Technical Basis for the IAEA Regulations for the Safe Transport of Radioactive Material (SSR-6)

Support for the IAEA project “Progress and Justification of the Technical Basis SSR-6”

Extended draft by Ronald B. Pope
Chapter 11 removed pending detailed and comprehensive edit based upon multiple review comments from criticality experts

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DRAFT
REFERENCES:
All citations to references are done on a chapter by chapter basis, as recommended by TRANSSC.
Each citation in the text (except in Chapter 11) is highlighted in gray, e.g. [12].

It is noted that most of the earlier reference citations are in sequence in each chapter (identified by [12] as noted above); however, with the latest additions of new documents (generally identified by (RBPWXY), where “callouts” in the text are identified as [RBPWXY]) they are not in proper sequence although they are cited at the end of the chapters according to these identifiers. In addition, where documents already cited in one chapter and already incorporated into the Sharepoint database have been used in another chapter, these are numbered with that chapter with a high reference number and are so identified as bolded blue citations (e.g. [23]). As a result, the references identified as (RBPWXY) and in bold blue text (e.g. [23]) will ultimately need to be sorted sequentially as they appear in the text. However, this task should only be undertaken when the text of the document is deemed by the Secretariat to be at the appropriate level of being made “final for uploading to the IAEA website”.

In the lists of references in this draft (except only in parts of Chapter 11), the availability of these documents is shown in brackets (i.e. ... \{ \}), in red with yellow highlighting.

All efforts have been made to have as many as possible references included in electronic format, which will make them available on an IAEA “Sharepoint” website. When the document is available on Sharepoint, it is designated in this draft based on the individual who place that document on Sharepoint. For example, those documents identified with “RPXXX” were placed on the sharepoint site by R Pope; whereas “CBXXX” were placed on the sharepoint site by Chris Bajwa, and “DMXXX” were placed on the sharepoint site by Dennis Mennerdahl). Thus, the first reference cited in Chapter 1 is shown as (RP001).

Late additions to be made to Sharepoint (as of January 2014) are identified as RBPXXX.

Where references have not yet been made available electronically, they are identified in the reference number (e.g. [22]) with green highlighting.

APPENDICES:
The call-out of appendixes in the text is in bold with green highlighting.

NOTES ON ISSUES, ETC.:
Where additional attention is needed in the text or issues have been identified, endnotes are cited, and these appear at the very end of the document.

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FOREWORD

This report contains the latest (June 2015) draft version of a compilation of information with respect to the technical bases for the International Atomic Energy Agency (IAEA) "Regulations for the Safe Transport of Radioactive Material – 2012 Edition", Specific Safety Standards No. SSR-6*.

The report is the culmination of an effort undertaken by the IAEA in the latter part of the first decade of the twenty-first century to develop comprehensive documentation on the bases for the technical requirements in the Regulations. In the opening remarks of TM-41001 (one of the meetings working on the development of the technical basis, which was convened in Vienna 14-18 March 2011), the following was noted:

"The Regulations for the safe transport of radioactive materials have been revised more than 10 times since its first publication in 1961. Member States and International Organizations have put lots of efforts, time and resources including human and financial resources. These efforts contribute to excellent safety record of transport of radioactive material over 50 years.

"Technical bases obtained from research, feedback, experience are the foundation of the regulations. They are of great value not only to the users but also to developers. Some approaches used in the early editions of the Transport Regulations have remained virtually unchanged up to the present day. They have proved to be sound.

"Currently these technical bases are stored in decentralized ways among Member States and international organizations. When long-serving experts retire, young professionals come or governments are reorganized, this valuable knowledge will face risk of loss. One of the purposes of this meeting is to collect, sort, transfer, and store these valuable knowledge for future use.

"In addition, easy access to technical bases could lead to better understanding of the regulations. That will facilitate the implementation of the Regulations.”

This compilation is structured essentially to the format of the chapters in the Transport Regulations.

Finally, an extensive set of references were used in developing this technical basis. References are cited within each chapter and listed at the end of each chapter. The listing of references includes (at least in this draft version) the source of each reference {in red bracketed text} as available to the compiler when assembling this interim draft. This extensive use and listing of references, along with making as many as them available on a Share Point site at the Agency, goes a considerable way in satisfying the desire as noted in the opening comments to TM-41001 of providing “easy access to technical bases could lead to better understanding of the regulations. That will facilitate the implementation of the Regulations.” However, as can be seen, some of the referenced documents are only available to individual experts in hardcopy. To have them readily available to future generations, scanning of the documents and storing them electronically may be necessary.

* The IAEA Transport Regulations are under continuous review and are periodically revised. The Transport Regulations were first issued in 1961, and have been re-issued multiple times since. The latest edition is dated 2012, and is identified as SSR-6. The various editions of the Transport Regulations (and similar various editions of supportive guidance documents) are cited, as appropriate, throughout this document.
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PREFACE

The IAEA Regulations for the Safe Transport of Radioactive Material were first published in 1961 and have been revised more than 10 times since; the latest edition was published in 2012, as SSR-6. For more than five decades, the IAEA, Member States and International Organizations have contributed a great deal of effort, time and resources including human and financial resources to the development, review and maintenance of these editions of the Regulations. The Regulations, in turn, serve as the primary basis for the development of international modal regulations and individual Member States regulations. These efforts have contributed to the excellent safety record of the transport of radioactive material over these 50 plus years.

The technical bases obtained from research, feedback, experience have served as the foundation for the regulations. As this effort moved from its 5th to 6th decade, it was recognized that some approaches used in the early editions of the Transport Regulations have remained virtually unchanged up to the present day; they have proved to be sound; while other approaches have evolved over the years as reflected by additions, deletions and changes to the regulatory requirements.

It was recognized that much of the technical details underlying the technical basis of the Regulations were stored in a decentralized and inconsistent manner among Member States, the IAEA, other involved international organizations, and various experts. Also, it was possible that if some of this information was not captured it could become lost to the transport safety community. For example, when long-serving experts retire, they are replaced by younger professionals; or when governments are reorganized, this valuable knowledge potentially faces risk of loss.

Thus beginning in 2010, the IAEA undertook an effort to capture as much of this information and knowledge as possible; with a goal of producing a “Technical Basis Document” (TecBasDoc). As this effort has proceeded it has become clear that the document will need to be a “living document”; growing with time as resources are spent searching out, capturing and summarizing relevant documents, and then maintaining the TecBasDoc as further reviews and revisions to the Regulations occur.

In addition, with the advent of the broad use of electronic media, it was decided to “capture” the relevant documents electronically wherever possible and make them available world-wide. Providing this easy access to technical bases could lead to better understanding of the regulations by regulators and users facilitating the implementation of the Regulations, and of those experts tasked with reviewing and proposing revisions to them. With respect to such reviews and proposals for change, it is anticipated that the TecBasDoc will improve the methodology and quality of review or revision by requiring that each proposal should show that an existing requirement was justified with a sound technical basis, and any proposed change would require an assessment justifying why a change or addition is needed before the proposal is submitted.

Various meetings of experts were undertaken as shown in Appendix 6. The ultimate goal of this effort is to have a “living document” that provides a narrative history of the development of the regulations, the technical bases for the decisions that were made regarding the requirements in the regulations, and the appropriate historical references, all of which will be available electronically, to provide the necessary explanations of the “whys” behind the current regulations.
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1. INTRODUCTION

1.1. Background

Beginning in 1957, the International Atomic Energy Agency (IAEA) was assigned the tasks of formulating regulations governing the transport of radioactive materials. As summarized in Appendix 1, since that time, the IAEA has vigorously pursued the development and maintenance of the Regulations for the Safe Transport of Radioactive Material, including cooperation and coordination with relevant international bodies responsible for ensuring safety of dangerous in transport.

The first edition of the Transport Regulations was issued in 1961 [1], and updates reflecting added knowledge, experience and technology have been issued periodically. The history of issuing the regulations and the supporting guidance and explanatory documents is shown in Table 1-1. In addition, during the early phases of the development of the regulations, an draft version of the regulations was issued to involved regulators, noting that “In presenting the attached regulations, the International Atomic Energy Agency aims to propose safety rules which can govern, in national or international traffic, the transport of radioactive materials through all means of transport” [RBP034].

Beginning in 2002, the advisory and explanatory documents were combined into a single document.

The IAEA works closely with Member States, involved international organizations and interested non-governmental organizations with a view to having the resulting Regulations reflect a worldwide view which has led to a generally consistent application of the requirements by the States and international modal organizations. By periodically issuing the Transport Regulations, the IAEA is providing a comprehensive, internationally agreed, consensus set of recommended safety provisions for the transport of radioactive material. The provisions focus in part on a detailed set of criteria for ensuring safety during all phases of transport by all modes and during interim storage during transport.

The Transport Regulations are developed and maintained through a cooperative process involving Member States, international governmental organizations, and international non-governmental organizations. Currently (2012) this process involves at the IAEA the Secretariat, led specifically by the Transport Safety Unit; four IAEA safety standards committees, where focus for transport safety lies specifically with the Transport Safety Standards Committee (TRANSSC); and an oversight commission, known as the Commission on Safety Standards (CSS). Through this process, all Member States and the relevant governmental organizations have the opportunity to be actively involved in advancing the Transport Regulations. A similar but less robust oversight and advisory structure preceded the formation of TRANSSC, starting in 1977 with the IAEA Standing Advisory Group for the Safe Transport of Radioactive Material (SAGSTRAM).

Each edition of the Regulations is approved for publication by the IAEA Board of Governors. This effort is now in its seventh decade. As shown in Table 1-1, fifteen revisions, amended editions, or supplements of the Regulations have been issued, and this effort is expected to continue beyond 2012. These revisions have been taking into account the world-wide experiences, new issues, new technologies, best practices, lessons-learned from their application, the demand for safer transport, and the need for harmonization. Problems, challenges and the demand for improvements drive the transport community to continuously review and revise the Regulations.

In the early part of the second decade of the Twenty-first Century, TRANSSC recognized that the scientific and technical heritage of these past several decades of development in transport
safety needs to be assessed with a view to preserving the valuable knowledge involved in this extensive, long-term effort. This document represents the results of an effort undertaken by the IAEA to preserve the knowledge of the technical basis.

Table 1-1. History of issuing transport safety regulatory documents*

<table>
<thead>
<tr>
<th>Edition</th>
<th>Regulations</th>
<th>Explanatory</th>
<th>Advisory</th>
</tr>
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<tbody>
<tr>
<td>1964 (1964 revised edition)</td>
<td>SS-06 (1965)</td>
<td></td>
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<tr>
<td>1986 (1985 Edition (Supplement))</td>
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<td>1988 (1985 edition (Supplement))</td>
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<tr>
<td>2012 (2012 edition)</td>
<td>SSR-6</td>
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** Notes on Certain Aspects of the Regulations

The development of this document involved a large number of experts, most of which had decades of experience in the development of the international Transport Regulations. Supporting this effort, the experts developed an extensive table that considered each paragraph in the Regulations and assessed whether they represented a requirement and how the technical basis for the requirement was (or maybe was not) established; and also developed a second table that lists a number of issues related to paragraph-related requirements, options and specifications. These two tables are not part of this document, but are available for use by and from the Secretariat.
The information and discussion contained in this document is based upon the 2012 edition of the Transport Regulations [2]. However, consideration was given to other features of earlier editions of the Regulations; where such considerations were deemed to be significant those earlier regulatory recommendations are identified.

Generally every requirement in the Transport Regulations was developed on a technical basis, though often the basis was not adequately documented, and/or the basis exists in a decentralized manner in many Member States with mature nuclear programmes. This limits the access of users to the technical basis for the requirements. It was recognized by TRANSSC that broad access to the technical basis could lead to a better understanding of the Regulations; where sharing and pooling of knowledge can contribute, in many cases, to the development and innovation of methods for ensuring transport safety and could, concurrently, prevent unnecessary efforts by newly-involved personnel attempting to "re-invent the wheel".

The 2012 edition of the Transport Regulations [2] is in the safety requirements category of the IAEA safety standards. Thus, the Regulations have to be consistent with the IAEA Safety Fundamentals [4] and with relevant general safety requirements, in particular the International Basic Safety Standards (BSS) [5]. A schematic view of the 2012 structure of the safety requirements is given in Appendix 2.

It is noted that, according to the 2012 edition of the Transport Regulations that the “These Regulations apply to the transport of radioactive material by all modes on land, water, or in the air, including transport that is incidental to the use of the radioactive material”. It further specifies that transport “comprises all operations and conditions associated with, and involved in, the movement of radioactive material; these include the design, manufacture, maintenance and repair of packaging, and the preparation, consigning, loading, carriage including in-transit storage, unloading and receipt at the final destination of loads of radioactive material and packages”. Thus, the Transport Regulations do not cover the physical movement of radioactive material from one place to another within a fixed facility.

1.2. Purpose

The purpose of this document is to provide a single, comprehensive source of information relative to preserving, as best as could be done in the 2011 – 2013 time frame, knowledge concerning the technical basis for the requirements set forth in the Transport Regulations. To the extent possible, the document looks back into historical documents to define the logic behind the requirements that were initially introduced into the Regulations, and how these requirements have changed with time.

1.3. Scope

[Text yet to be developed and added here.]

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1 The Transport Regulations [2] are supplemented by a number of safety guides on how to comply with them. The most relevant safety guide with respect to the technical basis is IAEA-TS-G-1.1, Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material [3].
1.4. Structure

[Text yet to be developed and added here.]

References for Chapter 1


2. GENERAL HISTORY

The Transport Regulations \[1\] are an IAEA safety standard, in the category specific safety requirements\[2\]. The IAEA Transport Regulations serve as the basis for regulations for consistent regulatory actions at both the international and domestic level through application by international modal organizations, and by individual member states As a result, they therefore have to be consistent with the Safety Fundamentals \[3\] and with the general safety requirements.

It must be noted that, according to the Transport Regulations, transport comprises all operations and conditions associated with, and involved in, the movement of radioactive material; these include the design, manufacture, maintenance and repair of packaging, and the preparation, consigning, loading, carriage including in-transit storage, unloading and receipt at the final destination of loads of radioactive material and packages.\[3\]

The development of the Transport Regulations began as early as the 1940s; while the international effort began in the 1950s. The following summary of the general historical background is based, in part, on documents that were produced by Pope in 2004 \[4,5\].

2.1. Early Efforts in the Development of International Regulations

In 1957, the Preparatory Commission of the IAEA \[6\] noted that the Agency might be able to obtain information on the work which has been done in, and consider the formulation of, the regulations governing the transport of radioactive materials. It further discussed how the Agency might staff and organize this effort. It also suggested that an advisory panel might be appointed, 'which might later be transformed into a standing committee'.

In response to this, the United Nations Economic and Social Council (ECOSOC) passed, in 1959, a resolution \[7\] that, among other things, requested the 'Secretary General, in light of the relevant recommendations contained in the report of the Committee of Experts', to continue the Committee of Experts, to explore 'the possibility of finding mutually acceptable performance tests for outer packages for certain classes or groups of dangerous substances', and—significantly—to "inform the International Atomic Energy Agency of the desire of the Council that the Agency be entrusted with the drafting of recommendations on the transport of radio-active substances".

These recommendations have guided the IAEA throughout the development and maintenance of the Transport Regulations. Specifically, the IAEA has:

- formulated and periodically updates regulations governing the transport of radioactive material;
- formed a transport safety committee, which it transformed in the late 1970s into a standing committee. It was initially identified as the Standing Advisory Group on the Safe Transport of Radioactive Material (SAGSTRAM), which later transitioned into the Transport Safety Standards Advisory Committee (TRANSSAC), and more recently renamed the Transport Safety Standards Committee (TRANSSC);
- developed performance tests for packages for radioactive materials; and
- The Transport Regulations are supplemented by a number of safety guides providing insight on how to comply with them. The most relevant safety guide with respect to the technical basis is IAEA-TS-G-1.1, Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material \[2\]. It is noted that, as of early 2012 when this chapter was developed, TS-G-1.1 was under review, and an updated revision is anticipated to be issued by the IAEA later in 2014.

3 The definition used in the BSS is slightly different: the deliberate physical movement of radioactive material (other than that forming part of the means of propulsion) from one place to another.

4 Radioactive materials constitute one of nine classes of dangerous goods as defined by the United Nations Committee of Experts in the UN Model Regulations \[8\]. Radioactive materials are denoted as “Class 7”.

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• worked, and continues to work, in undertaking these tasks in consultation with other concerned UN bodies, other international bodies and its own Member States.

Gibson and Messenger [9] provide an overview of what transpired at the IAEA in developing the first Regulations for the Safe Transport of Radioactive Material [10]. Two panels of experts were convened at the IAEA in 1959 for which the "recommendations made the two Panels were woven into a single set of regulations and these were approved by the Agency Board of Governors at their meeting on the 13th September 1960". The Board of Governors "recommended them to Member States and other international organisations as a basis for their own regulations"; and also propose to have the UN Committee of Experts include the recommendations in their future efforts on the transport of dangerous goods.

The previous efforts by a number of Member States in understanding the risks posed by radioactive materials in transport and in developing guidelines and requirements at the domestic level were instrumental in guiding the development of the 1961 edition of the Transport Regulations. Three examples of these efforts follow.

2.1.1. Early Efforts in the United States (US)

George [11] provides insight into the formation of the United States (US) Bureau of Explosive (BofE) in the early 1900’s, the initial efforts to control the transport of radioactive materials in the US, and the early efforts to develop regulations for packaging of radioactive materials by the US Interstate Commerce Commission (ICC). All of this served as one of the precursors to the international regulations.

As early as 1944, the ICC was undertaking the establishment of regulations for the transport of radioactive materials, and issued regulations as early as about 1946. In part, these regulations initially required that shipments consisting of large quantities of radioactive materials be shipped in containers approved by the BofE.

George included in his paper two examples of the early BofE permits. One of the permits [12], which was for spent (irradiated) fuel elements, was issued in 1959. The other permit [13], which was for other radioactive materials, was issued in 1962. Thus, these actions were occurring concurrently with the international deliberations that were underway, leading to the issuing of the IAEA Transport Regulations in 1961. Although some regulations had been established, the focus was in part on protecting property in general and film specifically. George noted that approvals of packagings would need to continue since "special types, sizes, and shapes of containers" will be required for "to provide for larger quantity shipments than are presently authorized". Thus, the permits that were used for providing safe transport of radioactive materials in the US in these early days were very prescriptive (specifying how to design a package rather than what must be accomplished in designing a package).

2.1.2. Early Efforts in the United Kingdom (UK)

Gibson and Messenger [9] reported that the carriage "of dangerous goods by sea in U.K. registered ships, or in other ships which load or discharge in U.K. waters" was governed by a general set of rules specified under the U.K Merchant Shipping Act of 1949; and guidance was issued in the application of these general rules, including the addition of radioactive substances, in 1957.

Experts from the U.K participated in the panels convened by the IAEA in 1959, and were supportive of the provisions included in the 1961 edition of the IAEA Transport Regulations. In June of 1961, the U.K. Radioactive Substances Advisory Committee recommended that the 1961 edition of the Transport Regulations [10] "be used as the basis for all U.K. regulations on this subject, and the appropriate government departments were asked to proceed accordingly".
Gibson and Messenger reported that by December 1962, practical "regulations on the I.A.E.A. model thus exist in the U.K. for the carriage of radioactive materials by sea and by rail"; that compatible regulations were being produced for road transport; and that the IATA regulations were being used.

2.1.3. Early Efforts in Canada

Martin [14] reported that the Canadian Board of Transport Commissioners requirements for radioactive material packaging for rail transport was at that time the most comprehensive in Canada. Requirements for road transport were then controlled at a provincial level, and none of the provinces prescribed any packaging requirements; whereas the requirements for air transport were the regulations prescribed by IATA; and requirements for transport by ship were established by the Department of Transport. In addition, he reported that the Canadian Atomic Energy Control Board (CAECB) had, at that time, certain powers pertaining to radioactive material packaging. He illustrated that they had a “multitude of regulatory bodies that are concerned with the transportation of radioactive materials”.

The CAECB had issued a circular in 1960 that established requirements similar to the initial, i.e. 1961, edition of the IAEA Transport Regulations [10]. For example, he noted that (a) the “maximum credible accident” concept was employed, but (b) this limited container design to no more than a 20 ft (6.1 m) drop “on a solid floor”. However, the circular did not provide any specific requirement for resisting a fire environment, it simply specified that the "shell shall be fabricated from steel or other material equivalent in strength and fire resistance".

Martin further reported that Canada had formed a committee to address a standardized set of package design requirements. This committee had concluded that "the general criteria of IAEA’s Pamphlet Nos. 6 and 7 are satisfactory bases for drafting of uniform regulations". Martin the elaborated on inadequacies the Committee has identified with the 1961 edition of the Transport Regulations, which included (a) the need to modify requirements for approvals of Type A and Type B package designs, and (b) the need to strengthen requirements for fissile materials. He concluded by stating that "...although our accident record is a favourable one, we cannot afford to relax our standards".

2.2. Specific early efforts to improve the Transport Regulations

Gibson [RBP035] indicates that, upon approving the initial edition of the regulations, the IAEA was directed to “arrange for the regulations to be reviewed from time to time in the light of experience”. The first meeting to initiate the review of the 1961 edition was undertaken by an IAEA Panel which met in Vienna, 11–22 March 1963. Gibson noted that the procedure that was agreed included two criteria, namely “(i) a practical need for an amendment had to be demonstrated, and (ii) any consequent amendment had to be technically justified. In other words, there was no intention to see change for its own sake”. This procedure has generally held true ever since as the regulations were reviewed for potential revision.

As work progressed in improving the Transport Regulations during those early years, many changes occurred. These changes included five key issues:

1. specifying more clearly package design requirements;
2. developing more comprehensively the concept of different types of packages (e.g. Type A, and Type B packages);
3. eliminating the early provisions for applying the concept of a "maximum credible accident" to the package design requirements;
4. emphasizing that the Regulations address “what” was required to satisfy the regulatory requirements, not “how” those requirements were to be satisfied; and

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5 The two documents referred to here, “pamphlets Nos. 6 and 7" are the first editions of IAEA Safety Series No. 6 [8] and Safety Series No. 7 [15], respectively.
5. establishing clarity in the Regulations and the interpretation thereof by specifying a set of definitions.

Each of these five changes, how they have developed over the years in a manner that has strengthened the understanding and application of the Transport Regulations, and the technical bases thereof, is summarized in the following.

### 2.2.1. Clearly specifying package design requirements

With regard to the package design requirements in these early permits, they typically described the packaging as a ‘container’, but didn’t explicitly establish quantified containment requirements nor did they establish specific design, testing and acceptance requirements. They typically established:

- a) established requirements for containing the radioactive materials being carried;
- b) defined constraints on how radiation shielding should behave under ordinary conditions and under severe fire and impact conditions;
- c) specified that it was to protect against criticality in the presence of other shipping containers of fissionable materials during transportation;
- d) simple requirements for managing generated heat, and
- e) specified how to design a package, not what requirements must be satisfied in the design.

Thus, in these early attempts at regulating the packaging of radioactive material, the four parameters currently deemed necessary to provide protection as specified in para. 104 of the 2012 edition of the Transport Regulations [1] (i.e. containment, radiation control, prevention of criticality and managing heat) were covered in these early domestic permits.

Rogers [16] elaborated on actions that were taken as a result of concerns that packaging controls needed to be codified. Rogers quoted Gibson of the UK as follows:

> "We can no longer expect competent authorities, port authorities and others to be content that a shipment is safe because we in the industry are satisfied that it is safe—safety not only has to be achieved but must be seen to be achieved”.

It was noted that, in 1957, a US domestic interagency committee was formed to coordinate the Federal Agency activities in this area and to support the ECOSOC guidance [7]. Rogers [16] further indicated that the IAEA convened two panels shortly thereafter (in 1959) to develop international transport regulations. The result of these initial efforts was the IAEA 1961 edition of the Transport Regulations [10].

### 2.2.2. Developing the concept of Type A and Type B packages

With respect to package test standards established in this first edition of the Transport Regulations, it identified (in Section 5 of the 1961 edition [10]) two types of packages – Type A and Type B packages – and specified packaging requirements as follows:

- The Type A package must be leakproof, securely closed by a positive fastening device, shielded adequately to prevent an external dose rate in excess of the values prescribed in the regulations and must prevent loss or dispersal of radioactive contents and retain shielding efficiency under conditions normally incident to transport (such as minor drops and spills) and under minor accident conditions.
- The Type B package must be designed so as to maintain its integrity under conditions normally incident to transport without loss or dispersal of radioactive contents and the package must retain shielding efficiency under conditions normally incident to transport and in the most severe accident which is considered credible for the mode of transport involved.

Thus the general concepts that exist today in the Regulations for Type A and Type B packages and the other package types that now exist – following a graded approach – were established through these early efforts; namely that:
• Type A packages must be designed to withstand normal conditions of transport (NCT), whereas
• Type B packages must be designed to withstand both NCT and also severe accident conditions of transport (ACT).

However, this first edition of the Transport Regulations specified that Type B packages must “be adequate to prevent any loss of dispersal of radioactive contents and to retain the shielding efficiency” under conditions “normally incident to transport and for the maximum credible accident relevant to the mode of transport”. This requirement that the Type B packages must adequately survive “the most credible accident” became a significant driver for modifying package design requirements in the second edition (i.e. the 1964 edition) of the Transport Regulations [17].

2.2.3. Eliminating the provision for applying the concept of a ‘maximum credible accident’
The first edition of the Transport Regulations established:
• test and acceptance criteria that were described generically. These criteria for Type A packages have not changed significantly since first conceived although the specification of ‘conditions normally incident to transport’ has since been quantified;
• accident test criteria for Type B packages were initially specified qualitatively as essentially the ‘maximum credible conditions’. These were very quickly modified to ‘accident conditions of transport’ and were quantified in later editions of the Transport Regulations; and
• acceptance requirements for Type B packages following exposure to both the test conditions ‘normally incident to transport and in the most severe accident’ were initially only specified qualitatively, but were also very quickly quantified in later editions of the Transport Regulations.

Steps were taken immediately following the publication of the first international Transport Regulations [10] to evaluate their adequacy and to elaborate on the international test and acceptance requirements for packages containing radioactive material. Much of this effort was documented in 1962 by Gibson and Messenger [9] and in 1963 by Messenger and Fairbairn [18].

One of the issues that was very quickly addressed was the requirement that Type B packages be able to withstand “the most severe accident which is considered credible for the mode of transport involved”, more frequently identified as the ‘maximum credible accident’ concept. Messenger and Fairbairn noted that:

“The requirement for the packaging to be able to withstand the ‘maximum credible accident’ is novel in the transport field; it has not been applied as a mandatory requirement to the carriage of non-radioactive dangerous goods, some of which, for example, cyanides, may be far more hazardous than many radioactive materials”.

This statement still holds true today; the package test requirements for Type B radioactive materials packages are more demanding than any of the packing group tests currently specified for Classes 1–6, 8, and 9 in the UN Model Regulations [8].

They further elaborated on the situation as perceived in 1963 as follows:

“In the absence of a reasonable borderline between ‘credible’ and ‘incredible’ accidents, some accident can always be postulated sufficiently severe or elaborate to defeat any packaging design. No accident, however extraordinary, can be ruled out as completely impossible. In fact many major accidents that have occurred had been thought incredible. On the other hand, no transport package can be designed to withstand every conceivable accident including combinations of both natural and man-made forces. Indeed, if such a package could be constructed, it would not be transportable.”
Appleton and Servant [19] addressed in 1965 the early requirement for applying the ‘maximum credible accident’ criteria to Type B packages, stating that:

“The conditions defined are somewhat vague and the concept of the maximum credible accident appeared particularly objectionable and so unpracticable that it was discarded. On the basis of experience and work done in the interim period in Member States a major effort was made during the recent revision of the Agency transport regulations to make such conditions more objective in terms of testing procedures.”

This thinking by those imminently involved in the early development and enhancement of the Transport Regulations led to an effort that would establish reasonable, yet meaningful, test requirements initially for both the Type A and Type B radioactive material packages and later for other package types as they were defined in the Regulations.

To accomplish this, specifically for Type B packages, Messenger and Fairbairn [18] discussed considerations that underlay their proposed quantitative tests. These considerations included:

- mishandling and tampering;
- impacts due to large drops when loading or to collision during transport;
- fire and damage by fire-fighting materials,
- immersion in water, and
- ‘smothering’ by debris or by other goods as a result of one of the above –

where it was judged that “impact and fire are the most likely to cause serious immediate damage”.

From these philosophical considerations arose the extensive efforts on the part of many Member States to define quantitatively various tests that represent a large portion of the severe accident environments that a Type B package might be exposed to during transport. As a result, the Transport Regulations follow a graded approach to the performance standards, characterized by (see para 106 of SSR-6 [1]):

- routine conditions of transport (incident free),
- normal conditions of transport (minor mishaps), and
- accident conditions of transport.

The full suite of these three conditions are applied in the Regulations to Type B packages as elaborated in Chapter 10 of this document, and to other packages such as fissile material packages as elaborated in Chapter 11 of this document. The “Tests for demonstrating ability to withstand accident conditions of transport” which are specified in paras 726 through 729 of SSR-6.

In preparation for issuing the 1985 edition of the Regulations [23], an advisory group (AG-225) was convened in 1979 [RBP040] to address specific basic safety principles associated with the transport regulations. One topic that was addressed was an attempt at clarifying what constitutes “extreme accident conditions”, and how the regulations have been structured to address such conditions. AG-225 concluded that an extreme accident is "one in which a package experiences an environment more severe than that represented by the Type B tests". AG-225 then established the following basic principles:

“(a) The probability of a package experiencing an accident condition more severe than that represented by the Type B tests should be low and such that the resulting risk is acceptable when compared with other involuntary risks accepted by society.

“(b) The total radiological risk from low probability, high consequence extreme accidents should be less than the total radiological risk from high probability, lower consequence events.”

Much of this Technical Basis Document addresses these risks and probabilities, and illustrates that the Type B tests satisfy these two basic safety principles.
2.2.4. Emphasizing ‘what’ is required not ‘how’ a requirement is to be satisfied

Gibson and Messenger [9] further elaborated on another issue relative to the development of the package design standards, noting the following:

“I.A.E.A. regulations confine themselves to specifying the objects to be attained by the packaging standards. They state what is to be achieved, but only suggest how it is to be achieved. Thus the designer is allowed the greatest possible freedom to develop new techniques for improving both safety and economy. The packaging standards are in fact defined in terms of transport conditions of differing severity under which the four radioactivity hazards must be so controlled as to afford the same high degree of safety.”

Fairbairn [20], in addressing that the Regulations prescribe “what” not “how”, commented that:

“In prescribing ‘what’ has to be achieved rather than ‘how’, the regulations encourage packaging/package design effort, especially regarding the use of new materials and improved constructional techniques. Also the fact that the transport of any specific radioactive material is not prohibited results largely from this basic principle.”

In this regard, Fairbairn noted that:

“...it is important that guidance be available as to 'how' certain regulatory requirements may be met, such guidance being given as 'a way' not 'the way'. Also, in order to promote understanding of the technical basis of any regulatory prescription and to help those concerned with further reviews, comprehensive information of a 'why' nature is required.”

The philosophy of emphasizing “what” has to be achieved has continued from 1961 to the present. The 2012 edition of the Transport Regulations [1] specifies “what” must be accomplished. In contrast, TS-G-1.1 [2] couples specific guidance to the relevant paragraphs of SSR-6, specifying “how” the specific provision in the Regulations may be satisfied (i.e. providing “a way” or “ways” that the provision can be accomplished), and/or “why” the provision was introduced.

2.2.5. Establishing Clarity through Definitions

Gibson and Messenger [9] introduced the need to be specific about the use of terminology in the Transport Regulations, including ‘receptacle’, ‘container’, ‘packaging’ and ‘package’; and they then proposed specific definitions for ‘packaging’ and ‘package’ following which they noted that:

“The indiscriminate use of these terms in the past, both in regulations and in U.K.A.E.A. domestic documents, has led to difficulties of interpretation and to some confusion. Transport regulations should be primarily concerned with complete packages, loads and consignments, and only to a minor extent with the constituent containers and packaging details. There is a need, we believe, for definitions of these terms to be embodied in regulations.”

Fairbairn [21] commented after the 1973 edition of the Transport Regulations was published, as follows:

“Many regulatory prescriptions relate to the packages in which radioactive materials are carried. During the formulation of such prescriptions it is necessary to be quite clear as to whether it applies to the package itself or to the packaging in which the radioactive material is carried.”

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6 At the time this discussion was prepared, the latest version of TS-G-1.1 [2] reflected the 2005 edition of TS-R-1[22]. Because of changes that have occurred in the Transport Regulations since that time, the paragraph cross-referencing used between the Regulations and the Guidance does not always match. Thus a user of the 2012 edition of SSR-6 and the 2008 edition of TS-G-1.1 will need to check paragraph numberings between the 2005 and 2012 editions of the Transport Regulations when striving to apply the guidance in TS-G-1.1. A revision of TS-G-1.1 is underway which will properly reflect the paragraph structure of the 2012 edition of the Transport Regulations.
He further elaborated, noting that:

“To help a user comprehend “what” is required, any regulation must be clear and concise. This is especially important in the case of the Agency’s transport regulations as besides being translated into various languages, they need to be converted by transport organizations into the form of the regulations used for dangerous goods as a whole.”

As a result of these discussions, the concepts for ‘packaging’ and ‘package’ were incorporated into the Regulations in the 1964 edition of the Transport Regulations [17], with specific definitions as currently in paras. 231 and 232 of the 2012 edition of SSR-6 [1].

The term ‘container’ is not used as a stand-alone term in the Regulations, but is always used as either ‘freight container’ or ‘intermediate bulk container’, which are also now defined in SSR-6 in paras 223 and 224. Also, the term receptacle is used in the Regulations to describe one or more components of a packaging (see para. 232 of SSR-6).

2.3. Growth of Importance of the Transport Regulations

With time, the importance of the Transport Regulations for governing the safe transport of radioactive material throughout the world has grown. The significance of this has often been recognized within the IAEA and by Member States and other involved international organizations. For example, an IAEA Advisory Group meeting convened in March/April 1977 [RBP033], the following was noted:

“...what we are doing is attesting to the value of these Regulations and saying to the Agency that although when they were first developed they were made mandatory only for the Agency’s own work, their value has far exceeded perhaps even the wildest hopes of the originators of the documents. The conclusion we have reached is that these Regulations have come of age and have outstripped this original constraint and the initial role of the Agency thought they might play. If the Chairman wants to send one overall recommendation I think it must be that the Agency recognize its role – if it has not already done so –recognize and accept the major role that these Regulations now play. They form the basis of a great deal of international commerce.”

It is fair to say that in the late 1970s and henceforth thereafter, such recognition of the key role the Regulations play, worldwide, has been accomplished.

References for Chapter 2


3. THE FUNDAMENTAL SAFETY PRINCIPLES

As the IAEA’s involvement matured in striving to provide world-wide safety in the use of radioactive materials, the preface by the nine sponsoring organizations of the current Fundamental Safety Principles [1] notes that it three separate documents were issued in the mid-1990s addressing fundamental principles and objectives of safety. These three documents dealt with (a) the safety of nuclear installations, (b) the principles of radioactive waste management, and (c) the radiation protection and safety of radiation sources.

In 1995, the IAEA Board of Governors "requested the IAEA Secretariat to consider, at an appropriate time, the revision of the three Safety Fundamentals texts with the aim of combining them in a unified set of principles representing a common safety philosophy across all areas of application of the IAEA safety standards". This request was satisfied by the issuance of the Fundamental Safety Principles [1] in 2006. The following addresses each of ten safety principles set forth in the latest Safety Principles document, and discusses how each relates to and is satisfied by the 2012 edition of the IAEA Transport Regulations SSR-6 [2].

As stated in Ref. [1], the fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation, without unduly limiting the operation of facilities or the conduct of activities (such as the transport of radioactive material) that give rise to radiation risks. Measures have to be taken:

- to control the radiation exposure of people and the release of radioactive material to the environment;
- to restrict the likelihood of events that might lead to a loss of control over a radioactive source;
- to mitigate the consequences of such events if they were to occur.

Ten safety principles have been formulated, on the basis of which safety requirements are developed. In this context, the term ‘safety’ means the protection of people and the environment against radiation risks, and the safety of facilities and activities that give rise to radiation risks. Safety is concerned with both radiation risks under normal circumstances and radiation risks as a consequence of incidents; safety measures include actions to prevent incidents and arrangements put in place to mitigate their consequences if they were to occur.

The fundamental safety principles are listed below, together with some comments related to their applicability to the transport of radioactive material.

1. The prime responsibility for safety must rest with the person or organization responsible for facilities and activities that give rise to radiation risks.

   Since transport comprises the design, manufacture, maintenance and repair of packaging, and the preparation, consigning, loading, carriage including in-transit storage, unloading and receipt at the final destination of loads of radioactive material and packages, there are a number of responsible parties. It is not evident to designate the person who has the prime responsibility for safety.

2. An effective legal and governmental framework for safety, including an independent regulatory body, must be established and sustained.7

   The transport regulations are drafted on the presumption that a national framework is in place that enables the government to discharge its responsibilities for transport safety.

   It must be noted that the transport of radioactive material is also regulated through international agreements for the different modes of transport (road, rail, air, sea, inland waterways). All these agreements are based on SSR-6.

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7 Note: In the IAEA transport regulations the term ‘competent authority’ is used instead of the term ‘regulatory body’.
3. Effective leadership and management for safety must be established and sustained in organizations concerned with, and facilities and activities that give rise to, radiation risks.

Section III of the Transport Regulations comprises provisions regarding management systems and compliance assurance. Additionally guidance on these subjects is given in TS-G-1.4 [3] and TS-G-1.5 [4].

4. Facilities and activities that give rise to radiation risks must yield an overall benefit (justification principle).

The principle of justification has to be applied at different levels. At the generic level, transport of radioactive material does not need to be justified as a practice in its own. It is the practice associated with the radioactive material that gives rise to transport activities that has to be justified, taking into account that transport activities may be needed.

This principle has also to be applied to the each individual shipment, mainly by the consignor.

5. Protection must be optimized to provide the highest level of safety that can reasonably be achieved (optimization principle).

Both normal exposure and potential exposure are considered.

