.1. INTRODUCTION

This appendix will provide examples of approaches that have been used successfully, while pointing out potential pitfalls associated with criticality safety assessments.

AX.2. REGULATORY REQUIREMENTS AND CRITICALITY ASSESSMENT

Fundamental to the safe transport of fissile material is the valid criticality safety assessment. The more directly this assessment is dependent on experimental data, the more reliable will be the results. Unfortunately, experiments of direct applicability to shipping configurations are uncommon. There does exist, however, a substantial body of data relevant to many problems associated with fissile material transport. Examples include the critical arrays of metallic uranium which were studied at the Oak Ridge National Laboratory, as well as measurements of the effect on array criticality of several materials important to transport. The publication TID 7028 [75] is a useful compilation of critical data for systems containing enriched uranium, plutonium, and uranium-233. There remain, however, many assessments necessary for transport for which no direct data exist and reliance needs to be placed on results of calculations. Improved confidence in the results of such calculations can be developed through investigation of the effect of input uncertainties on the calculated results.

The result of the criticality assessment should provide an appropriate margin of subcriticality to maintain safety under the various contingencies that may be encountered in transport. While rules of thumb, such as taking 80% of the mass that would be expected to be critical under specified conditions, are clearly useful, they should not be considered mandatory. These rules should not be considered as a substitute for the careful evaluation of those contingencies. Safety depends not on blind adherence to guidelines but on enlightened evaluation of those conditions appropriate to transport.

The variety of packagings in which fissile material may be shipped is so great that the extremes are scarcely related. Type A quantities may be placed in relatively light packaging while irradiated reactor fuel is shipped in massive packagings which are essentially invulnerable to accidents that occur in transport. Requirements for the criticality safety assessment of packages involve a comparable variety of considerations. While all packages must, of course, be evaluated for all designated conditions

of transport, smaller packages usually have loading limitations based on the neutron interaction among such packages. The massive packages containing spent fuel are generally less vulnerable to interaction but economics dictate a maximum reasonable loading for each package so that the single unit evaluation may be the controlling consideration.

For both the single unit assessment and the evaluation of interacting accumulations of packages, the important consideration for criticality safety is the neutron balance, i.e. the behaviour of the neutrons that result from fission. These neutrons have, for purposes of the assessment, three possible destinies. Some will leak out of the system under consideration (either a single unit or an accumulation of packages) and unless returned to the system by a reflector will be of no further significance; some will be captured in the various materials of the system and have no additional influence; and some will induce additional fissions in the material being transported. These fissions will return additional neutrons to the system. If the additional neutrons make up for the losses by leakage and by capture, the system will have a constant neutron population. This condition is defined as criticality. The neutron balance can be changed by altering the geometry of the system as this affects the fraction of neutrons that are lost by leakage. Criticality can be avoided by influencing the leakage from a single unit, for example by controlling the diameter of the fissile material container. Similarly, for any given number of packages the accumulation of packages cannot become critical if the packages are separated sufficiently. It should be noted that for an unlimited number of packages separation alone may not always be sufficient to maintain subcriticality. The presence of sufficient neutron absorbing material will, however, keep an unlimited accumulation of packages subcritical. Examples of such materials are cadmium, boron, and gadolinium. The steel from which many packagings are fabricated provides some limited capture of neutrons, so that for modest quantities of fissile material with effective neutron moderation, such as may be provided by insulating or shock mitigating material, these packages may be subcritical in unlimited numbers.

AX.3. METHOD VALIDATION AND SUBCRITICALITY CRITERIA

The economic imperative to maximize payloads, and thereby reduce transport costs, requires a high standard of nuclear criticality safety assessment. Assessments of complex package designs will usually involve the use of sophisticated computer codes, incorporating solution strategies of a deterministic or statistical nature. The performance of a computer code and the chosen nuclear data library should be evaluated by means of experiments selected for the purpose. As far as practicable, the experimental configuration should have neutron leakage and energy distribution similar to the transport container. Ideally, the influence on the neutron balance of materials of interest in such experiments should be of similar importance to that of

the same materials in the package design being assessed. In a well designed experiment used for validation, the measurement errors will be significantly less than the uncertainties in the code calculations.

Subcriticality criteria against which particular calculational results may be judged include a subcritical margin of, typically, several per cent in the multiplication factor. Most validation experiments to date have identified the critical state only. Calculational errors which cancel at a multiplication factor of unity may no longer do so at lower values which meet the subcriticality criterion. Where possible, it would be desirable to incorporate subcritical and reaction rate measurements in experiments designed to validate computer methods; the former provide a direct comparison with calculated subcritical values while the latter permit a more stringent test of calculational performance.

Testing against experimental measurements will identify errors associated with the code and the nuclear data library chosen for the assessment of the package concerned. Additional errors might be introduced in the process of assessment. Quality assurance procedures should minimize the possibility of developing an inappropriate computer model, i.e. a model which is inaccurate or fails to incorporate the state with the highest multiplication factor. Where code limitations make non-conservative assumptions in the model unavoidable, some allowance should be incorporated in the subcriticality criterion. In summary, the criterion for subcriticality should include:

- (i) An arbitrary minimum subcritical margin which does not depend on the accuracy of the code and chosen nuclear data. Thus, the margin would persist even if it were possible to calculate the multiplication factor exactly. The value of the margin is a matter for judgement, bearing in mind the sensitivity of the multiplication factor to foreseeable physical changes to the package.
- (ii) An allowance for systematic and random errors associated with the performance of the code and chosen nuclear data library; and
- (iii) An additional allowance, where appropriate, for non-conservative modelling approximations.

AX.4. THE INDIVIDUAL PACKAGE

The individual package is evaluated in isolation under the conservative assumption that water can leak into all void spaces of the package. In the evaluation, use may be made of special features which prevent leakage of water into or out of protected voids or regions within the package. An example of protected regions would be sealed cans of fissile material in oxide form. If, as a result of the package tests, it is demonstrated that the contents do not escape from the sealed cans, then the oxide may be considered confined to the can within the containment system. If the tests show the contents of the can are retained but water inleakage may occur,

then water must be considered in the containment system and within the oxide container. If the tests show the oxide does escape from its container, then it must be assumed to be distributed within the containment system in the configuration resulting in the largest value of the neutron multiplication factor.

After the containment system has been specified in its most reactive credible configuration, it needs to be evaluated for three reflector conditions. These are:

- (i) The containment system with closely fitting full (20 cm) water reflection;
- (ii) The undamaged package surrounded by a closely fitting 20 cm thick water reflector; and
- (iii) The damaged package with water inleakage to the package as consistent with the results of the package tests, surrounded by a closely fitting 20 cm thick water reflector.

In many cases the water reflected containment system will be found to be the limiting situation, but it is necessary to show this to be true.

Assessments of the criticality safety of packages can, in general, be accomplished most effectively through the use of computer codes. The code and the neutron cross-sections used need to be validated through test calculations of experimental systems of similar materials characterized by similar neutron energy distributions. As an example, as one searches for the most reactive distribution of oxide in a flooded containment system, the situation may vary from damp oxide reflected by water to oxide suspended in water at a number of concentrations. Each concentration will be characterized by a different ratio of hydrogen to fissile material, and the larger this ratio the lower the mean energy of the neutrons of the system. Thus the assessment might necessitate code and cross-section validation against water reflected oxide (or metal) systems as well as solutions at several concentrations.

An acceptable fissile material loading for a package will provide single package subcriticality under the conditions discussed above, and also have a sufficient margin of subcriticality for the damaged and undamaged package configuration to meet the interaction criteria of para. 567.

AX.5. ARRAYS OF PACKAGES

The evaluation of neutron interaction among fissile material packages determines the subcritical number N , the number of like packages that may be transported together, which is necessary for defining the transport index. The Regulations require the examination of both undamaged and damaged packages for this determination.

The array of undamaged packages contains five times N packages and is evaluated without additional moderation between packages, while the array of

damaged packages contains twice N packages and also has water in the space between packages to the extent that will provide the most reactive array. In each case the array is closely reflected by 20 cm water and must be subcritical with any arrangement of packages.

The requirement of 'any arrangement' is significant in that the greatest neutron multiplication factor attained by an accumulation of fissile material is in the geometry resulting in the lowest surface-to-volume ratio, since neutron leakage at that ratio is normally minimized. However, the geometry of the fissile material and containment system of packages may produce preferential neutron coupling in one or two dimensions of the array. For this reason it needs to be demonstrated, through a series of calculations, that the value N is the smallest number attainable for the package loading, as the array configuration is varied.

The examination necessitates a validated computational method. In the description of the package some details may be approximated without altering the neutron multiplication factor. These should be a matter of experienced judgement. For example, the two rolled hoops on the lateral surface of a 210 L drum need not be detailed in the calculations for an individual package. Moreover, in an array of packages the approximation does not change the result provided that the centre separation of packages is properly prescribed. Further, cylindrical packages, e.g. the 210 L drum, call for examination of triangular and square patterns of package arrangements. The calculation of the triangular patterned arrays can be approximated by a square pattern through an appropriate reduction in the centre spacing.