The principle of optimization of protection also has to be applied at different levels. It is applied in a generic way in many provisions of the transport regulations, although not in a mathematically rigorous way.

The principle has to be implemented by the different actors in the transport of radioactive material: consignor, carrier, and receiver.

6. Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm.

The BSS [5] sets individual dose limits for workers and for members of the public. They are applicable to the transport of radioactive material.

Limits for radiation levels at the surface of and at certain distances from packages and conveyances have been set, although not in a mathematically rigorous way (using models and scenarios). Several surveys have confirmed that the dose limits are complied with [6] and [7].

7. People and the environment, present and future, must be protected against radiation risks.

The objective of the Transport Regulations is to establish requirements that must be satisfied to ensure safety and to protect persons, property and the environment from the effects of radiation in the transport of radioactive material.

8. All practical efforts must be made to prevent and mitigate nuclear or radiation accidents.

The Transport Regulations comprise requirements dealing with the design of packages and on the accumulation of packages. There are also provisions dealing with operational controls (administrative, radiation protection, criticality).

9. Arrangements must be made for emergency preparedness and response for nuclear or radiation incidents.

Emergency preparedness and response is dealt with in Section III of the Transport Regulations. Additional guidance is given in TS-G-1.2 [8].

10. Protective actions to reduce existing or unregulated radiation risks must be justified and optimized.

This principle is not relevant for transport of radioactive material.
Most of the safety principles are reflected in the provisions of section III of the transport regulations (general provisions).

References for Chapter 3


4. SAFETY OBJECTIVES AND PRINCIPLES FOR TRANSPORT

[See Endnote i (i)]

4.1. Safety Objectives for Transport

The objective of the Transport Regulations [1] is to establish requirements that must be satisfied to ensure safety and to protect persons, property and the environment from the effects of radiation in the transport of radioactive material, without undue restriction of movement of radioactive material. This objective is consistent with the fundamental safety objective specified in para. 1.6 of the IAEA's Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards document [2].

This protection is achieved by requiring:

- containment of the radioactive contents;
- control of external radiation levels;
- prevention of criticality;
- prevention of damage caused by heat.

These requirements are satisfied by:

- applying a graded approach to contents limits for packages and conveyances and to performance standards applied to package designs, depending upon the hazard of the radioactive contents; the graded approach is also reflected in the application of the exemption concept and in the administrative arrangements (approvals);
- imposing requirements on the design and operation of packages and on the maintenance of packagings, including consideration of the nature of the radioactive contents;
- requiring administrative controls, including, where appropriate, approval by competent authorities.

The Regulations note that “the protection of property and the environment are assured when these Regulations are complied with.” [1]

4.2. Safety Principles for Transport

The application of key safety principles in order to satisfy the safety objectives is fundamental to the success of the Transport Regulations. One such principle is the application of a graded approach, where the greater the hazard posed by the contents of a package should shielding or containment be lost or criticality control be degraded, then a greater number of design, operational and administrative controls are imposed by the Regulations.

One example of the application of the graded approach is, in particular, applied in specifying performance standards for packages (i.e. the packaging plus its radioactive content) in different conditions of transport:

- excepted packages are to be designed to withstand routine conditions of transport (incident free);
- Type A packages are to be designed to withstand both routine and normal conditions of transport (including minor mishaps);
- Type B packages are to be designed to withstand routine, normal and accident conditions of transport; whereas
- Type C packages are to be designed to withstand severe accident conditions for air transport, beyond the accident conditions of transport which Type B packages are designed to withstand. [1]

This categorization of packages is elaborated in Chapters 9, with respect to the design and testing requirements, and also in Chapter 11. With regards to the categorisation of packages, it is noted that the transport regulations have also introduced three categories of industrial

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8 It must be noted that protection of property is not included in the fundamental safety objective.
packages (IPs) to accommodate the transport of low specific activity (LSA) material and surface contaminated objects (SCO). A graded approach has been applied to the ability of IPs to withstand a given category of transport conditions. IP-1 packages are identical to excepted packages except for the minimum dimension and marking and labelling requirements, and are therefore designed to withstand routine conditions of transport; whereas IP-3 packages are like type A packages for solid material, and therefore are designed to withstand both routine and accident conditions of transport.

A basic principle is that the content of packages is constrained such that the radiological consequences of an accident are limited, in particular the exposure of persons in the vicinity of an accident (external exposure and internal exposure due to the intake of radioactive material). This is explained in detail in the section on the derivation of $A_1$ and $A_2$ values (i.e. Section 6.2) and in Chapter 8 (on classification of materials).

The transport regulations define different categories of material (special form, low specific activity material, low dispersible material), the characteristics of which are taken into account to derive contents limits for packages containing that kind of material. In normal operation, the control of radiation is ensured by setting limits to radiation levels for packages and conveyances, the application of segregation distances, the use of warning labels and other operational provisions (e.g. exclusive use, application of radiation protection programmes). These are intended to ensure that the dose to members of the public resulting from transport of radioactive material does not exceed a fraction of the applicable dose limits.

When defining the scope of the transport regulations the graded approach is applied. Transport of radioactive material (a) that is not controllable (e.g. radioactive material in a person or an animal for the purpose of diagnosis or treatment, radioactive material that forms an integral part of a means of transport such as depleted uranium counterweights); or (b) that gives rise to negligible radiological risks are excluded from the field of application (e.g. objects with surface contamination below a certain level, consumer goods that are exempted from regulatory control, material with an activity concentration below the corresponding exemption level, consignments with a total activity lower than the corresponding exemption level). The initial establishment of these exemption levels was established during deliberations by AG-406 preceding the issuing of the 1985 edition of the Regulations.

The scope of the Transport Regulations includes consideration of those natural materials or ores which form part of the nuclear fuel cycle or which will be processed in order to use their radioactive properties. The Transport Regulations do not apply to other ores which may contain naturally occurring radionuclides, but whose usefulness does not lie in the fissile, fertile or radioactive properties of those nuclides, provided that the activity concentration does not exceed ten times the exempt activity concentration values.

In addition, the Transport Regulations do not apply to natural materials and ores containing naturally occurring radionuclides which have been processed (up to ten times the exemption levels in terms of activity concentration) where the physical and/or chemical processing was not undertaken for the purpose of extracting radionuclides (e.g. washed sands and tailings from alumina refining). However, such processed materials should not be intended for further processing for the removal of their radionuclides. Were this not the case, the Transport Regulations would have to be applied to enormous quantities of material that present a very low hazard. However, there are ores in nature where the activity concentration is much higher than the exemption values. The regular transport of these ores may require consideration of radiation protection measures. Hence, a factor of ten times the exemption values for activity

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9 These levels are set in the definition of contamination: the presence of a radioactive substance on a surface in quantities in excess of $0.4 \text{ Bq/cm}^2$ for beta and gamma emitters and low toxicity alpha emitters, or $0.04 \text{ Bq/cm}^2$ for all other alpha emitters.

10 Exemption levels as in Table I-1 of the BSS (moderate quantities) [2].
concentration was chosen as providing an appropriate balance between the radiological protection concerns and the practical inconvenience of regulating large quantities of material with low activity concentrations of naturally occurring radionuclides.

The application of the basic graded approach principle is found in many aspects of the Transport Regulations where, for example, the content of packages is limited such that the radiological consequences of an accident are limited, in particular the exposure of persons in the vicinity of an accident (external exposure and internal exposure due to the intake of radioactive material). In addition, the transport regulations define different categories of material (special form, low specific activity material, low dispersible material), the characteristics of which are taken into account to derive contents limits for packages containing that kind of material. This classification of materials is elaborated in Chapter 8.

Specific applications of the graded approach used in establishing the regulatory requirements are elaborated in detail in Appendix 3. Five specific areas are addressed therein, illustrating the application of applying the graded approach to:

1. package contents activity limits,
2. types of package designs,
3. package design performance standards,
4. package performance standards, and
5. package design approval requirements.

References for Chapter 4


5. GENERAL SAFETY REQUIREMENTS

Section III of the Transport Regulations comprises general provisions that find their justification in the General Safety Requirements that are applicable to all facilities and activities and that are specifically relevant for transport (e.g. see Chapter 3 of this Technical Basis Document). These provisions are structured to follow the principles set forth in the following series of the General Safety Requirements series on Safety Standards:

- GSR Part 1, Governmental, legal and regulatory framework for safety (2010) [1];
- GSR-R-2, Preparedness and Response for a Nuclear or Radiological Emergency (2002) [2];
- GSR Part 3 (Interim), Radiation protection and safety of radiation sources: international basic safety standards, interim edition 2011 [4]; which replaces SS 115 (1996) [5]; and
- GSR Part 4, Safety assessment for facilities and activities (2009) [6];

In addition, the following two documents in this series are under development:
- GSR Part 2, Leadership and management for safety, under development; to replace GS-R-3 (2006); and

5.1. General Requirement for Compliance Assurance

The Transport Regulations have been drafted on the presumption that a national framework is in place which enables the government to discharge its responsibilities for transport safety; in particular that a regulatory body (called competent authority in the Transport Regulations) has been set up to deal with transport safety (requirement 2 of GSR Part 3 [4]), and to ensure compliance with the established regulatory requirements.

The 2012 edition of the Transport Regulations specify, e.g. in para. 208 of SSR-6 [7] that “Compliance assurance shall mean a systematic programme of measures applied by a competent authority that is aimed at ensuring that the provisions of these Regulations are met in practice”. Thus, it is the responsibility of the competent authority to ensure compliance. The establishment of this responsibility was initially specified in the 1985 Edition of the Transport Regulations [8].

Discussion on the need for compliance assurance occurred at the March 1965 Panel Meeting [RBP037], where the need for inspection of the construction of packaging for large radioactive sources by the competent authority was raised. The Panel agreed that “the competent authority need only be satisfied that the specification of the design has been fully met – and an independent inspectorate could be used”. In the discussion “It was pointed out that though approved designs were bound to be accepted, there was no similar requirement for the acceptance of packaging manufactured to those designs without the competent authority being satisfied that they were properly constructed”.

With respect to a competent authority being satisfied that construction was adequate, the March 1965 Panel Meeting “pointed out that domestic movements raised no problem, it was in the international movement that difficulties might arise”; where this issue “was considered more relevant to the preliminaries to the authorisation of a shipment of irradiated fuel by the competent authority of the country of origin of the shipment”.

Deliberations such as documented in the March 1965 Panel Meeting [RBP037] appears to have led to the incorporation of compliance assurance text in Section C-6.3.1 of the 1967 Edition of the regulations [19] that there be:

“Either complete certification from the manufacturer, consignor or user that the constructional methods and materials used for the construction of the packaging are in accordance with the approved design requirements, or a document issued by the competent
authority of the country in which the packaging was constructed, stating that it has been provided with such complete certification from the manufacturer, consignor or user; ....”.

Guidance on compliance assurance was provided as early as 1973, where para. 1.09 of the first edition of Safety Series No. 37 [9] read “The IAEA Regulations define ‘competent authority’ and assign to him the responsibility for receiving applications for approval, evaluating compliance,.....”. This same comment was carried forward in 1982 into the second edition of Safety Series No. 37 [10].

In the 1973 Edition of the regulations [20], an inference to compliance assurance was made in the Subsection titled “QUALITY CONTROL IN FABRICATION AND MAINTENANCE OF PACKAGING” where para. 839 reads in a manner similar to Section C-6.3.1 of the 1967 Edition of the regulations. In turn, para. 150 of the 1973 Revised Edition (As Amended) in 1979 [21] provided the first specification for compliance assurance, preceded by a subheading of “Compliance Assurance”.


Further detailed guidance on this was provided in Safety Series No. 112 [12], published in 1994. This document dealt entirely with compliance assurance. It stated that:

“105. According to para. 117 of the Regulations, “Compliance assurance shall mean a systematic programme of measures applied by a competent authority which is aimed at ensuring that the provisions of these Regulations are met in practice”. Paragraph 210 states:

“The competent authority is responsible for assuring compliance with these Regulations. Means to discharge this responsibility include the establishment and execution of a programme for monitoring the design, manufacture, testing, inspection and maintenance of packaging, and the preparation, documentation, handling and stowage of packages by consignors and carriers, to provide evidence that the provisions of these Regulations are being met in practice.”

“106. While competent authorities are responsible for assuring compliance with the Regulations (which must include oversight and enforcement of all regulations), the prime responsibility for ensuring safety in transport rests with consignors and carriers, who must take account of all relevant safety regulations. Thus, consignors, carriers and any other users of the Regulations must comply with the actual regulations, and the competent authority should assure compliance with these regulations. The competent authority itself should comply with the IAEA Regulations, for example in such matters as issuance of approvals and the allocation of design identification marks for packagings.”

In retrospect, attributing the responsibility for compliance assurance to the competent authority as part of the regulatory functions is consistent with GSR Part 1 [3], and requirement 2 of GSR Part 3 [4]. It is also to be considered as an element of the management system of the competent authority (requirement 5 of GSR Part 3).

The handling of non-compliance with provisions of the Transport Regulations is also part of the regulatory functions.

5.2. General Requirement for Radiation Protection

The general provisions on radiation protection recall two of the basic principles of radiation protection, namely optimisation of protection and individual dose limits. The provisions for monitoring of compliance with the dose limits for workers are similar to those that are normally applied for workers in supervised and controlled areas in facilities.
A radiation protection programme is an element of the management system, the basis of which can be found in requirement 5 of GSR Part 3 [8] and in GS-R-3 [3]. Guidance on how to set up a radiation protection programme is contained in TS-G-1.3 [13].

As a precursor to the development of the 1985 edition of the regulations [8] and the inclusion of requirements for a radiation protection programme for transport, AG-225 [RBP040] was convened in 1979 with a view to reviewing basic principles in transport. AG-255 agreed that "The purpose of the IAEA Transport Regulations is to establish standards of safety which provide an acceptable level of control of the radiation hazards to persons, property and the environment that are associated with the transport of radioactive materials".

AG-225 deliberated on the following three issues of (a) justification, (b) optimization, and (c) recommended dose limitation. The Advisory Group noted that the "first two principles are related to protective measures and requirements of a practice, activity or source of exposure. The third principle is related to individual risk limitation as a boundary condition to the justification and optimization procedures".

**Justification**

With respect to justification of the practice, AG-225 [RBP040] documented that the transport of radioactive materials "includes some risk and the uses of those materials result in some benefits. Transport is essential in the use of radioactive materials and the benefits from transport more than offset the risks; therefore, transport of radioactive materials is justified". Elaboration on the basis for this statement of justification can be found elsewhere in the AG-225 report; e.g. see pages 8 and 9 of the AG report, and page 5 of AG-225 Paper 1 (prepared by Lester Rogers, included in AG-225 report as Attachment III).

**Optimization**

For optimization, Attachment III of AG-225 [RBP040] documented the following:

"Optimization of protection for a given practice is a general requirement of all concerned following the ICRP recommendation that all doses should be as low as reasonably achievable (ALARA principle). This recommendation means that for a particular practice, such as transport of radioactive materials, it is necessary to derive authorized dose limits or limitations not directly related to the ICRP dose equivalent limits, but on cost-effectiveness assessment of what appears to be reasonably achievable. The derived dose limitations for a particular practice would necessarily be well within ICRP overall dose-equivalent limits and would be expected, generally, to be only a small fraction of ICRP limits. This requirement is source-related and consists of increasing the level of protection to a point such that further improvements achieve exposure reductions which are less significant than the additional effort needed. ..."

Further Elaboration on the basis for the preceding optimization statement can be found elsewhere in the AG-225 report; e.g. see pages 13 – 15, and 20 of Annex III of AG-225 (a paper prepared by Lester Rogers).

**Dose Limitation**

Attachment III of AG-225 [RBP040] specified the following that is related to dose limitation:

"The technique employed to provide protection from hazards in the transportation of radioactive material are as follows:

"(a) Protection against radiation that penetrates through the package is provided through limitations on the radiation levels on the outside of packages, and on stowage and segregation provisions that require operational control by transport workers. ...

"(b) Protection from the release of radioactive materials is provided by design requirements for packaging and limitations on the types and quantities of radioactive material that may be shipped in specified types of packages."
"(d) “Protection from internal and external exposure due to contamination on the external surfaces of packages is provided by requirements for cleaning and limits on contamination levels.”

Attachment III of AG-225 further noted that “If all risks and all benefits of a practice are limited to one and the same group of individuals, the principles of justification of practice and optimization of protection would suffice to assure enough protection. In practice, however, risks and benefits are not equally distributed. Those who benefit from a practice may not be those who are exposed to the risks. In order to prevent that practices are found justified in spite of high risks to individuals, an individual risk limitation is needed as a boundary condition to the justification and optimization procedures. This risk limitation is achieved by the application of the primary dose equivalent limits recommended by ICRP.”

Attachment III of AG-225 also noted that “The principles set forth in the ICRP system of dose limitation have in fact been implemented intuitively in the regulations to a large degree”. A good portion of the remaining AG-225 document was devoted to re-examining the major radiation protection principles “in light of actual experience”. It continued by stating that “As far as is practicable, the provisions of the regulations for restricting exposure shall emphasize the intrinsic protection provided by design of the transport packaging, package and system and only secondarily on protection that depends on the actions of transport and storage workers”. To this end, it stated that “All relevant transport and storage personnel shall receive such instruction and training as are necessary concerning the hazards involved, the precautions to be observed and the procedures to be followed to assure compliance with the regulations”.

Further Elaboration on the basis for the preceding dose limitation statements can be found elsewhere in the AG-225 report; e.g. see pages 9 – 13, and 15 – 22 of Annex III of AG-225 (a paper prepared by Lester Rogers), as well as Annex V of AG-225.

A key principle to establishing a requirement of a radiation protection programme for transport, which was first set forth in the Foreword of the 1985 edition of the Transport Regulations [8], was specified by AG-225 [RBP040]. AG-225 agreed that “dose limits must be seen truly as limits which must not be exceeded by the combined contributions of exposure to which an individual may be exposed from all sources and practices now or in the foreseeable future. This ICRP principle also has important implications for defining limitations of radiation dose for transport and storage workers and members of the public due to transport of radioactive material”.

Radiation Protection Programme

Utilizing much of the work documented in the AG-225 report [RBP040], the Foreword of the 1985 edition of the Transport Regulations [8] was developed through a consultants services meeting convened in Vienna in 1984 [14]. The Foreword specified that “the radiation exposure to transport workers and to the general public is subject to the limitations stated in the ‘Basic Safety Standards for Radiation Protection, 1982 Edition’”. It further stated that “This revision of the Transport Regulations implements the 1982 Edition of the Basic Safety Standards for Radiation Protection which sets forth a new system of dose limitation, the components of which are: (1) justification of the practice, (2) optimization of protection for sources of exposure, and (3) individual dose limitation.” Each of these components was then elaborated upon in the Foreword.

In the next revision of the Transport Regulations, the requirement for a radiation protection programme was embedded into the document itself (e.g. see paras 233 and 301 through 307 of ST-1 (1996)) [15]; i.e. it truly became a requirement.

Para. 302 of the 2012 edition of the Transport Regulations [7] comprises the provisions for the establishment of a radiation protection programme. It is not stipulated which actors have to comply with it. The radiation protection programme shall incorporate the provisions of para. 562, which text needs to be revised in order to correctly apply the concept of ‘dose constraint’ in
subpara. (b) in order to satisfy the requirement that a dose constraint shall be a fraction of the dose limit (i.e. a fraction of 1 mSv for members of the public).

5.3. General Requirement for Emergency Response

The provisions on emergency response have been drafted on the presumption that the government has set up an integrated and coordinated emergency management system (requirement 43 in GSR Part 3 [4]). TS-G-1.2 [16], developed during the 2000-2002 time period, provides guidance on emergency planning and preparedness, and contains guidelines on planning and preparing for emergency response to transport accidents involving radioactive material.

5.4. The General Requirements from the Basic Safety Standards

The Basic Safety Standards (BSS), which were revised in 2011 [4] and issued as an interim edition, establishes 52 general requirements to be fulfilled in all facilities and activities giving rise to radiation risks. Activities include transport, which is defined in the BSS as "the deliberate physical movement of radioactive material (other than that forming part of the means of propulsion) from one place to another".

For certain facilities and activities, such as nuclear installations, radioactive waste management facilities and the transport of radioactive material, other safety requirements, complementary to these Standards, also apply.

Of the 52 general requirements in the interim 2011 edition of the BSS [4], all but one of the first 33 apply to the transport of radioactive material, as summarized in Table 5-1 below.

**Table 5-1. General Basic Safety Standards Requirements applicable to Transport.**

<table>
<thead>
<tr>
<th>Requirement Number</th>
<th>Requirement</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Application of the principles of radiation protection. Parties with responsibilities for protection and safety shall ensure that the principles of radiation protection are applied for all exposure situations.</td>
</tr>
<tr>
<td>2</td>
<td>Establishment of a legal and regulatory framework. The government shall establish and maintain a legal and regulatory framework for protection and safety and shall establish an effectively independent regulatory body with specified responsibilities and functions.</td>
</tr>
<tr>
<td>3</td>
<td>Responsibilities of the regulatory body. The regulatory body shall establish or adopt regulations and guides for protection and safety and shall establish a system to ensure their implementation.</td>
</tr>
<tr>
<td>4</td>
<td>Responsibilities for protection and safety. The person or organization responsible for facilities and activities that give rise to radiation risks shall have the prime responsibility for protection and safety. Other parties shall have specified responsibilities for protection and safety.</td>
</tr>
<tr>
<td>5</td>
<td>Management for protection and safety. The principal parties shall ensure that protection and safety is effectively integrated into the overall management system of the organizations for which they are responsible.</td>
</tr>
<tr>
<td>6</td>
<td>Graded approach. The application of the requirements of these Standards in planned exposure situations shall be commensurate with the characteristics of the practice or the source within a practice, and with the magnitude and likelihood of the exposures.</td>
</tr>
<tr>
<td>7</td>
<td>Notification and authorization. Any person or organization intending to operate a facility or to conduct an activity shall submit to the regulatory body, as appropriate, a notification or an application for authorization.</td>
</tr>
</tbody>
</table>
### Table 5-1. General Basic Safety Standards Requirements applicable to Transport.

<table>
<thead>
<tr>
<th>Requirement Number</th>
<th>Requirement</th>
</tr>
</thead>
<tbody>
<tr>
<td>8</td>
<td><strong>Exemption and clearance.</strong> The government or the regulatory body shall determine which practices or sources within practices are to be exempted from some or all of the requirements of these Standards. The regulatory body shall approve which sources, including materials and objects, within notified practices or authorized practices may be cleared from regulatory control.</td>
</tr>
<tr>
<td>9</td>
<td><strong>Responsibilities of registrants and licensees in planned exposure situations.</strong> Registrants and licensees shall be responsible for protection and safety in planned exposure situations.</td>
</tr>
<tr>
<td>10</td>
<td><strong>Justification of practices.</strong> The government or the regulatory body shall ensure that only justified practices are authorized.</td>
</tr>
<tr>
<td>11</td>
<td><strong>Optimization of protection and safety.</strong> The government or regulatory body shall establish and enforce requirements for the optimization of protection and safety, and registrants and licensees shall ensure that protection and safety is optimized.</td>
</tr>
<tr>
<td>12</td>
<td><strong>Dose limits.</strong> The government or the regulatory body shall establish dose limits for occupational exposure and public exposure, and registrants and licensees shall apply these limits.</td>
</tr>
<tr>
<td>13</td>
<td><strong>Safety assessment.</strong> The regulatory body shall establish and enforce requirements for safety assessment, and the person or organization responsible for a facility or activity that gives rise to radiation risks shall conduct an appropriate safety assessment of this facility or activity.</td>
</tr>
<tr>
<td>14</td>
<td><strong>Monitoring for verification of compliance.</strong> Registrants and licensees and employers shall conduct monitoring to verify compliance with the requirements for protection and safety.</td>
</tr>
<tr>
<td>15</td>
<td><strong>Prevention and mitigation of accidents.</strong> Registrants and licensees shall apply good engineering practice and shall take all practicable measures to prevent accidents and to mitigate the consequences of those accidents that do occur.</td>
</tr>
<tr>
<td>16</td>
<td><strong>Investigations and feedback of information on operating experience.</strong> Registrants and licensees shall conduct formal investigations of abnormal conditions arising in the operation of facilities or the conduct of activities, and shall disseminate information that is significant for protection and safety.</td>
</tr>
<tr>
<td>17</td>
<td><strong>Radiation generators and radioactive sources.</strong> Registrants and licensees shall ensure the safety of radiation generators and radioactive sources.</td>
</tr>
<tr>
<td>18</td>
<td>Not applicable</td>
</tr>
<tr>
<td>19</td>
<td><strong>Responsibilities of the regulatory body specific to occupational exposure.</strong> The government or regulatory body shall establish and enforce requirements to ensure that protection and safety is optimized, and the regulatory body shall enforce compliance with dose limits for occupational exposure.</td>
</tr>
<tr>
<td>20</td>
<td><strong>Requirements for monitoring and recording of occupational exposure.</strong> The regulatory body shall establish and enforce requirements for the monitoring and recording of occupational exposures in planned exposure situations.</td>
</tr>
<tr>
<td>21</td>
<td><strong>Responsibilities of employers, registrants and licensees for the protection of workers.</strong> Employers, registrants and licensees shall be responsible for the protection of workers against occupational exposure. Employers, registrants and licensees shall ensure that protection and safety is optimized and that the dose limits for occupational exposure are not exceeded.</td>
</tr>
<tr>
<td>22</td>
<td><strong>Compliance by workers.</strong> Workers shall fulfil their obligations and carry out their duties for protection and safety.</td>
</tr>
<tr>
<td>Requirement Number</td>
<td>Requirement</td>
</tr>
<tr>
<td>--------------------</td>
<td>-------------</td>
</tr>
<tr>
<td>23</td>
<td><strong>Cooperation between employers and registrants and licensees.</strong> Employers and registrants and licensees shall cooperate to the extent necessary for compliance by all responsible parties with the requirements for protection and safety.</td>
</tr>
<tr>
<td>24</td>
<td><strong>Arrangements under the radiation protection programme.</strong> Employers, registrants and licensees shall establish and maintain organizational, procedural and technical arrangements for the designation of controlled areas and supervised areas, for local rules and for monitoring of the workplace, in a radiation protection programme for occupational exposure.</td>
</tr>
<tr>
<td>25</td>
<td><strong>Assessment of occupational exposure and workers' health surveillance.</strong> Employers, registrants and licensees shall be responsible for making arrangements for assessment and recording of the occupational exposure and for workers' health surveillance.</td>
</tr>
<tr>
<td>26</td>
<td><strong>Information, instruction and training.</strong> Employers, registrants and licensees shall provide workers with adequate information, instruction and training for protection and safety.</td>
</tr>
<tr>
<td>27</td>
<td><strong>Conditions of service.</strong> Employers, registrants and licensees shall not offer benefits as substitutes for measures for protection and safety.</td>
</tr>
<tr>
<td>28</td>
<td><strong>Special arrangements.</strong> Employers, registrants and licensees shall make special arrangements for female workers, as necessary, for protection of the embryo or fetus and of breast-fed infants. Employers, registrants and licensees shall make special arrangements for protection and safety for persons under 18 years of age who are undergoing training.</td>
</tr>
<tr>
<td>29</td>
<td><strong>Responsibilities of the government and the regulatory body specific to public exposure.</strong> The government or the regulatory body shall establish the responsibilities of relevant parties that are specific to public exposure, shall establish and enforce requirements for optimization, and shall establish, and the regulatory body shall enforce compliance with, dose limits for public exposure.</td>
</tr>
<tr>
<td>30</td>
<td><strong>Responsibilities of relevant parties specific to public exposure.</strong> Relevant parties shall apply the system of protection and safety to protect members of the public against exposure.</td>
</tr>
<tr>
<td>31</td>
<td><strong>Radioactive waste and discharges.</strong> Relevant parties shall ensure that radioactive waste and discharges of radioactive material to the environment are managed in accordance with the authorization.</td>
</tr>
<tr>
<td>32</td>
<td><strong>Monitoring and reporting.</strong> The regulatory body and relevant parties shall ensure that programmes for source monitoring and environmental monitoring are in place and that the results from the monitoring are recorded and are made available.</td>
</tr>
<tr>
<td>33</td>
<td><strong>Consumer products.</strong> Providers of consumer products shall ensure that such products are not made available to the public unless their use by members of the public has been justified, and either their use has been exempted or their provision to the public has been authorized.</td>
</tr>
</tbody>
</table>

A review of these safety standards requirements shows that the extensive efforts that have led to the current Transport Regulations [7] address most, if not all, of the above listed requirements. Examples include:

- Requirements 2 and 3: the establishment of a legal and regulatory framework and specifying the responsibilities of a regulatory body is established by specifying the need for competent authorities (para. 207 of SSR-6), where detailed guidance has been provided with References [11], [12], and [17];
• Requirement 6: the use of the graded approach is discussed in detail in Section 2.2, Chapter 4 and Appendix 3 of this document;
• Requirement 7: notification of competent authorities is required, following a graded approach, in paras 557 through 560 of SSR-6 [7], hence the requirements of the interim edition of the BSS for notification and authorization are fulfilled by means of compliance with the Transport Regulations; and
• Requirements 10, 11, and 12: justification, optimization, and dose limitations were addressed specifically in the forward to the 1985 edition of the Transport Regulations [8], and requirements and guidance have significantly expanded since then as discussed in Section 5.2 above.

Furthermore, specific application of the interim edition of the BSS requires that no person or organization shall transport a source other than in accordance with the requirements of the BSS.

Schedule I of the interim edition of the BSS stipulates that the exemption values (in terms of total activity and in terms of activity concentration) used in the Transport Regulations are, usually, numerically equal to the levels for exemption for moderate amounts that are given in Table I-1 in the 2011 interim edition of the BSS [4] (see also, General Basic Safety Standards Requirement 8).

Special arrangement, as specified in the Transport Regulations, is intended to give flexibility to consignors to propose alternative (and more practicable) safety measures effectively equivalent to those prescribed in the Transport Regulations. It is based on the requirement that the overall level of safety resulting from additional operational control must be shown to be at least equivalent to that which would be provided had all applicable provisions been met. It does not relate to General Basic Safety Standards Requirement 28.

In addition, this makes possible both the development of new controls and techniques to satisfy the existing and changing needs of industry in a longer term sense and the use of special operational measures for particular consignments where there may be only a short term interest. Such new safety techniques may later be assimilated into specific regulatory provisions and therefore the application of this concept is useful for the further development of the Transport Regulations.

5.5. General Requirements for Quality Assurance and Management Systems

The 1973 Edition of the regulations [20] incorporated as para. 839 a requirement on "Quality Control in Fabrication and Maintenance of Packaging". This was supplemented by approximately two pages of advisory text in the 1973 edition of Safety Series No. 37 [9]. That text specifically addressed the need for a quality assurance programme, but the regulations at that time did not specify the need for a quality assurance programme.

AG-126 [RBP033] which was convened in December 1977, advised that "the Agency should amend the regulations under the 90-day procedure by adding the first sentence of para. 820 from the Advisory Document, Safety Series No. 37. This would make it clear that designers, manufacturers and users must have quality and compliance assurance programmes for both Type A and Type B packages." The text from para. 820 of Safety Series No. 37 read specifically that "The designer of any approved packaging should, prior to fabrication, establish a quality assurance programme (1-3) to ensure that the packaging is manufactured in accordance with the approved design, and should be prepared to provide the competent authority with complete certification that the approved design requirements have been fully implemented".

TC-406 (convened in 1981 in Tokyo) [22] recommended that the text in the regulations on quality assurance be revised to clearly identify that the elements necessary in a Quality Assurance Programme should provide a requirement that the Programme be established prior to fabrication. It further recommended that provisions to enable the Competent Authority to carry out inspections of the packaging during manufacture and use, as well as to assure the Competent Authority that the necessary standards are met.
As a result of this recommendation, the 1985 edition [8] of the regulations incorporated, as a general provision, para. 209 a quality assurance programme is required.

The provisions on quality assurance for transport safety are consistent with the requirements in GS-R-3, except that in more recent IAEA safety standards the term ‘quality assurance’ is no longer used. Para. 306 of the Transport Regulations were revised in the 2012 edition of the Transport Regulations [7] to take into account this change in terminology, specifying the use of a management system (see General Basic Safety Standards Requirement 5).

As stipulated in GS-R-3 [3], the term ‘management system’ reflects and includes the initial concept of ‘quality control’ (controlling the quality of products) and its evolution through quality assurance (the system to ensure the quality of products) and ‘quality management’ (the system to manage quality). The management system is a set of interrelated or interacting elements that establishes policies and objectives and which enables those objectives to be achieved in a safe, efficient and effective manner.

Further guidance has been developed in TS-G-1.4 [18] to assist those involved in transport in applying the management system concept.

The Transport Regulation provisions on training address the General Basic Safety Standards Requirements 26 and 28, and also find their basis in the general safety requirements on management systems [3].

References for Chapter 5


6. RADIATION PROTECTION

Radiation protection is provided by the Transport Regulations, in part, through specification of activity limits and classification of materials and of packages. The classification of materials is briefly discussed in this chapter and is further elaborated in Chapter 8, and the classification of packages is elaborated in Chapter 9. In addition, radiation protection is provided through the imposition of provisions on the maximum radiation levels allowed outside of packages, conveyances and within occupied spaces on conveyances during routine, normal and accident conditions of transport.

6.1. Exemption values

The exemption levels are those given in Table I-1 of the 2011 edition of the BSS [1] for moderate quantities. The criteria for exemption are given in schedule I of the BBS; the derivation of the exemption levels is detailed in EC-RP 65 [2]. However, the exposure scenarios that were used for the derivation of the exemption levels did not explicitly address the transport of radioactive material. Additional calculations were performed for transport specific scenarios. These transport specific exemption values were then compared with the values in the BSS. It was concluded that the relatively small differences between both sets did not justify the incorporation into the Transport Regulations of a set of exemption values different from that in the BSS, given that the use of different exemption values in various practices may give rise to problems at interfaces and may cause legal and procedural complications [4].

6.2. A\textsubscript{1} and A\textsubscript{2} values

The radioactive content of type A packages are limited based on activity limits for special form radioactive material (A\textsubscript{1}), and other than special form radioactive material (A\textsubscript{2}) values using the Q system, which is elaborated in detail in Appendix I of TS-G-1.1 (Rev. 1) [6].

The history and methodology of the development of the A\textsubscript{1} and A\textsubscript{2} values is elaborated in Appendix 3. A recommendation was put forward at AG-126 [RBP033] that “a more coherent basis for the A\textsubscript{1}/A\textsubscript{2} philosophy and regulatory requirements that relate to permissible dose rates and packages test requirements” was needed. This appendix describes the initial development of the Q-system leading to the content limit values in the 1985 edition of the Transport Regulations [7].

The adoption of the Q-system for defining A\textsubscript{1} and A\textsubscript{2} values was recommended by AG-365.2 [RBP042] in March 1982. Appendix 3 also describes the updates that were made that lead to the content limit values in the 1996 edition of the Transport Regulations [8] [13]. AG-365.2 noted that “The main grounds for the Working Group’s recommendation to accept the Q system are that a dosimetric route, viz. skin dose, is identified in the Q system which was not considered in the earlier A\textsubscript{1}/A\textsubscript{2} system, and that the method of determining dose from beta emitters is improved over earlier versions”.

\[\text{11 An update may be needed, to be consistent with dose conversion factors given in the BSS. See also the derivation of exemption levels for additional radionuclides [3]; a complete list of which is in Table I-1 of the BSS [1].}\]

\[\text{12 The list of radionuclides in table I-1 of the BSS [1], taken from [3] is not the same as in table 2 of the Transport Regulations [5]. Moreover, for some radionuclides different progenies have been used in the calculations, sometimes leading to different exemption levels.}\]

\[\text{13 Following the lessons learnt from the Fukushima accident, the Q system may need to be reviewed with a view to assessing the need to consider additional exposure pathways.}\]
The Q system is more extensively documented in Appendix I of TS-G-1.1 [4]; which provides the best documentation of the efforts involved in the updates. In addition to controlling the quantity of radioactive material in a Type A package, the content limit values $A_1$ and $A_2$ are also used as a reference for establishing contents limits of excepted packages, release limits for Type B packages, to characterise low specific activity material, and to establish thresholds for requiring added tests such as the deep water immersion test.

The Type A package contents limits are constrained such that the radiological consequences of an accident involving a Type A package are limited as follows:

- the effective or committed effective dose to a person exposed in the vicinity of a transport package following an accident should not exceed 50 mSv; and
- the equivalent dose or committed equivalent dose received by individual organs, including the skin, of a person involved in the accident should not exceed 0.5 Sv, or in the special case of the eye 0.15 Sv.

It is assumed that a person is unlikely to remain at 1 m from the damaged package (shielding completely lost) for more than 30 minutes. The dose rate at 1 m from the unshielded material is therefore to be limited to 100 mSv/h.

It is further assumed that about one thousandth of the content would be released and that a person would take up about one thousandth of the released content (i.e. about one millionth of the content would be taken up).

In the case of special form material only external radiation has been taken into account.

### 6.3. Low specific activity (LSA) material

Low specific activity (LSA) material is classified into three groups (LSA-1, LSA-2 and LSA-3). These groups were developed with a consideration of the radiation dose hazard presented by the material.

LSA material may be transported in limited quantities in one of three types of industrial packages (i.e. IP-1, IP-2 and IP-3), and in some limited cases unpackaged depending on the specific radiological and physical characteristics of the material. The restrictions are such that in case of an accident the radiological consequences are comparable to those of a type A package involved in an accident: dose rate of 10 mSv/h at a distance of 3 m from the unshielded material; intake of about one millionth of the content (assumed uptake of 10 mg).

Materials containing radionuclides in concentrations above the exemption levels have to be regulated. It is reasonable, following a graded approach to radiation protection, that materials containing radionuclides up to 30 times the exemption level may be exempted from parts of the transport regulatory requirements and may be associated with the category of LSA-1.  

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**Footnotes:**

14 Since the BSS was updated in 2011 [1], the list of radionuclides in the 2012 edition of the Transport Regulations [5] is now outdated. Many of the radionuclides in the 2011 edition of the BSS are not covered in the Regulations. An update is needed to make the activity limits in the Transport Regulations consistent with those in the current BSS, both in terms of radionuclides addressed and dose conversion factors. Otherwise, a user of the Regulations will need to calculate activity limit values for any of the radionuclides listed in the BSS that are not listed in the Regulations.