The effect on the neutron multiplication factor of a variation in a parameter of the fissile loading (such as material density) may differ greatly for packages of different size, even though the packages have similar design and composition. For example, a factor of eight reduction in uranium metal density in a 210 L drum with a transport index of zero will reduce the array neutron multiplication factor from 0.93 to 0.85. The same reduction in density to an infinite array of 57 L drums will cause an increase in the neutron multiplication factor from 0.94 to 1.33. The effect of parameter changes on the criticality assessment of arrays of superficially similar packages should not be generalized.

Given the importance of these evaluations to criticality safety in transport, the paucity of experimental data on the variety of packaging materials used, the unrestricted comingling of packages allowed, and the likely use of sophisticated computer codes as a substitute for understanding neutron behaviour in the circumstances of transport, the margin of subcriticality for individual packages and arrays of packages should include an arbitrary margin ranging upwards from 0.03.

AX.6. IRRADIATED FUELS

With the need to maximize irradiated fuel payloads, coupled with a move towards more highly rated fuels in some Member States, criticality assessments on

packages containing irradiated fuel in which advantage is taken of reduced reactivity due to irradiation are likely to become more common. The Regulations recognize the use of this reduced reactivity provided that the degree of irradiation is known with appropriate accuracy. If such information is not available, the normal assessment procedure is to assume that the fuel is unirradiated or has that irradiation which would result in the maximum neutron multiplication factor (para. A-568.1).

If advantage is taken of reduced reactivity as a consequence of irradiation, the assessment presented to the competent authority should satisfy the following conditions:

- (a) Procedures should be adopted to prevent the package having a higher reactivity than the calculated value under any foreseeable circumstances. Among the contingencies to be considered are the possibility of misidentification of the fuel at the time of loading into the transport container, and any possible errors in the evaluation of burnup.
- (b) The validity of the method of evaluating the effect of irradiation on the fuel composition should be established. One of the validated computer codes which exist for this purpose could be used.
- (c) The assessment should include an evaluation of any inherent uncertainties, so that the probability of criticality is demonstrated to be acceptably small.

TABLE AX-1. PERMISSIBLE MASS OF URANIUM-235 PER CONSIGNMENT

Uranium enrichment in weight per cent of uranium-235 not exceeding	Permissible mass per consignment in grams of uranium-235 for carriage not under special arrangement	Permissible mass per consignment in grams of uranium-235 for carriage under special arrangement
93	160	400
30	190	480
10	250	640
5	360	900
4	400	1 000
3	500	1 250
2	800	2 000
1.5	1360	3 400
1.35	1600	4 000
1	3400	8 500
0.92	6000	15 000

AX.7. EXAMPLES OF PACKAGE DESIGNS FOR FISSILE MATERIAL

Two examples are described here. In one case, carriage not under special arrangement is considered and in the other case carriage under special arrangement. In both cases, the criticality safety of the consignments does not depend upon the integrity of the packagings involved. For the use of these examples, however, by any consignor, the package design and possibly the shipment itself are subject to the approval of the competent authority of the country of origin and of any country through or into which the consignment will be transported (i.e. multilateral approval is required, see paras 710 and 716(c)).

AX.7.1. Packaging description

The criticality safety of these consignments does not depend upon the integrity of the packaging. Any packaging which complies with the other relevant requirements of the Regulations with respect to the non-fissile radioactive characteristics may therefore be used.

AX.7.2. Authorized radioactive contents

(a) *General conditions*

- (i) The total quantity of fissile material per package is limited to the permissible mass of uranium-235 according to the appropriate specific conditions below. If more than one enrichment is to be included in a single consignment, the permissible mass for that consignment shall be based on the highest enrichment present.
- (ii) If any plutonium or uranium-233 is present then an effective uranium-235 mass shall be determined by adding twice the mass of the plutonium and uranium-233 present to the mass of the uranium-235.
- (iii) Beryllium and deuterium shall not be present in excess of 1 kg each.
- (iv) The total mass of graphite present shall not exceed 7.7 times the total mass of uranium-235 (this is equivalent to a carbon to uranium-235 atomic ratio of not more than 150).
- (v) Mixtures of fissile material with substances having a higher hydrogen density than water, e.g. some hydrocarbon oils, shall not be present. This does not preclude the use of polyethylene for wrapping or packing.

(b) *Specific conditions*

- (i) For materials in unspecified form the total permissible mass per consignment is limited to that given in Table AX-1.

TABLE AX-2. PERMISSIBLE MASS OF URANIUM-235 PER CONSIGNMENT

Uranium enrichment in weight per cent of uranium-235 not exceeding	Permissible mass per consignment in grams of uranium-235 for carriage not under special arrangement	Permissible mass per consignment in grams of uranium-235 for carriage under special arrangement
3	560	1 400
2	1200	3 000
1.5	2800	7 000
1.35	4000	10 000

(ii) For materials which are distributed homogeneously and which do not form a lattice arrangement in the package, the total quantity present per consignment is limited to that given in Table AX-2.

AX.7.3. Shipment requirements

(a) *Transport index assignment*

The number 'N' for a particular package in the consignment, for the purposes of para. 567, will depend on the actual contents and is equal to the total permissible mass per consignment divided by the actual fissile mass present in the package.

The TI based on nuclear criticality control for the package is then obtained as provided in para. 429.

(b) *For carriage under special arrangement*

The quantity of material in a consignment must not exceed that defined under the appropriate specific conditions above. Transport must be direct to the consignee without any intermediate transit storage.

REFERENCES

- [1] INTERNATIONAL ATOMIC ENERGY AGENCY, Regulations for the Safe Transport of Radioactive Material: 1985 Edition, Safety Series No. 6, IAEA, Vienna (1985); see also the Supplement 1986, Safety Series No. 6 — Suppl. 1986, IAEA, Vienna (1986).
- [2] INTERNATIONAL ATOMIC ENERGY AGENCY, INTERTRAN: A System for Assessing the Impact from Transporting Radioactive Material, IAEA-TECDOC-287, IAEA, Vienna (1983).
- [3] UNITED STATES DEPARTMENT OF TRANSPORTATION, Guidelines for Selecting Preferred Routes for Highway Route Controlled Shipments of Radioactive Materials, Rep. DOT/RSPA/MTB-84/22, USDOT, Washington, DC (1984).
- [4] AMERICAN NATIONAL STANDARDS INSTITUTE, American National Standard for Nuclear Criticality Control of Special Actinide Elements, ANSI/ANS 8.15-1981, American National Standards Institute, New York (1981).
- [5] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, Series 1 Freight Container Specifications and Testing — Part 1: General Cargo Containers, ISO 1496/1-1984, ISO, Geneva (1984).
- [6] INTERNATIONAL ATOMIC ENERGY AGENCY, Basic Safety Standards for Radiation Protection, Safety Series No. 9, IAEA, Vienna (1982).
- [7] INTERNATIONAL ATOMIC ENERGY AGENCY, Discussion of and Guidance on the Optimization of Radiation Protection in the Transport of Radioactive Material, IAEA-TECDOC-374, IAEA, Vienna (1986).
- [8] UNITED STATES DEPARTMENT OF TRANSPORTATION, All About Radioactive Materials Packages — A Guide for Supervisors at Cargo Terminals, USDOT, Washington, DC (1978).
- [9] INTERNATIONAL ATOMIC ENERGY AGENCY, Emergency Response Planning for Transport Accidents Involving Radioactive Materials, IAEA-TECDOC-262, IAEA, Vienna (1982).
- [10] UNITED KINGDOM ATOMIC ENERGY AUTHORITY, Shielding Integrity Testing of Radioactive Material Transport Packaging, Gamma Shielding, Rep. AECF 1056, Part 1, UKAEA, Harwell (1977).
- [11] UNITED KINGDOM ATOMIC ENERGY AUTHORITY, Testing the Integrity of Packaging Radiation Shielding by Scanning with Radiation Source and Detector, Rep. AESS 6067, UKAEA, Risley (1977).
- [12] BRITISH STANDARDS INSTITUTION, Guide to the Design, Testing and Use of Packaging for the Safe Transport of Radioactive Materials, BS 3895:1976, GR 9, BSI, London (1976).
- [13] AMERICAN NATIONAL STANDARDS INSTITUTE, American National Standard for Leakage Tests on Packages for Shipment of Radioactive Material, ANSI N14.5-1977, ANSI, New York (1977).
- [14] AMERICAN NATIONAL STANDARDS INSTITUTE, American National Standard for Packaging of Uranium Hexafluoride for Transport, ANSI N14.1-1982, ANSI, New York (1982).

- [15] INTERNATIONAL ATOMIC ENERGY AGENCY, Neutron Monitoring for Radiological Protection, Technical Reports Series No. 252, IAEA, Vienna (1985).
- [16] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Protection Against Ionizing Radiation from External Sources used in Medicine, Publication 33, Volume 9, No. 1, Pergamon Press, Oxford and New York (1982).
- [17] UNITED KINGDOM ATOMIC ENERGY AUTHORITY, Atomic Energy Code of Practice, Rep. AECF 1030, UKAEA, Harwell (1977).
- [18] GIBSON, R., The Safe Transport of Radioactive Materials, Pergamon Press, Oxford and New York (1966).
- [19] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, Sealed Radioactive Sources — Classification, ISO 2919-1980(E), ISO, Geneva (1980).
- [20] ASTON, D., BODIMEADE, A.H., HALL, E.G., TAYLOR, C.B.G., The Specification and Testing of Radioactive Sources Designated as "Special Form" Under the IAEA Transport Regulations (CEC Study Contract XVII/322/80.6), Rep. EUR 8053, CEC, Luxembourg (1982).
- [21] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, Trunnions for Spent Fuel Element Shipping Casks, ISO/TC85/SC5/WG9 (draft), ISO, Geneva.
- [22] AMERICAN NATIONAL STANDARDS INSTITUTE, American National Standard for Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4,500 kg) or More for Nuclear Materials, ANSI N14.6-1978, ANSI, New York (1978).
- [23] KERntechnischer AUSSCHUSS, Lastanschlagpunkte in Kernkraftwerken, KTA 3905 (draft), KTA, Cologne (1985).
- [24] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, Series 3 Tank Containers for Liquids and Gases — Specification and Testing, ISO 1496/3-1981, Part 3, ISO, Geneva (1981).
- [25] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, Series 1 Freight Containers — Specifications and Testing, ISO 1496/1-1978, ISO, Geneva (1978).
- [26] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, Series 1 Freight Containers — Specifications and Testing, ISO 1496/1-1984, ISO, Geneva (1984).
- [27] CHEVALIER, G., GILLES, P., PHALIPPOU, C., POUARD, M., JOLYS, J.C., DRAULANS, J., LAFONTAINE, I., "L'arrimage de colis de matières radioactives en conditions accidentelles", Packaging and Transportation of Radioactive Materials, PATRAM '86 (Proc. Symp. Davos, 1986), Vol. 1, IAEA, Vienna (1987) 63.
- [28] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, Radioactive Materials — Packaging — Test for Contents Leakage and Radiation Leakage, ISO 2855-1976 (E), ISO, Geneva (1976).
- [29] AMERICAN NATIONAL STANDARDS INSTITUTE, American National Standard for Program for Testing Biological Shielding in Nuclear Reactor Plants, ANSI N18.9-1972, ANSI, New York (1972).
- [30] JANARDHANAN, S. et al., "Testing of massive lead containers by gamma densitometry", Industrial Isotope Radiography (Proc. Nat. Symp.), Bharat Heavy Electrical Ltd., Tiruchirapalli, India, February 26-27, 1976.

- [31] KRISHNAMURTHY, K., AGGARMAL, K.S., "Complementary role of radiometric techniques in radiographic practice", Industrial Radiography (Proc. Nat. Symp.), Bharat Heavy Electrical Ltd., Tiruchirapalli, India, February 26-27, 1976.
- [32] UNITED KINGDOM ATOMIC ENERGY AUTHORITY, Testing the Integrity of Packaging Radiation Shielding by Scanning with Radiation Source and Detector, Rep. AESS 6067 (1977), UKAEA, Risley (1977).
- [33] UNITED KINGDOM ATOMIC ENERGY AUTHORITY, Shielding Integrity Testing of Radioactive Material Transport Packaging, AECP 1056 Part 1 (1977), UKAEA, Risley (1977).
- [34] NUCLEAR ENERGY AGENCY OF THE OECD, Standard Problem Exercise on Criticality Codes for Spent LWR Fuel Transport Containers, CSNI Rep. No. 71, OECD/NEA, Paris (1982).
- [35] NUCLEAR ENERGY AGENCY OF THE OECD, Standard Problem Exercise on Criticality Codes for Large Arrays of Packages of Fissile Materials, CSNI Rep. No. 78, OECD/NEA, Paris (1982).
- [36] DAVIES, A.J., Description of Various Test Methods Used at Harwell to Comply with the IAEA Regulations for Radioactive Transport Containers, Ref. AERE-R 7003, Her Majesty's Stationery Office, London (1972).
- [37] DIXON, F.E., An Indexed List with Commentary of Experimental Work Connected with Transport Containers for Radioactive Material Performed at AERE Harwell, 1960-71, Rep. AERE-R 6797, United Kingdom Atomic Energy Authority, Harwell (1960).
- [38] Transport Packaging for Radioactive Materials (Proc. Sem. Vienna, 1976), IAEA, Vienna (1976).
- [39] Packaging and Transportation of Radioactive Materials (PATRAM), Proc. of Symp.: (Albuquerque, 1965), Sandia Labs, Albuquerque, NM (1965); (Gatlinburg, 1968), United States Atomic Energy Commission, Oak Ridge, TN (1968); (Richland, 1971), United States Atomic Energy Commission, Oak Ridge, TN (1971); (Miami Beach, 1974), Union Carbide Corp., Nuclear Div., Oak Ridge, TN (1975); (Las Vegas, 1978), Sandia Labs, Albuquerque, NM (1978); (Berlin (West), 1980), Bundesanstalt für Materialprüfung, Berlin (West) (1980); (New Orleans, 1983), Oak Ridge Natl Lab., Oak Ridge, TN (1983); (Davos, 1986), International Atomic Energy Agency, Vienna (1987).
- [40] INTERNATIONAL ATOMIC ENERGY AGENCY, Directory of Transport Packaging Test Facilities, IAEA-TECDOC-295, IAEA, Vienna (1983).
- [41] OAK RIDGE NATIONAL LABORATORY, Cask Designers Guide, Rep. ORNL-NSIC-68, UC-80, Oak Ridge Natl Lab., Oak Ridge, TN (1976).
- [42] DIGGS, J.M., POPE, R.B., TRUJILLO, A.A., UNCAPHER, W.L., Crush Testing of Small Type B Packagings, Rep. SAND 83-1145, Sandia Natl Labs, Albuquerque, NM (1985).
- [43] McCAFFERY, B.J., Purely Buoyant Diffusion Flames — Some Experimental Results, Rep. PB80-112 113, US National Bureau of Standards, Washington, DC (1979).
- [44] McADAMS, W.H., Heat Transmission, McGraw Hill, New York (1954).
- [45] INTERNATIONAL ATOMIC ENERGY AGENCY, Quality Assurance for Safety in Nuclear Power Plants, Safety Series No. 50-C-QA, IAEA, Vienna (1978).

- [46] UNITED STATES ATOMIC ENERGY COMMISSION, Recommendations for Revising Regulations Governing the Transportation of Radioactive Material in Passenger Aircraft (July 1974). [Available at the US Nuclear Regulatory Commission's Public Document Room, Washington, DC.]
- [47] INTERNATIONAL CIVIL AVIATION ORGANIZATION, Technical Instructions for the Safe Transport of Dangerous Goods by Air, 1986 Edition, DOC 9284-AN/905, ICAO, Montreal (1986).
- [48] UNITED KINGDOM ATOMIC ENERGY AUTHORITY, Securing Radioactive Materials Packages to Conveyances, Rep. AECF 1006 (1979), UKAEA, Risley (1979).
- [49] UNITED STATES DEPARTMENT OF ENERGY, Fuel Shipping Containers Tie-down for Truck Transport, RDT Standard F8-11T, USDOE, Washington, DC (1975).
- [50] OAK RIDGE NATIONAL LABORATORY, Cask Tie-down Design Manual, Structural Analysis of Shipping Casks, Vol. 7.J.TI. Evans, Rev. ORNL TM-1312, Oak Ridge Natl Lab., Oak Ridge, TN (1969).
- [51] AMERICAN NATIONAL STANDARDS INSTITUTE, American National Standard for Highway Route Controlled Quantities of Radioactive Materials — Domestic Barge Transport, ANSI N14.24-1985, ANSI, New York (1985); also JONES, R.H., WILMOT, E.L., "A standard for barge transport of Type B quantities of radioactive materials", Packaging and Transportation of Radioactive Materials, PATRAM '86 (Proc. Symp. Davos, 1986), Vol. 1, IAEA, Vienna (1987).
- [52] KNOTT, J.F., Fundamentals of Fracture Mechanics, Butterworths, London (1973).
- [53] BRITISH STANDARDS INSTITUTE, Unfired Fusion Welded Pressure Vessels, BS 5500:1985, BSI, London (1985) Appendix D, para. 2.2.1.
- [54] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Special Technical Publication No. 158, ASTM, Philadelphia, PA (1954).
- [55] HODGSON, J., BOYD, G.M., Trans. R. Inst. Naval Archit. 103 (1961).
- [56] TIPPER, C.F., The Brittle Fracture Story, Cambridge University Press, Cambridge (1962).
- [57] KOBAYASHI, A.S. (Ed.), Experimental Techniques in Fracture Mechanics, Society for Experimental Stress Analysis, The Department of Publications, State University of Iowa (1974).
- [58] HALL, W.J., KIHARA, H., SOETE, W., WELLS, A.A., Brittle Fracture of Welded Plate, Prentice-Hall, Englewood Cliffs (USA) (1967).
- [59] BOYD, G.M. (Ed.), Brittle Fracture in Steel Structures, Butterworths, London (1970).
- [60] RICHARDS, K.G., Brittle Fracture of Welded Structures, The Welding Institute, London (1971).
- [61] PELLINI, W.S., Evolution of Engineering Principles for Fracture-Safe Design of Steel Structures, Rep. NRL 6957, Naval Res. Lab., Washington, DC (1969).
- [62] PELLINI, W.S., Integration of Analytical Procedures for Fracture-Safe Design of Metal Structures, Rep. NRL 7251, Naval Res. Lab., Washington, DC (1971).
- [63] PELLINI, W.S., Criteria for Fracture Control Plans, Rep. NRL 7406, Naval Res. Lab., Washington, DC (1972).