15 These values of the equivalent dose or committed equivalent dose are compared with the dose limits for occupational exposure of workers over the age of 18 years:

- (a) An effective dose of 20 mSv per year averaged over five consecutive years (100 mSv in 5 years), and of 50 mSv in any single year;
- (b) An equivalent dose to the lens of the eye of 20 mSv per year averaged over 5 consecutive years (100 mSv in 5 years) and of 50 mSv in any single year;
- (c) An equivalent dose to the extremities (hands and feet) or the skin of 500 mSv in a year.
material. The factor of 30 was selected to take account of the rounding procedure used in the derivation of the BSS exemption levels and to give a reasonable assurance that the transport of such material does not give rise to unacceptable doses.

6.4. Surface contaminated objects (SCO)

Similar to LSA material, surface contaminated objects (SCO) may be transported in two types of industrial packages (i.e. IP-1 or IP-2), and in some limited cases unpackaged depending on the specific radiological and physical characteristics of the objects.

A differentiation is made between the two categories of SCO in terms of their contamination level. This differentiation defines the type of packaging to be used to transport these objects. As with the transport of LSA material, the level of radiological protection is equivalent to that of type A packages.

6.5. Classification as excepted package

The limits for radioactive material contents of excepted packages are such that the radiological hazard associated with a total release of contents is consistent with the hazard from a Type A package releasing part of its contents (see Appendix 3, section A.3.1).

6.6 Consistency of Transport Regulations with the Basic Safety Standards

[See Endnote ii]

6.7 Preventing excessive radiation levels outside of packages and conveyances

The transport regulations impose constraints on the level of radiation allowed outside of packages and conveyances. Specifically, the Regulations specify radiation level limits outside of packages and in some cases overpacks in paras 516, 527, 528, 573, 575, 579, 659, 671 and 820 (as well as indirectly through the transport index in para. 523); and outside of conveyances in paras. 566 and 573 [5].

In many package designs, these limits are satisfied through the use of shielding materials incorporated into the packaging elements. Historically, it is the responsibility of the package designer to ensure the adequacy of the shielding concept, while it is the responsibility of the consignor and/or carrier to ensure that, during transport operations, the regulatory radiation limits are not exceeded.

Detailed guidance on the inspection of biological shielding against gamma radiation for the density of the shield and its integrity is provided in TCSC-1056 [9], which replaced AECP 1056 [10]. This guidance addresses key issues that should be addressed by the designer and user of packages, including methods for ensuring shielding integrity, personnel qualifications and associated equipment requirements; preparation and procedures for testing of the adequacy of shielding; methods for dealing with flaws, establishing acceptance criteria, and repairing of shields; and ensuring safety during testing.

References for Chapter 6


[10] United Kingdom Atomic Energy Authority (UKAEA), Shielding integrity testing of radioactive material transport packaging, AECP 1056, Part 1, Risley, Warrington, United Kingdom (October 1977). {RBP039}


7. CONTROLS FOR TRANSPORT

When preparing a transport, and during the course of a transport, requirements and controls set up by the Regulations have to be met.

To the extent feasible, safety features are required to be built into the design of the package. By placing primary reliance on the package design and preparation, the need for any special actions during carriage (i.e. by the carrier) is minimized. Nevertheless, some operational controls are required for safety purposes and they are an important component for safety as they complement the requirements for package design and preparation.

The controls for transport are found primarily in Section V of SSR-6 \[1\], which specifies requirements for the following topics:

- Requirements before the first shipment and before each shipment (paras 501–503);
- Transport of other goods and other dangerous properties of contents (paras 504–507);
- Requirements and controls for contamination and for leaking packages (paras 508–514);
- Requirements and controls for transport of excepted packages and for transport of LSA material and SCO in industrial packages or unpackaged (paras 515–522);
- Determination of the transport index and the criticality safety index (paras 523–525);
- Limits on transport index, criticality safety index and radiation levels, and controls on transport for
  (a) packages and overpacks (paras 526–528),
  (b) consignments transported under exclusive use by road and rail (paras 571-574),
  (c) consignments transported by vessels (paras 575–576),
  (d) consignments transported by air (para. 577-579), and
  (e) consignments transported by post (paras 580–581);
- Assigning categories to packages (529);
- Assigning UN numbers and proper shipping names (para. 530 and Table 1);
- Requirements for marking and labelling of packages (paras 531–542);
- Requirements for placarding of
  (a) freight containers and tanks (paras 543–544), and
  (b) road and rail vehicles (paras 571–572);
- Consignor's responsibilities (paras 545–561);
- Requirements for transport and for storage in transit (paras 562–570);
- Customs operations and undeliverable consignments (paras 582–583); and
- Retention and availability of transport documents by carriers (paras 584–588).

As appropriate, the following sections of the chapter address the technical basis of these controls.

[See Endnote iii \[1\]]

Although not included in the “Controls” section of the Transport Regulations, administrative requirements (contained in Section VIII) are mostly linked to the approval of packages and shipments. The Regulations distinguish between cases where the transport can be made without competent authority package design or shipment approval and cases where some kind of approval is required. Whilst, in both cases, the Regulations place primary responsibility for compliance on the consignor and the carrier, the approval process provides a mechanism that can be implemented by the competent authorities to ensure that the design or shipment meets the relevant requirements and that the controls required for safety are adequate for the country and for the circumstances of the shipment.

7.1. Requirements and controls for contamination and for leaking packages

The Transport Regulations apply the optimisation principle to non-fixed contamination and also prescribe limits for non-fixed contamination on the surfaces of packages and conveyances under routine conditions of transport. The limits for the surfaces of packages derive from a
radiological model developed by Fairbairn for the 1961 Edition of the Transport Regulations [2]. In summary, the pathways of exposure were external beta irradiation of the skin, ingestion and the inhalation of re-suspended material. Consideration of radionuclides was limited to the most hazardous radionuclides in common use, namely Pu-239 and Ra-226 in the case of alpha emitters and Sr-90 in the case of beta emitters. These derived limits correspond to values that were generally accepted for laboratory and plant working areas and were thus conservative in the context of transport packages for which exposure time and handling time for workers were expected to be very much less than for workers in laboratories or active plants. Since this derivation, although there have been changes in radiological protection parameters, the transport contamination limits have not been changed.

In the 1970s, Loud [RBP060] reported on the monitoring of a specialized carrier in the U.S. of contamination on transport vehicles and associated terminals. Over a period of almost two years, based on 62 surveys at 27 terminals, and surveys of 127 power units and 1,640 trailers, the carrier had found the following:

- Contamination at terminals ranged from a single radioactive particle on the shop floor, to 14 m² of yard contaminated to 2,500 counts per minute beta-gamma;
- Radiation in excess of the 0.5 mr/h allowed by the U.S. DOT was detected on 3 power units and 16 trailers.
- One trailer had more than 400 times the smearable beta contamination allowed by regulations

Loud concluded that this record showed a potential for serious loss of control of radioactive contamination. All activities at terminals associated with maintenance of power units and trailers offer the potential for spreading radioactive contamination to shops and shop personnel. Evaluations such as this resulted in greater emphasis being placed on control of contamination from radioactive material packages during transport.

Due to the spent fuel packages and conveyances contamination issue that was raised in Europe in 1998–1999, the IAEA initiated a co-ordinated research project (CRP) on non-fixed surface contamination, and its results were issued as TECDOC-1449 [3]. The CRP developed the Basic Model to evaluate annual doses to workers and public from the non-fixed surface contamination of packages.

One of the conclusions states that the contamination limits in the transport regulations are conservative, especially for irradiated nuclear fuel package shipments; however, it was decided to retain the existing conservative limits for non-fixed contamination on the external surface of any package.

7.2. Determination of and limits on the transport index (TI) and the criticality index (CSI) [See Endnote iv iv]

7.2.1. Determination of and limits on the TI

Limits on the transport index (TI), which is a measure for the radiation level at a distance of 1 m from the surface of a package, are set to ensure the compliance with dose limits set in the BSS.

7.2.2. Determination of and limits on the CSI [See Endnote v v]

Fissile material and packages containing fissile material shall be classified under the relevant UN number as FISSILE, unless excepted by one of the provisions set up in the Regulations.

Typically, one or more of these provisions include consignment limits, regarding the mass of fissile nuclide to be transported in a consignment or on a conveyance.

More information about the history and the technical background of these limits is provided in Chapter 11 (Prevention of Criticality).
The Criticality Safety Index (CSI) is a number assigned to a package, overpack or freight container containing fissile material that is used to provide control over the accumulation of packages, overpacks or freight containers containing fissile material.

In order to comply with the general requirements for nuclear criticality control, limits are set for the maximum CSI in any one group of packages. In addition, a minimum spacing distance between groups of packages is specified. Prevention of nuclear criticality is then achieved by limiting neutron interaction between packages containing fissile material.

In the case of transport under exclusive use, the limits about the CSI may be exceeded because of the additional operational controls. In some instances, a multilateral approval of the shipment can also be needed, in order to enable the competent authorities concerned to judge the need for any special controls to be applied during transport.

More information about the history and the technical background of these limits is provided in Chapter 11 (Prevention of Criticality).

*Criticality Safety Index (CSI) and Transport Index (TI)*

The Criticality Safety Index (CSI) is a new concept defined for the first time in the 1996 edition of the Regulations. The 1967, 1973 and 1985 Editions of the Regulations used the “transport index” for both radiological control and control of criticality safety of packages containing fissile material. These editions of the Regulations defined the transport index (TI) so that a single number accommodated both radiological safety and criticality safety considerations (the Transport Index in the 1985 Edition was the lower value of the TI (as defined in the 1996 Edition and those which came later) and the CSI (as defined in the 1996 Edition and those which came later)). As the operational controls needed for radiological protection and for criticality safety are essentially independent, the 1996 Edition of the Regulations has separated the CSI from the TI, which is now defined for radiological control only. This separation into two indices enables a clear recognition of the basis for operational control of a fissile package and eliminates potential unnecessary restrictions caused by the use of one index.

*Criticality Safety Index (CSI) and “allowable number”*

In the 1967 and 1973 Editions of the Regulations, a so-called “allowable number” was defined for Fissile Class II packages and Fissile Class III packages and was the basis for the contribution of the prevention of criticality to the calculation of the Transport Index (TI).

7.3. Limits and controls on radiation levels

7.3.1. Limits on radiation levels

The requirement that the radiation level at the surface of an excepted package should not exceed 5 µSv/h was established in order to ensure that sensitive photographic material will not be damaged and that any radiation dose to members of the public will be insignificant.

The radiation level limits inherent in the definition of the categories (and associated labels) have been derived on the basis of assumed package/cargo handling procedures, exposure times for transport workers and exposure times for photographic film. Historically these were derived as follows [4]:

(a) **0.005 mSv/h at surface** – This surface limit was derived, not from consideration of radiation effects on persons, but from the more limiting effect on undeveloped photographic film. Evaluation of the effect of radiation on sensitive X ray film in 1947 showed that threshold fogging would occur at an exposure of 0.15 mSv and a limit was set in the 1961 Edition of the Transport Regulations of 0.1 mSv linked to a nominal maximum exposure time of 24 h. In later editions of the Transport Regulations (1964, 1967, 1973 and 1973 (As Amended)), the 24 hour period was rounded to 20 hours and the limiting dose rate of 0.005 mSv/h was taken as a rounded-down value to provide protection to undeveloped film for such periods of transport. This dose rate was applied as a surface...
limit for category I-WHITE packages, which would ensure there being little likelihood of radiation damage to film or unacceptable doses to transport personnel, without need for segregation requirements.

(b) **0.1 mSv/h at 1 m** – For the purposes of limiting the radiation dose to film and to persons the dose of 0.1 mSv discussed in (a) above was combined with the exposure rate at 1 m from the package and an exposure time of 1 hour to give the 10 times TI limitation of the 1964, 1967 and 1973 Editions of the Transport Regulations (10 ‘radiation units’ in the 1961 Edition). This was based upon an assumed transit time of 24 hours and the conventional separation distance of 4.5 m (15 feet) between parcels containing radium in use by the US Railway Express Company in 1947. The above limitation would yield a dose of approximately 0.1 mSv at 4.5 m (15 feet) in 24 h.

I **2.0 mSv/h at surface** – A separate limit of 2.0 mSv/h at the surface was applied in addition to the limit explained in (b) above on the basis that a transport worker carrying such packages for 30 minutes a day, held close to the body, would not exceed the then permissible dose of 1 mSv per 8 hour working day. While such doses would no longer be acceptable, the adequacy of the current radiation level limits, in terms of radiological safety, has been confirmed by a number of surveys where radiation exposure of transport workers has been determined [5-14] and by an assessment performed by the IAEA in 1985 [15].

However, it must be recognized that the permitted radiation levels around packages and conveyances do not alone ensure acceptably low doses and the Transport Regulations also require the establishment of RPPs (para. 302) and the periodic assessment of radiation doses to persons due to the transport of radioactive material (para. 308).

### 7.3.2 Radiation controls on transport by road and rail

In most cases the radiation level at any point on the external surface of a package is limited to 2 mSv/h. For road and rail transport, when transported under exclusive use, packages and overpacks are allowed to exceed 2 mSv/h if access to the enclosed areas in the vehicle is restricted. The restrictions are to prevent unnecessary or uncontrolled exposures of persons.

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.

In establishing the dose rate for a conveyance, account may be taken of additional shielding within the conveyance. However, the integrity of the shielding should be maintained during routine transport; otherwise compliance with the conveyance radiation limit may not be maintained.

### 7.3.3. Radiation controls on transport by vessel

Each mode of transport has its own unique features. In the case of transport by sea the possibility of journey times of weeks or months and the need for continued routine inspection throughout the journey might lead to significant exposures during the carriage of the radioactive material. Simply having the exclusive use of a hold, compartment or defined deck area, particularly the latter, was not felt to provide sufficient radiological control for high radiation level packages. Two further restrictions were therefore introduced for packages having a surface radiation level greater than 2 mSv/h: either they must be in (or on) a vehicle or they must be transported under special arrangement. Access and radiation levels are therefore controlled by the additional requirements relating to transport by rail and road or by controls relevant to particular circumstances prescribed by the competent authority under the terms of the special arrangement.

The simple controls on the accumulation of packages as a means of limiting radiation exposure or preventing criticality may not be appropriate for ships dedicated to the transport of radioactive material. Since the vessel itself may be transporting consignments from more than
one consignor, it could not be considered as being under exclusive use, and the usual limits for the transport index for conveyances and for the criticality safety index for conveyances containing fissile material might therefore be unnecessarily restrictive. Special use vessels employed for the transport by sea of radioactive material have been adapted and/or dedicated specifically for that purpose.

7.3.4. Radiation controls on transport by air

The Regulations prohibits the transport by air of vented Type B(M) packages, packages that require external cooling by an ancillary cooling system, packages subject to operational controls during transport and packages containing liquid pyrophoric materials. If venting were permitted, it would be difficult to design this to occur safely. Ancillary cooling and other operational controls would be difficult to ensure within an aircraft under normal and accident conditions. Any liquid pyrophoric material poses a special hazard to an aircraft in flight: where a radioactive substance which has the subsidiary hazard of pyrophoricity is also a liquid, there is a greater probability of a spill occurring, and it is therefore absolutely forbidden to transport such a substance by air.

Packages or overpacks having a surface radiation level greater than 2 mSv/h shall not be transported by air except by special arrangement. Because of the higher radiation levels than would normally be allowed, greater care is necessary in loading and handling. The requirement for such consignments to be transported by special arrangement ensures the involvement of the competent authority and allows special handling precautions to be specified, either during loading, in flight or at any intermediate transfer point.

7.3.5. Radiation controls on transport by post

For movement by post, the allowed levels of activity are only one tenth of the levels allowed for excepted packages by other modes of transport, for the following reasons:

- The possibility exists of contaminating a large number of letters, etc., which would subsequently be widely distributed, thus increasing the number of persons exposed to the contamination.
- This further reduction would result in a concurrent reduction in the maximum radiation level of a source which has lost its shielding, and this is considered to be suitably conservative in the postal environment in comparison with other modes of transport.
- A single mailbag might contain a large number of such packages.

7.4. Determining categories of packages

The radiation level limits inherent in the definition of the categories (and associated labels) have been derived on the basis of assumed package/cargo handling procedures, exposure times for transport workers and exposure times for photographic film. Historically these were derived as follows [4]:

(a) 0.005 mSv/h at surface – This surface limit was derived, not from consideration of radiation effects on persons, but from the more limiting effect on undeveloped photographic film. Evaluation of the effect of radiation on sensitive X-ray film in 1947 showed that threshold fogging would occur at an exposure of 0.15 mSv and a limit was set in the 1961 Edition of the Transport Regulations of 0.1 mSv linked to a nominal maximum exposure time of 24 h. In later editions of the Transport Regulations (1964, 1967, 1973 and 1973 (As Amended)), the 24 hour period was rounded to 20 hours and the limiting dose rate of 0.005 mSv/h was taken as a rounded-down value to provide protection to undeveloped film for such periods of transport. This dose rate was applied as a surface limit for category I-WHITE packages, which would ensure there being little likelihood of radiation damage to film or unacceptable doses to transport personnel, without need for segregation requirements.

(b) 0.1 mSv/h at 1 m – For the purposes of limiting the radiation dose to film and to persons the dose of 0.1 mSv discussed in (a) above was combined with the exposure rate at 1 m from the package and an exposure time of 1 hour to give the 10 times TI limitation of the 1964, 1967
and 1973 Editions of the Transport Regulations (10 ‘radiation units’ in the 1961 Edition). This was based upon an assumed transit time of 24 hours and the conventional separation distance of 4.5 m (15 feet) between parcels containing radium in use by the US Railway Express Company in 1947. The above limitation would yield a dose of approximately 0.1 mSv at 4.5 m (15 feet) in 24 h.

(c) 2.0 mSv/h at surface – A separate limit of 2.0 mSv/h at the surface was applied in addition to the limit explained in (b) above on the basis that a transport worker carrying such packages for 30 minutes a day, held close to the body, would not exceed the then permissible dose of 1 mSv per 8 hour working day. While such doses would no longer be acceptable, the adequacy of the current radiation level limits, in terms of radiological safety, has been confirmed by a number of surveys of radiation exposure of transport workers as discussed in Section 7.3.1 above.

However, it must be recognized that the permitted radiation levels around packages and conveyances do not alone ensure acceptably low doses and the Regulations also require the establishment of Radiation Protection Programmes (RPPs) and the periodic assessment of radiation doses to persons due to the transport of radioactive material.

7.5. Assigning and use of UN Numbers

[See Endnote vi]

7.6. Requirements for marking and labelling of packages

Markings are placed on each packaging, part of which are unique to each packaging (e.g. serial number); whereas the labels are placed on packages when prepared for transport and provide information on the actual radioactive contents and the external radiation levels.

7.6.1. Marking of packages

Identification of the consignor or consignee

To retain the possibility of identifying the consignee or consignor of a package for which normal control is lost (e.g. lost in transit or misplaced), an identification marking is required on outside of the packaging. This marking may consist of the name or address of either the consignor or consignee, or may be a number identifying a way-bill or transport document which contains this information. Each overpack should be so marked unless the markings on all the inner packages are clearly visible within the overpack.

UN number

The UN number marked on the package and the overpack, when appropriate, and indicated in the documents is important information in the event of incidents and accidents. The UN number corresponding to the approval certificate issued by the competent authority of the country of origin of design gives the information about package type that is needed for emergency management.

UN numbers can also be used for compliance situations, performance checks and controls, data collection and other statistical purposes should the competent authority find merit in this application.

The UN numbers 2977 and 2978 should be used instead of LSA material shipping numbers, to help the emergency response team to address the specific hazards raised by uranium hexafluoride in the event of an accident involving a severe fire; a fire on a uranium hexafluoride cylinder raises more severe hazards than a fire on other LSA material [16]. It is also considered that when an accident occurs involving uranium hexafluoride transported under special arrangement, it is better that the emergency response teams are quickly informed that uranium hexafluoride is involved in the accident.

The implementation of the 1996 Edition of the Transport Regulations could lead to multiple labelling and marking as a consequence of divergence between approvals issued by different
competent authorities. To avoid having to change the marking and labelling at border crossings, only one single set of information should be applied. The 2005 Edition of the Regulations (and those which came later) specifies that the marking shall be in accordance with the certificate of the country of origin of design.

**Gross mass**

Packages exceeding 50 kg gross mass are likely to be handled by mechanical rather than manual means and require marking of the gross mass to indicate the possible need for mechanical handling and observance of floor loading and vehicle loading limits. To be useful in this respect, the marking is required to be legible and durable.

**Durability of marking**

Markings should be durable in the sense of being at least resistant to the rigours of normal transport, including the effects of open weather exposure and abrasion, without substantial reduction in effectiveness.

**Type of package**

All Type B(U), Type B(M), Type C and fissile material package designs require competent authority approval. Markings on such packages provide a link between the individual package and the corresponding national competent authority design approval (via the identification mark), as well as information on the kind of competent authority design approval.

Although no competent authority approval is required for industrial or Type A packages whose contents are not fissile material, the designer and/or consignor should be in a position to demonstrate compliance with any cognizant competent authority. The package marking therefore should identify the organization responsible for designing the package. This marking assists in the inspection and enforcement activities of the competent authorities. Where the designer is also the consignor, the mark may also provide, to the knowledgeable observer, valuable information in the event of an accident.

The 1996 Edition of the Transport Regulations introduces the requirement to identify industrial packages with a mark. The design of the mark is consistent with other similar marks in that it includes the word ‘Type’ together with the appropriate industrial package description (e.g. Type IP-2). The design of the mark also avoids potential confusion where, in other transport regulations, the abbreviation IP may be used for a different purpose. For example, the ICAO Technical Instructions use IP to mean Inner Packaging; for example ‘IP.3’ to denote one out of ten particular kinds of inner packagings.

**Serial number**

The marking with a serial number is required because operational management system and maintenance activities are oriented towards each packaging and the corresponding need to perform and verify these activities on an individual packaging basis. The serial number is also necessary for the competent authority’s compliance assurance activities and for application of the provisions dealing with transitional arrangements.

**Trefoil**

The marking of a Type B(U), Type B(M) or Type C package with a trefoil symbol resistant to the effects of fire and water is needed to ensure that such a type of package can be positively identified after a severe accident as carrying radioactive material.

7.6.2. Labelling of packages

It is noteworthy that the initial edition of the regulations (1961) showed an image of a “skull and crossbones” on the labels (see pages 24 and 25 of Reference [19]). Gibson [RBP035] notes that the initial panel convened in March 1963 to review the 1961 edition of the regulations made “a clear cut recommendation to delete the skull and crossbones from all IAEA transport..."
labels”. Thus the designs of the labels specified in the 1964 edition [20] were essentially the design that has been used ever since.

Packages, overpacks and freight containers can be characterized as handling or cargo units. Transport workers need to be made aware of the contents when such units carry radioactive material and need to know that potential radiological and criticality hazards exist. The labels provide that information by the trefoil symbol, the colour and the category (I-WHITE, II-YELLOW or III-YELLOW), and the fissile label. Through the labels it is possible to identify (a) the radiological or criticality hazards associated with the radioactive content of the cargo unit and (b) the storage and stowage provisions which may be applicable to such units.

The radioactive material labels used form part of a set of labels used internationally to identify the various classes of dangerous goods. This set of labels has been established with the aim of making dangerous goods easily recognizable from a distance by means of symbols. The specific symbol chosen to identify cargo units carrying radioactive material is the trefoil.

The content of a cargo unit may, in addition to its radioactive properties, also be dangerous in other respects, for example corrosive or flammable. In these cases the regulations pertaining to this additional hazard must be adhered to. This means that, in addition to the radioactive material label, other relevant labels need to be displayed on the cargo unit.

**Labelling for radioactive contents**

In addition to identifying the radioactive properties of the contents, the labels also carry more specific information regarding the contents (i.e. the name of the nuclide and the activity). In the case of fissile contents, the total mass of fissile nuclides may be used in place of activity. This information is important in the event of an incident or accident where content information may be needed to evaluate the hazard. The more specific information regarding the contents is not required for LSA-I material, because of the low radiation hazard associated with such material.

Yellow labels also show the TI of the cargo unit (i.e. package, overpack, tank and freight container). The TI information is essential in terms of storage and stowage in that it is used to control the accumulation and ensure proper separation of cargo units. The Transport Regulations prescribe limits on the total sum of TIs in such groups of cargo units.

**Labelling for criticality safety**

The Criticality Safety Index (CSI) is a value used for accumulation control of packages needed for criticality safety purposes (see section A.16.1.16). The control is provided by limiting the sum of the CSIs.

To facilitate such control, the CSI is required to be displayed on a label which is specifically designed to indicate the presence of fissile material in the case of packages, overpacks or freight containers where contents consist of fissile material not excepted.

The CSI label is additional to the category labels (categories I-WHITE, II-YELLOW and III-YELLOW), because its purpose is to provide information on the CSI, whereas the category label provides information on the TI and the contents. The CSI label, in its own right, also identifies the package as containing fissile material.

Like the TI, the CSI provides essential information relevant to storage and stowage arrangements in that it is used to control the accumulation and ensure proper separation of cargo units with fissile material contents.

**7.7. Requirements for placarding**

Placards, which are used on road and rail vehicles, large freight containers and tanks, are designed in order to clearly identify the hazards of the dangerous goods. Displaying the placards on all four sides of the freight containers and tanks ensures ready recognition from all directions. The size of the placard is intended to make it easy to read, even at a distance. To prevent the need for an excessive number of placards and labels, an enlarged label only may be
used on large freight containers and tanks where the enlarged label also serves the function of a placard.

The display of the UN number can provide information on the type of the radioactive material transported, including whether or not it is fissile, and information on the package type. This information is important in the case of incidents or accidents resulting in leakage of the radioactive material in that it assists those responsible for emergency response to determine proper response actions.

During deliberations preceding the publishing of the 1985 edition of the Regulations [21], AG-406 [RBP049] reviewed the requirements for placarding and proposals that had been made to adjust those requirements. For example, the minimum dimensions, background colour, etc. requirements for the diamond shaped placard in Fig. 5 of the 1985 edition were established by AG-406; while also suggesting (as is shown in the figure) that the use of the word “RADIOACTIVE” is optional.

AG-406 also proposed the additional dimensions that are shown for the rectangular placard shown in Fig. 6 of the 1985 edition of the Regulations; and further it recommended the text that is contained in para. 443 of the 1985 edition.

7.8. Requirements for transport and for storage in transit

7.8.1. Controls on segregation

Operational controls that are applied in the transport of radioactive material can include the use of segregation distances.

In preparing to issue the 1985 edition of the Regulations [21], AG-406 [RBP049] considered the need for strengthening the requirements for segregation of radioactive material from transport workers and members of the public. AG-406 concluded that the Regulations needed to establish reference doses to ensure adequate radiation protection. The reference doses agreed were: 100 mrem/y for members of the public, and 500 mrem/y for transport workers. Draft text for this requirement was developed, which ultimately became para. 205 of the 1985 edition of the Regulations.

The history of the parameters used in the derivation of segregation tables is that originally a fraction of the dose limit was chosen in each case (for workers and for members of the public) and what was considered to be a realistic model was used to derive the tables of segregation distances for each mode of transport. It was noted that real data were sparse and that these data should be reviewed. With the production of more realistic data [9, 14] it has become apparent that the models are very conservative. So conservative, in fact, that as the dose limits have been reduced the model and dose criteria have, on several re-examinations, been considered to provide adequate segregation [17]. By comparing all aspects of the practice (not simply segregation) with appropriate dose constraints for transport (as a whole — not just for one transport operation) the use of the current tables has been deemed to provide an adequate level of safety.

An example of such a review was carried out during the preparation of the 1996 Edition of the Transport Regulations. The model and dose criteria were examined in light of the developing philosophy of dose constraints as amplified in Ref. [18]. A dose constraint of 0.7 mSv was considered appropriate for exposure of a critical group of the public to direct radiation from sources such as radioactive material in transport. This constraint was envisaged as being applicable to global transport operations in general rather than the operations of one particular consignor. Over a series of three technical meetings information on assessed exposures to members of the public was actively collected and evaluated. The assessment of this information demonstrated that exposures being received by members of the public from these operations were far below the dose criterion used in the modelling and the appropriate dose constraint. The conclusion of these studies was that the existing segregation tables and the other provisions
of the Transport Regulations together provide for an appropriate level of radiological safety. However, these evaluations were not adequately reflected in the associated guidance publication. It is considered that the current segregation tables are consistent with the use of appropriate dose constraints. For example, the postulated public dose presented in the tables relate to a 1 mSv dose with a very pessimistic model (exposures are actually estimated to be of the order of tens of µSv), not (as was intimated in the 1996 guidance publication) a realistic model. [[[NOTE: further research is needed for the basis for this text. The reference citation in TS-G-1.1 (2008) is incorrect.]]]

Although not a radiation protection issue, an evaluation of the effect of radiation on fast X-ray films in 1947 [4] determined that they may show slight fogging after development when exposed to doses exceeding 0.15 mSv of gamma radiation. This could interfere with the proper use of the film and provide incorrect diagnostic interpretation. Other types of film are also susceptible to fogging although the doses required are much higher. Since it would be impracticable to introduce segregation procedures which vary with the type of film, the provisions of the Transport Regulations are designed to restrict the exposure of undeveloped films of all kinds to a level of not more than 0.1 mSv during any journey from consignor to consignee.

The different time durations involved for sea transport (in terms of days or weeks) and air or land transport (in terms of hours or days) mean that different tables of segregation distances are used, so that the total film exposure during transit is the same for each mode. More than one mode of transport and more than one shipment may be involved in the distribution and ultimate use of photographic film. Thus, when segregation distance tables are being established for a specific transport mode, only a fraction of the limit prescribed in para. 562 should be committed to that mode. In road transport a driver may ensure sufficient segregation from photographic film carried in other vehicles by leaving a clear space of at least 2 m all around the vehicle when parking.

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.

7.8.2. Controls on Stowage

One of the reasons for limiting the accumulation of packages in groups, or in conveyances and freight containers is to prevent the creation of higher than acceptable radiation levels as a result of the additive effects of radiation from the individual packages. For consignments not carried under exclusive use, this is done by placing a limit on TI. The theoretical maximum dose rate at 2 m from the surface of a vehicle carrying a TI of 50 was historically calculated as 0.125 mSv/h, and considered to be equivalent to 0.1 mSv/h since the maximum was unlikely to be reached. Experience has confirmed the acceptability of these values.

References for Chapter 7


8. CLASSIFICATION OF MATERIALS

The classification of materials has significantly changed and matured since the first edition of the Transport Regulations [1]. For example, in the early editions of the Regulations (i.e. the 1961, 1964 and 1967 editions) the individual radionuclides were classified into groups based on radioactivity as well as classifications based on physical properties. Between that time period and the issuing of the 2009 edition of the Regulations, the terms “classify” and “classification” were not used significantly, but the materials were still ordered according to radioactivity and physical properties. However, with the 2009 edition [2], and also the 2012 edition of the Regulations (SSR-6) [3], classification was imposed in detail according to specific physical characteristics of the radioactive material.

The following subsections provides a summary of how the concept of classification of materials evolved through the various editions of the transport regulations, illustrating how these classifications changed, and providing some insight into why the changes were made.

8.1. Classification of material in the 1961 edition of the Transport Regulations

In that first edition, the materials were classified as follows:

- Group I – very high radioactivity
- Group II – high radioactivity
- Group III moderate radioactivity
- Large radioactive sources
- Non-friable massive solids that are non-soluble in water and non-reactive with air or water
- Pyrophoric radioactive material
- Explosive radioactive material
- Radioactive materials of low specific activity
- Low Specific Activity material
- Fissile materials

From the radiological perspective, each radionuclide was assigned a group (i.e. Group I, II or III), where the quantity allowed in a package was limited except, if that quantity was exceeded, then it was to be treated as a “large radioactive source”. These materials were required to be transported in Type B packagings, the design of which was to be approved by the competent authority of the country in which the shipment originated.

From the physical characteristics perspective:

- small quantities of the non-friable solids could be transported in Type A packagings; however if a specified activity limit was exceeded, they were required to be transported in Type B packagings;
- pyrophoric radioactive materials were required to be transported in Type B packagings; and
- explosive radioactive materials were only permitted to be transported under special arrangements.

In addition, characteristics of materials with low specific activity were specified. When a material satisfied those characteristics, their transport was allowed in strong, leak-proof packages or in vehicle compartment specially designed to prevent leakage under normal conditions of transport. The low specific activity material classification was developed with a consideration of the radiation dose hazard presented by the material.

Materials containing radionuclides in concentrations above the exemption levels must be regulated. However, it is reasonable that materials containing radionuclides slightly above exemption levels may themselves be exempted from parts of the transport regulatory requirements; these may be associated with the low end of the low specific activity material spectrum, or with the exemptions noted in the penultimate paragraph of this sub-section. This approach is intended to allow packaging and transport flexibility while concurrently giving reasonable assurance that the transport of such materials does not give rise to unacceptable doses.
Exemptions from parts of the regulatory requirements were specified for Groups I, II and III materials based on the maximum activity in a package and other features of the package; while the transport of instruments with radioactive materials as components and the transport of empty packages were further exempted from many of the regulatory requirements.

The technical basis for the classifying of the materials in terms of activity limits for packages is summarized in Section A.3.1 of Appendix 3. Additional information on the basis of the initial classification of material can be found in References [4], [5].

8.1. Classification of LSA material in the 1961 edition of the Transport Regulations

The concept of low specific activity (LSA) material was introduced in Section 14.1 of the 1961 edition of the Regulations. This concept was mentioned briefly in the 1961 edition of Safety Series No. 7 [22]; however, no technical basis was elaborated at that time. The initial regulations required that such materials be "packed in strong, industrial, leak-proof packages or loaded in vehicles or compartments specially designed to ensure that there will be no leakage under conditions normally incident to transport" (para. 14.2 of the 1961 edition of the Regulations).

8.2. Classification of material in the 1964 edition of the Transport Regulations

As elaborated in Appendix 3, the grouping and limits changed to a larger number of groups in the 1964 edition of the Transport Regulations [6] to eight groups. Section A.3.1 of Appendix 3 shows that Aikens [7] provided a summary of the activity limits for the eight different groups as they applied to the three package types that then existed in the Regulations (i.e. Exempt from Requirements, Type A Packaging, and Type B Packaging).

Concurrently, the 1964 edition retained the concept of large radioactive sources and low specific activity materials; however, the concept of special form radioactive material was introduced in place of the concept of 'massive non-friable solid' with a different set of limits [8]. Specifically, Gibson [RBP035] notes that the March 1963 review panel leading to the 1964 edition of the regulations addressed this issue as follows:

"The 1961 regulations allow higher activity content in a package when the material is in the form of a 'massive non-friable solid'. The panel has agreed to extend this relaxation to encapsulated substances irrespective of the form of the substance itself. A specification has been produced describing what is meant both by 'encapsulation' and 'massive non-friable solid' and this amendment should be of great practical value to the consignors of commercial radioisotopes."

The concept of exemptions from parts of the regulatory requirements was expanded to include not only instruments but also of manufactured articles with radioactive materials.

8.2.1. Classification of contaminated solid, non-radioactive objects in the 1964 edition of the Transport Regulations

A specification for "Objects of non-radioactive material, externally contaminated with radioactive material" with contamination levels specified (see para. A.2.8(e) of the 1964 edition of the Regulations [6]) was added to the options for low specific activity material. This was the progenitor of what was to become low level solids and then, later, surface contaminated objects.

8.3. Classification of material in the 1967 edition of the Transport Regulations

In the 1967 edition [9], the number of groups was changed from eight to seven. Fairbairn [8] noted, in the discussion of the work of various panels convened to address the Regulations, that the panels convened for the 1964 and 1967 editions of the Regulations. As already illustrated in the last subsection, the 1964 "Panel listed radionuclides within Groups I – VII"; whereas "in the 1967 issue of the regulations this was changed to Groups I – VII, the previous Groups VII and VIII having a common contents limit".
8.3.1. Classification of LSA material including contaminated solid, non-radioactive objects in the 1967 edition of the Transport Regulations


8.4. Classification of material in the 1973 edition of the Transport Regulations

The 1973 edition of the Transport Regulations [10] first introduced the concept of A1 and A2 values for specifying the contents limits in packages. Fairbairn [8] noted that with this A1 /A2 system "...each nuclide has two Type A package limits, A1 curies when in special form and A2 curies when not in special form". As elaborated in Section A.3.1 of Appendix 3, Fairbairn discussed how the A1 values were derived using a rather limited, but conservative model where it was assumed that the source was intact but had completely released from its packaging and associated shielding; whereas, the A2 values were derived assuming that the material was completely released from its packaging where the value was calculated using the then most recent ICRP guidance [11], while imposing the assumption that internal exposure from intake would be $10^{-6}$ of the package contents. The introduction of the A1 and A2 values for each radionuclide in the 1973 edition of the Regulations established a more rigorous technical basis for classifying the radioactive materials for shipment; and also "enabled the panel responsible for the 1973 revision to dispense with the artificial concept of the 'large radioactive source'"[8].

Paras 401 – 421 of the 1973 edition of Safety Series No. 37 [12] elaborate on the technical basis for establishing the A1 and A2 values. It states that the "replacement of the 20-Ci Type A package limit for special form, ..., by an A1 curie limit for each nuclide represents an increase in safety with regard to emitters of gamma photons in excess of 1 MeV and also with regard to alpha, beta and neutron emitters for which assessments based on specified models were made". Because of the advanced sophistication of the model used, a number of limits increased from what they were in the 1967 edition while others decreased.

Furthermore, for the A2 values, the limits were previously established based on the "most toxic member of the Group (......), the resulting A2 values represent a relaxation of Type A package limits for most nuclides".

8.4.1. Classification and technical basis of LSA material and low level solids in the 1973 edition of the Regulations

The 1973 edition of the Regulations introduced for the first time the concept of low-level solid radioactive materials (see para. 120 of the 1973 edition of the Regulations [10]). This included: non-radioactive solids contaminated with limited quantities of radioactivity at levels to not exceed a value of 20 µCi/cm² for beta and gamma emitters and low toxicity alpha emitters indicated in Table XI, and 2 µCi/cm² for other alpha emitters.