- [64] SHANNON, J.L., Jr., BROWN, W.F. Jr., Progress in fracture mechanics, Mach. Des. (1970) 133.
- [65] MANSON, S.S., Thermal Stress and Low-Cycle Fatigue, McGraw Hill, New York (1966).
- [66] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Standard Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels (PELLINI NIL DUCTILITY TEST), Rep. E208-69, ASTM, Philadelphia, PA (1975).
- [67] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Standard Methods for Notched Bar Impact Testing of Metallic Materials, Rep. E23-72, ASTM, Philadelphia (1978).
- [68] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, The Charpy V-Notch Impact Test on Metals, ISO/R148, ISO, Geneva (1983).
- [69] INTERNATIONALLY ORGANIZATION FOR STANDARDIZATION, The Charpy U-Notch Impact Test on Metals, ISO 83, ISO, Geneva (1976).
- [70] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Special Technical Publication 381, ASTM, Philadelphia, PA (1964).
- [71] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Special Technical Publication 410, ASTM, Philadelphia, PA (1967).
- [72] BRITISH STANDARDS INSTITUTE, Methods of Test for Plane Strain Fracture Toughness (K₁ subscript C) of Metallic Materials, BS 5447:1977, BSI, London (1977).
- [73] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Standard E399-78A, Philadelphia, PA (1979).
- [74] BRITISH STANDARDS INSTITUTE, Guidance on Some Methods for the Derivation of Acceptance Levels for Defects in Fusion Welded Joints, BSI PD 6493:1980, BSI, London (1980).
- [75] UNITED STATES DEPARTMENT OF ENERGY, Critical Dimensions of Systems Containing U-233, U-235 and Pu-239, Rep. TID 7028, USDOE, Washington, DC (1964).

This publication is no longer valid
Please see <http://www-ns.iaea.org/standards/>

BIBLIOGRAPHY

AMERICAN NATIONAL STANDARDS INSTITUTE/AMERICAN NUCLEAR SOCIETY, ANSI standards produced by subcommittee ANS-8, ANSI.

ANDERSON, J.A. et al., PAT-2 Safety Analysis Report, Rep. SAND 81-0001, Sandia Natl Labs, Albuquerque, NM (1981).

ANDERSON, J.A., "Correlation between measured gas leaks and possible loss of contents from radioactive materials packagings", Packaging and Transportation of Radioactive Materials, PATRAM '83 (Proc. Symp. New Orleans, 1983), Oak Ridge Natl Lab., Oak Ridge, TN (1983), 646.

ASTON, D. et al., The Specification and Testing of Radioactive Sources Designated as 'Special Form' Under the IAEA Transport Regulations, Rep. EUR 8053 EN, CEC, Luxembourg (1982).

BIERMAN, S.R., DURST, B.M., CLAYTON, E.D., Criticality Experiments with Subcritical Clusters of 2.35 Wt% and 4.29 Wt% U-235 Enriched UO₂ Rods in Water with Uranium or Lead Reflecting Walls, Rep. NUREG/CR-0796 (PNL-2827), Battelle Pacific Northwest Labs, Richland, WA (1979).

BURIAN, R.J., et al., Validity of Scale Modeling for Large Deformation in Shipping Containers, Rep. BM1-2040, Battelle Columbus Labs, OH (1979).

BUTTERFIELD, R.S., "An assessment of the containment of the UK 250 litre plutonium nitrate package", Packaging and Transportation of Radioactive Materials, PATRAM '80 (Proc. Symp. Berlin (West)), Bundesanstalt für Materialprüfung, Berlin (West) (1980).

CARTER, R.D., Concrete and Criticality (Transactions of the ANS Meeting, June 1978), American Nuclear Society, New York (1978).

CARTER, R.D., KIEL, G.R., RIDGWAY, K.R., Criticality Handbook, ARH-600, Atlantic Richfield, Richland, WA (1968-73).

CHALMERS, J.H., WALKER, G., PUGH, J., Handbook of Criticality Data, United Kingdom Atomic Energy Authority Rep. AHSB (S) Handbook 1 (1st Revision) Her Majesty's Stationery Office, London (1967).

CHIVAS, T.C., GEORGE, A.F., "Factors influencing the design and assessment of elastomer seals for nuclear transport flasks", Packaging and Transportation of Radioactive Materials, PATRAM '83 (Proc. Symp. New Orleans, 1983), Oak Ridge Natl Lab., Oak Ridge, TN (1983).

COMMISSION OF THE EUROPEAN COMMUNITIES, Stowing of Packages Containing Radioactive Materials on Conveyances, Repts EUR 8057, EUR 9103 and EUR 9195, CEC, Luxembourg (1982-84).

Criticality Array Data and Calculational Methods, Nucl. Technol. **30** 97 (August 1976).

Criticality Control in Chemical and Metallurgical Plant (Proc. Symp. Karlsruhe, 1961), Organisation for Economic Co-operation and Development, Paris (1961).

Criticality Control of Fissile Materials (Proc. Symp. Stockholm, 1965), International Atomic Energy Agency, Vienna (1966).

CURREN, W.D., BOND, R.D., "Leakage of radioactive powders from containers", Packaging and Transportation of Radioactive Materials, PATRAM '80 (Proc. Symp. Berlin (West)), Bundesanstalt für Materialprüfung, Berlin (West) (1980).

Deutsche Normen, Kritikalitätssicherung bei der Verarbeitung und Handhabung von Kernbrennstoffung, DIN 25403, Deutsches Institut für Normung, e.V., Berlin (West).

INTERNATIONAL MARITIME ORGANIZATION, Guidelines on Securing Arrangements for the Transport of Road Vehicles on RO-RO Ships, BC2611, (draft), Annex 3.

DRAULANS, J., LAFONTAINE, I., AUGIN, B., GILLES, P. "Stowing of packages containing radioactive materials on conveyances in normal and accident conditions", Packaging and Transportation of Radioactive Materials, PATRAM '83 (Proc. Symp. New Orleans, 1983), Oak Ridge Natl Lab., Oak Ridge, TN (1983) 1540.

DRAULANS, J., LAFONTAINE, I., CHEVALIER, G., GILLES, P., JOLYS, J.C., POUARD, M., Stowing of Packages Containing Radioactive Materials on Conveyances, Rep. CEA-DAS-225, CEA Centre d'études nucléaires de Fontenay-aux-Roses (1986).

DUFFEY, T.A., Scaling Laws for Fuel Capsules Subjected to Blast, Impact and Thermal Loads, Rep. SC-RR-70-134, Sandia Labs, Albuquerque, NM (1970).

GIBSON, R., "The Safe Transport of Radioactive Material", Pergamon Press, Oxford and New York (1966).

GOODIER, J.N., THOMSON, W.T., Applicability of Similarity Principles to Structural Models, Rep. NACA-TN-933 (1944).

Guide de Criticité, Rep. CEA-R3114, Commissariat à l'énergie atomique, Paris (1967).

HEINICKE, W., MÜLLER, C.H., THOMAS, W., WARNEMUENDE, R., WEBER, W., Handbuch zur Kritikalität, Gesellschaft für Reaktorsicherheit, Garching (1970-1979).

HEUMOS, K., KOWALEWSKY, H., "Investigation of gas leakage from sealing construction at containments for radioactive materials", Packaging and Transportation of Radioactive Materials, PATRAM '80 (Proc. Symp. Berlin (West)), Bundesanstalt für Materialprüfung, Berlin (West) (1980).

INTERNATIONAL MARITIME ORGANIZATION/INTERNATIONAL LABOUR ORGANISATION, Guidelines for Packing Cargo in Freight Containers on Vehicles, London (1985).

INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, Sealed Radioactive Sources — Leak Test Methods, ISO/TR 4826, ISO, Geneva (1979).

KATAYAMA, Y.B., Leaching of Irradiated LWR Fuel Pellets in Deionized and Typical Ground Water, Rep. BNWL-2057 (1976), Battelle Pacific Northwest Labs, Richland, WA (1976).

KATAYAMA, Y.B., BRADLEY, D.J., HARVEY, C.O., Status Report on LWR Spent Fuel IAEA Leach Tests, Rep. PNL-3173 (1980), Battelle Pacific Northwest Labs, Richland, WA (1980).

KATAYAMA, Y.B., BRADLEY, D.J., HARVEY, C.O., Status Report on LSR Spent Fuel Leach Tests, Rep. PNL-3473 (1980), Battelle Pacific Northwest Labs, Richland, WA (1980).

KIRCHNER, B.P., BLUM, B.T., "Importance of leak flow mode assumptions for interpretation of package leak tightness tests and influence of test conditions on conclusions", Packaging and Transportation of Radioactive Materials, PATRAM '80 (Proc. Symp. Berlin (West)), Bundesanstalt für Materialprüfung, Berlin (West) (1980).

KLOTZKIN, G., VALENTINE, R.F., STACHEV, J.C., FLANAGAN, C.A., The Reactivity Effect of Replacing a Water Reflector with Lead (Transactions of the ANS Meeting, 1969), Vol. 12, American Nuclear Society, Chicago, IL (1969).

KOPONEN, B.L., Reactivity Enhancement in Transportation and Storage Arrays due to Fissile Material Density Reductions, Nucl. Technol. **34** 242 (July 1977).

KOPONEN, B.L., WILCOX, T.P., HAMPEL, V.E., Nuclear Criticality Experiments from 1943 to 1978, an Annotated Bibliography, Rep. UCRL-52769, California University, Livermore (1979).

KOWALEWSKY, H., "Criteria for and proof of leak-tightness concerning casks for dry transport and intermediate storage of spent LWR-fuel elements", Packaging and Transportation of Radioactive Materials, PATRAM '83 (Proc. Symp. New Orleans, 1983), Oak Ridge Natl Lab., Oak Ridge, TN (1983).

KOWALEWSKY, H., "Leak testing and activity leak rate evaluation — practical problems, first approaches to some systemization and outstanding problems", Packaging and Transportation of Radioactive Materials, PATRAM '80 (Proc. Symp. Berlin (West)), Bundesanstalt für Materialprüfung, Berlin (West) (1980).

LAKE, W.H., "Containment system evaluation", Packaging and Transportation of Radioactive Materials, PATRAM '83 (Proc. Symp. New Orleans, 1983), Oak Ridge Natl Lab., Oak Ridge, TN (1983).

LEISHER, W.E.B., BURNS, F.P., RODACY, P.J., TRUJILLO, A.A., "In-situ permeation tests of elastomer O-ring seals", Packaging and Transportation of Radioactive Materials, PATRAM '83 (Proc. Symp. New Orleans, 1983), Oak Ridge Natl Lab., Oak Ridge, TN (1983).

LEISHER, W.E.B., WEISSMAN, S.H., TALLANT, D.R., KUBO, M., "Helium/solid powder O-ring leakage correlation experiments", Packaging and Transportation of Radioactive Materials, PATRAM '83 (Proc. Symp. New Orleans, 1983), Oak Ridge Natl Lab., Oak Ridge, TN (1983).

MISHIMA, J., SUTTER, S.L., OWZARKI, P.C., SCHWENDIMAN, L.C., "Pressurised powder release through micro-openings on faulted containers", Packaging and Transportation of Radioactive Materials, PATRAM '80 (Proc. Symp. Berlin (West)), Bundesanstalt für Materialprüfung, Berlin (West) (1980).

O'DELL, R.D., Nuclear Criticality Safety, Rep. TID-26286, Office of Information Services, USAEC, Oak Ridge, TN (1974).

PAXTON, H.C., Criticality Control in Operations with Fissile Material, Rep. LA-3366 (Rev.), Los Alamos Sci. Lab., Los Alamos, NM (1972).

Proceedings on the International Symposium for Packaging and Transportation of Radioactive Materials (PATRAM), 1965, 1968, 1971, 1974, 1978, 1980 and 1983.

PRUVOST, N.L., KOLAR, O.C., Proceedings of the Livermore Array Symposium, CONF-680909 (1968).

Reports from the Criticality Data Center, ORNL-CDC-1 to 5 and Y-CDC-6 to 13, 1967-1971.

SANDIA NATIONAL LABORATORIES, Environment Experience by Cargo During Normal Rail and Truck Transport, Rep. SC-M-71-241, Sandia Natl Labs, Albuquerque (1971).

THOMAS, A.F., ABBEY, F., Calculational Methods for Interacting Arrays of Fissile Material, International Series of Monographs in Nucl. Energy, 108, Pergamon Press, Oxford and New York (1973).

THOMAS, J.T., "Evaluation of criticality criteria for fissile class II packages in transportation", Transport Packaging for Radioactive Materials (Proc. Sem. Vienna, 1976), IAEA, Vienna (1976).

THOMAS, J.T., Nuclear Safety Guide, Rep. TID-7016, Revision 2, NUREG/CR-0095, Oak Ridge Natl Lab., TN (1980).

Thomas' Stowage: The Properties & Stowage of Cargo, Nautical Press (1983).

Transactions of the American Nuclear Society Meetings, Nuclear Criticality Safety Division (twice annually) (also available through the Nuclear Criticality Information System, Lawrence Livermore National Laboratory).

TUCK, G., OH, I., Benchmark Critical Experiments on Low-Enriched Uranium Oxide Systems with H/U = 0.77, Rep. NUREG/CR-0674, Battelle Pacific Northwest Labs, Richland, WA (1979).

UNITED KINGDOM ATOMIC ENERGY AUTHORITY, Securing Radioactive Materials Packages to Conveyances, AECF 1006 (Revised), UKAEA, Harwell (1979).

YOSHIMURA, H.R., HUERTA, M., Analysis, Scale Modelling, and Full-Scale Test of Shipping Containers for Radioactive Materials, Rep. SAND 81-1314C, Sandia Labs, Albuquerque, NM (1981).

**LIST OF MEETINGS RELATING TO THE THIRD EDITION
OF SAFETY SERIES NO. 37
AND OF PARTICIPANTS AT THOSE MEETINGS**

- AG-445 Advisory Group on the Explanation and Application of the Agency's Transport Regulations, 21–25 May 1984.
- CM1 Consultants Meeting on Safety Series Nos 7 and 37, addressing the Provisions for Quality Assurance and Compliance Assurance, and the Topic of Nuclear Criticality, 3–7 September 1984.
- TC-458.2 Technical Committee on Assuring the Implementation of the Regulations for the Safe Transport of Radioactive Materials, 12–16 November 1984.
- CM2 Consultants Meeting to develop a Guide on Applying the Individual Dose Limitations and the Optimization Requirements in Transport Operations, 3–7 December 1984.
- CM3 Consultants Meeting to prepare Explanatory and Advisory Material on Criticality Assessment for the Safe Transport of Radioactive Materials, 21–25 January 1985.
- AG-445.2 Advisory Group to Revise Advisory Material for the Application of the IAEA Transport Regulations, 8–12 July 1985.
- CM4 Consultants Meeting to assist the Agency in preparing the Final Draft of Safety Series No. 37, 2–6 December 1985.

Designating Member States or International Organizations and names of Participants	Meeting						
	AG-445	CM1	TC-458.2	CM2	CM3	AG-445.2	CM4
ARGENTINA							
Biaggio, A.L.*	X					X	
AUSTRIA							
Mayr, K.*	X		X				
Patek, P.*	X					X	
BELGIUM							
Baekelandt, L. ⁺			X			X	
CANADA							
Dicke, G.*						X	
Eyre, P.*							
Hurda, A.						X	
Jackson, B.J.*			X				
Joseph, D.*	X					X	
Kotler, J.						X	
McLellan, J.*			X				
Taylor, W.R.*	X					X	
Thaller, R.A.*							
White, M.C.*							
FINLAND							
Vilkamo, S.	X						
FRANCE							
Blum, P.	X					X	
Geffroy, J.*							
Gilles, P.*							
Grenier, M.*	X	X	X			X	
Hamard, J.						X	
Ringot, C.	X						
Tanguy, L.						X	
Warniez, P.* ⁺		X				X	
GERMANY, FEDERAL REPUBLIC OF							
Alter, U.						X	
Collin, F.W.* ⁺	X	X	X			X	X
Kowalewsky, H.*							

Designating Member States or International Organizations and names of Participants	Meeting						
	AG-445	CM1	TC-458.2	CM2	CM3	AG-445.2	CM4
GERMANY, FEDERAL REPUBLIC OF (cont.)							
Ridder, K.*	X		X				
Schulz-Forberg, B.*							
Schweer, H.H.*							
Tamberg, T.*							
Wehner, G.*	X						
Wieser, K.*	X	X	X				X
INDIA							
Nandakumar, A.N.*			X				
Subrahmanian, G.	X						
Seetharamiah, P.							X
ITALY							
De Marco, I.*							X
Orsini, A.*	X						X
JAPAN							
Abe, H.*	X		X				X
Akiyama, H.							X
Fukazawa, A.*							
Fukuda, S.*							
Futamura, Y.	X						
Honna, R.*							
Kasai, A.					X		
Kataoka, K.*							
Kubo, M.*	X		X				X
Matsuo, Mr.*							
Nagahama, H.*			X				X
Nemoto, K.*							X
Nomura, M.*			X				X
Ohashi, M.*							
Ohno, T.							X
Onodera, A.*	X						
Sasaki, T.*							
Sato, M.*							X
Shibata, K.			X				X

Designating Member States or International Organizations and names of Participants	Meeting						
	AG-445	CM1	TC-458.2	CM2	CM3	AG-445.2	CM4
JAPAN (cont.)							
Shimada, H.*							
Suzuki, H.*							
Takeuchi, K.*							
Yasogawa, Y.*						X	
Yasunaka, T.*							
Yoshimura, S.	X						
NETHERLANDS							
Selling, H.			X			X	
POLAND							
Art, J.						X	
SPAIN							
Muñoz-González, M.J.						X	
SWEDEN							
Dufva, B.	X						
Pettersson, B.G.* ⁺			X			X	
Svahn, B.*	X		X			X	
SWITZERLAND							
Brélaz, P.*			X			X	
Stalder, F.						X	
UNITED KINGDOM							
Baker, C.*						X	
Blackman, D.J.		X	X				
Burgess, M.*						X	
Butterfield, R.S.*	X					X	
Cheshire, R.			X			X	X
Evans, M.C.					X		
Goldfinch, E.P.*	X		X			X	
Grover, J.R.*							
Hart, J.D.*							
Jankowski, G.*			X			X	
Jones, D.K.*							
Lindsay, G.E.*							
Livesay, E.*							

Designating Member States or International Organizations and names of Participants	Meeting						
	AG-445	CM1	TC-458.2	CM2	CM3	AG-445.2	CM4

UNITED KINGDOM (cont.)