Concurrently, the 1973 edition contained as part of its definition of LSA material (see para. 121(g)), the specification of "Objects of non-radioactive material contaminated with radioactive material, provided non-fixed surface contamination does not exceed ten times the values given in Table XI and the contaminated object or the contamination on the object, if reduced to the minimum volume under conditions likely to be encountered in transport, such as dissolution in water with subsequent recrystallization, precipitation, evaporation, combustion, abrasion, etc., would have an average estimated specific activity of no more than $10^{-4}$ A2/g".

These two definitions, one classified as a low level solid and the other classified as LSA material were the precursors to the ultimate classification, introduced in the 1985 edition of the Regulations as surface contaminated objects (SCO).

Technical basis for the LSA materials as defined in the 1973 edition of the Regulations
Paras 338 to 346 of the 1973 edition of Safety Series No. 37 [12] elaborates on the basic principles used for establishing the specifications for LSA material. It makes special note that the criteria were established such that "the specific activities of which are so low that it is inconceivable that, under any circumstances arising in transport, a sufficient mass of such materials could be taken into the body to give rise to a significant radiation hazards". The basic principle was extended to other solid materials on the basis of a specific model elaborated in para. 339. It was concluded that material with a specific activity not exceeding $10^{-4} \text{ A}_2 / \text{g}$ could be regarded as inherently safe, as long as the activity is uniformly distributed throughout the material, and there is no conceivable mechanism by which the specific activity could be increased during transport.

**Technical basis for the low level solids as defined in the 1973 edition of the Regulations**


In elaborating on LSA material, para. 345 of the 1973 edition of Safety Series No. 37 states that the contaminated objects described in para. 121(g) of that edition (as described above) "are, strictly speaking no low specific activity materials at all". That concern ultimately led to reclassifying such objects as SCO in the 1985 edition of the Regulations [15].

Paras 354 to 364 of the 1973 edition of Safety Series No. 37 notes that "this category extends the concept of non-inherently safe low specific activity materials in Section I of the Regulations, para. 121 (d) and (g), to materials with a distributed activity of up to twenty times the levels specified there, liquids and solids being specifically excluded". It then discusses the basis for the non-radioactive material contaminated with radioactive materials, noting that (a) this allows a twenty-fold increase in the maximum surface contamination levels over what was then specified for low specific activity materials; (b) this increase means these "materials are therefore potentially very hazardous, which is why packaging specification must be laid down"; and (c) "the contamination must be in a 'non-readily dispersible form'".

**8.5. Classification of material in the 1973 Revised Edition (As Amended), 1979, of the Transport Regulations**

Essentially no changes were made in classification of materials in the 1973 Revised Edition (As Amended), which was issued in 1979 [13]. As noted in the Foreword to this document, only minor amendments, mainly of an editorial nature, plus a number of changes of detail were implemented. However, it is noted that paras 4.12 – 4.16 of the Second Edition of Safety Series No. 37 [14] provides additional insight into the manner by which the original $A_1$ and $A_2$ values were derived.

**8.6. Classification of material in the 1985 edition of the Transport Regulations**

**8.6.1. Basis for $A_1$ and $A_2$ values in the 1985 edition of the Transport Regulations**

The establishment of a sound basis for specifying the $A_1$ and $A_2$ activity values for specific radionuclides preceded the publication of the 1985 edition of the Transport Regulations [15]. The result of this extensive effort, known as the Q system, provided a much more solid basis for the values. Appendix I of the 1990 edition of Safety Series No. 7 [16] provided a detailed discussion of the technical basis for the $A_1$ and $A_2$ values derived using the initial Q system. In addition, Section 6.2 above, and Appendix 2 further address the basis of the Q system.

**8.6.2. Classification of Low Specific Activity (LSA) Material and Surface Contaminated Objects (SCO)**

In the 1985 edition, extensive changes were made to what were formerly classified as low specific activity (LSA) material and low-level solid radioactive (LLS) material. These
classification were replaced by (a) low specific activity (LSA) material, defined into three groups (LSA-I, LSA-II and LSA-III); and (b) surface contaminated objects, defined into two groups (SCO-I and SCO-II). The classification of these materials and objects were specified initially (in the 1985 edition) \([15]\) in the definitions section of the regulations; but the classification has since been transitioned (in the 2012 edition) \([3]\) of the regulations (SSR-6) to paras 408 – 411, and 412 – 413, respectively.

In order to correct minor editorial errors that occurred in the 1985 edition, and to address changes of details that were identified and agreed to, the 1985 edition was supplemented with change pages in 1986 and 1988; and an amended version incorporating these editorial and minor changes was issued in 1990. However, none of these changes materially affected the classification of materials.

A brief explanation of the basis for the changes introduced in 1985 is provided in Safety Series No. 7 \([16]\) paras. E-131.1 – E-131.10 for LSA material, and in paras. E-144.1 – E-144.4 for SCO. This explanatory text describes typical types of materials or objects that may be included in each of the categories, but it does not provide a full explanation of the technical basis for the parametric thresholds for the LSA materials or surface contaminated objects.

The following fourth-level subsections provide further insight into the basis for the 1985 classification of LSA material and SCO.

**1979 Deliberations on Low Hazard Materials**

In a technical committee meeting convened in 1979 (TC-272 \([RBP026]\)), it was recommended that the concepts of LLS and LSA should be reviewed, taking into account (a) the need to ensure adequate shielding is retained for these materials to limit the external radiation dose rates following an accident, and (b) the specific activity might be increased to unsafe levels following an accident. In order to protect against excessive radiation levels following an accident, TC-272 recommended that the material without any shielding be constrained to having a radiation level at 3 meters no greater than 1 rem/h (which was recognized was the same requirement imposed on Type A shipments). The concern over the specific activity increasing to unsafe levels following an accident was ultimately addressed by the classification of LSA material into the three categories (LSA-I, LSA-II and LSA-III), where added protection was to be provided by imposing enhanced package requirements of IP-2 for LSA-II solids and of IP-2 or IP-3 for LSA-III solids depending upon the manner by which they were transported.

**1980 Deliberations on Low Hazard Materials**

A two-week Advisory Group meeting was convened at the Agency in September 1980, designated as AG-266 \([RBP025]\). One primary topic of this meeting dealt with multiple issues that had been raised concerning low level solids and low specific activity materials. The meeting dealt with many issues including the definition of LSA and LLS, packaging standards, liquids and slurries, and external radiation levels (pg 64 of \([RBP025]\)).

It was agreed, for example, that the use of the term “inherently safe” for these materials was inappropriate, and therefore the term "low risk" was recommended; i.e., "LSA/LLS materials should present a low risk in both normal transport and in accidents (including fires)". In order to solve then-existing problems, it was further agreed to rearrange the categories of LSA/LLS into two different categories:

1. materials in which the activity is generally dispersed throughout the contents and in which the concentration is limited to fairly low levels, and
2. slightly contaminated objects.

AG-266 specified that the “more hazardous the material the more packaging or transport controls one can impose”; and that in the event of an accident, the radiological consequences for these materials and objects should be no more adverse than "the radiological consequences of Type A or Type B accident conditions". 
The advisory group also concluded that it found no reason to change the value of $10^{-4} \text{A}_2/\text{g}$ for solids and gases as was then applied in para 121(d) of the 1973 Revised Edition (As Amended) of the Transport Regulations.\[13\] It recommended a factor of ten reduction in this value for liquids. Ultimately, this value was retained as the threshold for LSA-II material for solids and gases, but was reduced by an order of magnitude to $10^{-5} \text{A}_2/\text{g}$ for liquids (see e.g. para. 131(b) of the 1985 edition\[15\]).

AG-266 further considered the need to establish a radiation level limit for these materials, and recommended a value of 1 rem/h at 3 m (i.e. 10 mSv/h at 3 m). Ultimately this value was incorporated as a requirement in para. 422 of the 1985 edition.

In all of the AG-266 documentation, the decisions made ultimately led to what was ultimately incorporated into the 1985 edition of the Regulations, but no technical justification for the values chosen was provided. Most of those values resulted from recommendations made at earlier meetings and in submissions by Member States to the Agency.

However, it is noteworthy that AG-266 was still identifying the materials of concern as LSA/LLS materials.

1982 Deliberations on Low Hazard Materials – Defining as Low Specific Activity (LSA) Material and Surface Contaminated Objects (SCO)

A consultants meeting was convened in February 1982[See Endnote vii\[16\]] which, among other recommendations proposed identifying the materials as Low Specific Activity (LSA) materials and Surface Contaminated Objects (SCO). The next month, AG-365.2 [RBP042] formed a working group (WG) to specifically assess the proposed changes for low hazard materials. The report of that consultants meeting, and a document summarizing Member States comments on that report (identified as AG-365.2 Working Papers No. 9 and 6 respectively) served as the basis for discussion by the WG.

Working Group 3 of AG-365.2 [RBP052] proposed, and AG-365.2 agreed to the following:

- “Surface Contaminated Objects (SCO)” would replace an earlier proposed terminology of “Low Contaminated Objects”.
- Retain the unshielded radiation level requirement because “it is a property of the total quantity of the material placed in a simple package rather than a property of the SCO material itself”; and transfer the requirement the Transport Arrangements area of the Regulations.
- With minor changes, it was recommended that the definition of SCO-I be taken from the February 1982 consultants report. The Working Group recommended that:
  - SCO would only include an object of non-radioactive material having radioactive distributed on it surface;
  - Separate the limits for fixed contamination on SCO-II from the limits for non-fixed contamination. SCO-I and SCO-II were to have specifications for non-fixed and fixed contamination following a graded approach, with
    - the limits for non-fixed contamination being a factor of ten lower for SCO-I than for SCO-II; and
    - the limits for fixed contamination being a factor of 20 lower for SCO-I than for SCO-II.
- The definitions of the three types of LSA materials were taken, with minor changes, from the second draft of the regulations, considering the February 1982 consultants evaluation.

AG-365.2 further recommended that additional effort was needed to better define “non-fixed contamination”.

Radiation Level Limit for LSA Materials and SCO

During the early deliberations in preparation for issuing the 1985 edition of the regulations [15], an Advisory Group meeting (AG-144) was convened in December 1977 [RBP038]. At that
time, the materials and objects were being identified as LSA material and LLS, respectively, and it was deemed that LLS was a greater hazard than LSA material. AG-144 noted that imposing the impact and crush tests required for Type A packaging for LLS "clearly put LLS, as intended, between LSA and Type A in regard to risk to the public, but loss of shielding could occur. It is believed that an external radioactivity limit (i.e. dose limit) be added rather than requiring special testing". This appears to be the beginning justification for imposing a limiting value of 10 mSv/h at 3 m from the unshielded LLS which was ultimately established in the 1985 edition for both LSA materials and SCO.

For LSA material, special note is made in E-131.3 [16] that for adequate radiation protection, the amount of LSA materials in a package must be limited so that the unshielded radiation level does not exceed a limiting value of 10 mSv/h at 3 m as specified in para. 517 of SSR-6 [3]. This limit was introduced because the specific radiation level is a property of the quantity of material rather than a property of the material itself. Similarly, the extent to which uniform distribution within the LSA material is required varies depending upon each LSA category (see E-131.4) of Reference [16].

Further elaboration on this limit was provided in 2002 in TS-G-1.1 [17]. Specifically, the limit on the quantity of LSA material or SCO that can be carried in a single IP package is such that it shall be "so restricted that the external radiation level at 3 m from the unshielded material or object or collection of objects does not exceed 10 mSv/h (1 rem/h)".

The basis for this restriction is that, since "industrial packages used for transporting LSA material and SCO are not required to withstand transport accidents, a provision was initiated in the 1985 edition of the Regulations to limit package contents to the amount which would limit the external radiation level at 3 m from the unshielded material or object to 10 mSv/h. Geometrical changes of LSA material or SCO as a result of an accident are not expected to lead to a significant increase of this external radiation level. This limits accident consequences associated with LSA material and SCO to essentially the same level as that associated with Type A packages, where the A1 value is based on the unshielded contents of a Type A package creating radiation levels of 100 mSv/h at a distance of 1 m." [17]

Thus, in the event that the "unshielded material or object or collection of objects exceeds 10 mSv/h (1 rem/h)"), the material or objects must be transported in Type A or Type B packages, as appropriate, and in that event the materials or objects should not be classified as LSA material or SCO for the purposes of transport. [17]

Activity Limits for LSA Materials

[See Endnote viii]

Account was also taken by AG-266 [RPB025] deliberating on the possibility that, for liquids, the potential exists for volume reduction (e.g. through evaporation) which could result in an increase in specific activity. AG-266 therefore recommended, based on expert judgement, establishing for LSA-II materials a specific activity limit for liquids that is a factor of ten lower than for solids – e.g. see para. 409(b) of SSR-6 [3].

Leaching Test for LSA-III Material

In the 1973 edition of the Regulations [10] the definition of LSA material included a requirement in para. 121(e) for one type of material that read:

"Materials in which the activity is uniformly distributed and which, if reduced to the minimum volume under condition likely to be encountered in transport, such as dissolution in water with subsequent recrystallization, precipitation, evaporation, combustion, abrasion, etc., would have an average estimated specific activity of no more than 10^{-9}A$_2$/g."
In the 1985 edition of the Regulations [15], when LSA material was structured into three different categories (i.e. LSA-I, LSA-II, and LSA-III), a “leaching test” was imposed in para. 603 on LSA-III materials. That test was specified as follows:

“603. Solid material representing no less than the entire contents of the package shall be immersed for 7 days in water at ambient temperature. The volume of water to be used in the test shall be sufficient to ensure that at the end of the 7 day period the free volume of the unabsorbed and unreacted water remaining shall be at least 10% of the volume of the solid test sample itself. The water shall have an initial pH of 6-8 and a maximum conductivity of 1 mS/m (10 µmho/cm) at 20 °C. The total activity of the free volume of water shall be measured following the 7 day immersion of the test sample.”

This test is coupled in the 1985 edition of the Regulations with an acceptance requirement specified in para. 501 that reads: “501. LSA-III material shall be a solid of such nature that if the entire contents were subjected to the test specified in para. 603, the activity in the water would not exceed 0.1 A2.”

It is noteworthy that this test is identical to the test imposed on special form material in para. 612(a) of the 1985 edition of the Regulations; although the acceptance requirement for the special form material is different as specified in para. 503(c) of the 1985 edition of the Regulations.

[See Endnote ix *]

The leach requirement for LSA-III material has remained within the regulations, where in the 2012 edition (SSR-6) [3], the requirement is found in para 409(c)(ii), which reads “The radioactive material is relatively insoluble, or is intrinsically contained in a relatively insoluble matrix, so that, even under loss of packaging, the loss of radioactive material per package by leaching when placed in water for 7 days would not exceed 0.1A2”. The test specification for this is found in para. 703 of SSR-6, and reads essentially the same as para. 603 of the 1985 edition of the Regulations.

In a paper presented at PATRAM 2013, Nitsche, et. al. [RBP048], a study was reported evaluating the need for the leaching test. The results of the study that focused primarily on the “need and justification of the LSA-III leaching test which have often [been] questioned in the past”.

Nitsche, et. al. concluded that:

“For packages with LSA-III material it can be concluded that the achieved high level of transport safety is not connected with the currently required limited solubility of the material demonstrated by the leaching test but is resulting from the other required material properties. Therefore the performance of the leaching test for LSA-III material does not contribute to the requested safety level and can be omitted without decreasing the level of transport safety. This would help to simplify the Transport Regulations and to overcome difficulties regarding different interpretations and implementations of the leaching test in practice especially for radioactive waste.”

Contamination Limits for SCO

During the early deliberations in preparation for issuing the 1985 edition of the regulations [15], AG-144 was convened in December 1977 [RBP038]. Discussion occurred with respect to LLS, and it was agreed that “LLS is not intended for other than solids (e.g. single ‘block’) and leakage should not be a concern. It is, therefore, recommended that there be no leak test requirement (for LLS) beyond that contained in the specification for the material”. This appears to be the beginning justification for imposing limiting values on the fixed and non-fixed contamination of SCO, which were ultimately established in the 1985 edition for SCO.
The classification of an object as SCO-I depends in part on the amount of fixed contamination on that object. Specifically, it is required that “the fixed contamination on the accessible surface averaged over 300 cm² (or the area of the surface if less than 300 cm²) does not exceed 4 × 10⁴ Bq/cm² for beta and gamma emitters and low toxicity alpha emitters, or 4000 Bq/cm² for all other alpha emitters”.

The basis for that limit is summarized in para. E-144.2 [16]. It was assumed that since SCO-I could be shipped unpackaged, in an accident, SCO-I could be scraped by other packages or objects or through its own movement; and that scraping could result in 20 percent of the fixed contamination being freed. By constraining the amount of fixed contamination on an SCO-I to 4 X 10⁴ Bq/cm², through arguments presented in E-144.2 the level of safety for SCO-I would be equivalent to that of a Type A package.

The classification of an object as SCO-II imposes a limit on fixed contamination that is 20 times higher than for SCO-I. This results from considerations that SCO-II must be shipped in an industrial package, not unpackaged (see para. E-144.3 [16]).

AG-266 [RBP025] considered and adopted an earlier recommendation of TC-272 [RBP026] that, for surface contaminated objects, the radioactive contamination on such objects should have limits specified for both fixed and non-fixed contamination.

During the 1982 deliberations of AG-365.2 [RBP043, RBP052] it was noted that additional effort was needed to better define “non-fixed contamination”. Working Group 3 of AG-365.2 [RBP053] recommended the definitions for fixed and non-fixed contamination that were incorporated into paras 123 and 124 of the 1985 edition of the Regulations [15]. In addition, the Working Group recommended that activity levels per cm² below which an object would not be considered contaminated; values that were incorporated into para. 122 of the 1985 edition of the Regulations.

8.7. Enhancing the Radiation Protection Basis for LSA and SCO Classifications

In 1996, SAGSTRAM determined that the classification of LSA material and SCO did not necessarily have a clearly defined radiation protection basis. In 1997, the IAEA initiated a coordinated research programme to assist in developing a dose-based approach for the LSA and SCO requirements [RBP024]. Six countries participated in this CRP: Brazil, Canada, France, Germany, UK and USA. Each country carried out work that as outlined in agreements with the IAEA, with the work directed toward meeting the specific objectives of the agreement and also contributing to achieving the overall objective of the CRP. Following the establishment of the CRP in 1997, the first Research Co-ordination Meeting (RCM) was held in December 1997. The second RCM was held in March 1999, with the final RCM held in (date????). [see Endnote X8]

8.8. Classification of material in the 1996 edition of the Transport Regulations

In the 1996 edition of the Regulations [18], two additional material classifications were added: (1) low dispersible radioactive material (LDRM), and (2) uranium hexafluoride (UF6).

8.8.1. Updated Basis for A₁ and A₂ values in the 1996 edition of the Transport Regulations

As work went forward for issuing the 1996 edition of the Transport Regulations, Para. I.3 of the 2008 edition of TS-G-1.1 [19] reads that “In anticipation of the publication of the 1996 Edition of the Transport Regulations, the latest ICRP recommendations and data in the form of coefficients for dose per unit intake [dose coefficients] [I.8] were incorporated into the Q system by L. Bologna (ANPA, Italy), K. Eckerman (ORNL, USA) and S. Hughes (NRPB, UK). Their results served as a basis for updating the A₁ and A₂ values”. In this quote from TS-G-1.1 (2008), reference I.8 is the ICRP 1990 Recommendations of the ICRP, Publication 60 [20].
8.8.2. The LDRM Classification

The LDRM classification was added to allow large quantities of radioactive material that are not prone to airborne dispersion to be carried in Type B packages transported by air. The threshold for materials other than LDRM where transport by air is not allowed in Type B package is specified in para. 416 of the 1996 edition, which reads as follows:

“416. Type B(U) and Type B(M) packages, if transported by air, shall meet the requirements of para. 415 and shall not contain activities greater than the following:
(a) for low dispersible radioactive material - as authorized for the package design as specified in the certificate of approval,
(b) for special form radioactive material - 3000 $A_1$ or 100 000 $A_2$, whichever is the lower; or
(c) for all other radioactive material - 3000 $A_2$.”

For materials that can be demonstrated to satisfy the requirements for LDRM, they can be transported in a single Type B package in a quantity greater than the threshold specified in para. 416(a) and para. 416(b), subject to certification approval by the relevant competent authority(ies). However, for materials that cannot be demonstrated to satisfy the requirements for LDRM, if they are to be transported in a single package in a quantity greater than the threshold specified in para. 416(a) and para. 416(b), they must be transported in a package satisfying the more robust Type C package design requirements.

The requirements that must be satisfied for a material to be accepted as LDRM are specified in para. 605 of the 1996 edition of the Regulations [18].

As elaborated in Paras 225.2 and 225.3 of TS-G-1.1, 2008 edition [19], 225.2, the LDRM requirements were established to ensure that the material would not give rise to significant potential releases or exposures even when subjected to high velocity impact and severe thermal environments such as might be expected to occur during a severe aircraft accident. Under these conditions, only a limited fraction of the LDRM would be expected to become airborne. As a result, the potential for radiation exposure from inhalation of airborne material by persons in the vicinity of such an accident would be very limited. These criteria are derived so as to be consistent with other safety criteria.

[see Endnote xi]

8.8.3. The UF$_6$ Classification

In deliberations preceding the publication of the 1985 edition of the Regulations [15], AG-406 [RBP049] considered the subsidiary hazard posed by UF$_6$ "presented by the interaction of UF$_6$ with water or atmospheric water vapour". AG-406 recommended specific additional text be added to the Regulations to address this hazard. This additional text is found in the 1985 edition in paras 105, 208 and 407.

As concerns grew concerning the subsidiary hazard posed by UF$_6$, the specific text found in the 1985 edition of the Regulations was made generic to all subsidiary hazards, and further deliberations led to a specific classification of UF$_6$, which was added in the 1996 edition of the Regulations. These steps were taken in order to introduce special requirements for protecting people and the environment against the chemical, not the radiological, nature of the material. Although not specifically elaborated as a classification, requirements for testing of UF$_6$ packages beyond what had previously been required were introduced at that time.

Extensive deliberations led to the addition of UF$_6$-specific requirements. For example, a consultants services meeting convened in June 1995 addressed many issues relating to the safety of transport of UF$_6$ [21]. The consultants noted that the following issues were considered:

"(a) "the potential for chemotoxic threat of released UF6 and its reaction products with air and water vapour to workers, emergency responders, the public and the environment;"
"(b) "the types of rupture which can be expected from the different sizes of unprotected cylinders;"
"(c) "the nature and extent of the threat, and the level of protection currently provided compared to that provided by the 'UN Recommendations' for other similarly toxic and corrosive chemicals; and

"(d) "the inherent nature of large, thick-walled cylinders to provide resistance to the thermal exposure, and the nature of the releases of UF6 from large cylinders after significant thermal exposure which are believed to be significantly less violent than those experienced with liquefied flammable chemicals, often called a BLEVE (boiling liquid expanding vapour explosion)."

Consideration of these issues led to a series of recommendations that were later considered by IAEA regulatory review panels; consideration which ultimately led to the addition of UF6 as a specifically-classified material in the Regulations as summarized in Section 10.6.

8.9. Emphasis on classification of material in the 2009 and 2012 editions of the Transport Regulations

Essentially no changes were made with regard to the classification of material in the 2005 edition of the Regulations. However, in the 2009 edition of the Regulations [2], significant emphasis was placed on the classification of material. Specifically, the title of Section IV of the Regulations was changed to read "ACTIVITY LIMITS AND CLASSIFICATION"; where paras 408 through 419 elaborate the requirements for (a) LSA material, (b) SCO, (c) special form radioactive material, (d) LDRM, (e) fissile material, and (f) UF6. This change, which was carried forward into the 2012 edition of the Regulations [3] in paras 408 through 420, was introduced to clarify that materials (and also packages, see Chapter 9 of this document) are classified according to the above. It is noted that this then leads to the assignment of UN Numbers according to the classification of material and sometimes package used (see Chapter 7, Section 7.5 for more details on assignment of UN Numbers).

8.10. Technical basis for the requirements for a material classified as a special form radioactive material

As noted in Section 8.2, the classification of "special form radioactive material" was introduced in the 1964 edition of the Regulations in place of the earlier classification of 'massive non-friable solid' with a different set of limits. The requirements for special form radioactive material are found in the 2012 edition of the Regulations, SSR-6 [3] in paras 239, 415, 602-604 and 802. Specifically, para. 239 specifies that Special form radioactive material means "either an indisperisable solid radioactive material or a sealed capsule containing radioactive material". In turn, para. 415 specifies that "Radioactive material may be classified as special form radioactive material only if it meets the requirements of paras 602–604 and 802." The requirements in paras 602 – 604 were established with a view to ensuring that a Type A package containing special form radioactive material would not release or disperse its radioactive contents during a severe accident, by leakage from the sealed capsule or by dispersion/leaching of the radioactive material itself, even though the packaging may be destroyed. This minimizes the predicted hazards from inhalation or ingestion of, or from contamination by, the radioactive material.

An extensive discussion of the tests "for encapsulation of special form material" was documented in the mid-1964 Panel Meeting [RBP036]. The deliberations focused on looking at each critical property of the radioactive material in "massive solid non friable form would be looked at to see if a parallel test was required on a capsule". Paras 602 through 604 of SSR-6 [3] specify requirements for special form radioactive material, referencing in turn tests specified in paras 704-711 of SSR 6. The specific requirements include (a) a minimum dimension, (b) and impact test, (c) a percussion test, (d) a bending test, (e) a heat test, and (f) a leaching test. The Panel also recommended that "a contents leakage test should follow each test done on a sample".

The requirement that the special form radioactive material would not break or shatter and would not release or disperse its radioactive contents by leakage from the sealed capsule or by dispersion/leaching of the radioactive material itself during a severe accident, even though the
packaging may be destroyed was imposed with a view to minimizing the predicted hazards from inhalation or ingestion of, or from contamination by, the radioactive material. The tests in para. 603 of the Regulations were established pragmatically to address what a special form radioactive material in a Type A package might experience in a severe accident condition.

The impact test is the drop of a specimen representing the special form material from a height of 9 m as represented by the 9 m drop test for packages (see Section 10.4.3 for the technical basis for this test).

In 1979, TC-272 \[RBP026\] considered the adequacy of the special form tests. TC-272 determined that “There were no additional environmental or accident forces which could be identified as being representative of transport operations that were not adequately covered by the existing test requirements”.

8.10.1. Limiting Dimension for Special Form Radioactive Material
Para. 602 of SSR-6 specifies that special form material shall have at least one dimension of not less than 5 mm. This limit was established during the mid-1964 Panel Meeting \[RBP036\], noting that it “was referred to as being relevant not only to radioactive materials special form but also encapsulated special form”; and as a result it was agreed to “include suitable statement on minimum dimension”, and that a specification of a test for this requirement was not needed.

Thus, in developing the requirements for special form radioactive material, it was agreed that the material must be of a reasonable size to enable it to be easily salvaged or found after an incident or loss; hence the restriction on minimum size. Para. 602.1 of TS-G-1.1 \[19\] notes that the figure of 5 mm as specified in para. 602 of SSR-6 is arbitrary but practical and reasonable, it was specified considering the type of material normally classified as special form radioactive material.

8.10.2. Percussion Test for Special Form Radioactive Material
The mid-1964 Panel Meeting \[RBP036\] deliberated on what it then called the impact test (as opposed to the drop test). The placement of the specimen on a lead base was elaborated, noting that it should be placed on a solid surface. What is termed SSR-6 as the specimen being “struck by the flat face of a mild steel bar” was described as a hammer in the Panel’s discussion. Key issues that needed to be addressed were “the hammer shape and size”, and “whether the capsule was to be subjected to a simple impact force or impact associated with shear”. The Panel agreed that “the hammer should provide simple impact without shear”.

In the 1985 Edition of the Regulations, an alternative to the percussion test specified in para. 608 was added in para. 611, allowing the use of the Class 4 impact test specified in ISO2919-1980(E). This resulted from a recommendation by AG-365.2 \[RBP042\]. This inclusion was justified by AG-365.2 on the basis of consistency with the ISO and for clarification. That alternative test is found in SSR-6 \[3\] at para. 709.

8.10.3. Bending Test for Special Form Radioactive Material
[see EndNote xii]

8.10.4. Heat Test for Special Form Radioactive Material
The mid-1964 Panel Meeting \[RBP036\] noted that an earlier agreement specified that “the temperature to which the capsule should be heated was 800 °C: the test was designed to test the materials and brazing/welding of the capsule: the capsule should be complete for transport (i.e. with contents simulating the material it would carry)”. The panel agreed that liquids and gases inside a special form capsule were not appropriate unless “the contents of the sample capsule …. as tested should simulate as closely as possible the radioactive materials to be carried”.

Discussions led to the conclusion that:
“The sample capsule shall be heated to 800 °C and shall be held at that temperature for 10 minutes. There shall be no melting, subliming or ignition of any capsule material and when cold it shall be subjected to the contents leakage test.”

In the 1985 Edition of the Regulations, an alternative to the heat test specified in para. 610 was added in para. 611, allowing the use of the Class 6 temperature test specified in ISO2919-1980(E). This resulted from a recommendation by AG-365.2 [RP042]. This inclusion was justified by AG-365.2 on the basis of consistency with the ISO and for clarification. That alternative test is found in SSR-6 [3] at para. 709.

8.10.5. Leaching Test for Special Form Radioactive Material
[See EndNote xiii xiii]

8.11. Technical basis for the requirements for a material classified as a fissile material
Chapter 11 of this Technical Basis Document provides extensive details on the technical basis for the classification of fissile material and fissile excepted material.

References for Chapter 8


Management, San Francisco (2013). \{RBP048\}

9. CLASSIFICATION OF PACKAGES

Initially, in the first editions of the Regulations, five basic types of packagings were specified. The specifications for the packages, which were graded according to the risk posed by the contents, were:

(a) **Exempt and Empty Packages**: packages that were exempt from requirements, containing small quantities of radioactive material and having a minimal set of package design requirements;

(b) **Strong, leak-proof packages**: packages for materials that were judged to be "inherently safe", e.g. low specific activity materials such as radioactive ores, where the regulations provided for their carriage in "bulk or in strong, leak-proof package;"

(c) **Type A packagings**: containing larger, but limited, quantities of radioactive material while being designed to be resistant to normal conditions of transport;

(d) **Type B packagings**: containing larger quantities of radioactive material while being resistant to both normal and accident conditions; and

(e) **Large radioactive source packages**: containing large radioactive sources, where the activity exceeded the limits for Type B packages, and where additional requirements were imposed on materials compatibility, containment system pressure and pressure relief systems, protection of valves, and proper design of attachments and tiedowns [1].

As the Regulations have matured, more package types were introduced, the classification titles of some have changed and expanded, while the two categories with the highest risk contents as defined in the early regulations (i.e. Type B packagings and packages containing large radioactive sources) were coalesced in the 1973 edition of the Regulations [2] into a single category. Beginning in 1996 [3], ten different package classifications were specified.

The extension and changes to the number and classification of package types involved extensive deliberations by expert panels concerning the nature of the contents and the need to provide flexibility in the Regulations, while concurrently ensuring adequate protection by each package type. The history of specification of classes of packages, as it has grown with time, is illustrated in Table 9-1.

Of the package classifications existing in the Transport Regulations since the 1996 edition (see Table 9-1), all but the H(U) and H(M) package classifications are specified in para. 231 of SSR-6 [4], whereas the H(U) and H(M) package classifications are established by way of para. 832 of the SSR-6.

The "classifying" of packages was not detailed specifically in the early edition of the Regulations; whereas, beginning in 2009, Section IV of the Regulations specifically called for classification of packages (see paras 421-434 of TS-R-1 (2009) [5] and paras 421-433 of SSR-6 [6]). However, SSR-6 does not address all of the package types that exist; it only addresses (1) excepted packages, including empty packages; (2) Type A packages; (3) Type B(U) packages; (4) Type B(M) packages; and (5) Type C packages.

The technical bases for these classifications of packages are discussed in detail in Chapter 10 and Appendix 3 of this Technical Basis document. The technical bases for the remaining package classifications – i.e. industrial packages and packages for UF6 – are also discussed in detail in Chapter 10 and Appendix 3.
<table>
<thead>
<tr>
<th>Package Type</th>
<th>Edition of the Transport Regulations</th>
</tr>
</thead>
<tbody>
<tr>
<td>Exempt Package / Empty Package</td>
<td>X</td>
</tr>
<tr>
<td>Excepted Package / Empty Package</td>
<td></td>
</tr>
<tr>
<td>Strong, leak-proof package</td>
<td></td>
</tr>
<tr>
<td>Strong industrial package</td>
<td></td>
</tr>
<tr>
<td>Industrial Package Type 1 (IP-1)</td>
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<tr>
<td>Industrial Package Type 2 (IP-2)</td>
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<tr>
<td>Industrial Package Type 3 (IP-3)</td>
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<tr>
<td>Type A</td>
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</tr>
<tr>
<td>Type B</td>
<td>X</td>
</tr>
<tr>
<td>Large radioactive source packages</td>
<td></td>
</tr>
<tr>
<td>Type B(U)</td>
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<tr>
<td>Type B(M)</td>
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<tr>
<td>Type C</td>
<td></td>
</tr>
<tr>
<td>H(U) (specific approval specified for UF₆ contents)</td>
<td></td>
</tr>
<tr>
<td>H(M) (specific approval specified for UF₆ contents)</td>
<td></td>
</tr>
</tbody>
</table>

* Note: This table does not include the specific package types that address fissile material. The types of packages for fissile material are Type AF, Type B(U)F, Type B(M)F, and Type CF. The requirements for these are discussed in Chapter 11 of this Technical Basis document.

**References for Chapter 9**


{RP022}
10. PACKAGE DESIGN AND TESTING

This chapter provides insights into the technical bases for the package design and testing requirements that are embodied in the Regulations. A significant amount of documentation is available on the technical basis for establishing the requirements and also for later justifying their continued use. The following looks at, and provides where possible the historical background and insight into the requirements ranging from the minimal set for routine conditions of transport to the much more robust set of tests simulating accident conditions of transport and concluding with the most demanding requirements that are imposed on Type C packages and some fissile material packages.

The package design and testing requirements follow the graded approach as discussed in Appendix 3.

Over the years, the packaging and transport of plutonium has been singled out as a unique material, requiring special packaging design and test requirements and operational requirements for plutonium. Some countries imposed such requirements, and have modified them over the years. It is worth noting that AG-126 [RBP033] addressed this issue in 1977, stating that

“The working party discussed the whole problem of transport of plutonium and especially the point of whether such transport is actually a special problem.

“The working party concluded there are no scientific reasons to consider plutonium as a special material. Nevertheless, a careful comparison of risk assessment studies of transport of plutonium and transport of other radioactive materials would be useful.”

Specific efforts with respect to plutonium are addressed in multiple section of Chapter 10 as well as in Appendices 3 and 4.

10.1. Technical basis for routine conditions of transport for packages

The following provides some insight into the technical bases for the routine conditions of transport.

The environments relating to routine conditions of transport for which package designers might need to consider include the following issues: (a) acceleration, (b) vibration, (c) vibration resonance, and (d) static and dynamic stresses. In addition, the package designers need to establish relevant ambient temperatures and ambient pressures. The Regulations apply a number of criteria to packaging performance standards with respect to “routine conditions of transport”, but generally do not specify any package testing requirement associated with such conditions.

The performance requirements in SSR-6 [1] relative to routine conditions of transport are (emphasis is placed on the “routine conditions of transport” phrase using underlining in the text):

- Para. 508 – states that “The non-fixed contamination on the external surfaces of any package shall be kept as low as practicable and, under routine conditions of transport, shall not exceed” specified limits.
- Para. 520(a) – For LSA-I and SCO-I, this paragraph specifies that “All unpackaged material other than ores containing only naturally occurring radionuclides shall be transported in such a manner that under routine conditions of transport there will be no escape of the radioactive contents from the conveyance nor will there be any loss of shielding”.
- Para. 566(b) – states that “The radiation level under routine conditions of transport shall not exceed” specified limits at the surface and 2 m from the surface of the conveyance.
- Para. 573(a) – specifies that a vehicle under exclusive use shall have an enclosure “that, during routine conditions of transport, prevents the access of unauthorized persons to the interior of the enclosure”; and that the package or overpack shall be made secure “so that its position within the vehicle enclosure remains fixed during routine conditions of transport”.

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• Para 613 – requires that "The package shall be capable of withstanding the effects of any acceleration, vibration or vibration resonance which may arise under routine conditions of transport without any deterioration in the effectiveness of the closing devices on the various receptacles or in the integrity of the package as a whole".
• Para 616 – states that "The design of the package shall take into account ambient temperatures and pressures that are likely to be encountered in routine conditions of transport".
• Para. 617 – states that "A package shall be so designed that it provides sufficient shielding to ensure that, under routine conditions of transport and with the maximum radioactive contents that the package is designed to contain, the radiation level at any point on the external surface of the package would not exceed the values specified in paras 516, 527 and 528, as applicable, with account taken of paras 566(b) and 573".
• Paras 627(c) and 628(c) – require that, for portable tanks and tanks, that they "are designed so that any additional shielding that is provided shall be capable of withstanding the static and dynamic stresses resulting from handling and routine conditions of transport and of preventing more than a 20% increase in the maximum radiation level at any external surface of the" portable tanks or tanks, as applicable.
• Para. 629(c) – requires that, for freight containers, they "shall be designed such that if subjected to the tests prescribed in that document and to the accelerations occurring during routine conditions of transport they would prevent" specified limits on loss of contents and increase in radiation levels.
• Para. 682 (a) – specifies that fissile material packages would remain subcritical consistent with "Routine conditions of transport (incident free)".

The technical basis for routine conditions of transport for accelerations, vibration, vibration resonance, static stresses, dynamic stresses, ambient temperatures, and ambient pressures are as follows.