Macdonald, H.F.*							
McLean, K.*							
O'Sullivan, R.A.* ⁺	X					X	
Pecover, C.		X	X				
Pollitt, N.*	X						
Shaw, K.B.* ⁺	X		X	X		X	
Spiller, G.T.*							
Williamson, S.*	X		X				
Young, C.N.*	X						X

UNITED STATES OF AMERICA

Allen, G.C.*	X						
Andersen, J.A.*							
De Maria, E.*							
Eckerman, K.*							
Gonzales, A.*							
Grella, A.*	X				X		
Hopkins, D.*						X	
Jefferson, R.M. ⁺						X	X
Lake, W.*							
Luna, R.E.*							
McClure, J.D.*					X		
Munro, J.J. III*							
Rawl, R.R.*	X		X				
Smith, D.						X	
Thomas, J.T.*		X				X	
Yoshimura, H.R.*							

**COMMISSION OF THE
EUROPEAN COMMUNITIES**

Marchal, M.A.			X			X	
---------------	--	--	---	--	--	---	--

**INTERNATIONAL AIR
TRANSPORT ASSOCIATION**

Abouchaar, J.						X	
---------------	--	--	--	--	--	---	--

Designating Member States or International Organizations and names of Participants	Meeting								
	AG-445	CM1	TC-458.2	CM2	CM3	AG-445.2	CM4		

**INTERNATIONAL CIVIL
AVIATION ORGANIZATION**

Cox, J.*

**INTERNATIONAL FEDERATION OF
AIRLINE PILOTS ASSOCIATION**

Venet, M. X

**INTERNATIONAL MARITIME
ORGANIZATION**

Wardelmann, H.* X X

**UNION INTERNATIONALE DES
PRODUCTEURS ET DISTRIBUTEURS
D'ENERGIE ELECTRIQUE**

Blythe, R.A. X

Evans, D.T.* X

**INTERNATIONAL ATOMIC
ENERGY AGENCY**

Pope, R.*⁺ X X X X X X X X

Koponen, H.*⁺ X

Levin, I.⁺

Barker, R. X

Raisić, N. X

Wenzel, U. X

* Served as a co-ordinator or contributor of text.

⁺ Provided final review of the text, June to September 1986.

INDEX

(by paragraph and table number)

A

Accident Conditions: 106.2, 142, 438, 462.2, 527.1, 542.1, 542.2, 547.3, 548.1, 548.6, 548.11, 553.1, 553.3, 558.1, 622.2, 627–629, 701.7 App. IV, 9.2, App. IX

Activity Limits: 101.2, Section III, 301.3, 304

A₁: 301, 303–306, 701.1, 701.11

A₂: 131.11, 301, 303–306, 407.4, 442, 548.1, 548.3, 548.5, 548.6, 548.8, 548.18, 627.1, 701.1, 701.11, 716.2

Air (transport by): 116.1, 132.2, 206, 460.4, 462.1, 473–475, 478.3, 515–517, App. I, App. II, 2, App. VI, App. VII, 1, Table AVII-1

Agency: 106.2, 116.3, 116.4, 201.1, 202.2, 207, 701.4, App. IV

Ambient Conditions: 562.6, 708.2

Approval Certificate: 210.8, 210.10, 315, 402.1, 701.6, 711, 716.2, 717, 724, 726–729, 730.2, 730.3

Approval — General: 116.1, 211.1, 315, 407.3, 555.2, 601.14, Section VII, App. IV, 4.1, 4.3, 6.1, 9.1, 13.1, Table AIV-1

Approval — Unilateral: 702.1

Approval — Multilateral: 715, 730, App. X, 7

Approval — Special Arrangement: 211.2, 720–721

B

Basic Safety Standards: 201.1

C

Carrier: 116.2, 128, 210.1, 210.7, 210.8, 210.14, 429, 462.1, 462.2, 469.2, 473, 561, 701.1, 701.11, App. V

Categories: 204, 435

Category: 204, 421.2, 435, 447.3, 555.2, App. IV, 2.3, App. VI

Combustible: 131.11

Competent Authority: 116, 130.1, 203.1, 209.1, 209.3, 209.4, 210, 211.1, 402.3, 438, 471, 472.1, 472.4, 483.4, 483.7, 484.2, 503.2, 547.7, 558.2, 628.1, 701.1–701.7, 701.11, 702, 705, 708.1, 708.3, 711, 713–714, 715, 724, 726–729, 730, App. IV, 1, 2.1, 3.1, 5.1, 6.1, 6.3, 13, App. X, 6, 7

Compliance Assurance: 101.2, 117, 210, 602.1, 701.11, App. IV, 1.1

Consignee: 116.2, 408.1, 429, 447.1, 469.2, 483.4, 484, 701.11, App. X, 7.3

Consignment: 128, 210.8, 407.3, 447, 473, 478.3, 701.11, 716.1, App. IV, 10, App. VI, App. X, 7, Tables AX-1&AX-2

Consignor: 116.2, 128, 144.3, 209.3, 210.1, 210.2, 210.7, 210.8, 408.1, 428.1, 429, 447, 462.1, 469.2, 477, 483.4, 484, 701.1, 701.11, App. IV, 2.1, 2.2, 6.3, App. V, App. X, 7

Containment: 101.1, 132, 208, 401.2, 401.4, 421.3, 460.5, 512.4, 516, 518.5, 527.1, 533, 548.1, 548.11, 548.14, 550, 553.1, 562.2, 562.3, 563.1, 566.2, 615, 617, 622.2, 627.2, 627.3, 627.7, 627.10, 701.7, 701.11, App. IV, 2.3.2, 2.3.5, App. X, 4, 5

Contamination: 101.2, 122-124, 131.12, 204, 210.8, 408-413, 417.1, 421.2, 472.1, 508.3, 512.1, 537.2, 548.15, 721, App. II, App. IV, 10, App. VIII - Art. 8, 11, 12

Fixed: 123-124, 144.3, 408.1, App. II

Non-Fixed: 123-124, 144.3, 408.1, 408.2, 413.1, 417.2, 420.2, 421.1, 421.2, 425, App. II

Conversion: Table IV-1

Conveyance: 101.2, 123-124, 128, 131.10, 144.3, 205.1, 410.1, 412, 413.1, 425, 428.1, 434, 462.1, 462.3, 462.4, App. II, 2, 4, App. IV, 9.1, App. VII, 1, 2.2

Limits: 131.11

Cooling: 132.2, 408.1, 408.2, 410.2, 601.7, 701.11, App. II-1, App. VIII - Art. 5, Table AVIII-1

Criticality: 101.1, 129.2, 129.3, 315, 401.6, 407.1, 429, 472.2, 479, 506.4, 561, 562.1, 562.3, 563.1, 566.1, 568.2, 601.16, 617.1, 617.11, 627.3, 627.8, 701.11, App. IV-2.2, 2.3, 2.3.4, 2.3.5, App. X

Customs: 483

D

Dangerous Goods: 130.2, 407.2, 407.7, 436.3, 460.5, Table IV-3, 471, 512.4, 519.3, 519.5, App. I, Table AVI-1, App. VIII - Art. 1

Deck Area: 465

Decontamination: 408.1, 408.2, 413.1, 508.1, 508.3, App. II-1, 4, App. IV, 1.2, App. VIII - Art. 11, 12

Dose Limitation: 201.3, 202.1

E

Emergency: 207, 208, 210.3, 410.2, 429, 701.11, 721, App. IV, 9.1, App. VII, 1, Table AVIII-1

Empty Packaging: 421, 447.2, 447.3

Excepted: 105.1, 142, 209.3, 301.2, 301.3, 407.6, 413.1, 416-420, 421, 435, 447.2, 447.3, 701.11, App. IV, 2.3.1

Exclusive Use: 128, 209.5, 210.8, 407.4, 466, 469.1, 471, 473, 475, 555.1, 701.11, 721, App. VIII

F

Fabrication: 210.8, 401.3, App. IV, 7.1

Filter: App. II-3

Fissile: (*See also Criticality*) 101.1, 116.1, 129, 131.1, Section III, 315, 407.6, 472.2, 561-563, 566, 568, 617.1, 622.2, 627.2, 627.3, 627.8, 701.1, 701.2, 701.11, 710, 715, 716.1, 724, App. IV, App. X, 2, 4, 5, 7, 7.2, 7.3