### 10.1.1. Potential routine condition of transport requirements for accelerations

The issue of accelerations during routine conditions of transport were addressed in Appendix IV of the 2008 Edition of TS-G-1.1 [2], and in Appendix IV of the new guidance document replacing the 2008 Edition of TS-G-1.1 [3]. The guidance in these appendixes addresses potential sources of acceleration (and deceleration), noting that such events as minor impacts, rail shunting, heavy seas and turbulence or rough landings may (or may not) be viewed as routine. It then addresses the type of retention systems (often known as tiedown systems) that could be used in transport, and then suggests a technical basis for addressing accelerations and designing the needed retention systems. The data shown in Tables IV.1 (accelerations for package retention systems) and IV.2 (acceleration factors for package retention system design for specific packages) of these two IAEA guidance documents were developed considering information from many of the references cited therein, including specifically References [4–10], which provide a sound basis for addressing these issues.

Reference [10], which was published in 1990, has since been replaced with Reference [RBP005].

This document (in both German and English) provides, for road vehicles, methods for calculating tie down forces. The inertial forces generated during road transport are specified as being 0.5 g longitudinal for acceleration, 0.8 g longitudinal for deceleration, and 0.5 g lateral. No vertical acceleration is specified other than that of gravity.

The UK Transport Container Standardisation Committee (TCSC) has provided a detailed guide, TCSC 1006, on accelerations [4] which also serves as a potential technical basis for defining acceleration values that can be applied to package designs. Included in this document are tables of acceleration factors for the four modes of transport, where the values range from 0.5 g to 3 g, depending upon the mode of transport and the direction of the resultant force relative to the motion of the conveyance. In a separate table, the g-values provided in Appendix IV of TS-G-1.1...
are provided for comparison, where the values range from 1 g to 6 g. For example, the US NRC specifies in its regulations (in §71.45 of [6]) that

“If there is a system of tie-down devices that is a structural part of the package, the system must be capable of withstanding, without generating stress in any material of the package in excess of its yield strength, a static force applied to the center of gravity of the package having a vertical component of 2 times the weight of the package with its contents, a horizontal component along the direction in which the vehicle travels of 10 times the weight of the package with its contents, and a horizontal component in the transverse direction of 5 times the weight of the package with its contents.”

TCSC 1006 also provides tables showing the frequency of occurrence of a range of acceleration values; these are given in amplitude of the g-forces in the three directions, and the total number of cycles that can be expected per 1000 hours for two different types of packages, a flask and an ISO freight container.

A comprehensive review of package retention systems (tiedowns) was provided by Gonzales [11] in 1988. Gonzales summarized the existing (i.e. in the late 1970s) recommended acceleration values for tiedown design emanating from the IAEA and four United States entities. The acceleration values ranged from 0.5 g to 7.5 g (see page 3 of Reference [11]). Gonzales included the detailed appendixes that provided (a) the documentation of a proposed American National Standards Institute (ANSI) N14.2 standard on tiedowns for truck transport of radioactive materials, (b) an analysis of tiedowns using both U.S. Department of Transportation (DOT) and ANSI design criteria, and (c) a French guide for “Sizing of Type B Package Stowage Systems on the Basis of Criteria Related to Hypothetical Traffic Accident Conditions.

10.1.2. Potential routine conditions of transport requirements for vibration, vibration resonance, static stresses, and dynamic stresses

Because the shock and vibration requirements for normal transport are not quantified in the Transport Regulations, it was determined that package designers needed to understand the shock and vibration environments that different packages could experience during different modes of transport. A series of studies were performed at Sandia National Laboratories during the late 1970s and early 1980s to quantify these environments for large packages transported by road and rail [12-16]. These studies showed that:

- the vibration environments could result in maximum accelerations
  - for truck transport ranging from 0.19 to 2.0 with a frequency range of 0 to 1900 Hz, and
  - for rail transport ranging from approximately 0.2 to 0.4 with a frequency range of 0 to 350 Hz, and 0.1 to 0.52 g, with a frequency range of 0 to 750 Hz,
- the shock environment could result in peak accelerations superimposed on vibration
  - for truck transport ranging from 1.6 to 7.0 g with a pulse duration ranging from 20 to 83 ms,
- for rail transport, 4.7 g with a pulse duration of 14 ms, and
- for rail coupling as performed in the United States, 2.5 to 39 g with pulse durations ranging from 8 to 50 ms.
- TCSC 1006 [4] guide on the securing/retention of radioactive material packages on conveyances establishes acceleration factors in terms of “g” values for assessing fatigue for three modes of transport (road, rail and sea, see Table 3 of TCSC 1006), where the “g” values range from 0.2 to 0.65 depending upon the mode and the direction of resultant force (longitudinal, lateral, or vertical); the lower values of “g” occur in the longitudinal direction, mid-range values occur in the lateral direction, and the highest range of values occur in the vertical direction. The appendix in TCSC 1006 also provides methodology for assessing the fatigue strength of tiedown systems. Tables 5 and 6 in this appendix illustrate five load cases for two examples (a flask and an ISO 20-foot freight container) for static stresses in the packaging tiedown system; where the total number of cycles per 1000 hours ranges from zero to as high as 11 X 10⁶.
The TCSC guide (TCSC 1042) on the design of transport packaging for radioactive material notes [17] that fatigue failure of tiedown systems that arises from vibration and the natural frequency of the vehicle suspension needs to be considered. For evaluating the effects of vibration, it suggests that:

“Packages may be tested using, for instance, a transit test which simulates the likely worst case scenario that the package will encounter in practice. If a route is chosen that involves many van changes then the severe conditions often experienced during handling can be reproduced.

“The higher frequency vibrations often involved with air transport may be simulated using vibration test equipment available at many commercial test laboratories. An alternative is simply to ship out test packages on air routes which simulate the conditions the design is likely to be subjected to, repeating such shipments; possibly on the same package, until confidence is achieved in its performance.”


10.1.3. Relationship between acceleration, vibration and packaging tie downs

Para. 638 of SSR-6 [1] specifies that “tie-down attachments shall be so designed that, under normal and accident conditions of transport, the forces in those attachments shall not impair the ability of the package to meet the requirements of these Regulations”.

This technical basis for this requirement derives in part from deliberations that occurred during the March 1965 Panel Meeting [RBP037]. The Panel noted that “There was some discussion on the vehicle ‘end’ of the tie-down system; unless this was part of the assembly of packaging and vehicle to meet requirements (which would remain one unit during the transport operation) any attempt to provide for tie down would be difficult since during international transport the vehicle could change e.g. road vehicle, train and ship”. The Panel agreed that “any statement in respect to the ‘vehicle end’ of the tie down system should be in advisory section”.

Although the panel considered the benefit that might be derived from the deformation of the vehicle, but such provisions were not ultimately accepted in the regulations. For example, Annex II of the 1967 Edition of the Regulations [22] discusses lifting of packages and the attachments used (paras I-1.1 through I-1.5), and para. I-2.2 specifically requires that the integrity of the package would not be degraded by acceleration and vibration. Similarly, para. III-1.5 specifies that “Any tie-down attachments on the package shall be so designed that forces in those attachments shall not impair, during transport, the ability of the package to meet the requirements of the regulations”. This is essentially the same requirement specified in para. 638 of SSR-6.

The 1973 edition of the regulations [23] specified in para. 526 that “Consignments of radioactive material shall be securely stowed”. During deliberations by an advisory group convened in December 1977 (AG-144) [RBP038], data on accidents as a result of human error, tiedown failures and non-secured packages were reviewed. It was noted that the regulations at that time did not specify design criteria for tiedown devices. AG-144 concluded “that the submitted data contains sufficient evidence to show that the absence of adequate package tiedowns and stowage can result in accidents”; recommended that the text of para. 526 be modified to specifically require adequate tiedowns for consignments. It similarly recommended that additional text be added to the advisory material addressing the need for adequate tiedowns; and that “examples of suitable tiedown principles and methods should be added to the advisory material.

As a result of these recommendations, the requirement for tiedowns was added (as para. 527 in the 1985 edition of the regulations [24]) which, in para. 638 of SSR-6 [1] now reads as follows:

“638. Any tie-down attachments on the package shall be so designed that, under normal and accident conditions of transport, the forces in those attachments shall not impair the ability of the package to meet the requirements of these Regulations.”
Similarly, the Third Edition of Safety Series No. 37 [101] included three paragraphs (paras A-527.1 through A-527.3) advising on proper tie-downs, and an appendix (Appendix VII) specifying acceleration values and calculation methods for package tie-down forces.

10.1.4. Potential routine conditions of transport tests for ambient temperatures and pressures

Para. 616 of SSR-6 [1] requires that the design of the package take into account ambient temperatures and pressures that are likely to be encountered in routine conditions of transport.

Routine temperature and pressure conditions for surface transport

Para. 615.1 of TS-G-1.1 [2] provides guidance that changes in ambient temperature and pressure should be considered in ensuring the safety features of the package are not impaired. Para. 615.2 of TS-G-1.1 elaborates as follows (for surface modes of transport):

"An ambient pressure range of 60–101 kPa and an ambient temperature range of −40°C to 38°C are generally acceptable for surface modes of transport. For surface movements of excepted package(s), industrial packages Types IP-1, IP-2 and IP-3, and Type B(M) packages solely within a specified country or solely between specified countries, ambient temperature and pressure conditions other than these may be assumed providing they can be justified and that adequate controls are in place to limit the use of the package(s) to the countries concerned."

Thus, for a country where excessive cold or hot temperature environments, or surface transport routes where excessively high elevations which can result in lower ambient pressures are possible, the ambient pressure and temperature range for materials and for design of containment systems may need to account for higher or lower values in these parameters. For example, if the transport is to occur in a country where ambient temperatures can exceed 38°C, such as in many desert climes, then the potential for such higher ambient temperatures need to be considered as part of the routine conditions of transport.

For domestic transport, alternate values of ambient temperature and pressure ranges may be assumed if they can be justified by the package designer with its competent authority, or are specified differently by the competent authority. For example:

- for domestic transport in the UK, TCSC 1042 [17] reads that “the package must be designed for an ambient temperature range of -40°C to +38°C unless the Competent Authority specifies otherwise”; and
- for domestic transport in the USA, in specifying load combinations for the structural analysis of shipping casks for radioactive material, NRC Regulatory Guide 7.8 [18] reads that “...this guide presents a range of ambient temperatures from -20°F (-29°C) to 100°F (38°C) as part of the initial conditions. In the contiguous United States, there is a 99.7 percent probability that any hourly temperature reading will fall within this range.”

Routine temperature and pressure conditions for transport by air

Paras 620 and 621 of SSR-6 specifies that, for air transport, ambient temperatures of −40°C to +55°C, and ambient pressures leading to a pressure differential between internal pressure (of not less than maximum normal operating pressure) and external pressure of 95 kPa.

As is noted in para. 618.1 of TS-G-1.1 [2], the ambient temperature range specified for air transport is established to satisfy the carriage requirements of the International Civil Aviation Organization (ICAO), where the temperature range accounts for those values that experience has shown can be attained in commercial aircraft service; and have been documented as a requirement in the ICAO Technical Instructions [19]. In keeping with the guidelines of ICAO, para. 618.2 notes that “In designing the containment, the effect of ambient temperature extremes on resultant surface temperatures, contents, thermal stresses and pressure variations should be considered to ensure containment of the radioactive material”.

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As is noted in para. 619.1 of TS-G-1.1, the pressure differential requirement for air transport was adapted from a similar requirement in the ICAO Technical Instructions [19], and is intended to account for those values that experience has shown can be attained in commercial aircraft service; and have been documented as a requirement in the ICAO Technical Instructions. The reason for this more demanding requirement for air transport is seen in para. 619.2 of TS-G-1.1:

“Pressure reductions due to altitude will be encountered during flight (see para. 578.1). The pressure differential that occurs at an increased altitude should be taken into account in the packaging design. The pressure differential of 95 kPa plus the MNOP (see paras 228.1–228.3) is the pressure differential to be accommodated, without leakage, by the package designer. This design specification results from a consideration of aircraft depressurization at a maximum civil aviation flight altitude together with any pressure already inside the package, with a safety margin.”

Further, para. 578.1 of TS-G-1.1 notes that:

“There may be a considerable reduction in ambient air pressure at the cruising altitudes of aircraft. This is partially compensated for by a pressurization system, but that system is never considered to be 100% reliable.”

Thus, for domestic air transport, alternate values of ambient temperature and pressure ranges may be assumed if they can be justified by the package designer with its competent authority. For example, TCSC 1089 [20] reads, for excepted packages transported in the UK that “ambient temperature and pressure conditions other than these may be assumed providing they can be justified and that adequate controls are in place to limit the use of the package(s) to the assumed conditions”. TCSC 1089 also states that “In designing the containment, the effect of ambient temperature extremes on resultant surface temperatures, contents, thermal stresses and pressure variations should be considered to ensure containment of the radioactive material”.

10.2. Technical basis for the target for mechanical testing of packages

Although the terminology “unyielding target” is not used in the Transport Regulations, such a target is required for the mechanical tests specified in paras 705, 722, 725, 727, 735 and 737 of the 2012 edition of the Regulations [1]. The use of such a target relates to mechanical tests for normal conditions of transport, accident conditions of transport and the test for Type C packages. The terminology “unyielding target” and “essentially unyielding surface” is elaborated in paras 717.1–717.2 of TS-G-1.1 [2, 3], where minimum target characteristics are recommended in para. 717.2.

The regulatory requirement is not specific as to what constitutes an adequate target for drop testing, it provides a generic, qualitative requirement that the target “shall be a flat, horizontal surface of such a character that any increase in its resistance to displacement or deformation upon impact by the specimen would not significantly increase damage to the specimen”.

In establishing the basis for a drop test a drop from 9 m onto a “hard surface” was discussed by Grange on page 93 of Safety Series No. 7 [21]. It states that, after considering various direct and glancing impacts at velocities between 50 to 80 km/h, “it is felt that a direct drop of 30 ft. (9.2 m) on a hard surface which is representative of glancing blows of 60 mph (100 km/hr) at 60° to the normal or 45 mph (75 km/hr) at 45°, is a reasonable test”.

In 1967, the Transport Regulations [22] specified that the drop would be onto a target that was a “rigid, smooth, flat and horizontal surface”. It then elaborated by providing an example as follows:

“One example of such a target is the upper surface of a block of material of sufficient mass to absorb all shocks, when any package is dropped upon it, without appreciable movement. The upper surface of the block may be covered with a steel plate to protect that surface.”

In the 1973 edition of the Transport Regulations [23], the requirement was modified to read:
“The target shall be a flat, horizontal surface of such a character that any increase in its resistance to displacement or deformation upon impact by the specimen would not significantly increase the damage to the specimen”.

This regulatory terminology for the target for drop tests has remained unchanged through the 2012 edition of the Transport Regulations [1].

The term “unyielding target” was first used in the 1985 edition of the guidance document Safety Series No. 37 [24]. The specification that has been used since then (e.g. see paragraph 717.2 of TS-G-1.1 [2]), was proposed by Appleton and Servant [25] and Fairbairn and George [25], who described the example target as comprising “a concrete block of mass at least ten times that of the sample package, faced by a mild steel plate at least 1.25 cm thick in intimate contact with it, the whole being set on firm soil”.

This example of an acceptable target seems to have first appeared during deliberations by a panel of experts in early 1964, where they elaborated on what could be used as a model for the target, see page 26 of Ref. [RBP001]. The results of the deliberations was that “One example of a surface of this type is considered to be that provided by a steel plate at the upper surface of a block of concrete of mass at least 10 times that of a sample package that is being tested. The block should be set on firm soil and the steel plate on its upper surface should be at least 1.25 cm thick, and wet floated so as to be in intimate contact with the concrete.”

Fairbairn and George then elaborated on this, discussing, for example, the dimensions of packages and target, using the American Standards Materials Handbook [27]. This document has been withdrawn in 1993 and replaced by ASTM D5276-98(2009) Standard Test Method for Drop Test of Loaded Containers by Free Fall.

The example target described by Fairbairn and George in 1966 is essentially that provided today (in paragraph 717.2 of TS-G-1.1 [2, 3]), although (a) the steel thickness has been increased from 1.25 cm to 4 cm, and (b) mass has been increased from ten times to 100 times the mass of the specimen being tested. These changes are based upon experience. Since this is guidance, rather than a regulatory requirement, the final judgment with regard to the adequacy of the target used in a given package test is with the competent authority responsible for approving the specific package design being tested.

As summarized by Pope [28, 29], considerable attention has been paid as the Regulations matured to specifying the target to ensure “minimum loss of energy of the falling package due to mass movement or damage to the target surface”; i.e., the intent is to strive for the condition where all of the mechanical energy is transferred to the packaging components. Typical targets used for testing packages can be found in References [30–32]. Reference [30] contains details on targets that were available in 1983 in France, Germany, Italy, Japan, the UK and the USA.

Reference [31] contains details on targets that were available in 1991 in Argentina, Canada, France, Germany, Italy, Japan, the UK and the USA. Reference [32] contains details on targets that were available in 2001 in Australia, the Czech Republic, France, Germany Romania, Russia, the UK and the USA.

10.2.1. Assessment of the Adequacy of the Unyielding Target

In 1977, AG-126 [RBP033] recommended that “drop tests onto actual targets (e.g. soil) are recommended to demonstrate the relative severity of the 9 m impact test to the public”. A significant portion of this subsection, as well as portions of Appendices 3 and 4 address the efforts that had already been undertaken when this recommendation was made, and the many efforts that were undertaken following this recommendation.

Multiple studies have been performed comparing the structural response of packages impacting unyielding and more realistic targets at various speeds, including speeds greater than that resulting from the 9 m drop test. The primary result of such studies was the comparison between the damage from the “real life” impacts with that damage that was determined or could
be expected to occur with the same package impacting the unyielding target at the 9 m drop velocity. The following references \[33 – 46, RBP015, RBP020\] provide an extensive background for the adequacy of the technical basis of the target specified in SSR-6 and TS-G-1.1, as well as the overall concept of imposing the mechanical tests with package orientations on impact “so as to suffer maximum damage” (e.g. see para. 727 of SSR-6).

In 1980, McClure, et. al. \[33\] studied the results that were then available of various drop tests of packages onto essentially unyielding targets and yielding targets. It was concluded that “the current IAEA regulatory drop (impact) test provides a damage equivalence to severe impact-like transport accidents”. This conclusion was directly related to the hardness of the target, velocity of impact, and structural hardness of the packages tested, where having the impact target unyielding provided a very severe mechanical stress environment for the package. In part, the results presented by McClure, et. al. were based on work previously reported by Shappert, et. al. \[34\] and by Bonzon and Schamaun \[35\].

Blythe, et. al. \[36\] studied the effect of different targets on the behaviour of flasks impacting various targets at velocities ranging from 13.3 m/s (30 mph) to 26.6 m/s (60 mph). The targets studied were (a) the IAEA regulatory target (unyielding), (b) granite, (c) hard limestone, (d) Portland limestone, (e) hard concrete, (f) medium concrete, (g) soft concrete, and (h) engineered bricks. The authors concluded that for the specimens tested “the harder rocks exhibit no reduction in impactor damage when compared with the IAEA target. However, engineered targets (such as concrete and bricks) appear to exhibit significant reductions in impactor damage.”

Gablin, et. al. \[37\] provided information on two tests of a relatively soft package impacting two different targets with a drop height of 9 m. One target was essentially unyielding, whereas the other was described by the authors as “semi-yielding”, consisting of three steel plates 22.2-mm thick, placed on a 203-mm thick road bed constructed over an existing railroad track spur line. Based on the results of the two tests, the authors concluded that the semi-yielding target produced much less damage to the package, and therefore “to obtain a perfectly rigid target, a very large amount of mass and a very rigid material are necessary so that essentially all the energy goes into container deformation”.

Gonzales \[RBP021\] provided the results of an experimental study looking at the effects of different targets and impact velocities on the damage to a model half-scale truck transportation cask, and the penetration of the model into yielding targets. The study considered impact velocities ranging from 30 to 74 ft/s (9.1 to 22.6 m/s) onto (a) an unyielding target, (b) an 18-in thick (46-cm thick) concrete slab representing an airport runway, (c) a 9-in thick (23-cm thick) concrete slab representing a highway, (d) native desert soil, and (e) un-compacted soil. The results of these tests are summarized in Table 10-1.

Gonzales’ data clearly illustrate that the impact at 44 ft (equivalent to a 9-m drop) onto the unyielding target results in greater damage to the model cask than in impacts onto yielding targets at greater velocities.

Ammerman \[RBP015\] undertook a “mathematically rigorous method is developed for relating impacts with yielding targets to lower velocity impacts with unyielding targets. The method correctly models the mechanics of the impact and the conversion of kinetic energy to strain energy. An important result shown by the example problem is that apparent target hardness depends on the stiffness of the impacting package. For a cask with impact limiters a 26.8 m/s impact onto hard soil results in equivalent forces as a 13.9 m/s impact onto an unyielding target. For the same cask without the impact limiters a 26.8 m/s impact onto hard soil is equivalent to a 1.74 m/s impact onto an unyielding target. This is one reason why non-technical members of the public often have difficulty realizing the severity of the regulatory impact. For most people, objects such as trucks and bridge columns appear to be very hard, but to many radioactive material shipping packages these objects are relatively soft.” He concluded that the results of his study “helps to explain how the regulatory impact accident provides a high degree of safety to the public”.

Interim Draft -Revision as of 1 June 2015
Table 10-1. Summary of Test Results of Drops of a Half-Scale Cask onto Various Targets (See Gonzales [RBP021]).

<table>
<thead>
<tr>
<th>Target</th>
<th>Impact Velocity (ft/s)</th>
<th>Maximum Increase in Radius of the Cask (in.)</th>
<th>Test Unit Penetration Depth into the Target (in.)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Unyielding Target</td>
<td>44</td>
<td>0.09</td>
<td>0</td>
</tr>
<tr>
<td></td>
<td>44</td>
<td>0.01</td>
<td>0.25</td>
</tr>
<tr>
<td>Concrete Runway</td>
<td>66</td>
<td>0.03</td>
<td>4</td>
</tr>
<tr>
<td></td>
<td>88</td>
<td>0.08</td>
<td>8</td>
</tr>
<tr>
<td>Concrete Highway</td>
<td>44</td>
<td>0</td>
<td>4</td>
</tr>
<tr>
<td></td>
<td>88</td>
<td>0.02</td>
<td>19</td>
</tr>
<tr>
<td>Native Desert Soil</td>
<td>66</td>
<td>0</td>
<td>25</td>
</tr>
<tr>
<td></td>
<td>88</td>
<td>0</td>
<td>36</td>
</tr>
<tr>
<td>Un-compacted Soil</td>
<td>110</td>
<td>0</td>
<td>92</td>
</tr>
</tbody>
</table>

Itoh, et al. [38] documented the drops of spent fuel casks weighing more than 100 t, without impact limiters, from heights of 1.5, 7.5 and 17 m. The drops were performed onto reinforced concrete. The tests showed that the casks retained their containment integrity even for a drop height of 17 m. An analysis of a drop from 7.8 m of a full scale NFT-32B package onto a reinforced concrete target simulating the rigidity of a loading wharf showed that the structural effects resulting from the 9 m regulatory drop test onto an unyielding target were more severe than were the effects when the package impacted the simulated loading wharf from a height of 17 m.

Droste [39] summarized a number of assessments which have shown that the impacts of casks onto various targets with drop heights exceeding 9 m resulted in retention of containment integrity. These included: (a) a drop from 200 m (impact velocity of 225 km/h, i.e. 140 mph) of a half scale model of the ‘TN-8/9’ spent fuel cask and of a ‘BRECO KR 100/200’ spent fuel cask onto 40 cm of concrete reinforced by steel mats over a 60 cm layer of pit gravel; (b) three drops from heights of ~800 m of ‘Mosaik’ type packages onto compressed sand covered by 40 cm of gravel and 30 cm of concrete; (c) a drop from 19.5 m of a full scale spent fuel transport and storage cask, the ‘CASTOR Ic’, onto a target layer simulating a typical heavy truck road; (d) a drop from 13 m of a full scale ‘CASTOR Ic’ onto a layered target consisting of aluminium honeycomb layers and steel plates with a total target thickness of 2 m; (e) a drop from a height of 25.5 m of a 1 : ¾ scale model of the ‘C 30’ cask onto a representative concrete building structure; (f) three drops from heights of 100, 185 and 200 m of type ‘18B’ Pu nitrate packages onto a target consisting of 20 cm concrete over 60 cm gravel; and (g) analyses of different type B package designs onto real targets for drop heights up to 17.5 m onto real targets, e.g. a concrete highway.

Droste [39] also provided information on an impact at different speeds of an “18 B” package designed for transport of 10 L of Pu nitrate solution. The drops, which were onto unyielding targets from heights of 100, 185 and 200 m resulted in package deformation but no leakage. A high speed impact of a modified “18 B” package design with stiffer spacers was performed at SNL for BAM at a velocity of 129 m/s (287 mph) onto an unyielding target. For this very high-speed impact, loss of containment integrity occurred. It was reported that “These investigations have shown that for this type B package large margins of safety beyond the 9 m regulatory drop...
test existed, but ‘type C package quality’ was not provided. An enhanced crush test was also applied onto this package by BAM for the first time (9 m drop of a 2 tons, 161 m steel plate onto the package), which caused more severe damage than a 200 m drop test; this was one of the initiatives leading to the ‘crush test’ as a mechanical test for ‘lightweight’ packages, instead of the 9 m drop test.”

Sert, et al. [40] summarized analyses that were performed of two packages designed for the transport of spent fuel and plutonium dioxide powder, respectively, where they were dropped from heights of 8–50 m, onto simulations of real targets. The targets that were simulated included (a) various types of soils, including clay, sand and rock; (b) various metallic targets; (c) reinforced concrete structure; and (d) another package of the same design. These simulations showed that “drops onto realistic targets, present in the real environment of transport, do not call into question the maintenance of the safety functions of the studied typical package designs”.

Vaughan and Farrington [41] reported on drop tests performed with two SAFKEG type B packages (a 2816A and a 2816C), which were dropped from 500 m (1650 ft) onto a concrete target. The assessment of the package after the drop showed that the package was significantly deformed at the point of impact but that the lid was retained and the outer packaging elements completely encased the containment vessel, and that the package provided complete containment of the contents. Specifically, both of the “tests showed that these two packages, which were designed to pass the Type B package 9 m drop test, performed well under the much more severe test of impacting at near terminal velocity on a concrete target, with no loss of containment”.

Yoshimura, et al. [42] summarized the test of a 3.05 t cask for transport of irradiated capsules that was dropped from a height of 610 m onto hardpan desert soil. The cask sustained less damage than a second cask of the same design sustained when it was dropped 9 m onto an unyielding target. It was also reported on another test of a 7.4 t cask which was dropped from 610 m onto hardpan desert soil. Although a comparison of this cask’s behaviour during a regulatory drop test was not available, post-test examination showed some structural deformation, the cask was difficult to open, but there was no damage to the inner basket which would have contained spent nuclear fuel.

Akamatsu, et. al. [RBP032] reported on a finite element analysis of the impact of a 115 ton TN24 cask onto yielding targets and an unyielding target from various drop heights. The study established the drop heights needed for the various yielding targets to achieve the same impact as would be achieved for a 9 m drop onto an unyielding target, as summarized in Table 10-2.

<table>
<thead>
<tr>
<th>Target</th>
<th>Drop Height (m)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Unyielding Target</td>
<td>9</td>
</tr>
<tr>
<td>Soil (N-value = 10; soft)</td>
<td>71</td>
</tr>
<tr>
<td>Soil (N-value = 50; hard)</td>
<td>33</td>
</tr>
<tr>
<td>Asphalt (road surface)</td>
<td>51</td>
</tr>
<tr>
<td>Concrete (road surface)</td>
<td>47</td>
</tr>
</tbody>
</table>

These values compare favourably with similar data obtained by Gonzalez [RBP021] that are summarized in Table 10-1.
Ammerman [43] reviewed experimental and analytical research directed towards demonstrating the severity of the regulatory 9 m impact onto an unyielding target. The work compared the regulatory impact with higher velocity impacts onto other types of targets, including soil, concrete roadways, concrete runways, bridge columns, sand, water and transportation vehicles. Impact velocities in the studies have ranged up to 110 m/s (246 mph). It was concluded that the response of a specimen to impact on a yielding target depends on the stiffness of the impacting package. For most target types and most radioactive material packages, the regulatory impact from 9 m onto an unyielding target encompasses nearly all “real life” impacts that were studied.

Ammerman, et. al. [44] summarized analyses that considered various generic cask designs impacting a rigid (essentially unyielding) target at speeds from 13 to 54 m/s (30–20 mph). One of the cask designs was dropped onto an unyielding target from heights of 9, 20 and 36 m, with impact velocities of 13.4, 20 and 27 m s\(^{-1}\) (i.e. 30, 45 and 60 mph respectively). An accompanying assessment was performed with both pre-test finite element analysis and testing of a minimally designed cask. Here the purpose was to baseline the analytical methods and to also compare analytical results with test results. The combined test and analyses demonstrated that the use of the unyielding target results in robust package responses while also providing a stable method for evaluating individual package designs analytically and comparing those results with tests using the unyielding target.

In Lopez, et al. [45], the analytical assessment was described of the NCI-21PF UF\(_6\) package impacted at 30 mph onto an unyielding target, and for impacts of the same package at higher speeds onto yielding targets. The targets considered included soils, concrete and rock. It was determined that, to obtain the same level of damage that would result from the regulatory impact, velocities would need to range between (a) 35 m/s (78 mph) and 207 m/s (462 mph) for different types of soils, (b) 14 m/s (31 mph) and 21 m/s (46 mph) for different types of concrete, and (c) 13 m/s (30 mph) and 42 m/s (97 mph) for different types of rock. Assessment of the probabilities of exceeding regulatory conditions in accidents revealed only a limited number of circumstances under which regulatory conditions may be exceeded and the likelihood of UF\(_6\) being dispersed by impact from this package was determined to be small.

Malesys [46] described how an FS 47 packaging for the carriage of Pu oxide powder was dropped 50 m onto a concrete structure (representative of a pier in a port), as compared with 9 m onto an unyielding target as required by the Regulations. The test did not lead to unacceptable consequences for the package. It was reported that, effectively, this drop was less severe than a regulatory 9 m drop onto an unyielding surface.

Sandia National Laboratories [RBP020] has provided a summary analysis demonstrating the effects of target hardness, considering the differences between package impacts onto an unyielding target as compared with package impacts onto a concrete target, both dropped from 30 ft (9.1 m). This summary compares the behavior of a passenger van dropping on both an unyielding target and a concrete target, and provides a similar comparison of a radioactive material cask undergoing drops onto the same two targets.

The summary concludes the following:

“It is expected that the van impact onto the concrete target will result in similar damage to the van as the drop onto the unyielding target. This is because, for a van, a concrete roadbed is nearly an unyielding target. This result will reinforce what people already believe. It is highly probable that the amount of damage to the van will be similar to the amount seen in very severe highway collisions, but not greater than the amount seen in ‘worst case’ accidents. The result that will provide the demonstration on the severity of the regulatory test will be the comparison between the regulatory impact of the DHLW 1/2-scale model and the impact of this cask onto the concrete target. For this cask, the results of the two tests will not be the same. There will be significant damage to the concrete target. This will communicate the point that what constitutes a rigid target depends on the stiffness of the object impacting it. For
things that people are used to dealing with, such as vans and bodies, the concrete target is essentially rigid. For RAM casks it takes a much more substantial target to be essentially rigid. Targets of this type are rarely found in the ‘real world’.

“Actual results were nearly indistinguishable damage in the two van drop tests and no damage to the DHLW cask from impact onto the concrete target [the cask punched a hole through the concrete].”

An assessment looking at the impact from a different viewpoint was reported in 1991 by Bergmann and Ammerman [RBP047], where the parameters affecting the package behaviour during eccentric impacts were evaluated. The primary concern was that eccentric impact on one end of a cask can result in rotation of the cask after initial impact leading to a secondary impact, a situation often referred to as “slapdown”. It was shown that in “a slapdown event, the rotational acceleration during the primary impact can cause accelerations at the nose and tail which are greater than at the nose during primary impact”. Specifically, the results of the analysis showed that “the cask with an aspect ratio greater than 2 has nose and tail accelerations greater than the side drop acceleration at all values of friction (between the cask and the target) considered”. It was found that increasing the friction force results in decreasing peak nose accelerations while concurrently increasing peak tail accelerations. This study illustrates that the requirement of performing the mechanical tests with package orientations on impact “so as to suffer maximum damage” is a means of ensuring the behaviour of a specific package design under alternate damaging scenarios is adequately addressed (see para. 727 of SSR-6).

All of the preceding assessments have uniformly illustrated that the choice of an unyielding target provides for a very severe mechanical impact test environment, such that impacts onto “real life” targets require much greater velocities to obtain the same level of damage. In addition, it is noted that the Regulations require that the 9 m drop test be performed so that the specimen shall drop onto the target so as to suffer maximum damage. This results in a very conservative approach to mechanical testing.

The concept of maximum damage appears to have been deliberated by the early 1964 panel of experts, where on page of Ref. [RBP001] it states that “There was a suggestion that the requirement should be that the package should fall ‘so as to suffer the maximum damage’ instead of ‘on the weakest part’ or ‘so as to stress the weakest part. This was agreed.”

10.3. Technical basis for normal conditions of transport package tests

Deliberations at IAEA panels, on what should be specified for normal conditions of transport, which were initially applied to Type A and Type B packages, occurred from the beginning of the development of the Regulations. Appleton and Servant [25] wrote that:

“The effects of the transport environment on packages under either normal or accident conditions, were complex and no series of tests could simulate those conditions completely and accurately”. They then wrote that “The aim was to produce tests of a severity which would provide for packages of an adequate standard without attempting to cover every detailed effect. This would also have the result of reducing to a minimum the number of tests required, with a consequential saving in the expenditure of time and money on testing.”

Fairbairn and George [26] elaborated on the thinking behind the requirements as they existed in 1966. Specifically:

- The water spray test was intended to expose the package to a heavy rain, followed by a free drop from 1.2 m (4 ft); and that packages with outer layers consisting of metal, wood, ceramic or plastic were to be exempted from the test.
- The free drop test was intended to provide a means of examining the resistance to limited shock. It was noted that the drop tests in some countries “have included a graduation in height of drop which decreases with the weight of the package [47, 48] or the contents [49]. However, in 1966, the drop height for all packages was a fixed value of 1.2 m (4 ft). In
addition, in “... view of the potential vulnerability of lighter packaging whose outer parts are constructed of fiberboard or wood”, it was required that a separate sample be exposed to a free drop of 0.3 m (1 ft) onto each corner (or if a cylinder, onto each quarter of each rim).

- The compression test was intended to represent stacking of cargo onto the package equivalent to five times the weight of the package.
- The penetration test was intended to assess whether a package would adequately resist “penetration of the vessel and any outer packaging by a relatively sharp object”. However, it was noted that “penetration by the bar of the outer parts of the packaging would not, in itself, constitute failure”.

Fairbairn [50] provided a summary of the early history. He summarized the concepts embodied in the first issue of the Regulations [51], and then wrote that “…the General Revision Panel for the 1964 regulations decided that it was clearly necessary to help package designers by defining the above conceptual requirements in terms of specific design principles and test requirements – a Package Panel was accordingly set up to do this work”. Efforts by involved countries and through the International Organization for Standardization (ISO) resulted in proposals being brought forward by the UK to the Panel, as reported by Messenger and Fairbairn [52].

Fairbairn [50] further noted that the definition of requirements for “normally incident to transport” and “minor accidents” is very complex, since they cover “as it does both the world environment and all modes of transport”. The initial proposal addressed 18 different tests, which the Panel – following extensive discussion – reduced to just the four tests that have been specified in the Regulations since the 1964 Edition. He noted that the experience gained with the 1964 requirements remained unchanged for the 1967 Edition, but minor changes were then agreed for the 1973 Edition. These were:

(a) that each of the other normal conditions of transport tests be preceded by the water spray test,
(b) the addition of grading of the drop height for the free drop test, considering the weight of the package, and
(c) the addition of the 9 m drop test being applied to Type A packages containing liquids and gases (currently specified in para. 725(a) of the 2012 Edition of SSR-6 [1]).

This latter requirement was justified on the basis of the results of a UK study [53].

Appleton and Servant [25] note that a very comprehensive UK document by Messenger and Fairbairn [52] was used in the early deliberations. Initially, to address the large number of environments “which come under the heading of ‘shocks’ the use of a ‘tumbler drum’ test” was considered, but discarded. The water spray test was intended to evaluate the wetted strength of the package; therefore a drop onto a firm target was incorporated. The free drop through a few feet onto a suitable target was accepted as an appropriate test for all packages except pressurized gas cylinders. Additional drops at reduced height onto the corners of fibreboard or wooden packages were added to ensure “racking and other stresses” were addressed.

The compression test was added to address concerns of “over-stowing” by other packages or of compressive loads under inertia. For the penetration test, the “penetration force selected was of reasonable magnitude, and not meant to be destructive, but sufficient to evaluate the protection afforded to the containment vessel by all outer layers of packaging.

Additional details of the discussions leading to the set of normal conditions of transport requirements and their associated criteria for failure are elaborated on pages 45 – 52 of Ref. [54].

In 1979, TC-272 [RBP026] addressed tests for Type A packages, responding in part to concerns over experience with these packages in crush environments. Following deliberations on actual experience, and consideration of documents provided to the technical committee, it noted that
"the level of protection provided by today’s Type A packagings generally exceeds the ‘minimum Type A packaging’. In addition the TC stated that “most experience (which has been generally favourable) has been gathered on commercial Type A packages which are usually stronger than required by the regulations’. TC -272 concluded that the Type A tests should remain as already required and that there was no need to add a crush test to the Type A package test requirements.

The current Regulations specify that all four of the tests may be performed on a single specimen as long as – for each of the drop, penetration and stacking tests – the water from the water spray test has penetrated to the maximum depth (see paras 720 and 721 of SSR-6 [1]). In this regard, Appleton and Servant wrote as follows:

“The above tests represent different effects, all of which may be imposed upon a single package. However it is not considered necessary to apply all the tests to a single sample package and a separate sample may be used for each test with the exception that at least one of the samples should be subjected to two tests applied consecutively.”

10.3.1. Additional normal conditions of transport for Type A packages containing liquids or gases

Appleton and Servant [25] and Fairbairn and George [26] address the additional tests for Type A packages designed for liquids and gases (see para. 725 of SSR-6 [1]). Appleton and Servant noted that "The additional test in respect of liquids is for those type A packages which do not include absorbent material; it comprises a very severe drop test of the package. For type A packaging designed to carry gases in excess of 20 curies the same severe drop test is applied to the unpackaged containment vessel."

The current test requirement for liquid and gaseous contents are more severe than earlier specified in that two separate tests are generally required: (a) the free drop from 9 m, and (b) a penetration test with the bar dropped from 1.7 m.

[see EndNote xiv ]

In considering the drop tests for Type A packages with liquid or gaseous contents, deliberations addressed whether the package or the unpackaged containment vessel should be subjected to the 9 m drop test. In early 1964, the deliberations of a panel of experts [RBP002] resulted in the following summary of discussion. For liquids contents:

“It was eventually proposed and accepted that the containment vessel of a Type A package designed for liquids would be required to be metal and that it would be separately subjected to a 9 m drop test unless it contained sufficient absorbent material to absorb twice the volume of the liquid contents. The provision in respect of the provision of absorbent material outside the shield and the limitation of the dose rate, in the case of liquid escape, to not exceed 1000 mr/hr at the package surface would be retained.”

Following further discussion, it was concluded that:

“Eventually, however, it was proposed and accepted to reinstate the subject of the package to the drop test of 9 m and not the separate containment vessel, for Type A packagings designed for liquids, since there would now be the added requirement of a metal containment vessel.”

For gaseous contents:

“It was eventually accepted that the additional test for Type A packagings designed for gases in quantities in excess of 20 curies should be the requirement that the containment vessel be subjected separately to a drop of 9 m.”
Para. 529(c) of the 1985 edition of the Regulations [24] reads:

“For packages in which the liquid volume is greater than 50 mL, either:

(i) be provided with sufficient absorbent material as prescribed in subpara. 529(a); or
(ii) be provided with a containment system composed of primary inner and secondary outer containment components designed to ensure retention of the liquid contents within the secondary outer containment components, even if the primary inner components leak.”

During AG-406 deliberations [RBP049], the requirement of having either absorbent material or a primary and secondary containment system was extensively deliberated. AG-406 agreed that the designer should have the alternative (i.e. it emphasized that “or” should be included between the two different requirements); however, the threshold for this requirement was recommended at AG-406 to be 1 L rather than 50 mL. However, the 50 mL constraint was removed almost immediately, in the 1988 Supplement to the 1985 edition of the Regulations [RBP050], so that now any quantity of liquid in a Type A package is required to satisfy either the absorbent material or the primary/secondary containment components requirement.

10.3.2. Addressing other normal conditions of transport through general requirements

Fairbairn and George [26] then commented that, in the interest of simplifying the regulations and minimizing both the number of tests and the cost of testing, issues relating to dry heat, cold, damp heat, bumping and vibration were dealt with in the general packaging requirements without specifying specific test requirements.

They also addressed the potential need for sequencing the Type A tests, and concluded that this was not needed, other than sequencing the mechanical tests each preceded by the water spray test.

10.3.3. Requiring normal conditions of transport tests for packages other than Type A packages

The normal conditions of transport tests are imposed, following a graded approach, on industrial packages: IP-1 package designs do not have any normal conditions of transport tests requirements; IP-2 package designs are required to be able to withstand the normal condition free drop and stacking tests; whereas IP-3 package designs are required to be able to withstand all four of the normal condition of transport tests.

Type B and Type C package designs are required to be able to withstand all of the normal conditions of transport tests (i.e. paras 719 – 724 of SSR-6 [1]) in addition to the other more severe test environments that are imposed on these designs which are addressed in the following sections. This was instituted in the 1964 Edition of the Regulations: "With regard to test requirements, the revised regulations require that Type B packaging be capable of withstanding the four tests for Type A packaging .... in addition to the tests designed to simulate accident conditions" [26].

10.4. Technical basis for accident conditions of transport package tests

This section addresses the technical basis for the regulatory requirements for testing radioactive materials package to accident conditions of transport as specified in the 2012 Edition of SSR-6 [1]. The basis of development of these tests and their technical justification is summarized in the following subsections. A summary of follow-on studies that further justify the technical basis of the accident conditions of transport used in the Regulations is elaborated in Appendix 4 of this Technical Basis Document.

A number of documents were assembled, presented at PATRAM 2007, and then published in the international journal Packaging, Transport, Storage & Security of Radioactive Material that
addressed many of these topics to assess the adequacy of the test requirements with respect to covering a large percentage of actual accident environments that might be expected. These studies considered the results of tests and analyses that were performed to expose packages to environments that exceeded the regulatory requirements. The results and findings from these studies are referenced, as applicable in the sections below. In addition, much of this overall effort was briefly summarized by Pope [55].

10.4.1. Comprehensive Assessments of Accident Environments

In addition, a number of comprehensive studies of the accident environments were undertaken in the 1970s at SNL. Clarke, et. al. [56] considered the accident environments posed by road, rail and air transport to small packages. This document provided extremely detailed results for (a) the calculated environment, (b) a prediction of the probability of exceeding any given set of fire and impact levels, and (c) a methodology for transforming the data generated into a measure that could be more clearly related to experience (i.e. the likely number of accidents of a given severity in a given distance travelled).

The data from [56] served as a basis for more detailed accident environment studies, for both small and large packages, in some cases coupling them to the qualification criteria for packages. Dennis, et. al. [57] considered fire, impact, crush, immersion and puncture for large packages in road and rail transport in the US using historical accident rates that had been documented from 1966 through 1972. A large amount of predictive data was developed. These data were used to predict the cumulative probabilities that a regulatory test requirement parameter would not be exceeded if a truck or rail vehicle were involved in a reportable accident. The probability that the fire duration would not be exceeded was about 99.8 percent of the cumulative percent of occurrence. The cumulative percent that the impact velocity would not be exceeded was well above 99.5 percent. For large packages with wall thicknesses greater than 5 cm, the cumulative percent that the puncture is not expected to occur was in excess of 99.97 percent.

In the mid-1970s, McClure [58] used the data from Clarke, et. al. [56] to predict the percentage of accidents for small packages transported in a given mode, where the accident environment would be expected to be equal to or less than the package qualification standards (i.e. a fire environment of 800 °C for a duration of 30 minutes, an impact environment with package striking an unyielding target at a velocity equivalent to the 9 m drop test, a puncture environment equivalent to regulatory 1 m drop onto the puncture bar, and a 15 m water immersion test). His results are summarized in Table 10-3.

<table>
<thead>
<tr>
<th>Transport Mode</th>
<th>Accident Environment Categories</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Impact</td>
</tr>
<tr>
<td>Air</td>
<td>79%</td>
</tr>
<tr>
<td>Influence**</td>
<td>Primary</td>
</tr>
<tr>
<td>Road</td>
<td>100%</td>
</tr>
<tr>
<td>Influence**</td>
<td>Secondary</td>
</tr>
<tr>
<td>Rail</td>
<td>100%</td>
</tr>
<tr>
<td>Influence**</td>
<td>Secondary</td>
</tr>
</tbody>
</table>

* The report also produced data on the anticipated crush environment; however, since the Regulations at that time did not include the dynamic crush test, those data were part of the body of data considered in determining the need for such a test.

** McClure assessed the individual accident environment categories for each mode and determined which were more significant in terms of the influence on the package behaviour in an accident, and categorized them as either “primary” (of most significance package integrity) or “secondary” (of lesser significance to package integrity).
McClure’s summary of data shows that, for road and rail transport, essentially all accidents (at least 99 percent) are covered by the regulatory test requirements, and the fire environment was of primary concern. However, for air transport, the summary showed that only 79 to 98 percent of anticipated air accidents were covered, and that both impact and fire were of primary concern. In fact, these data were one of the sources used in establishing test criteria for the US-specific requirements for a package transporting plutonium by air (see Section 10.6 below).

In addition, in the late 1970s, McClure undertook a study to assess the probabilities, based on actual experience, of a spent nuclear fuel shipping cask being exposed to an accident environment during transport that might lead to a failure of the cask [59]. Failure was defined in this study as release of radioactive materials. McClure determined that this probability for transport by road in the U.S.A.; concluding that the information that:

“...is contained in this report ... allows one to assess, at least in a crude fashion, the likelihood that a spent fuel cask will be in an accident which will cause cask failure. This analysis suggests that accidents with the potential for release occur about once in every 500 million miles travelled”.

10.4.2. Defining Specific Representative Accident Scenarios for Type B Packages

The tests for demonstrating a package’s ability to withstand accident conditions of transport are covered in SSR-6 by paras 726 – 729. One feature of these test requirements is that a specimen representing the package design shall be subjected to the cumulative effects (as specified in para. 726) of two mechanical tests (para. 727) and one thermal test (para. 728). Then either the specimen already exposed, or a separate specimen, shall be subjected to a 15 m depth water immersion test (para. 729), and – if required by the quantity of radioactive material in the package - to the enhanced water immersion test at a 200 m depth (para. 730).

Historical Background in the Development of Accident Scenarios

Fairbairn [60] wrote that when “the 1961 issue of the regulations was prepared information on transport accidents was not readily available”; this led the Panel to use the ‘maximum credible accident’ concept in the first edition of the Regulations “as a ‘stop-gap’ until such information became available”. Section 2.4 of this document addressed in detail the elimination, in later editions of the Regulations, of this concept.

One concept that was emphasized from the beginning, and is stressed here deals with a Type B package that has been involved in an accident. In this event, Fairbairn and George [26] noted the concept was (and still is) that “…it was decided that Type B packaging must be capable of withstanding the accident to an extent that safe recovery of such packages was feasible within overall emergency plans and procedures. It was never conceived that after such recovery the packaging should be suitable for re-use; in other words, Type B packaging was never required to withstand a succession of so-called ‘maximum credible accidents’.”

The deliberations leading to the requirements in the 1964 Edition of the Regulations considered multiple inputs [26, 52]. These deliberations looked at environments that could be expected in transport accident by land, sea and air. This “led to the decision by the Packaging Panel to simulate the damaging effects of transport accidents by the combining of the mechanical and thermal tests as resulted in being specified in the 1964 Edition of the Regulations”.

With respect to the consecutive nature of the accident-simulating tests (i.e. that the cumulative effects of two mechanical tests and one thermal test shall be applied sequentially on the same specimen - see para. 726 of SSR-6), Appleton and Servant [25] and Fairbairn and George [55] addressed this issue in their early documents. This feature of sequencing or pyramiding of the tests on a single specimen was established early, and has been retained over the years. However, it has been extended such that the ‘order in which the specimen is subjected to the drops shall be such that, on completion of the mechanical test, the specimen shall have suffered
such damage as will lead to the maximum damage in the thermal test which follows’ (see paragraph 728 of SSR-6 [1]).

As early as early 1964, the concept of sequencing of the mechanical tests followed by the thermal test was deliberated and agreed. On page 28 of Ref. [RBP001] it reads that “it was emphasized that the introduction should make it quite clear that the sample package should be subjected consecutively to the mechanical and thermal tests, and in that order”.

The concept of testing in a sequence so as to have maximum damage in the following fire test also has been carried forward to testing in the orientation of the package in the mechanical and pressure tests. As early as 1973, the Regulations specified that each mechanical test would have the package drop onto the target ‘so as to suffer maximum damage’. This concept is also retained in today’s Regulations.

Fairbairn [60] also elaborates on the sequencing of the Type B tests on the same specimen (see para. 727 of the RSS-6 [1]). He wrote that the deliberations

“...led to the decision by the Panel to simulate the damaging effects of transport accidents by the combination of the mechanical and thermal tests as specified in the 1964 issue of the regulations. The decision to ‘pyramid’ these two tests on the same sample aimed to facilitate the movement of Type b packages by all modes of transport. When devising these tests, the Packaging Panel decided not to add a water immersion test to the pyramid, but rather to prescribe, as a separate packaging design principle, additional to all the Type A packaging design principles to be met by the Type B packaging, that the containment vessel must remain intact at a depth of 15 m of water.”

In 1963, Messenger and Fairbairn [52] documented the process used in the early days to define accident test conditions, then made proposals for the next revision. They listed five main elements of concern, which were:

- "Mishandling and tampering
- Impacts due to large drops when loading or to collision during transport
- Fire and damage by fire-fighting materials
- Immersion in water
- ‘Smothering’ by debris or by other goods as a result of one of the above"

Following detailed documentation of the analysis of each of these five phenomena, including modal considerations, they proposed that, for the practical purposes of evaluating and approving a design of Type B packagings, the accident environment:

“shall be regarded as comprising:
(i) Head-on collision at 30 m.p.h. (or 30 ft free fall, which is equivalent) with a rigid structure, followed by—
(ii) Exposure for 30 minutes to a liquid fuel fire, the mean effective temperature of which is 800 °C, with no quenching after such exposure until the temperature of the interior as measured during test ... has started to fall, followed by—
(iii) Immersion in water, provided that ingress of water to the package constitutes a hazard.”

One year later, Appleton and Servant [25] documented the results of deliberations at the IAEA leading to the 1964 edition of the IAEA Transport Regulations [21] were documented. The experts used the above-cited information provided by Messenger and Fairbairn [52]. During these discussions they noted that three important aims were kept in mind as the detailed standards were developed. These were that:

- an adequate standard of containment and shielding needed to be maintained,
- no reduction in standards below those already being attained should result, and
• there should be no restriction on the initiative of the designer of packaging in choosing new
materials or employing new methods.

They also summarized the early deliberations on specific test requirements with a view to
replacing the use of the qualitative ‘maximum credible accident’ requirement. The issues
considered included the concerns noted by Messenger and Fairbairn about mishandling,
tampering, the possible need for a requirement for tamper indicating seals, and smothering of
the package in accident debris. They noted that it “is almost impossible to devise tests to
represent such possibilities”, and therefore such seals “cannot prevent willful interference, but
they constitute a deterrent and, if tampering has occurred, [provide] a telltale that warns those
concerned to handle or open the package under precautions”. Additionally, for ‘smothering’, they
concluded that this would only be of concern for large packages with contents generating
significant quantities of heat, and that appropriate analyses for the ‘smothering’ phenomenon
should be undertaken as part of a package design. Fairbairn and George [26] wrote, that
smothering “…from a safety point of view is of no real concern unless the package contains a very
large heat source such as irradiated fuel”.

Until the Type C package requirements were instituted, smothering was not considered an
issue; however for Type C packages, they must be capable of withstanding burial in a relatively
low thermal conductivity environment (see para. 670 of SSR-6 [1]).

Appleton and Servant [25] further noted that the aim of the Regulations would be to produce
tests of a severity which would provide for packages of an adequate standard without
attempting to cover every detailed effect. This would also have the result of reducing to a
minimum the number of tests required, with a consequential saving in the expenditure of time
and money on testing. They went on to state that one goal was to have the packaging standards
so developed that alternative calculative methods could be derived in order to avoid actual
testing. This was deemed of importance in particular for specimens of type B packaging which
may be very expensive.

A significant body of data has since been developed and documented on the adequacy of the

10.4.3. Technical Basis for Drop I (9 m Drop Test)

The test for demonstrating a package’s ability to withstand the major impact accident condition
of transport (i.e. the second mechanical test) is covered in TS-R-1 by para. 727(a). The following
subsections address (a) the historical background in the development of the 9 m drop test onto
an unyielding target accident scenario, and (b) later studies that have been undertaken to better
evaluate the adequacy of the 9 m drop test accident scenario.

Historical Background in the Development of the 9 m Drop Accident Test Scenario

Initial discussions by a panel of technical experts convened in early 1964 [RBP001] led to a
proposal that the first drop test should be from9 m onto a 30 cm wide beam target, where the
beam would be mounted on a rigid horizontal surface. Discussion later in 1964 [54] resulted in
the following conclusion: “…if the beam was removed the effect of shear could not be studied.
However, it was pointed out that there was still the pedestal test which gave some shear effect,
whereas the drop onto a flat target would test overall structural integrity which was the other
important factor.”

As reported by Fairbairn and Messenger [52], the “beam” referred to here was one that was
being used for testing of safes in the United Kingdom. They state:

“161. Manufacturers of fire-resistant safe cabinets have carried out drop tests immediately
after severe fire tests, while the cabinets were still hot, the drop tests being immediately
followed by further fire tests. Such tests, as reported in several manufacturers’ catalogues,
were on cabinets weighing between 250 and 1,400 lbs. Heights from 12 ft to 30 ft (generally
the latter) were used, there being no apparent connection between the height of drop and the
weight of the cabinets. The “target” consisted of rubble, which would be much more yielding than solid concrete. They go on to recommend a drop test in the following manner:

“162. The United Kingdom Atomic Energy Authority have used a target consisting of a rigidly mounted steel beam 1 ft wide with a maximum deflection of 0.01 under a 50 ton static load at the centre (as in the normal environmental impact test proposed in Part B). For packages that were not intended to be secured to the vehicle, a 30 ft drop, representing a 30 mile/hour direct impact, has been used. Transient peak accelerations as high as 3,000 to 4,000 g have been recorded in such tests. These high accelerations are due to the unyielding nature of the target which was selected primarily from the point of view of test reproducibility and target life.”

Messenger and Fairbairn further indicated that for the 1964 Edition, the then available data on impacts was used. Heights, frequencies and probabilities of the potential for drops from cranes especially in seaports as well as those for collisions and other mechanical accidents (derailments, falls, impacts with stationary structures, aircraft-to-ground crashes, impacts with other vehicles, etc.) were all considered, in addition to modifying factors that were needed to establish realistic environments. For example, it was recognised for collisions between vehicles that “it is misleading to add their speeds” because “the additional kinetic energy would be absorbed by damage to the second vehicle”. It was therefore concluded that: “collisions between vehicles may be regarded as no more severe than collisions at similar speeds with permanent structures”. Other modifying factors considered included:

(a) a propensity for such heavy vehicles to adhere to statutory speed limits,
(b) the potential for braking of vehicles before impact,
(c) noting that many impacts will not be “head-on”, but rather will involve only a glancing blow, and
(d) the ameliorating effects of the crushing of the carrying vehicle structure before impact of the package into some more solid structure.

In establishing the recommended impact test requirements, they took note of the requirement then imposed by the US Atomic Energy Commission (AEC) that required a flask carrying irradiated nuclear fuel to be able to withstand a fall of 15 ft (5.47 m) onto an “unyielding horizontal flat surface”. This was in contrast to some packages used for transporting military stores in the UK that were then required to withstand a 40 ft (12.2 m) fall onto a steel plate wet-floated onto 18 in (0.46 m) thick concrete. At that time, fire-resistant safes were required to be dropped from heights ranging from 12 ft (3.6 m) to 30 ft (9 m) onto “rubble”.

As noted above, another test configuration that was considered and that was then being used by the United Kingdom Atomic Energy Authority (UKAEA) was a 30 ft (9m) drop onto a rigidly mounted 1 ft (0.3 m) steel box beam.

Elaborating on the discussion relative to the mechanical tests, Appleton and Servant [25] stated that the “impact effects are considered to be of two kinds, structural shock and shear respectively. Instead of merely striking a flat surface a package might fall from a height onto a protruding object such as a small package or post, in collision it might be struck by a relatively small projectile or whilst being carried on a vehicle might hit a bridge abutment at an acute angle”. It was also stipulated that: “those effects [of structural shock accompanied by shear] could be obtained by dropping a sample package through a suitable height onto appropriate targets”. After discussing the alternatives, the experts agreed to adopt a two-drop test concept that was proposed by the USA: first a drop from 30 ft (9 m) onto a flat target, and second a drop from 3 ft (0.9 m) onto a pedestal target. The alternative, the single UK-proposed drop onto a rigid box beam was not accepted due to both testing and analytical complexities.

“While one drop test on to such a target {a rigidly mounted rolled steel beam 1 ft wide} subjects a package to both impact and shear, for the purpose of specifying a test which is more amenable to the application of calculative methods, the Agency’s mechanical test separates these effects by requiring that the package be dropped twice {a 30 ft (9 m) drop onto a flat horizontal surface,
and a 3 ft (1 m) drop on to the end of a 6 in. (15 cm) diameter mild steel bar or punch} [26]. The discussion during these deliberations was captured in the mid-1964 Panel Meeting report [RBP036] wherein it is noted in conclusion that “It was eventually agreed that to maintain the same accident conditions for all weights of packaging, and to facilitate the calculative methods, the fall onto the beam would be replaced by a fall onto a flat surface; the shear effect being provided by the second fall of the same sample package onto the pedestal”.

The choice of the 30 ft (9.1 m) drop test height resulted “from practical judgement, first that in the course of transport Type B packages are unlikely to suffer higher drops on to very hard targets such as dock wharves, and second that a part of the impact during collisions at high speeds will be absorbed by the vehicles”. For the first mechanical test it was therefore agreed that the drop height should be 30 ft (9.1 m), and that the target for this test was to be a flat, horizontal surface. They note that “the target mass could be related to the falling package in order to provide the same relative structural shock”. “The choice of 30 ft for the impact part of the mechanical test results from practical judgment, first that in the course of transport Type B packages are unlikely to suffer higher drops on to very hard targets... and second that a part of the impact during collisions at high speeds will be absorbed by the vehicles”. [26].

10.4.4. Technical Basis for Drop II (1m Drop onto Puncture Bar)

The test for demonstrating a package’s ability to withstand the puncture probe accident condition of transport (i.e. the second mechanical test) is covered in SSR-6 by para. 727(b). The following subsections address (a) the historical background in the development of the 1 m puncture probe test (where the probe is mounted onto an unyielding target) accident scenario, and (b) later studies that have been undertaken to better evaluate the adequacy of the 1 m puncture probe test accident scenario.

Historical Background in the Development of the 1 m Puncture Probe Accident Test Scenario

For the second mechanical test (what was then called the pedestal test, now commonly called the punch or puncture test) the target surface specified for the 9m drop test was to be used as the foundation for the pedestal bar. In preparation for issuing the 1964 edition of the transport regulations [61], it was agreed that the test would consist of a drop from a height of 3 ft (0.91 m) onto the end of a 6 in (15 cm) diameter mild steel bar or punch with edges rounded off to a radius not exceeding 6 mm, which was intended to evaluate the package integrity for the effects of shear in an accident (Fairbairn and George [26]). They noted that the “use of a 6 mm, as opposed to say 0.6mm radius may be criticized as unduly blunting the punch; on the other hand, it must be recognized that if no radius were allowed this shear test would be very severe indeed, particularly for very heavy packaging”.

The specific parameters for the 1 m puncture probe test were discussed by technical experts at a panel meeting [RBP002] convened in early 1964. For example, it was noted that “There was a proposal that in the second method of doing this test, e.g. the two impacts, the height of drop of the second impact should be reduced to 1 metre and the target should be defined as ‘a circular area of not more than 180 cm²”’. At this point, the experts were considering a 9 m drop onto a beam and a second drop onto a pedestal. Following further discussion [RBP001] “It was proposed that under this heading there should be two components” the second of which would be “a 1 m drop onto the 15 cm diameter pedestal.”

At a similar panel meeting convened later in 1964, it was noted that the regulations needed define the height, diameter and material of fabrication of the pedestal. The panel’s report [54, RBP036] noted that:

“The height was discussed in respect of the tendency to buckling which might occur under very heavy packages, 60 cm as currently in the annex was considered to be generally too large; 20 cm was considered to be more appropriate. However, it was pointed out that for some packages in order to ensure testing for penetration up to the containment vessel it would be necessary to have a longer pedestal. It was said that for a range of pedestal lengths between 20
cm – 1 metre, of thickness 15 cm diameter, there was little difference in buckling effect under heavy impact; however it was agreed that the shorter the pedestal the more straight forward the test.

“It was eventually agreed that the pedestal height would be reduced to 20 cms for all weights of packaging but that this would be qualified ...[to read] ... unless a larger bar would cause greater damage when a bar of sufficient length as to cause maximum damage shall be used.”

Following extensive deliberations, the experts concluded, for the 1 m “puncture test” that “The target shall be the upper end of a solid mild steel bar of circular section and 15 cm in diameter. It shall be 0.6 m long and rigidly mounted perpendicularly on the foundation described in above. The upper surface of the target shall be flat and horizontal with its edges rounded off to a radius of not more than 6 mm.” This requirement was incorporated into the 1964 Edition of the Regulations [62], with the addition of specification of tolerances of ± 0.5 cm on the diameter of the probe, and that the “bar shall be 20 cm long unless a longer bar would cause greater damage; and in that case a bar of sufficient length as to cause maximum damage shall be used”.

Fairbairn and George [26] further commented on the second mechanical test as follows:

“The part of the mechanical test specifically designed to test shear is essentially a punch test, and requires the package to be dropped so as to suffer maximum damage on to the end of a 6 in. diameter mild steel bar rigidly mounted to a foundation complying with the target prescribed for the impact drop”. The target surface of the bar is described in detail (as quoted above) and then it is noted that “...the bar must be of a length which will cause maximum damage to the package. This punch test helps to assess the resistance of packaging containment features to penetration during transport and to the containment of any low melting point shielding material such as lead.”

10.4.5. Technical Basis for Drop III (Dynamic Crush Test)

The test for demonstrating a package’s ability to withstand the dynamic crush accident condition of transport (i.e. the second mechanical test) is covered in SSR-6 by para. 727(c). The following subsections address (a) the historical background in the development of the dynamic crush test accident scenario, and (b) later studies that have been undertaken to better evaluate the adequacy of the dynamic crush test accident scenario.

Historical Background in the Development of the Dynamic Crush Accident Test Scenario

The inclusion of the dynamic crush test came about as a result of deliberations by international panels of experts convened in the late 1970s and early 1980s. In 1977, the potential need for a crush test was identified by AG-126 [RBP033] and AG-144 [RBP038]. For example, the AG-144 report states that “With regard to crush forces, the WG noted that the US studies identified crush forces to be a primary factor in rail and road accidents. The US data applies primarily to the transport of small packages (less than 500 kg); new data on larger packages is still being collected and analysed. The Working Group considered that a possible case has been made for additional Type B tests...”. Ag-144 concluded that “the existing Type B tests are adequate for rail, road and sea transport, even without the crush test, but that the addition of a crush test may be appropriate to close an apparent gap in that regulatory coverage, i.e. light packages that do not experience, during impact testing, forces equivalent to crush”.

In 1979, deliberations continued on the potential need for a crush test at TC-272 [RBP026]. A working group (WG) at this meeting agreed “that a crush environment did exist in transport” and the WG then considered whether “the crush environment was sufficient to need protection against and if the tests could be said to satisfy this apparent need”. Multiple documents had been provided and were evaluated (see pages 19 and 20 of the TC-272 meeting report). Among its considerations, the WG examined various proposals for testing with a view to setting forth a specific recommendation for testing in the event that “a risk assessment and cost-benefit study should support the inclusion of a crush test”.

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The WG of TC-272 WG concluded that:
1. The crush environment was a highly probable event,
2. There was no evidence that a crush event had caused a significant problem in a transport accident involving radioactive materials.
3. The crush forces that exist in normal transport were not sufficient to warrant imposing a crush test of Type A packages.
4. Although it appeared at that time that the increase in safety resulting from the introduction of a crush test for Type B packages could be marginal, further investigation into the crush environment was warranted.
5. The packages most vulnerable to a transport crush environment were determined to be those that were lightweight and small; large packages such as those used for transporting spent fuel would not be vulnerable to crush.

Although further investigation was deemed warranted at that time, the TC-272 WG recommended a static compression crush test.

In 1981, the recommendation for the crush test was addressed in detail at the meeting of the IAEA Technical Committee on Transport Package Test Standards (TC-406) [62], which was convened in Tokyo, Japan, September / October 1981. This meeting was preceded by earlier technical committee, consultants services meetings and advisory group meetings as well as meetings of the Standing Advisory Group on the Safe Transport of Radioactive Material (SAGSTRAM); many of which discussed the issue of the need for a new crush test.

During the TC-406 meeting, one working group addressed the issue of crush testing in detail, made recommendations to the plenary body of TC-406 which were endorsed for inclusion in the Regulations. These recommendations, with minor editorial changes, resulted in what ultimately became the dynamic crush test (i.e. the mechanical test drop III, para. 727 of RSS-6 [1]). The working group determined, based on the data and information available in 14 different documents, that "The fact that the crush environment exists has been established .... The fact that crush environments can damage soft type-B packages has been established ....". The working group then recommended that it was recommended that the test should be required for packages that:

- "do not contain special form material;
- contain radioactivity greater than 1000 $A_2$;
- have a mass less than 500 kg; and
- an overall density less than 1500 kg/m$^3$, the density being based on the outside envelope of the package".

After justifying each of these criteria, the working group then recommended that the test itself would consist of "the specimen shall be subjected to a dynamic crush test by positioning the specimen on the target so as to suffer maximum damage by the drop of a mass onto the specimen. The crush energy shall be equivalent to that of dropping a mass of 500 kg from 9 m onto the specimen. The mass shall consist of a solid mild steel plate 1m x 1m and shall fall in an horizontal attitude. The height of the fall shall be measured from the underside of the plate to the highest point on the specimen. The target on which the specimen rests shall be as defined in para. 708."

Nehrig, et. al. [63] provides a summary of the TC-406 discussions and what transpired previously and following the TC-406 meeting. This document further summarizes data from various experiments on light-weight, low density packages that were performed to illustrate the need for a dynamic crush test. Ultimately, the recommendation from TC-406 was considered by working group WG-2 of the 1982 IAEA Advisory Group AG-365 which, in turn, proposed a slightly modified text with respect to which packages would be covered. That was introduced into the 1985 revision to the Regulations [24] as the specified crush test environment. This paper notes AG-365 recommended that the test should be required for packages that:

- contain normal form material,
- have contents greater than 1000 $A_2$. 

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• have a mass less than 500 kg, and
• have a density less than 1000 kg/m$^3$.

The second meeting of the Advisory Group performing the comprehensive review of the regulations that was convened in 1982 (AG-365.2 [RBP042]) considered the proposals for a crush test. The AG agreed that a "crush test for relatively lightweight, low to moderate density Type B packages intended to contain significant amounts of normal form radioactive material" should be added to the regulations. This recommendation was the result of extensive deliberations by a working group at the AG, which considered the report of TC-406.28 (September/October 1981) and seven additional papers provided by various Member States, which included (a) a consultants review by S. Williamson, (b) two papers submitted by the Federal Republic of Germany, (c) a paper submitted by France, and (d) three papers submitted by the US. One of the US papers is found as [RBP043]\(^\text{16}\).

Following extensive deliberation, the WG at AG-365.2 [RBP051] recommended a crush test as proposed by TC-406; and the plenary of AG-365.2 agreed with that proposal.

During the AG-406 [RBP049] deliberations, convened in 1983, the density of the packages to which the test applied was briefly discussed. One Member State had recommended that the density threshold be raised from 1000 kg/m$^3$ to 1500 kg/m$^3$; however that proposal was not accepted by the Advisory Group.

As noted in Section 6.1 above, in the mid-1970s, McClure [58] used the Clarke, et. al. [47] data to predict the percentage of accidents for small packages transported in a given mode, where the accident environment would be expected to be equal to or less than the package qualification standards. His results for the then existing package tests were summarized in Table 1. However, the crush environment was not addressed in the regulations at that time, but McClure also evaluated the crush environment for road, rail and air transport. He noted, for surface modes of transport (having addressed the road and rail modes of transport, but not water mode), that "The protection levels provided by existing qualification criteria for surface transport, i.e. truck and rail, is almost complete. .... nearly 100% of all truck and rail accidents are equal to or less than the severity of existing qualification criteria. It must be recalled that crush is an environment that is not specified in existing qualification criteria."

Pope et al. [64] summarised an early test in the US where a highway vehicle carrying various radioactive material packages was impacted into a massive barrier at a speed of 66 km/h (41 mph). The vehicle carried 33 packages of six different types which, at the time (circa mid-1960s), were used for the transport of U-irradiated nuclear materials in the USA. Many of the packages experienced significant dynamic crushing and were severely deformed. In a second series of tests performed in the UK, three road vehicles loaded with various combinations of excepted, type A and type B packages were impacted into a 41 t barrier at velocities ranging between 35 to 111 km/h (22–69 mph). At higher impact velocities, many of the light weight, low density excepted and type A packages were significantly damaged. These tests were part of the efforts that ultimately led to identifying the potential need for a dynamic crush test in the transport regulations.

One test that was performed in Germany that added to the data considered during the deliberations on dynamic crush [65] elaborated on an evaluation of the resistance of light packagings against stresses caused by crush forces. Tests were performed on the 18B, L-10 and

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\(^{16}\) Of the seven papers considered by the AG-365.2 WG assigned to deliberate on crush, only [RBP043] was found by consultants in their review of the Agency’s archives. Reference [RBP043] included one of the other papers considered by the WG (i.e. “Highway Vehicle Impact Studies: Tests and Mathematical Analyses of Vehicle, Package, and Tiedown Systems Capable of Carrying Radioactive Material”, by O. A. Kelley and W. C. Stoddard (ORNL, 1970)) Reference [RBP043] also includes excerpts from some of the other papers considered at that meeting.
FL-10 package designs where each test consisted of dropping a “crush load of 2.000 kg from a height of 9 m onto the packaging lying flat on the unyielding target”. All three package designs experienced significant deformation, but in all cases the containment system maintained its integrity although failure in a subsequent thermal test could be expected.

A paper by Pope and Wille [66] provides expanded elaboration on the information and data used in the deliberations in the late 1970s and early 1980s leading to the specification of the dynamic crush test. The test specification resulted, in part, from consideration of the results of: (a) early analyses by J. D. McClure of SNL in the USA (e.g. see [58], [67], and [68]); (b) testing of various packages to different crushing environments performed from the mid-1960s through the early 1980s by various groups including: (i) Amersham International, UK (see [69]), (ii) the US AEC and the US Department of the Army (DOA) (see [70]), (iii) SNL, USA [70 – 72]); (iv) the CAECB, Canada [73]; and (v) Bundesanstalt für Materialforschung und-prüfung (BAM)/West Germany [74].

Chevalier, et. al. [75] examined the relationship between kinetic energy of a conveyance carrying radioactive material packages (where the larger the vehicle, the greater the kinetic energy would be at a given speed), and the mass of the package(s) carried. This assessment then compared the results to the energy imparted to a package due to the 9 m drop test and the 1 m puncture test. The conclusion was that for packages under 500 kg, “the aggression level required by the regulations seems insufficient in comparison with possible accidents, and experience shows that the packages in service in France display a resistance capacity closer to probably accidents than the minimum requirements of the regulations.” This study added to the body of data supporting the addition of the dynamic crush test.

The dynamic crush test, as added to the regulations in 1985 (see para. 548 of Reference [24]), was applied to low density, light-weight packages with radioactive contents greater than 1000°A₂; where this requirement applied to fissile and non-fissile contents alike. That same requirement is included in SSR-6 (see para. 659 of Reference [1]) for non-fissile contents. However, in the 1996 edition of the regulations, the dynamic crush test was imposed on all light-weight, low-density packages containing fissile material, without regard to the activity of the contents (see para. 682 of Reference [82], and para. 685 of Reference [1]). The basis for imposing that requirement on fissile material packages, even for contents less than 1000 A₂, is elaborated in CS-2452 [RBP045]. CS-2452 notes that the Third Revision Panel leading to the 1996 edition of the regulations had recommended (and the consultants at CS-2452 agreed) that “the most limiting of the mechanical tests in para 727(a) or 727(c) should be performed”; where the sub-paragraph references 727(a) and 727(c) are to the 9 m drop test and the dynamic crush test, respectively.

10.4.6. Technical Basis for Accident-simulating Thermal Test

The test for demonstrating a package’s ability to withstand exposure to a severe thermal environment accident condition is specified in para. 728 SSR-6. The following subsections address (a) the historical background in the development of the accident-simulating thermal test, and (b) later studies that have been undertaken to better evaluate the adequacy of this test.

**Historical Background in the Development of the Accident-simulating Thermal Test**

In considering potential fire environments, Messenger and Fairbairn [52] indicate that the IAEA’s Panel considered the frequencies, probabilities and many other factors that can work together in defining the environment a package might experience in a severe accident. These included:

- types of fuel, quantities of fuel, rate of spillage of fuel and dispersal of spilled fuel;
- possible ranges of temperatures in a fire, and associated effects of size of fuel source and effects of oxygen supply (wind);
- duration of fires; and
• size and mass of the package.

It was recognised that the maximum temperatures achieved in a fire are typically the result of a "local torching", which would not provide a significant threat to large packages due to the localized nature of the heat source. Further, they noted that melting of materials could be a reasonable indicator of effective or average flame temperatures, for which it was shown that large fires in railway accidents had resulted in the following:

• zinc (with melting point of 419°C) was melted,
• aluminium (with melting point of 660°C) was partially melted,
• glass (with melting point of about 1000°C) sagged but was not melted, and
• steel (with melting point of 1500°C) was not melted.

After consideration of the data presented, and tests (both open fire and oven environments) that were then being used and noting that some of these tests precluded any intervention (i.e. quenching of combustion of burning packaging elements) following thermal exposure until package temperatures had begun to drop, it was recommended exposure "to a furnace temperature of 800 °C for 30 minutes with no quenching until after the temperature of the interior has started to fall".

In elaborating on the discussion relative to the thermal test, Appleton and Servant [25] stated that there "was considerable discussion on the kind of fire to which a package might be exposed. The majority opinion was in respect of a large conflagration as might occur when a tank of petrol or kerosene spilled and took fire, but reference was also made to 'torching' flames from a ruptured compressed gas tank vehicle. Temperatures in the order of 1000 °C were considered relevant". They further noted that reported tests in open fires provided thermal environments very similar to those attained in hot wall, 800 °C oven tests. It was further noted [25, 26] that the basis for the average temperature was initially established using work of various individuals, including that of Bader [76] where, following the detailed analysis of a number of open pool fire tests and consideration of work of others, he concluded that "an exact prediction of temperatures expected in a particular fire cannot be made. Examination [of data] which shows the wide range of fire environments measured in 'similar' fires, indicates the difficulty one would have in predicting the temperatures expected in a given fire. On the other hand, the range of fire temperatures to be expected can be stated with some certainty, and over a large number of tests, the fire temperatures will produce an average. This average turns out to be approximately 1850 °F".