Freight Container: 128, 130, 428.1, 447.1, 465, 523.2, 523.4, 527.2, App. I

G

Gas: 132.2, 407.2, 503.2, 512.1, 519.2, 520, 537.5, 548.1, 548.5, 548.7-548.9, 548.13, 548.14, 558.3, 617.5, 617.6, 628.20, 628.25, 628.27

Group: 129.3, 407.4, 518.2, 519.3-519.5, App. VI

H

Heat: 101.1, 132.2, 209.5, 401.5, 472.3, 515, 528, 530, 543.1, 543.3, 543.4, 545, 546.1-546.3, 546.7, 546.8, 547.1, 555.2, 556.2, 601.7, 628, 701.11, App. IV, 8

I

Identification Mark: 438, 519.5, 701.11, 703, 724, 726-729, 730.3

Industrial Package: 407.4, 407.6, 413.1, 518-523, 617.1, 622.2, 701.11, App. IV, 2.3.1, 2.3.2

Industrial Type Packages: (*See Industrial Package*)

Insolation (solar radiation): 543.1, 546, 628.7, 628.11, 628.16, 708.2

Inspection: 210.3, 210.7-210.12, 401, 402.2, 410.1, 436.5, 438, 483.1, 483.4, 483.7, 537.4, 537.5, 617.1, 628.7, App. IV, Table AIV-1, 1.2, 2.1, 2.2, 2.3, 3.1, 6.1, 9, 13, App. V, App. VIII - Art. 11, 12, App. IX

L

Label: 101.2, 209.4, 209.5, 210.8, 407.8, 420.2, 435, 436.3, 442, App. IV, 9.4, 10

Leaching: 503.1, 508.3, 604

Leakage (Loss): 129.3, 401.4, 402.4, 408, 410, 460.5, 483.5, Table V-1, 503, 517, 519.3, 535, 537.1, 537.4, 548.1–548.9, 548.11–548.15, 550.3, 553.1, 553.3, 558.3, 566.1, 601.17, 604, 606, 617.1, 617.3–617.7, 628.7, 628.10, 628.29, 701.11, App. IV, 2.2, App. X, 2, 3, 4, 5

Low Specific Activity (LSA): 131, Fig. I-1, 144.1, 301.3, 407.4, 422, 428.1, 519.3, 519.4, 601.1

M

Maintenance: 144.2, 209.5, 210.4, 210.8, 210.9, 436.5, 438, 469.2, 627.2, 701.11, App. IV, 1.2, 2.2, 9.2, 9.4, 13.1, 13.2, 14, App. V, App. VIII - Art. 1, Table AVIII-1

Marking: 101.2, 210.8, 410.2, 436, 438, 519.4, 519.5, 527.3, 537.8, 701.11, 703, 724, 726–729, 730.3, App. IV, 2.1, 9.4, 11, App. VI, App. VIII - Art. 8&9

Mass: 129.1, 143.1, 421.3, 436.1, 506.4, 519.4, 519.5, 523.1, 526.1, 527.2, 548.1, 566.2, 601.12, 601.13, 611.1, 617.6, 618, 622.3, 622.5, 623.2, 627.1, 627.6, 627.9, 627.11, 701.11, 713–714, App. III - Table AIII-2, App. VII, Fig. AVII-1, App. X, 2, 7.2, 7.3, Tables AX-1&AX-2

Maximum Normal Operating Pressure: 132, 553

Multilateral: 715, 730, App. X, 7

N

Normal Conditions: 132.2, 407.6, 462.1, 462.2, 469.3, 518.4, 521.2, 523.3, 527.1, 537.1, 537.2, 543.3, 545, 546.6, 546.7, 548.1, 548.6, 548.10, 553.1, 553.3, 558.1, 563.1, 621, 622.1, 628.16, 701.7, App. IV, 9.2, App. VII-1

Notification: 210.8, 475, 715

Number “N”: App. X, 5, 7.3

O

Operational Controls: 132.2, 210.8, 211.1, 211.2, 558.1, 558.2, 701.11, 708.2, 721, App. IV, 3.1

Other Dangerous Properties: 105.1, 116.1, 407, App. IV, 2.3

Overpack: 428.1, 433, 435, 447.1, 466, 469.1, 469.3, 475, 547.6

P

Package Design: 116.1, 129.3, 202.1, 208, 209.1, 210.4, 210.6, 210.8, 210.10, 211.1, 211.2, 401.1, 402.1, 402.3, 402.4, 407.3, 407.6, 506.2, 508.1, 518.2, 519.3, 526.2, 528, 530, 535, 536, 537.4, 537.8, 538, 542.1–542.3, 548.12, 550.2, 553.1, 555.2, 563.1, 601.2, 601.6, 601.14, 601.21, 627.14, 628.1, 628.2, 701.5, 701.8, 701.11, 705, 708, 710, 713–714, 715, 716.2, 717, 720, 724, App. IV, 1.2, 2.1, 2.2, 2.3.1, 4.1, 5, 6.1, 9.2, 13.2, App. V, App. VII, Table AVII-1, App. IX, App. X, 3, 5, 7

Package — Excepted: 105.1, 142, 209.3, 301.2, 301.3, 407.6, 413.1, 416, 421.1, 435, 447.2, 447.3, 701.11, App. IV, 2.3.1

Package — Industrial: 407.6, 518–523, 622.2, 701.11, App. IV, 2.3.1, 2.3.2

Package — Type A: 116.1, 142, 301.3, 407.6, 413.1, 526–540, 543.1, 617.1, 622.2, 701.11, 724, App. IV, 2.3.2, App. X, 2

Package — Type B: 116.1, 402.4, 407.6, 413.1, 542–548, 617.1, 622.2, 627.12, 701.1, App. IV, 2.3.4, 2.3.5

Package — Type B(U): 438, 550–556, 701.11, 705, 708.2, 715, 724

Package — Type B(M): 438, 473, 558, 701.11, 708, 715, 716, 724

Packaging: 132.2, 209.5, 210.1, 210.8–210.10, 315, 401.1–401.3, 401.5, 401.6, 407.4, 407.6, 407.7, 420, 421, 422, 433.2, 436.4, 436.5, 447.2, 447.3, Section V, 506, 508, 512.1, 512.2, 515, 519.2–519.5, 526.2, 527.1, 537.3, 537.5, 537.8, 542.4, 543.1, 547.4, 548.2, 553.1, 562.2, 562.4, 568.2, 601.2, 601.3, 601.9, 601.14, 601.16, 601.20, 617.1, 617.12, 622.1, 627.5, 627.11, 627.12, 627.13, 701.6, 701.9, 701.11, 715, 716.2, 720, 721, 730.3, App. IV, 1.3, 2.1, 2.2, 4.1, 5.1, 9.1, 9.4, App. X, 2, 5, 7, 7.1

Placard: 209.5, 210.8, 435

Placarding: (*See Placard*)

Post: 476, 477, App. I

Pressure: 132, 401.2, 402.3, 407.3, 407.5, 407.6, 410.2, 516, 518.2–518.4, 519.4, 519.5, 537.4, 537.5, 538, 548.7–548.9, 548.14, 553.1, 553.3, 553.4, 558.1, 617.5, 617.6, 628.2, 628.7, 628.9, 628.11, 628.16, 628.29, 628.30, 629.1, 629.3–629.5, 630, 701.7, App. II, 1, 3, App. IX

Q

Quality Assurance: 101.2, 136, 209, 210.4, 210.9, 210.10, 401.1, 402.1, 402.2, 518.2, 701.9, 701.11, App. IV, App. X, 3

R

Radiation — Control: 101.1, 407.1, 561, App. VIII - Art. 1, 2, Table AVIII-1

Radiation — Dose equivalent: 428.3, Table IV-1, App. VI

Radiation — Dose rate: 205, 470.1, 472.1, App. II, 4, App. VIII - Art. 2

Radiation — Exposure: 201.3, 203.1, 410.2, 421.3, 460.1, 470.1, 478.1, 483.2, 558.3, 561, App. II, 1, 4, App. VI, App. VIII, Table AVIII-1, Art. 2, 7&10

Radiation — Level: 101.1, 123–124, 131.2, 131.10, 204, 209.5, 401.3, 410.2, 416, 418, 420.2, 421.1, 421.3, 422, 428.1, 428.3, 433, Table IV-2, 434, 460.3, 469.1, 470.1, 471, 472.1, 478.2, 478.3, 521.4, 523.4, 537.6–537.8, 542.1–542.5, 553.3, 617.9, 617.10, App. II, 2, 3, 4, App. IV, 2.3.1, 10, App. VI

Radiation — Protection: 201, 202, 205.1, 416, 428.1, 460.1, 469.2, 469.3, 471, 472.1, 472.3, 472.4, 472.6, 483.1, 483.6, 558.2, 627.3, 701.1, 701.11, 721, App. VI, App. VIII

Radiation — Shield: 401.3, 410.2, 536, 627.3, 701.11

Rail (transport by) 116.1, 462.1, 469, 547.3, App. I, App. II, 2, 4, App. VII, 1, Table AVII-1

Responsibility: 116.1, 447, 462.1, 561, 701.1, 701.3, 701.11, App. IV, Table AIV-1, 3.1, App. VI, 1.3, 2.1, 3.1, 3.2, 11