The average temperature of 1850 °F proposed by Bader equates to 1010 °C. The average fire temperature of 1010 °C is, of course, higher than the 800°C that was ultimately used in the early Regulations, and continues to be used today. Fairbairn and George [26] stated that severe transport fires "seldom last more than half an hour, ... and information on the temperatures attained suggests that although flame temperatures of liquids such as petrol can be about 1000 °C, such peak temperatures are reached only very locally by metallic material involved in the fire".

The notes of the Panel Meeting convened in mid-1964 [RBP036] emphasized that the aim of the thermal test "was not to reproduce the conditions likely to arise in the transport fires (this was almost impossible) but to provide a means of ensuring packaging of a reasonable standard". The Panel agreed that:

1) "the area of the flame was important in order to provide black body emissivity";
2) "such an area was related to the package size in order to have 3 feet of luminous flame width around the package in order to simulate black body conditions";
3) "it would be necessary to define (a) coefficient of emissivity of flame or furnace walls, and (b) coefficient of absorption of heat by package surface";
4) "one test should be specified – leaving it open for other tests to be used if they were of comparable effects"; and
5) "there should be no endeavour to cover the worst conditions in transport otherwise we were back at the maximum credible accident concept".

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After extensive discussion on these and other related issues, the Panel agreed to the following text:

Following much deliberation, the experts felt it necessary to consider all factors in establishing the thermal test condition, not just the maximum average attainable temperature in a “perfect” fire situation. The ramifications of accounting for multiple, “real-life” parameters were considered including, inter alia: (a) radiant, conductive and convective heat inputs; (b) exposure scenarios (e.g. the height of the base of the package above the fuel reservoir, the almost unlimited number of positions that a package could have in an fire during an accident, and the types of fuel that could be involved in fires), it was recognized that the specification of the thermal test required definition of:

- an effective source (i.e. flame) temperature and effective flame thickness where, for pool fuel fires, this requires consideration of such parameters as – fuel type, size of package, mass of package, size of pool (too small and the flame is not luminous, too large and the flame suffers from oxygen starvation), location of package above the pool, and wind effects;
- emissivity coefficient of the heat source (i.e. the flame and its luminosity);
- absorption coefficient of the package surface;
- duration of exposure;
- support of the package at specified height; and
- whether the package should be cooled following termination of heat source exposure.

These deliberations resulted in inclusion of the statement in the Regulations that any thermal test shall be considered as satisfactory provided that the parameters for satisfying the test were then specified in terms of:

- source temperature (800 °C),
- duration of test (30 min),
- source emissivity (0.9),
- package surface absorptivity (0.8),
- flame thickness of not less than 0.7m (2 ft) and not more than 3m (10 ft),
- the flame must surround the package during the entire test, and
- there would be no intervention after exposure to the thermal source until the inner components of the package began to cool.

Servant [54] elaborated on the deliberations of the Panel of technical experts that convened in mid-1964, noting the many issues considered that are associated with the thermal test. On pages 2 through 14 of Ref. [54] deliberations based on inputs from the U.S., the U.K. and the Eurochemic Company are documented in detail. Issues addressed included (a) open pool fire tests versus oven tests, (b) the size of the package versus the size of the open pool and the size of the oven, (c) the thickness of a luminous flame, (d) the temperature to be reached by the package in the test and the length of time it should be required to remain at that temperature, (e) the heat input to the package, (f) the coefficient of emissivity of the flame or furnace wall, (g) the coefficient of absorption of heat by the package surface, (h) whether high humidity could depress the flames in an open pool test, (i) the effect of wind upon flames in an open pool fire, (j) the height of the bottom of a package above the fuel reservoir, (k) the choice of fuel for a pool fire, (l) the depth of the fuel in the pool and its effect on the height of the walls of the pool retention system, and (m) whether to allow mechanical cooling of the package immediately following termination of the fire test.

From this extensive discussion, it was recommended by that panel of experts that the text for the fire test for the 1964 Edition of the Regulations should be as follows:

“Any thermal test employed shall be considered satisfactory provided that the heat input to the package is not less than that which would result from the exposure of the whole package to a
radiation environment of 800 °C for 30 minutes with an emissivity coefficient of 0.9 assuming the surfaces of the packages had an absorptive coefficient of 0.8”.

Fairbairn and George [26] discussed the positioning of the package so that its lower surface would be 1 m above the surface of the burning fuel, and that the package should be supported “such that it does not prevent direct exposure of any significant area of the package to the heat generated”, with a view to ensuring maximum damage to the test package. They further emphasized that an open-fire test method or appropriate furnace test methods which are “equally considered to meet the requirements of the general specifications. There are two main advantages in giving this open-fire test; first it can be conducted with relative ‘home-made’ facilities without the need for much detailed work by highly qualified scientific personnel, and second, the conditions of an open-fire have the merit of being seen to be similar, in their essential aspects, to those of an actual transport fire.

With minor changes in wording this is essentially the test that exists in paragraph 728 of SSR-6 [1] today; and much of the discussion contained in Appleton and Servant [25] has been included in the advisory material on this test contained in TS-G-1.1 [2, 3].

They further discussed the fire duration, noting that “... when the actual heat input to the interior of the package is examined it can be shown, particularly for large packages, that a test involving a 30 min period of exposure to heat input, and a subsequent natural cooling period until the innermost temperature has started to fall before any artificial cooling is applied, might well be more severe in its effect on the package than one in which heat is applied for 60 min according to a specified time-temperature curve with artificial cooling applied immediately afterwards”.

Another topic addressed in the 1964 panel discussions [RBP004] was whether or not to allow artificial cooling of the package following thermal exposure. It was agreed that this would not be allowed. Specifically, the experts noted that “there is a considerable body of opinion that the post exposure conditions up to thermal equilibrium, prohibiting the use of artificial cooling, should also be specified.”

In preparation for the issuing of the 1985 edition of the regulations [24], the thermal test was re-considered at various meetings. The technical committee that was convened in August 1979 [RBP026] considered the adequacy of the test as then specified in para. 720 of the 1973 regulations as amended 1979 [102]. Initially, a working group (WG) assigned this task identified to possible deficiencies: (1) the adequacy of the test in terms of providing adequate protection against real fires, and (2) the perceived need for more closely specifying the provision as a test and not a calculational method.

Data then available on thermal environments were considered by the TC-272 WG. The WG proposed text alternative to that in the regulations then effect [102]. This text, found on page 32 of the TC-272 report [RBP026], and an associated summary of on-going thermal study programmes served as a starting point for further evaluations and discussion leading to the 1985 edition of the regulations [24].

10.4.7. Technical Basis for the Water Immersion Tests

The tests for demonstrating a package’s ability to withstand water immersion is covered in SSR-6 by paras 729 – 730. The following two subsections address (a) the 15 m water immersion test that is imposed on all Type B and Type C packages; and (b) 200 m immersion test imposed on all Type C packages, and on those Type B packages containing more than $10^5$ A₂ quantities of radioactive material.

Historical Background for the 15 m Water Immersion Test

During the deliberations of technical experts prior to issuing the 1964 Edition of the Regulations, discussions ensued as to what depth of water should be specified for the
immersion test. The second technical panel meeting [54, RBP036] noted that at an earlier meeting a depth of 50 m was introduced. Fairbairn, at this second meeting that:

“The U.K. has carried out the review and it was found that 15 m was a more appropriate figure for the depth of water at quays and wharves at which packages of radioactive materials could be dropped during loading or unloading. (It was subsequently learned that the 50 m was a figure in respect of an offshore loading anchorage in Naples Bay where a special radioactive material consignment was loaded). There was considerable discussion and it was thought that the type B tests were adequate to ensure that the containment vessel would withstand the 50 m depth without referring to that figure.

“IT was eventually agreed however to keep the requirement but to reduce the depth to 15 m.”

Fairbairn and George [26] elaborated further on the test as follows: "For practical purposes the depth of water in harbours, rivers and canals in which packages might be dropped is unlikely to exceed 50 ft; recovery up to such a depth is most probable." They then noted that "As with Type A packaging, the Agency’s revised regulations specify certain packaging design requirements not covered by test procedures, for example that the containment vessel must remain intact at a dept of 15 m in water" (i.e. 50 ft of water).

IAEA Safety Series No. 7 [77], states that "While immersion at depths greater than 15 m is possible, this value was selected to envelop the equivalent damage from most transportation accidents. In addition, the potential consequences of a significant release would be greatest near a coast or in a shallow body of water. The eight hour time period is sufficiently long to allow the package to come to a steady state from rate dependent effects of immersion (e.g. flooding of exterior compartments)."

During the early deliberations in preparation for issuing the 1985 edition of the regulations [24], an Advisory Group meeting (AG-144) was convened in December 1977 [RBP038]. One topic considered was the duration of 8 hours required for the 15 m immersion test. AG-144 noted that "Although arbitrary it is considered that the duration (of 8 hours) is sufficient to indicate leakage. The requirement of 15 meters was derived from depths of water that may exist in port."

Historical Background for the Enhanced 200 m Water Immersion Test

This section addresses the technical basis for testing packages to the enhanced water immersion test specified for certain Type B packages and for Type C packages as specified in Section VII of the 2012 Edition of SS-6 [1], in para. 730.

The inclusion of the 200 m water immersion test for certain packages came about as a result of deliberations by international panels of experts convened in the late 1970s and early 1980s. The recommendation for the 200 m immersion test emanated from the 1981 meeting of the IAEA Technical Committee on Transport Package Test Standards (TC-406) [62]. This meeting was preceded by earlier technical committee, consultants services and advisory group meetings as well as meetings of the Standing Advisory Group on the Safe Transport of Radioactive Material (SAGSTRAM).

During the TC-406 meeting, one working group addressed in detail the issue of the potential need for a water immersion test deeper than the 15 m immersion test that was then applied to all Type B packages. The working group considered seven documents relating to this issue. The proposal for deep water immersion was, at that time, intended for a package containing "irradiated nuclear fuel with radioactivity greater than 37 PBq (10^6 Ci), other than special form radioactive material". Discussions at the working group led to the conclusion that the intent of the 200 m immersion test was to facilitate recovery of the package if it we lost from a vessel, or if a vessel with the package were to sink, on the continental shelf. For this reason, the acceptance requirement was not containment such as no more than an A2 quantity release in a week, but
rather that it would withstand the immersion "without rupture", which would thereby facilitate recovery of the package with its solid contents.

AG-365.2 [RBP042], which convened in March 1982, recommended adoption of the immersion test as proposed by TC-406 with minor modifications. It emphasized it must be kept in mind that "the purpose of the requirement is to assist in recovery of a spent fuel cask submerged to depths of 200 m, and is not required for either criticality or health safety reasons, the acceptance criteria have been retained as 'rupture' and reference to criticality deleted".

The advisory/explanatory material that was prepared during this period of time and later included in TS-G-1.1 (ST-2) [78] noted that "Various risk assessments have been carried out over the years for the sea transport of radioactive materials. It cites two of them as being typical [79, 80]. Nagakura, et. al., [79] considered the transport of spent fuel by sea in areas around Japan. This study considered the probabilities of accidents and associated accident scenarios were considered, the possibility of release of radioactive material from casks involved in such accidents, considering corrosion of the casks in sea water, and then estimated radiation doses to individuals and the population. Immersion depths of 500 m and 2 000 m were selected. The estimated doses were found to be at or below 1 percent of the safe limit for the public as recommended by the ICRP, and at or below 0.1 to 1 percent of the man-year/year national dose of radiation from natural effects in Japan. This was part of the justification for limiting the deep immersion test to the more shallow depths commonly associated with the continental shelf.

Another early study that illuminated the potential effects of loss of packages of radioactive materials at sea was prepared by Rhoads and Heaberlin [81], which is a summary of Heaberlin, et. al. [80]. This study considered the loss of a package containing spent nuclear fuel and plutonium into the ocean – both on the continental shelf and in deeper water. Estimates were made assuming ultimate failure of the package, of radioactive material release, uptake by marine life, and subsequent consumption of the marine life by man. Releases were assumed to range from immediately upon submersion to as long as 10 000 years, and all aspects of the study were viewed to be conservative in nature. The results showed that the exposure could be approximately 1 000 times higher if the package were lost on the continental shelf and not recovered, as compared to loss in the deep ocean and not recovered. The study concluded by stating that "For many of the loss scenarios, recovery of the shipping container before a significant portion of its contents are released also a very real possibility. Despite this conservatism, doses that have been calculated seem relatively modest. The maximum population dose calculated is comparable to the dose that population could be expected to receive from natural background sources".

Early studies such as those reported in these three documents [79 – 81] served as a basis for deciding that the deep water immersion test should be limited to a 200 m depth; that recovery at greater depths could be undertaken but would not be necessary.

10.5. Technical basis for industrial packages IP-1, IP-2 and IP-3

The single designation of "packagings for materials that were judged to be 'inherently safe'" and were initially transported in "strong industrial packages" was expanded in the 1985 edition of the Regulations [24] to three types of industrial packages, i.e. to the IP-1, IP-2 and IP-3 packages. This expansion followed a graded approach which was based on the hazard posed by the contents.

The structure of the industrial package classification was the result of panels of experts, convened during the efforts leading to the 1985 Edition; where the experts considered the diversity of forms of materials previously determined to be "inherently safe"; and agreed during their deliberations that a more prescriptive approach was needed.

For example, the technical committee that was convened in 1979 (TC-272 [RBP026]) considered what were then identified as LSA and LLS materials, and the use of strong industrial packaging for these materials.
The technical committee acknowledged that “There are at present no regulatory requirements for the quality of packaging for LSA materials other than it should be strong enough to prevent the loss of material during transport. The IMCO subcommittee on the Carriage of Dangerous Goods has however proposed that packaging for LSA materials should be tested to the UN Group III tests (i.e. for goods presenting only minor danger). These tests require, inter alia, a drop test of 0.8 meters. IMCO has agreed to defer action on this matter pending advice from the IAEA.” TC-272 further noted that “there is a need for some minimum packaging standards for LSA materials”. Thus, TC-272 deliberations can be viewed as a factor in initiating a closer look at the packagings used for these materials.

TC-406 [62] (convened in Tokyo in 1981) continued the deliberations on the package requirements that should be established for what were then called ‘low hazard materials”. The second draft of the new edition of the regulations that had been produced prior to the convening of TC-406 listed four integrity levels for LSA materials and SCO. The TC explored reducing the number of integrity levels from four to three. At that time, Levels 1 and 2 were very similar and imposed no test requirements, and also concluded that “there is only little difference in the two sets of requirements in terms of radiological safety”.

As a result, the relevant TC-406 WG recommended that “a merge of Levels 1 and 2 would be possible. The merge is to be done such that to the combined levels 1 and 2 the present Level 2 requirements apply. Such a merge would marginally increase safety”. The WG also noted that (as discussed in Section 8.6.2 of this Technical Basis Document), the materials to which these package requirements were to be applied were for (a) three types of Low Specific Activity (LSA) materials, identified as LSA-I, LSA-II and LSA-III (see paras 226, and 408-411 of SSR-6 [1]); and (b) two types of Surface Contaminated Objects (SCO), identified as SCO-I and SCO-II (see paras 241, and 412-414 of SSR-6).

Ultimately, TC-406 recommended the packaging levels for LSA material and SCO that are summarized in Table 10-4.

In turn, AG-406 [RBP049], which was convened in 1983 to establish the changes recommended for the 1985 edition of the Regulations agreed that the industrial packages to be used for transporting LSA material and SCO should be structured into three levels, denoted as Industrial Package Type 1, Type 2 and Type 3, where the package design requirements followed a graded approach with:

(a) the general requirements for all packagings and packages required for IP Type 1,
(b) the requirements for IP Type 1 plus some of the requirements for Type A packages for IP Type 2, and
(c) the requirements for IP Type 1 plus most of the requirements for Type A packages; where
(d) an additional constraint was proposed for LSA-II liquids transported by air.

For comparison, the packaging integrity levels ultimately specified in the 1985 edition of the regulations [24], and also specified in each edition since including in SSR-6 [1], are summarized in Table 10-5.
Table 10-4. TC-406 Recommendation for Packaging Levels for LSA Material and SCO*

<table>
<thead>
<tr>
<th>Material</th>
<th>Packaging Level*</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Exclusive Use</td>
</tr>
<tr>
<td>LSA I</td>
<td>I</td>
</tr>
<tr>
<td></td>
<td>I</td>
</tr>
<tr>
<td>LSA II</td>
<td>I</td>
</tr>
<tr>
<td></td>
<td>II</td>
</tr>
<tr>
<td>LSA III</td>
<td>II</td>
</tr>
<tr>
<td>SCO I (LCO I)</td>
<td>I</td>
</tr>
<tr>
<td>SCO II (LCO II)</td>
<td>II</td>
</tr>
</tbody>
</table>

* Note: in 1981, TC-406 was identifying SCO as LCO. In addition, TC-406 was using Roman numerals to identify different levels of packaging whereas when the regulations were published the packaging levels were (and continue to be) identified using Arabic numerals.

Table 10-5. Industrial Package Integrity Requirements for LSA Material and SCO*

<table>
<thead>
<tr>
<th>Contents</th>
<th>Industrial Package Type</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Exclusive use</td>
</tr>
<tr>
<td>LSA-I</td>
<td></td>
</tr>
<tr>
<td>Solid**</td>
<td>IP-1</td>
</tr>
<tr>
<td>Liquid</td>
<td>IP-1</td>
</tr>
<tr>
<td>LSA-II</td>
<td></td>
</tr>
<tr>
<td>Solid</td>
<td>IP-2</td>
</tr>
<tr>
<td>Liquid</td>
<td>IP-2</td>
</tr>
<tr>
<td>LSA-III</td>
<td>IP-2</td>
</tr>
<tr>
<td>SCO-I**</td>
<td>IP-1</td>
</tr>
<tr>
<td>SCO-II</td>
<td>IP-2</td>
</tr>
</tbody>
</table>

* This table depicts the industrial package integrity requirements as specified in the 1985 edition of the regulations [22], and have remained unchanged since that time.

** Beginning with the 1985 edition of the regulations, under certain limiting conditions, LSA-I and SCO-I may be transported unpackaged.

Ultimately, a constraint on any liquid transported by air, not just LSA-II liquids, was incorporated into the 1985 edition of the Regulations as para. 517.

Thus, three different types of packages associated with the transport of these LSA material and SCO designations resulted from these deliberations. These three types of packages were ultimately identified as IP-1, IP-2 and IP-3 (with “IP” standing for Industrial Package”); with increasing design requirements imposed when moving from the IP-1 to IP-2 package types, and then when moving from IP-2 to IP-3 package types. The type of industrial package to be used for a given LSA material or SCO depends upon:
(a) the physical nature of the contents, i.e. liquid or solid (see e.g. Table 10-5 below),
(b) whether they are shipped under exclusive use or not under exclusive use (see e.g. Table 10-5 below), and
(c) the conveyance activity limits depending upon the mode of transport (see e.g. Table 6 of SSR-6).

All of these criteria merge together to form a comprehensive set of graded requirements for protection of man and the environment during the transport of LSA material and SCO.

10.6. Technical basis for Type C package tests

This section addresses the technical basis for testing packages to the Type C radioactive materials package test requirements intended to represent environments that could occur in aircraft accidents as specified in Section VII of the 2012 Edition of SSR-6 [1], in paras 734 – 737. Also addressed are the relevant package design requirements associated with these tests, as specified in paras 669 – 672 of SSR-6.

As noted earlier in Section 3.2, the requirements for Type C packages were added to the Regulations in the 1996 Edition of the Regulations [82]. The addition of this new type of package resulted from international deliberations prompted by the establishment in the US of specific package requirements for air transport of plutonium. Work performed in the 1970s by McClure [58] using data provided by Clarke, et. al. [56] (see Table 10-1 above) showed that the then existing package test standards for the hypothetical accident conditions did not appear to sufficiently address concerns for air transport.

During the early deliberations in preparation for issuing the 1985 edition of the regulations [24], an Advisory Group meeting (AG-144) was convened in December 1977 [RBP038]. One topic considered was the adequacy of the thermal test then specified. AG-144 noted that "the IAEA thermal test was adequate for surface (including marine) transport but that it may not fully cater for air transport insofar as its duration is concerned".

In 1974, Clarke, et. al. [83] summarized the results of a study on accident severities for air transport, by (a) categorizing environments into categories ranging from "Minor" to "Extreme", and (b) identifying the "most likely number of times an aircraft will experience an accident more severe than the severity level indicated". The category immediately below "Extreme" was defined as "Extra Severe". Extra Severe entailed exposure to an 1850 °F (1000 °C) fire for time periods between 40 and 70 minutes, and an impact environment equivalent drop heights of 50 to 100 ft. For the Extra Severe category, it was projected that an aircraft might experience that event once in 10 billion aircraft miles.

The basis for the US-specific requirements was established as a result of efforts by personnel at the US NRC and SNL who evaluated various parameters associated with air accidents. These efforts [84–87] ultimately resulted in a set of very stringent package design requirements for air transport of plutonium (in §71.74 of [6]) that satisfied US law [88]. Later studies essentially confirmed the adequacy of the earlier studies. For example, von Reismann and Hartman [86] evaluated the engine fragment threat to small containers carried on commercial jet aircraft in the mid 1970s. For example, a later study by Harding and Pierce [RBP013] determined that the "probability of a high-energy rotor burst fragment from four generic aircraft engines striking one of the containment vessels aboard a transport aircraft is approximately 1.2 x 10^-9 strikes/hour"; which is an extremely low probability.

To support this effort in the US, the ad hoc Committee on the Transportation of Plutonium by Air of the Aeronautics and Space Engineering Board of the US National Research Council evaluated the work of the NRC and SNL and concluded that "The committee is confident that the qualification criteria described or referenced in this report will result in a packaged container that will not rupture in the crash and explosion of a high-flying aircraft" [89].
Following the implementation of these requirements in US domestic regulations, international discussions ensued with regard to the adequacy of the Type B package design requirements for large quantities of radioactive material transported by air. Ultimately, revision panels agreed that a new type of package, with package test requirements more robust than those for Type B was required for such air transport. This then led to establishing a threshold above which these test requirements would be applied and the package so designed would be designated a Type C package requiring competent authority approval.

The Type C package requirements in the IAEA Transport Regulations only apply to radioactive material that is to be transported by air in quantities greater than 3000 A₁ or 100 000 A₂ (whichever is lower) if the material is special form, or 3000 A₂ if the material is other than special form, unless the material has been qualified as low dispersible radioactive material and has been authorized for air shipment in its design certificate (see, e.g. para. 433 of the 2012 Edition of SSR-6 [1]).

With respect to the crush environment that may exist in an aircraft accident, McClure and von Reismann [67], and McClure and Hartman [90] evaluated the potential for crush of small packages having a mass of 500 kg or less in aircraft accidents. This study showed that the crush loads could range from about 311 KN (70 000 lbs) for scenarios with few conservatisms, to as high as 939 KN (211 000 lbs) for scenarios with multiple conservatisms. The test requirements for a Type C package include the imposition of the dynamic crush test specified in para. 727(c) of SSR-6 irrespective of the mass or density of the package being transported by air subject to paras 417, 418 and 669 of SSR-6.

With respect to the requirement in Para. 670 of SSR-6 which specifies that a Type C package shall be capable of withstanding burial in a relatively low conductivity medium for a time period sufficient for steady state temperatures in the package to be achieved, Malesys [46] summarized an assessment of the thermal behaviour of a cask buried in soft ground (like a swamp). He noted that TN International studied from 1994 to 1998 the burial of a flask in soft ground like a swamp. The study assessed the probability of such an accident, and the evaluation of the depth to which the flask sinks, and the gradual heating of the components of the flask. He noted that the gradual heating arises from the loss of the heat dissipation capability of a package when it is buried, and the fins that surround the flask lose their efficiency. It was determined that the first sensitive components were the elastomeric sealing O-rings; and that in the case of partial sinking, sinking of half the flask should not cause the sensitive components to overheat. It was also demonstrated that even in the case of a complete sinking of the flask loaded with the maximum authorised heat load would not lose its leaktightness before 2 days. It was concluded that this duration is quite acceptable when compared to the response time needed to set up a cooling system. Although this assessment focused on large flasks generating significant quantities of heat, the tests can be related to Type C packages which would probably be much smaller and also would not generate such significant levels of heat.

It is noteworthy that the Type C package test requirements [17] (see paras 669-672 of the 2012 Edition of RSS-6) were initially instituted in the IAEA Regulations for radioactive materials in quantities in excess of the values specified in para. 416 of the 1996 Edition of the Transport Regulations [82]. However, these requirements now also apply to fissile material packages transported by air irrespective of the activity in the package (see para. 683 of SSR-6).

10.7. Technical basis for uranium hexafluoride package tests

This section addresses the technical basis for testing requirements for packages designed to contain 0,1 kg or more uranium hexafluoride (UF₆) as specified in the 2012 Edition of SSR-6 [1].

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[17] The test requirements established by the US NRC for packages of plutonium transported by air are more robust than those established for Type C packages. For example, the NRC requirement is for an impact onto an unyielding target at a velocity of 128 m/s (§71.74(a)1 of [8]); whereas the impact velocity specified for a Type C package is 90 m/s (para. 737 of SSR-6 [1]).
in paras 632(a), 632(b) and 6302(c). UF₆ has been transported for decades with no significant transport incidents that have resulted in serious consequences from either the radiological or the chemical nature of the material.

The IAEA Transport Regulations (SSR-6 [1]) contain specific design, testing and package certification requirements for UF₆ that were first introduced in the 1996 Edition of the Regulations (at the time identified as ST-1) [82]. As a result of these requirements, UF₆ is the only material with specific requirements imposed through the Transport Regulations that are based on the chemical and not the radioactive nature of the material. In earlier IAEA publications, all mention of UF₆ in regulatory, explanatory and advisory text has been as an example of a radioactive material with other dangerous properties.

The UF₆-specific requirements introduced two new requirements: (a) an additional drop test and (b) a thermal test, which apply to the transport of non-fissile and fissile excepted UF₆. The other requirements were already in effect through the cylinder standards issued by the American National Standards Institute (ANSI) [91] and the International Standards Organisation (ISO) [92], or through common practice in the industry.

The transport of UF₆ takes place in cylinders that conform to ANSI N14.1 and ISO7195 standards [91, 92]. Furthermore, USEC-651 (formerly ORO-651) [93] has been used throughout the industry as guidance for handling of UF₆. Historically, non-fissile or fissile excepted UF₆ has been transported as LSA material.

The ANSI standard for UF₆ cylinders was first issued in 1971, is periodically updated, and has been used by the industry worldwide. The ISO standard was first issued in 1993 as an international alternative for the ANSI standard. In spite of slight differences between the two standards, e.g. format, both standards cover essentially the same cylinders.

The sinking of the Mont Louis near the coast of Belgium in 1984 and an accident in a conversion plant at Gore, Oklahoma, USA in 1986, triggered action at the IAEA to develop guidance for UF₆ transport; specifically to address the potential chemical hazard for the non-fissile and fissile excepted material in the form of natural and depleted UF₆. The transport of enriched UF₆ had to comply with the fissile requirements in the regulations, in addition to the general requirements only, and those packages were deemed to provide sufficient protection from a chemical hazard standpoint.

Biaggio and Vietri [RBP055] reported in 1987 on a study of the physical and chemical risks of transporting UF₆. It was concluded that “The present practice in the transport of UF₆ [8] ensures a level of safety equal or higher than the level of safety recommended for the transport of similar dangerous goods [16]”. ¹⁸

Ringot and Hamard [RBP058], a paper presented in 1988 (prior to the adoption of the additional tests for UF₆), addressed the equivalency of risk from the toxic and radiological perspective. After considering the chemical and physical nature of UF₆ in the transport environment, and its potential behavior in packages experiencing the hypothetical accident condition tests, they concluded the following:

“The regulations concerning the transport of radioactive substances provide a high degree of safety which is equivalent for the toxic and radiological risks in the transport of more than 1% enriched UF₆. This is not the case for less than 1% enriched UF₆, for which the chemical risk of the UF₆ is the main one. Two approaches then become possible to guarantee an acceptable level of safety. The first approach would be to apply the same constraints as for chemical products involving a similar type of danger, for example

hydrofluoric acid or chlorine, i.e. to follow the UNO recommendations for class 8. This is the approach which has been adopted in developing TEC-DOC 423. The other approach would be to apply criteria equivalent to those used for the radiological risk.

“It is this second approach which we recommend be examined in the long term. This approach is consistent with the one used by the IAEA to develop the regulations for the transport of radioactive substances, i.e. of first setting an objective then establishing criteria for reaching it, indeed it appears to be liable to guarantee a clear level of safety while leaving full scope for innovation.”

The perceived need on the part of some investigators for a thermal test of packages containing UF₆ was based on the chemical risk of UF₆ and/or the reaction products (especially HF), rather than on the radiological risk. Specifically, when UF₆ is released in air, it reacts with water vapour and forms uranium oxyfluoride and hydrogen fluoride. The reaction products have three toxic effects: (a) the uranium in the uranium oxyfluoride can act as a heavy metal poison, (b) the hydrogen fluoride can cause acid burns on the skin or in the lungs if it becomes concentrated, and (c) the fluorides can both cause fluoride poisoning if intakes are sufficiently large. For further information see NUREG-1391 [RBP054].

Several technical and consultants meetings were organised, resulting in TECDOC-423 [94] and TECDOC-608 [95], and an IAEA Coordinated Research Project (CRP) was started in 1992. The CRP dealt with the issue of UF₆ packages being able to withstand the IAEA Thermal Test without rupture. The CRP however did not reach conclusive results, and a summary document of the CRP drafted, but because of the inability of the participants to reach a consensus conclusion, it was not published [96]. Specifically, the unpublished CRP report concluded the following:

“The CRP provided much valuable data concerning the behaviour of large UF₆ cylinders in fires. However, the data are not sufficiently definitive that consensus could be reached by all participants that failure would or would not occur when a large UF₆ cylinder is exposed to the regulatory, 30 minute thermal test. Indeed three of the CSIs felt that failure is likely, whereas three did not.”

All this ultimately led to the incorporation of the specific requirements for UF₆ in the 1996 Edition of the regulations [82]. Relevant meetings in the final stages of the decision making were a CSM in June 1995 [97], the Revision Panel Meeting (RPM) in September 1995 [98] and the TRANSSAC meeting in February 1996 [99]. An overview of the history and the important events in this development is described in a PATRAM 2007 paper [100].

Para. 632 of SSR-6 reads:

“632. Each package designed to contain 0.1 kg or more of uranium hexafluoride shall be designed so that it will meet the following requirements:

(a) Withstand, without leakage and without unacceptable stress, as specified in ISO 7195 [12], the structural test as specified in para. 718, except as allowed in para. 634;

(b) Withstand, without loss or dispersal of the uranium hexafluoride, the free drop test specified in para. 722;

(c) Withstand, without rupture of the containment system, the thermal test specified in para. 728, except as allowed in para. 634.

In turn, para.634 of SSR-6 reads:

“634. Subject to multilateral approval, packages designed to contain 0.1 kg or more of uranium hexafluoride may be transported if the packages are designed:

(a) To international or national standards other than ISO 7195 [12], provided an equivalent level of safety is maintained; and/or

(b) To withstand, without leakage and without unacceptable stress, a test pressure of less than 2.76 MPa as specified in para. 718; and/or
(c) To contain 9000 kg or more of uranium hexafluoride and the packages do not meet the requirement of para. 632(c).
In all other respects, the requirements specified in paras 631–633 shall be satisfied."

The structural test specified in para. 718 of the Transport Regulations is just a copy of the requirements in ANSI/ISO standards; it does not constitute anything new.

The free drop test specified in para. 722 is a new requirement. It was added for UF₆ in comparison with earlier editions of the Transport Regulations (i.e. Safety Series No. 6 editions). The general requirements from Safety Series No. 6 did result in an IP1/IP2 package, for which no drop test is required."

The thermal test requirement as specified in para. 728 of the Transport Regulations is new. It was added for UF₆ as compared to the earlier Safety Series No.6 editions. In the process of developing the regulatory requirements and associated guidance for UF₆ transport, this requirement has been very controversial.

The test temperature in the regulations for other dangerous goods is 600°C, whereas the test temperature in the IAEA regulations is 800°C. The logic would call for a test temperature of 600°C, but the IAEA decided to select "its own" thermal test with a test temperature of 800 °C.

Since most natural and depleted UF₆ is transported in 48 inch cylinders (mainly in cylinders identified as “48Y cylinders”), this cylinder type has been the subject of investigation by several experts in several countries. Six experts from six countries contributed to an IAEA Coordinated Research Project (CRP), but the outcome of this project was not conclusive. Three experts concluded failure would be possible for exposure to a 30-minute, fully engulfing 800°C fire; and three experts concluded survival could be expected for exposure to a 30-minute, fully engulfing, 800°C fire.

The CSM in June 1995 [97] recommended to not incorporate the thermal test into the regulations, but the RPM [98] and the TRANSSAC [99] meetings did not accept this recommendation.

Further, as a result of the recommendations of the review panel meeting (RPM) [98] and the TRANSSAC [99] meeting, the CSM [97] recommended to accept the thermal mass of cylinders with 9000 kg UF₆ or more as sufficient for survival of the thermal exposure to the thermal test condition of para. 728.

The RPM [98] and TRANSSAC [99] meetings accepted this recommendation, but with the requirement that approval of the package design was to be subject to multilateral approval. Currently, such multilateral approval is required in North America and Russia; whereas unilateral approval is allowed in other countries including those in Europe and Japan.

In 2001, Doare, et. al. [RBP059] provided a summary of the release calculations made as a result of the TENERIFE programme. It was concluded by the researchers in France involved in that programme, that:

"The toxic effects of the HF from the hydrolysis reaction of the UF₆ are preponderant on the toxic or radiological effects of the UO₂F₂ in the case where the enrichment of UF 6 is less than 5%.

"The analysis carried out shows that the UF 6 releases from a type 48Y or a type 30B full container exposed to a long duration fire leading to the bursting of the container may have irreversible effects for populations up to 4 900 m from the release point within the meteorological conditions used. The reaching distance for 50% lethal consequences is, according to the analysis, 2 300 m.

"For package design approved according to a H(M) certificate (this would be the case for type 48Y containers not equipped with fire protection for which the risks of rupture due to
a severe fire are more probable), the feasibility of setting up in approximately half an hour a security perimeter located at 5 kilometres from the accident should be considered.

“...In addition, it remains to be confirmed that the possible leakages of UF6 from the valve and plug connections of a 48Y container, that occur a long time before the rupture, have a sufficiently limited flow so that a safety perimeter placed at least 500 metres from the container is adequate.”

Following conclusion of the CRP, and the publication of assessments such as that provided by Doare, et al. [RBP059], Werkoff, et. al. [RBP056] issued in 2006 an assessment of the results of the TENERIFE experimental programme upon which much of the work of the CRP was based, and also of the preliminary modelling which was undertaken as part of the CRP.

Werkoff specifically addressed the controversy that continued between the various participants in the CSM [97]. Werkoff, et. al. provided the following conclusions:

“In conclusion, for a 48Y cylinder filled to the nominal value, at the end of exposure to the regulatory 30 min fire test, solid, liquid and gaseous phases coexist inside the cylinder. The value of the pressure depends on the distribution of UF6 between phases. Estimation of this distribution is a complex problem that has not been adequately resolved. The TENERIFE tests were terminated before the most important portion of UF6 phase change in the cylinder could occur, by stopping the tests well before 1800 s.

“Thus, the terms of the controversy can be expressed as follows:

a. Either the existing data and analyses related to temperatures inside the TEN4 cylinder are correct and the pressure evolution after the cessation of heating led to the conclusion that, under the conditions of the IAEA fire test, the pressure would not reach the value expected to rupture the cylinder at the end of the regulatory period of 30 min. However, after the cessation of heating, the pressure would continue to increase for a very long time.

b. Or the existing data and analyses related to temperatures inside the TEN4 cylinder are incorrect and the oscillations of the conductivity (reported in Fig. 4), used in the simulations presented by Niel et. al. 2, 3 and Sert and colleagues, 5, 6 have a physical meaning. Then the pressure could reach the value expected to rupture the cylinder at the end of the regulatory period of 30 min.”

It is worth noting, that in publishing the work by Werkoff, et. al. [RBP056], a “Note from the Editor” of the international journal – Packaging, Transport, Storage & Security of Radioactive Material (RAMTRANS) – was included at the end of the published paper. That note stated: “As noted by the author in the abstract, the subject matter of the present study has been the subject of some controversy for several years. This was emphasised during the refereeing process. While the content and conclusions may not be universally accepted, it has been decided that the interests of technology and fairness would be best served by publishing the paper in its present form. The result may be to promote further discussion on all aspects of the controversy.”

Industry experience in applying the new regulatory requirements for UF6 packaging is found in a paper by Dekker [RBP057]. Dekker noted that: “a consortium of UF6 producers/users has worked together on the design and development, testing and certification of technical solutions for modification and optimisation of the existing packages to comply with TS-R-1.”

Dekker concluded, in part, that:

- “The thermal requirement for UF6 packages aims to avoid chemical consequences of an accident involving a large fire.
- Industry has developed additional technical measures for use, where and when necessary.
- The use of thermal protection is costly and increases conventional safety-related handling risks and the dose to transport workers.
- TS-R-1, through paragraph 632(c), allows for taking credit of the thermal mass of large UF6 cylinders as a risk-informed consideration.
TS-R-1 recognises multilateral approvals H(M) and unilateral approvals H(U); both are currently available for use in the next years, although in different regions."

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Note, source at end of a reference citation with blue shading, RBP has hardcopy that could be scanned; but time was spent adding text rather than finding and scanning these documents.


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11. PACKAGING AND TRANSPORT OF FISSION MATERIAL

Earlier text is has been reviewed by criticality experts and
is under revision based upon comments received.
12. APPROVAL AND STATUTORY REQUIREMENTS

Text yet to be developed
### 13. ACRONYMS

[See EndNote xviii](#)

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<th>Description</th>
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<tr>
<td>AEC</td>
<td>Atomic Energy Commission</td>
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<td>ACT</td>
<td>Accident Conditions of Transport</td>
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<td>ADN</td>
<td>European Agreement concerning the International Carriage of Dangerous Goods by Inland Waterways</td>
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<td>ADR</td>
<td>European Agreement concerning the International Carriage of Dangerous Goods by Road</td>
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<td>Advisory Group</td>
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<td>(The United States) Bureau of Explosives</td>
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APPENDIX 1 – Developing and applying the IAEA Transport Regulations – Implementing an international transport safety regime

The provisions for transport safety found in the IAEA Transport Regulations [1] become part of the basis used by the United Nations Economic Commission for Europe (UNECE) in issuing, every two years, an updated version of the UN Recommendations on the Transport of Dangerous Goods – Model Regulations [2]. These requirements in the IAEA Transport Regulations the UN Model Regulations, in turn, are essentially translated directly into the international modal organizations regulatory documents for transport of dangerous goods.