Reviews: 203.2, 210.3, 210.5, 210.6, 408.2, 601.20, App. IV, Table AIV, 1, 2.1, 2.2, 3.1, 3.2, 4.1, 4.3, 5.1, 5.3, 6.1, 11, 13.1

Road (transport by): 116.1, 462.1, 469, 478.3, 547.3, 721, App. I, App. VII, 1, Table AVII-1

S

Segregation: 205, 206, 406, 460, Table IV-3, 478.2, 479, 483.3, 483.5, App. IV, 11, App. VI

Serial Number: 438, 715, App. IV, 13.2

Shielding: 131.2, Fig. I-1, 401.3, 410, 422, 469.2, 519, 521.3, 521.4, 523.2, 523.4, 523.5, 527.1, 528, 536, 537.6, 537.8, 537.9, 542.2, 542.4, 542.5, 550.1, 553.1, 601.16, 617, 622.2, 627.2, 627.3, 628.2, 628.10, 628.29, 701.11, 713–714, App. IV, 2.2, 2.3.2, 2.3.5

Shipment: 106.1, 116.1, 128, 206, 209.5, 210.6, 210.8, 211.1, 211.2, Section IV, 401, 402, 465, 701.1, 701.6, 701.7, 701.9, 701.11, 716, 717, 720, 721, 724, App. IV, 1.1, 1.2, 1.3, 2.3.6, App. X, 7, 7.3

Shipping: 210.8, 447.2, 721, App. IV, 2.2, 7.2, Table AIV-1, App. X, 2

Special Arrangement: 141, 209.5, 211, 421.2, 471, 475, 701.9, 701.10, 701.11, 720, 721, 724, App. X, 7, 7.3

Special Form: 142, 210.6, 503, 601.1, 604, 627.1, 701.1, 701.7, 701.9, 701.11, 702, 724, App. IV, 2.3.4

Specific Activity (See Low Specific Activity): 131.2, 131.3, 131.5, 131.9–131.12, 143, 407.4, 421.2, 548.5, 701.11, Tables AIII-1&2

Storage in Transit: Section IV, 478–479, 483.5, 483.7, 528, 561, 701.11, App. II-2, App. IV, Table AIV-1, 1.2, 2.1, 7.2, 10, App. VIII - Art. 1, 13, App. X, 7.3

Surface Contaminated Objects (SCO): 144, 425, 428.1, 519.3, 519.4

T

Tank: 421.1, 421.2, 428.1, 518.2, 521, 527.2, App. VIII - Art. 13

Tank Container: 521.1, 521.3, 521.4

Temperature: 402.3, 407.5, 410.2, 512.3, 512.4, 515, 516, 518.2, 518.5, 528, 538, 542.5, 543.1, 543.3, 543.4, 545, 546.1, 546.3, 546.7, 546.8, 547.5, 548.7, 553.1, 555, 556.2, 562.6, 611.2, 627.8, 628.2, 628.7, 628.9, 628.11, 628.13, 628.14, 628.16, 628.17, 628.22, 628.24–628.30, 708.2, App. VIII-Art. 5, App. IX

Tests (mechanical and thermal): 210.10, 401.2–401.5, 402.1–402.4, 407.6, 407.7, 462.2, 503, Table V-1, 506.2, 518.2, 518.3, 519.3–519.5, 521.4, 523.4, 533, 537.3–537.5, 537.7–537.9, 542.1–542.3, 542.5, 543.3, 543.4, 546.6–546.9, 553.1, 553.4, 555.2, 558.1, 562.6, 566.2, Section VI, 601, 602, 604, 606, 608, 611, 615, 617, 618, 621, 622, 623, 627, 628, 629, 630, 701.3, 701.9, 701.11, App. II, 3, App. IV, Table AIV-1, 1.2, 2.2, 9, 13, App. V, App. VII, 1, App. IX, App. X, 3, 4

Tie-down: 462, 527, 701.11, 721, App. VII

Transport Documents: 101.2

Transport Index (TI): 146, 209.5, 210.8, 315, 428, 429, 433, 466, 472.2, 561, 566.1, App. VI, Tables AVI-1&2, App. X, 5, 7.3

Transport Organizations: 130.2, 406, 407.3, 407.8, 407.9, 460.2, 701.4, App. VI

Transport Workers: 201.2, 204, 410.1, 460.2, 558.2, 558.3, App. II, 2.4

U

Ullage: 512.4, 518.5, 538

Unilateral: 702.1

UN Numbers: 447.2

Unpackaged: 425, 428.1, 566.1

V

Vehicle: 128, 209.5, 210.8, 436.1, 469, 470, 471, 478.3, 483.5, 519.5, 527.2, 528, 547.3, 628.3, 721, 726–729, App. V, App. VII, 2.1, 2.2, 2.3

Venting: 132.2, 558.1–558.3, 708.3

Vessel: 401.2, 407.5, 407.6, 465, 472.1, 472.6, 518.2–518.4, 553.1, 701.1, App. VIII, App. IX

W

Water: 116.1, 208, 407.2, 408.2, 410.2, 503.2, Table V-1, 548.1, 550.2, 562.3, 563.1, 611.2, 617.6, 621, 628.2, 628.3, 628.26, 628.29, 629.1, 629.2, 630, 701.11, App. II, 1, App. VIII - Art. 13, App. X, 4, 5, 7.2

White Label: *See Categories*

Y

Yellow Label: *See Categories*

This publication is no longer valid
Please see <http://www-ns.iaea.org/standards/>

HOW TO ORDER IAEA PUBLICATIONS

An exclusive sales agent for IAEA publications, to whom all orders and inquiries should be addressed, has been appointed in the following country:

UNITED STATES OF AMERICA BERNAN – UNIPUB, 4611-F Assembly Drive, Lanham, MD 20706-4391

In the following countries IAEA publications may be purchased from the sales agents or booksellers listed or through major local booksellers. Payment can be made in local currency or with UNESCO coupons.

ARGENTINA	Comisión Nacional de Energía Atómica, Avenida del Libertador 8250, RA-1429 Buenos Aires
AUSTRALIA	Hunter Publications, 58 A Gipps Street, Collingwood, Victoria 3066
BELGIUM	Service Courrier UNESCO, 202, Avenue du Roi, B-1060 Brussels
CHILE	Comisión Chilena de Energía Nuclear, Venta de Publicaciones, Amunategui 95, Casilla 188-D, Santiago
CHINA	IAEA Publications in Chinese: China Nuclear Energy Industry Corporation, Translation Section, P.O. Box 2103, Beijing IAEA Publications other than in Chinese: China National Publications Import & Export Corporation, Deutsche Abteilung, P.O. Box 88, Beijing
CZECHOSLOVAKIA	S.N.T.L., Mikulandska 4, CS-116 86 Prague 1 Alfa, Publishers, Hurbanovo námestie 3, CS-815 89 Bratislava
FRANCE	Office International de Documentation et Librairie, 48, rue Gay-Lussac, F-75240 Paris Cedex 05
HUNGARY	Kultura, Hungarian Foreign Trading Company, P.O. Box 149, H-1389 Budapest 62
INDIA	Oxford Book and Stationery Co., 17, Park Street, Calcutta-700 016 Oxford Book and Stationery Co., Scindia House, New Delhi-110 001
ISRAEL	Heiliger and Co., Ltd, Scientific and Medical Books, 3, Nathan Strauss Street, Jerusalem 94227
ITALY	Libreria Scientifica, Dott. Lucio de Biasio "aeiou", Via Meravigli 16, I-20123 Milan
JAPAN	Maruzen Company, Ltd, P.O. Box 5050, 100-31 Tokyo International
PAKISTAN	Mirza Book Agency, 65, Shahrah Quaid-e-Azam, P.O. Box 729, Lahore 3
POLAND	Ars Polona-Ruch, Centrala Handlu Zagranicznego, Krakowskie Przedmiescie 7, PL-00-068 Warsaw
ROMANIA	Ilexim, P.O. Box 136-137, Bucharest
SOUTH AFRICA	Van Schaik Bookstore (Pty) Ltd, P.O. Box 724, Pretoria 0001
SPAIN	Díaz de Santos, Lagasca 95, E-28006 Madrid Díaz de Santos, Balmes 417, E-08022 Barcelona
SWEDEN	AB Fritzes Kungl. Hovbokhandel, Fredsgatan 2, P.O. Box 16356, S-103 27 Stockholm
UNITED KINGDOM	Her Majesty's Stationery Office, Publications Centre, Agency Section, 51 Nine Elms Lane, London SW8 5DR
USSR	Mezhdunarodnaya Kniga, Smolenskaya-Sennaya 32-34, Moscow G-200
YUGOSLAVIA	Jugoslovenska Knjiga, Terazije 27, P.O. Box 36, YU-11001 Belgrade

Orders from countries where sales agents have not yet been appointed and requests for information should be addressed directly to:



**Division of Publications
International Atomic Energy Agency
Wagramerstrasse 5, P.O. Box 100, A-1400 Vienna, Austria**

This publication is no longer valid
Please see <http://www-ns.iaea.org/standards/>

INTERNATIONAL
ATOMIC ENERGY AGENCY
VIENNA, 1987

SUBJECT GROUP: II
Nuclear Safety and Environmental Protection/Radiological Safety
PRICE: Austrian Schillings 480,-