When a revised version of the IAEA Transport Regulations is issued, steps are taken at both the international and domestic level to implement the requirements therein as regulations governing these transport activities.

At the international level, the International Maritime Organization (IMO) issues its International Maritime Dangerous Goods (IMDG) Code [3], for maritime transport of all nine classes of dangerous goods; and the International Civil Aviation Organization (ICAO) issues its Technical Instructions for the Safe Transport of Dangerous Goods by Air [4], also for the air transport of all nine classes of dangerous goods. Both of these documents obtain their provisions for radioactive material transport from the IAEA Transport Regulations (currently identified as SSR-6 [1]) by way of the UN Model Regulations. The two documents are essentially applied worldwide by all States that are parties to the Safety of Life at Sea (SOLAS) and the Chicago Convention. In turn, the International Air Transport Association (IATA) issues its companion document, Dangerous Goods Regulations, which are binding upon all commercial air carriers.

For the other modes of transport, three international organizations deal with them on a regional basis:

- The UNECE’s Committee on Inland Transport maintains the European Agreement concerning the International Carriage of Dangerous Goods by Road (ADR) [5]. The ADR is mandatory for all members of the European Union and for any other State that is signatory to the Agreement. As of the beginning of 2012, there were 46 States signatory to the Agreement.
- The UNECE’s Committee on Inland Transport also maintains the European Agreement concerning the International Carriage of Dangerous Goods by Inland Waterways (ADN) [6]. The ADN is mandatory for all members of the European Union and for any other State that is signatory to the Agreement. As of the beginning of 2012, there were 17 States signatory to the Agreement.
- The Intergovernmental Organisation for International Carriage by Rail (OTIF) issues, bi-annually, and maintains Appendix C – Regulations concerning the International Carriage of Dangerous Goods Rail (RID), [7]. As of the beginning of 2012, the RID was mandatory for the 46 members of OTIF.

All three of these documents also obtain their provisions for radioactive material transport from the IAEA Transport Regulations by way of the UN Model Regulations.

Finally, at the domestic level, individual States apply the requirements, as applicable, coming from the international regulatory documents and/or adopt and implement provisions into their domestic regulations for designing and testing packages for transporting radioactive material safely. Thus, SSR-6 and/or the UN Model Regulations serve as the basis for these domestic regulatory documents.

The process of flowing the transport safety requirements from the IAEA Transport Regulations to international and domestic documents that make the requirements legally-binding is depicted graphically in Figure A1-1.
Thus, it is evident that the IAEA Transport Regulations document is key to promulgating regulatory requirements throughout the world for the safe transport of radioactive material.

References for Appendix 1


[7] Intergovernmental Organisation for International Carriage by Rail (OTIF), Convention concerning International Carriage by Rail (COTIF); Appendix C – Regulations concerning the International Carriage of Dangerous Goods Rail (RID), Bern (2011). {RP009}
APPENDIX 2 – The structure of the safety requirements

Figure A2-1 illustrates the structure of the safety standards at the time this document was assembled (2012).

The long-term set of Safety Standards includes a unified Safety Fundamentals (SF1), a General Safety Requirements (GSR) in seven parts applicable to all facilities and activities with a graded approach, complemented by a set of six facilities and activities Specific Safety Requirements (SSRs). The specific safety requirements for transport of radioactive material are found in the 2012 edition of the Transport Regulations (SSR-6) [1].

As illustrated at the bottom of the figure, the Safety Requirements are implemented through a set of general and specific safety guides.

References for Appendix 2

APPENDIX 3 – The technical basis for applying the graded approach

The design and testing of radioactive material packages is governed to a great extent by applying the concept of the graded approach, where the demands placed on the design are governed by the risk posed to humans and the environment of the radioactive contents, in the event that (a) the contents be released during transport, (b) accompanying shielding experiences a failure during transport resulting in an increase in radiation level; (c) if the contents are fissile material, the contents experience changes such that criticality could occur, or (d) the heat resulting from exposure to the normal conditions of transport or accident conditions combined with any heat generated by the contents result in degradation of the containment, shielding or criticality control components that would result in unacceptable consequences.

The concept of applying a graded approach dates back to the early development of the Regulations. For example, in 1979 Fairbairn [1] wrote that

“*The range of potential hazard of radioactive material requiring transport is very wide indeed and raises the problem of whether regulations should prescribe packaging design requirements linked with defined limits for package contents, or package design requirements, for example heat dissipation, whose implementation will restrict the contents. When the regulations were first developed, it was decided to use both approaches; this resulted in the Type A and B package concepts.*”

Fairbairn [1] further stated that:

“*While the Types A and B package design prescriptions provide for the safe transport of many nuclear materials whose potential hazard is in the medium and high part of the range, it is important that provision be made for the transport of materials of low potential hazard. For purposes of the regulations, such materials are classed either as ‘Low Specific Activity’ (LSA), ‘Low Level Solid’ (LLS) or ‘Items exempt from specific prescriptions’ by virtue of their small radioactive content. Containment standards less stringent than those set for a Type A package and closer to the standards for industrial or commercial packaging used for certain chemicals is justified for such materials provided that the material in the package or load is ‘inherently safe’.*”

Thus, during the early development of the regulations, considerations were given to at least four different types of packages, i.e. those:

(a) packages exempt from specific prescriptions due to their small radioactive content,
(b) containing materials of low potential hazard,
(c) containing materials of medium potential hazard (i.e. Type A packages), and
(d) containing materials of high potential hazard (i.e. Type B packages).

The four categories for packaging and shipping requirements were then described by Rogers [2] as shown in Table A.3-1.

Thus, the graded approach is used extensively throughout the Transport Regulations such that, the greater the risk posed by the contents of a package, the greater and more robust are the design and test requirements imposed on that package. The following outlines different ways by which the design and testing requirements satisfy the graded approach.

A.3.1. Applying a graded approach to package contents activity limits

At the outset of the development of the Regulations, it was decided that a radionuclide-specific limit would established for the maximum quantity of material that could be placed in a Type A package. The limit was first established (in the 1961 Edition of the Regulations) [3] on the basis of three groups of radionuclides. The grouping and limits changed thereafter (in the 1964 and 1967 editions of the Regulations) [4, 5] to a larger number of groups. Concurrently, the concept
of special form radioactive material with a different set of limits was implemented, moving away from the 1961 concept of a 'massive non-friable solid'.

Table A.3-1. Graded approach to packaging and labelling being used in the US in 1962 [2].

<table>
<thead>
<tr>
<th>Category</th>
<th>Types and quantity of radioactive material</th>
<th>Packaging Requirements</th>
<th>Labelling and Placarding Requirements</th>
</tr>
</thead>
<tbody>
<tr>
<td>First</td>
<td>Low hazard potential</td>
<td>Exempt</td>
<td>Exempt from labelling; vehicles to carry a yellow placard</td>
</tr>
<tr>
<td>Second</td>
<td>For each of the three radiotoxicity classifications, low to moderate hazard potential if released from a package</td>
<td>Specification packaging that withstands normal handling and minor accidents</td>
<td>Type A package label; vehicles to carry a yellow placard</td>
</tr>
<tr>
<td>Third</td>
<td>For each of the three radiotoxicity classifications, substantial hazard potential if released from a package</td>
<td>Retain contents “in the most severe accident considered credible” – designated a Type B package</td>
<td>Specification labelling, vehicles to carry a yellow placard</td>
</tr>
<tr>
<td>Fourth</td>
<td>For each of the three radiotoxicity classifications, high degree of hazard potential and serious contamination potential if released from a package; plus fissile materials or special nuclear materials</td>
<td>Specially approved Type B packages</td>
<td>Red label; vehicles to carry a red placard</td>
</tr>
</tbody>
</table>

In preparation for the issuing of the 1973 Edition of the Regulations [6], the concept of placing individual radionuclides in groups was eliminated, when the currently-used concept of $A_1$ and $A_2$ limiting values for each radionuclide was developed. A table listing those values first appeared in the 1973 Edition of the Regulations.

1961 Edition: For Type A packages, a maximum activity limit was initially established by grouping radioactive materials. For example, in the US, Rogers [2] reported that the domestic regulations in force in 1962 utilized three groups for the radioisotopes themselves, and four categories for packaging and shipping requirements. The three groups for the radiotoxicity classification were described as:

(a) Group I --------- Very high radiotoxicity,
(b) Group II -------- High radiotoxicity, and
(c) Group III -------- Moderate radiotoxicity.

Fairbairn and Dunning [7] documented the basis for the three-Group radionuclide structure used for Type A package limits in the 1961 Edition of the Regulations [3]. This three-Group structure, similar to what was being used domestically in the US at that time, was agreed by early IAEA panels using data provided by the UK, where the groups were determined using limits for short-term exposure [7]. The values determined used “the basic biological data: that were then available from the International Commission on Radiation Protection (ICRP)” [8].
**1964 Edition:** Efforts followed this first publication to enhance the quality of the basis for the activity limits. Fairbairn, Morley and Kolb [9] elaborated on those efforts, which led to the activity limits established for the 1964 revision of the Transport Regulations process. The revised classification "was derived primarily to permit the determination of appropriate activity limits for Type A packaging". The model utilized the following key assumption:

"...it has been assumed that as a result of a median accident one-thousandth of the radioactive contents of a Type A package will escape into the transport environment. It has further been assumed that in turn not more than a thousandth of this escaping fraction will find its way either by inhalation, ingestion or injection into the body or any one person; that is, an intake not exceeding one-millionth of the contents of the package."

It is noteworthy that this assumption has passed the test of time, and is one of the bases for the current methodology used for calculating activity limits for Type A packages (see e.g. paras I.30 and I.31 of TS-G-1.1 [10]).

As a result of these efforts, an eight-Group structure was established for radionuclides that were not in special form, and it was at this time that the "special form radioactive material" concept was introduced into the Regulations with a single activity limit.

Aikens [11] provided a summary of the activity limits for the three package type then specified in the 1964 Edition of the Regulations [4] as is depicted in Table A.3-2. This illustrates that not only activity limits were established for packages exempt from requirements and for Type A packages, but at this time the Regulations established activity limits for Type B packages (an approach which has since been eliminated from the Regulations).

<table>
<thead>
<tr>
<th>Toxicity Group</th>
<th>Exempt from Requirements (mCi)</th>
<th>Type A Packaging (Ci)</th>
<th>Type B Packaging (Ci)</th>
</tr>
</thead>
<tbody>
<tr>
<td>I</td>
<td>0.01</td>
<td>0.001</td>
<td>20</td>
</tr>
<tr>
<td>II</td>
<td>0.1</td>
<td>0.05</td>
<td>20</td>
</tr>
<tr>
<td>III</td>
<td>1.0</td>
<td>3.0</td>
<td>200</td>
</tr>
<tr>
<td>IV</td>
<td>1.0</td>
<td>20.0</td>
<td>200</td>
</tr>
<tr>
<td>V</td>
<td>1.0</td>
<td>20.0</td>
<td>5,000</td>
</tr>
<tr>
<td>VI</td>
<td>1.0</td>
<td>1000</td>
<td>50,000</td>
</tr>
<tr>
<td>VII</td>
<td>1000</td>
<td>1000</td>
<td>50,000</td>
</tr>
<tr>
<td>VIII</td>
<td>1000</td>
<td>1000</td>
<td>50,000</td>
</tr>
</tbody>
</table>

**1967 Edition:** With the 1967 Edition of the Regulations [5], the radionuclide classification for those materials not in special form was changed from an eight-Group to a seven-Group structure. Aspinall and Fairbairn [12] documented the evaluation that served as the basis for these changes. This was summarized by Fairbairn [13] as follows:

"The [1964] Panel listed radionuclides within Groups I-VIII; in the 1967 issue of the regulations this was changed to groups I-VII, the previous Groups VII and VIII having a common contents limit of 1000 C. .... The 1964 issue extended the ‘massive non-friable solid’ concept of the 1961 issue and earlier regulations to a ‘special form’ concept, which, by facilitating the use of encapsulation to specified standards, enabled any radionuclide in Groups I, II or III to qualify on grounds of containment, for 20-Ci Type A package limit."
Thus, for a given radionuclide, the activity limit for that material could be obtained for a Type A package, both for the material in special form and in other than special form.

The early application of the graded approach to activity limits for all three package types specified in the 1964 edition of the Regulations.

For each of the editions of the Regulations that classified activity limits using groups (i.e. the 1961, 1964 and 1967 editions), a table listing the individual radionuclides and group to which each belonged was provided (e.g. in the 1967 edition of the Regulations, Table II of Annex I of that document provided that listing).


“...has been developed for purposes of the 1973 issue of the regulations to the ‘A1, A2’ system, whereby each nuclide has two Type a package limits, A1 curies when in special form and A2 curies when not in special form. This system was proposed by France....”.

Fairbairn noted that the A1 values were derived using a rather limited, but conservative model where it was assumed that the source was intact but had completely released from its packaging and associated shielding. In contrast, the A2 values were derived assuming that the material was completely released from its packaging where the value was calculated using the then most recent ICRP guidance [14], and continued the assumption that internal exposure from intake would be 10^{-6} of the package contents. Thus, as noted in the third column of Table A.3-2 above, the radionuclides were no longer grouped, but two individual Type A package limits were established for each radionuclide.

A summary of the basis for the 1973 values of A1 and A2 was provided in the First Edition of Safety Series No. 37 [15]. It noted that "While the replaced of the Group classification system by the A1/A2 system results in lower package limits for some nuclides, for the majority it results in a relaxation of the previously restrictive package limits set on the basis of the most toxic nuclide in the Group".

The eight-Group classification used in the 1964 edition of the Regulations [4], the change to a seven-Group classification in the 1967 edition of the Regulations [5], and the change to activity limits individualized to each radionuclide as first established in the 1973 edition of the Regulations [6] are summarized in Table A.3-3. The significant changes that were made between these three editions are clearly shown.

1985 Edition: The activity limits for Type A packages contained in the 1985 Edition of the Regulations [16] were based on what has become known as the "Q-system". The development of the Q system was undertaken by Macdonald and Goldfinch [17] during the years preceding the publication of the 1985 Edition of the Regulations. The 1985 version of the Q system took account of the then most recent recommendations that were contained in ICRP Publications 26 and 30 [18, 19]. In addition, the radiological protection criteria underlying the derivation of the A1 and A2 Type A package contents limits were more clearly defined.

Three fundamental assumptions underlay the initial Q system:

(a) The effective dose equivalent to a person exposed in the vicinity of a transport package following an accident should not exceed the annual dose limit for radiation workers, namely 50 mSv (5 rem);
(b) The dose equivalents received by individual organs, including the skin, of a person involved in the accident should not exceed 0.5 Sv (50 rem), or in the special case of the lens of the eye 0.15 Sv (15 rem); and
(c) An individual is unlikely to remain at 1 m from the package for more than 30 minutes.

<table>
<thead>
<tr>
<th>1964 IAEA Regulations</th>
<th>1967 IAEA Regulations</th>
<th>1973 IAEA Regulations</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Special Form Radioactive Material</strong></td>
<td><strong>Special Form Radioactive Material</strong></td>
<td><strong>Special Form Radioactive Material</strong></td>
</tr>
<tr>
<td>Groups I-IV: 20 Ci</td>
<td>5000 Ci*</td>
<td>Individual values for each radionuclide, specified as A1</td>
</tr>
<tr>
<td>Other than Special Form Radioactive Material:</td>
<td>Other than Special Form Radioactive Material:</td>
<td>Other than Special Form Radioactive Material:</td>
</tr>
<tr>
<td>Group I–1 mCi</td>
<td>Group I–1 mCi</td>
<td>Individual values for each radionuclide, specified as A2</td>
</tr>
<tr>
<td>Group II–50 mCi</td>
<td>Group II–50 mCi</td>
<td></td>
</tr>
<tr>
<td>Group III–3 Ci</td>
<td>Group III–3 Ci</td>
<td></td>
</tr>
<tr>
<td>Group IV–20 Ci</td>
<td>Group IV–20 Ci</td>
<td></td>
</tr>
<tr>
<td>Group V–20 Ci</td>
<td>Group V–20 Ci</td>
<td></td>
</tr>
<tr>
<td>Group VI–1000 Ci</td>
<td>Group VI–1000 Ci</td>
<td></td>
</tr>
<tr>
<td>Group VII–1000 Ci</td>
<td>Group VII–1000 Ci</td>
<td></td>
</tr>
<tr>
<td>Group VIII–1000 Ci</td>
<td>Group VIII–1000 Ci</td>
<td></td>
</tr>
</tbody>
</table>

* The 5000 Ci limit applied to special form material that satisfied specified test requirements for the material or were contained in a capsule that satisfied specific requirements, and was not used as the containment vessel.

In addition, five exposure pathways were considered along with associated realistic assumptions the manner in which exposure could occur. The five pathways were:

1. External dose due to photons (gamma or X-rays) assuming complete loss of package shielding.
2. External dose due to beta emitters assuming dose rate variation with maximum beta energy.
3. Internal dose due to inhalation of non-special form or dispersible material released from a damaged package.
4. Skin contamination and ingestion doses resulting from handling a damaged package containing non-special form material.
5. Submersion dose due to gaseous isotopes assuming a 100% release of the package contents into a storeroom or cargo-handling bay.

Finally, special assumptions were made concerning special form alpha emitters and tritium and its compounds.

The A1 and A2 values were evaluated by a team in the UK and then verified by independent analysis by an expert in the US.

A detailed description of the effort, including the assumptions and sources in data, was provided as Appendix I of the 1990 Edition of Safety Series No. 7 [20]; and was updated with new information in Appendix I of the 2002 Edition of TS-G-1.1 (ST-1) [21].

1996 Edition and beyond: Since the initial development of the Q-system and its implementation in the 1985 Edition of the Regulations, the process has been enhanced, and the values updated to reflect later guidance from the ICRP. These additions and changes to the Q system have resulted in a better understanding and enhanced confidence in the values of A1 and A2, and in some greater stability with time of the activity limit values.

The latest activity limit values that have been published are contained in Table 2 of the 2012 Edition of the Regulations [22], and the details for their development was updated in Appendix I of Revision 1 of TS-G-1.1 [10]. The 2012 edition of the Transport Regulations [22] and the proposed revision of TS-G-1.1 [23] further update these values and the description of the Q system.
Table A.3-4 shows how the activity limit values have changed over time for two typical radionuclides (Co-60 and Pu-239). It shows that since the 1996 Edition, the values have been stable, prior to that they fluctuated as the methodology changed and was developed and the guidelines from ICRP changed.

Table A.3-4. Examples of Changes in $A_1$ and $A_2$ Values with time.

<table>
<thead>
<tr>
<th>Edition of the Transport Regulations</th>
<th>Activity Values</th>
<th>Pu-239</th>
<th>Co-60</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>$A_1$ (TBq)</td>
<td>$A_2$ (TBq)</td>
</tr>
<tr>
<td>1964</td>
<td></td>
<td>.74</td>
<td>.0003</td>
</tr>
<tr>
<td>1967</td>
<td></td>
<td>.74</td>
<td>.0003</td>
</tr>
<tr>
<td>1973</td>
<td></td>
<td>.07</td>
<td>.00007</td>
</tr>
<tr>
<td>1985</td>
<td></td>
<td>2</td>
<td>.0002</td>
</tr>
<tr>
<td>1996</td>
<td></td>
<td>10</td>
<td>.001</td>
</tr>
<tr>
<td>2009</td>
<td></td>
<td>10</td>
<td>.001</td>
</tr>
<tr>
<td>2012</td>
<td></td>
<td>10</td>
<td>.001</td>
</tr>
</tbody>
</table>

A.3.2. Applying a graded approach to types of package designs

Initially, in the first editions of the Regulations, five basic types of packagings were specified. The specifications for the packages, which were graded according to the risk posed by the contents, were:

(a) packagings exempt from requirements, containing small quantities of radioactive material and having a minimal set of package design requirements;
(b) packagings for materials that were judged to be “inherently safe”, e.g. low specific activity materials such as radioactive ores, where the regulations provided for their carriage in “bulk or in ordinary industrial packaging;
(c) Type A packagings, containing larger quantities of radioactive material while being designed to be resistant to normal conditions of transport;
(d) Type B packagings, containing larger quantities of radioactive material while being resistant to both normal and accident conditions; and
(e) packages containing large radioactive sources, where the activity exceeded the limits for Type B packagings (see Table 2 of SSR-6 [22]), and where additional requirements were imposed on materials compatibility, containment system pressure, pressure relief systems, protection of valves, and proper design of attachments and tiedowns [13].

As the Regulations have matured, more package types were introduced while the two categories with the highest risk contents as defined in the early regulations (i.e. Type B packagings and packages containing large radioactive sources) were coalesced into a single category. The packaging structure specified today in SSR-6 is:

(a) Excepted Package – equivalent to the earlier packagings exempt from requirements;
(b) Industrial Package Type 1 (Type IP-1) – part of the previous category of packagings for inherently safe materials;
(c) Industrial Package Type 2 (Type IP-2) – also part of the previous category of packagings for inherently safe materials;
(d) Industrial Package Type 3 (Type IP-3) – also part of the previous category of packagings for inherently safe materials;
(e) Type A package – equivalent to the earlier Type A package;
(f) Type B(U) package – part of the Type B packaging and packages containing large radioactive sources;
(g) Type B(M) package – also part of the Type B packaging and packages containing large radioactive sources;
(h) Type C package – a new category of package for the transport of certain radioactive material by air;
(i) H(U) approval – part of a new category of package approval used for transporting uranium hexafluoride; and
(j) H(M) approval – the other part of a new category of package approval used for transporting uranium hexafluoride.

All but the last two types are specified in, e.g. para. 231 of SSR-6\(^{[2]}\); whereas the latter two types are established by way of para. 832 of SSR-6\(^{[2]}\).

The extension of the number of packaging types involved extensive deliberations by expert panels concerning the nature of the contents and the need to provide flexibility in the Regulations, while concurrently ensuring adequate protection by each package type.

First, for example, the single designation of "packagings for materials that were judged to be 'inherently safe'" expanded to three types of industrial packages, the IP-1, IP-2 and IP-3 packages, which were first introduced in the 1985 Edition of the Regulations\(^{[16]}\), and which follow a graded approach based on hazard posed by the contents. This industrial package structure was the result of panels of experts, convened during the efforts leading to the 1985 Edition; where the experts considered the diversity of forms of materials previously determined to be "inherently safe"; and agreed during their deliberations that a more prescriptive approach was needed. These deliberations resulted in specifications for:

(a) three types of Low Specific Activity (LSA) material, identified as LSA-I, LSA-II and LSA-III (see paras 226, and 408-411 of SSR-6\(^{[2]}\));
(b) two types of Surface Contaminated Objects (SCO), identified as SCO-I and SCO-II (see paras 241, and 412-414 of SSR-6\(^{[2]}\));
(c) the three types of industrial packages, identified as IP-1, IP-2 and IP-3, with increasing design requirements imposed when moving from the IP-1 to IP-2 and from IP-2 to IP-3 package types; and
(d) incorporating contents and package design into a further grading on their use and operational constraints depending upon
   (i) the physical nature of the contents (i.e. gas, liquid or solid),
   (ii) whether they are shipped under exclusive use or not under exclusive use, and
   (iii) conveyance activity limits depending upon the mode of transport (see Table 6 of SSR-6\(^{[2]}\)).

All of these criteria merge together to from a comprehensive set of graded requirements for protection of man and the environment during the transport of these materials.

Coupled with these is a restriction that any LSA material or SCO must have an external radiation level at 3 m from the unshielded material or object or collection of objects that does not exceed 10 mSv/h (see para. 517 of SSR-6\(^{[2]}\)). This constraint was established to ensure adequate radiation protection since, in the case of an accident, the shielding integrity of the industrial

\(^{19}\) NOTE: this appendix does not specifically address the bases for imposing certain test requirements on uranium hexafluoride packages. The tests imposed are essentially some of the tests imposed on other types of packages, with the addition of a pressure test that is specific to uranium hexafluoride packages. The rationale for imposing those tests on uranium hexafluoride packages is addressed in Chapter 10 of this document.
packages cannot be ensured since the most demanding tests imposed are on the IP-3 package type, and these are only for normal conditions of transport \[2\].

Second, as another example, the Type C package was added to the Regulations in the 1996 Edition of the Regulations \[24\]. This package type results from international deliberations prompted by requirements implemented in the US in 1975 in response to domestic legislative action (i.e. Public Law 94-79) that no license could be issued for “any shipments by air transport of plutonium in any form”, except for that contained in a medical device designed for individual human application, until “a safe container has been developed and tested which will not rupture under crash and blast-testing equivalent to the crash and explosion of a high-flying aircraft” \[25\]. As a result of this domestic law, the US Nuclear Regulatory Commission (NRC), working with personnel from Sandia National Laboratories (SNL), evaluated air accidents and ultimately established a set of very stringent package design requirements for air transport of plutonium \[26 – 31\].

It is noted that the test requirements established by the US NRC for packages of plutonium transported by air are more robust than those established for Type C packages. For example, the NRC requirement is for an impact onto an unyielding target at a velocity of 128 m/s (§71.74(a)(1) of \[32\]); whereas the impact velocity specified for a Type C package is 90 m/s (para. 737 of SSR-6 \[2\]). The technical basis for the package test requirements for Type C packages are discussed in Section 7 of this document.

A.3.3. Applying a graded approach to package design performance standards

Chapter 10 of this document addresses the technical bases for the testing requirements for radioactive materials and radioactive material packages. After a package design has been exposed to testing requirements, the Regulations require that they must be demonstrated to satisfy specified performance standards such as defined leakage rate limits and external radiation level limits.

The following addresses the technical bases for these post-test performance test standards.

In many cases the Regulations apply a graded approach to the performance standards that are based on the type of package. The design and test requirements for the types of packages themselves follow a graded approach determined by the hazard posed by the contents as discussed in Section 3.2.

The type of package performance standards are those for containment and shielding parameters and, in the case of fissile materials, criticality prevention parameters that a package design must satisfy after it is exposed to the test requirements specified for that specific design.

Table A.3-5 summarizes the graded approach to these performance requirements.

Table A.3-5 illustrates that, as the risk of the contents increases, the tests become more robust and/or the post-test acceptance criteria for containment and shielding become more robust. For larger quantities of radioactive material to be transported by air, where the material is not qualified as a low dispersible material, then the accident condition tests are enhanced beyond those used for Type B packages while the post-test acceptance criteria remain the same as for Type B packages. This results in the specifications for Type C package.

The technical basis for the graded approach to acceptance criteria for the release rates and external radiation levels, for all but the Type C packages, was established early on and is discussed in detail by Fairbairn \[13\]; Aikens \[11\]; MacDonald and Goldfinch \[17\]; Swindell \[33\]; Aspinall, Gibson and Morley \[34\]; and Fairbairn and George \[35\].
Table A.3-5. The Graded Approach to Package Test Acceptance Parameters.

<table>
<thead>
<tr>
<th>Package Type*</th>
<th>Test Requirements</th>
<th>Acceptance Parameter</th>
<th>Relevant SSR-6 paragraphs</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Containment</td>
<td>Shielding</td>
</tr>
<tr>
<td>Excepted</td>
<td>None</td>
<td>None</td>
<td>None</td>
</tr>
<tr>
<td>Industrial Type IP-1</td>
<td>None</td>
<td>None</td>
<td>None</td>
</tr>
<tr>
<td>Industrial Type IP-2</td>
<td>Normal Condition free drop and stacking tests</td>
<td>No loss or dispersal of radioactive contents</td>
<td>No more than 20% increase in the maximum radiation level at any external surface of the package</td>
</tr>
<tr>
<td>Industrial Type IP-3</td>
<td>All Normal Condition tests</td>
<td>Restrict loss of contents to not more than (10^{-2}) (A_2)/h</td>
<td>Radiation level 1m from package surface does not exceed 10 mSv/h</td>
</tr>
<tr>
<td>Type B(U) and Type B(M)</td>
<td>All Normal Condition tests</td>
<td>Restrict loss of contents to not more than (10^{-2}) (A_2)/h</td>
<td>Radiation level 1m from package surface does not exceed 10 mSv/h</td>
</tr>
<tr>
<td>Type C</td>
<td>All Normal Condition tests</td>
<td>Restrict loss of contents to not more than (10^{-1}) (A_2)/h</td>
<td>Not specified</td>
</tr>
<tr>
<td>Enhanced Accident Condition Tests for Air Transport***</td>
<td>Restrict loss of contents to not more than (10^{-2}) (A_2)/h</td>
<td>Radiation level 1m from package surface does not exceed 10 mSv/h</td>
<td>669, 734-737</td>
</tr>
</tbody>
</table>

* Packages containing UF6 (which are not elaborated upon in this appendix) must satisfy specific requirements which pertain to the radioactive and fissile properties of the material, and generally must also withstand (a) without leakage or unacceptable stress, a hydraulic pressure test (para. 718 of SSR-6); (b) without loss or dispersal of the UF6, the normal condition free drop test (para. 722 of SSR-6), and (c) without rupture, the accident condition thermal test (para. 728 of SSR-6).

** Further grading results within the accident condition tests. For example, if a package contains more than \(10^3\) \(A_2\) of radioactive material, it is required to withstand, without rupture, exposure to an enhanced water immersion test (paras 660 and 730 of SSR-6).

*** The enhanced accident condition tests for air transport include (a) a burial test (para. 670 of SSR-6), (b) a puncture-tearing test (para. 735 of SSR-1), (c) an enhanced thermal test extending exposure from 30 to 60 minutes (para. 736 of SSR-1), and (d) an enhanced impact test with impact velocity of 90 m/s (para. 737 of SSR-6).

The preceding illustrates how the type of packages and their associated contents and modes of transport follow a graded approach depending upon the risk posed by the contents. It also illustrates the extent to which this approach has been applied in the Regulations.

**The graded approach to allowed package release limits**

The 1979 technical committee meeting TC-272 [RBP026] reviewed, for adequacy, the leakage test requirements following exposure of packages to normal and accident conditions of transport. It noted the following with respect to the requirement for Type A packages (i.e. for packages exposed only to the normal conditions of transport tests):

"The performance criteria for Type A and Type B packages are different in that Type A packages are required to suffer no loss of containment, whilst for Type B packages a finite leak rate figure is laid down in the Regulations."
“From the point of view that a Type B package must withstand accident conditions without a similar requirement for Type A packages, it appears that the requirements are conceptually or logically wrong. Also, from the radiation protection point of view, leakage cannot be said to be unacceptable, at least as long as the activity leakage is below a suitably established level. The concern is always the resultant dose, not the exposure pathway in isolation.”

Following deliberation on this issue, TC-272 decided to maintain the qualitative leakage rate requirement for Type A packages, and the quantitative requirement for Type B package.

TC-272 also considered the then existing factor of $10^3$ difference in allowed post-test leakage rates from Type B(U) and Type B(M) packages. Paras 233 and 243 of the 1973 edition of the regulations as amended 1979 [36] allowed $10^{-3} \text{A}_2$/h release rate from Type B(U) packages following exposure to the hypothetical accident condition tests, whereas it required a $10^{-6} \text{A}_2$/h release rate limit for Type B(M) packages.

TC-272 noted that it "was unable to find explanation for this difference except for engineering considerations and the fact that Type B(M) packages prior to being shipped require the competent authority approval of each country affected by the shipment and therefore different leak rate limits could, in principle, be allowed". It then suggested that the leak rate for both Type B(U) and Type B(M) packages should be based "on one and the same dose criteria"; but it noted that at the time of this meeting no sound recommendation could be made. [See EndNote xix xix]

The graded approach to general package requirements when transported by air

Prior to the 1985 edition of the Regulations [16], the requirements for packages were independent of mode of transport. However, during the deliberations leading to the 1985 edition, the representatives of ICAO and IATA stressed that the normal air transport environment was different from the surface transport environments (i.e. by road, rail or water), and there was a need to add specific requirements that would be imposed on packages transported by air. This resulted in the subsection titled “ADDITIONAL REQUIREMENTS FOR PACKAGES TRANSPORTED BY AIR”, with the addition of paras. 515 – 517. These paras imposed alternative requirements on (a) accessible surface temperatures (para. 515), (b) exposure to alternate ambient temperatures (para. 516), and (c) a reduced pressure requirement if the package contained liquids (para. 517).

The discussions involving these changes are documented in part in AG-406 [RBP049]. For example, the Advisory Group noted the following:

"WG 3 has considered at some length the issue of reduced pressure for carriage of liquids by air. ICAO has implemented a requirement for all other hazardous liquids in terms of a pressure differential, which is different from the specifications in SS 6 in terms of reduced ambient pressure. Recognizing that the requirements are already in place, the WG proposed that a paragraph be added in the prescription for air transport requiring the pressure differential test specified by ICAO. The WG hesitates to add mode specific requirements but recognizes that the requirement is already there as a result of the ICAO action."

In addition, the change to ambient temperature ranging from -40 °C to 55 °C was addressed by Working Group 3 of AG-406, where it noted the change was recommended but it met with strong disagreement, where the AG-406 report documents:

"The WG was divided on whether the SS 6 requirement of -40 °C to 38 °C with insolation and -40 °C to 70 °C design requirements for the package components includes the ICAO requirement. The WG proposed the following paragraph (by 7 to 5 vote):

“544 (new) For design of packagings for transport by air it is necessary to consider effects of ambient temperature of -40 °C to 55 °C."
A.3.4. Applying a graded approach to package performance standards based on physical and chemical properties of contents

The graded approach is also applied in SSR-6 [2] to the package contents with respect to both its physical and chemical properties.

For the physical form of the contents:

- For all packages, para 614 of SSR-6 requires that the packaging and any components be physically compatible with each other and with the contents of the package.

- For excepted packages, the activity limit classification of excepted packages is graded according to whether the contents are (a) in special form or in other than special form, and (b) in solid, liquid or gaseous form (see Table 5 of SSR-6).

- For industrial packages, the characterization of LSA material is graded considering whether or not the material is a solid; where, specifically, only solids can be characterized as LSA-III (see para. 409 of SSR-6). In turn, the package requirements for LSA material are partly structured as to whether the material is solid, liquid or gas; where, if the shipment is not under exclusive use a more robust package is required for liquid and gaseous LSA material than for solid LSA material (see Table 5 of SSR-6). Also, the activity conveyance limits for LSA material depends in part on whether the material is solid, liquid or gas, and whether or not it is combustible (see Table 6 of SSR-6).

- For Type A packages, additional requirements are imposed on the package design if the contents are liquid or gas (see paras 649-651 of SSR-6).

For the conveyance limits for LSA material and SCO, initially specified in para. 427 and Table Vi of the 1985 edition of the Regulations [16], these were initially proposed in Working Paper No.54 for AG-406 [RP049], and agreed by AG-406. [See EndNote xx xx]

For the chemical properties of the contents:

- For all packages, para. 614 of SSR-6 requires that the packaging and any components be chemically compatible with each other and with the contents of the package.

- For Type A packages, in addition to para. 614, para. 644 of SSR-6 specifies that the design shall account of any chemically-generated gas. This requirement also applies to Type B and Type C packages.

- For packages containing uranium hexafluoride, additional design and testing requirements were added in the 1996 Edition of TS-R-1 [24] to account for the chemical hazard posed by the material (see paras 631 – 633 of SSR-6 [2], and para. 629.1 of TS-G-1.1 [10]).

A.3.5. Applying a Graded Approach to Package Design Approval

There is a two-phased graded approach that is applied to the approval of package designs, again based on the risk posed by the contents of the package. As specified in para. 802 of SSR-6 [2], a number of package designs require approval by a competent authority. These are (a) packages containing more than 0.1 kg of UF₆, Type B packages, Type C packages, and all packages containing fissile material. Para. 801 of SSR-6 specifies that for the other packages (i.e. excepted packages, industrial packages and Type A packages); the consigner must have available documentary evidence of the compliance of the design with applicable requirements. The intent here is to ensure that those packages containing materials posing a higher risk undergo a closer scrutiny by the competent authority before that design is used for transporting the radioactive materials for which it is designed.

References for Appendix 3


APPENDIX 4 - Follow-on studies into the adequacy of the accident conditions of transport

A.4.1. Follow-on studies into the adequacy of the accident scenarios

The basic principles elaborated by Messenger, Fairbairn, Appleton and Servant have continued in all later revisions of the Transport Regulations. In addition to the studies encapsulated or referenced by these gentlemen, many studies have been performed over the years that further justify the efficacy of the tests established. Many of these studies focused on only one of the tests (e.g. on the 9 m drop test, or the more recently instituted dynamic crush test, or the 30-minute, 800°C thermal test); whereas others have focused on the suite of tests or on alternative scenarios to the established tests.

Those studies that focused on, and provide technical basis for the individual tests, are addressed in the appropriate sections (6.3 through 6.8) below; whereas, those studies that looked more broadly at the suite of tests are briefly summarized here. These tests and/or analyses were performed, either looking at the response of packages to severe, accident-simulating tests or at what additional mechanical and thermal stresses beyond those provided by the regulatory test standards are needed to fail packages. Examples of these tests include high-speed impact, extended fire and even explosive testing involving scale model and full-scale packages and vehicles in various countries, including France, Germany, Japan, the UK, and the US.

Sert et al. [1] reported on analyses of various configurations of packages impacting other packages from drop heights of 16 m. The analyses showed that the packages maintained their integrity and that any the ancillary equipment (e.g. frame and rack) present on the packages improves the absorption of impacts while concurrently increasing the safety margins.

A study was undertaken by a dual-national team from Japan and the U.S. (Yamamoto, et. al. for Japan, and McClure, et. al. from the U.S. [RBP029]. This study evaluated "the safety of a large number of plutonium transport operations for the international transport of plutonium by maritime cargo vessels for selected routes. The analysis centres on conventional cargo vessels and their accident history in order to provide an estimate of the probability of accident occurrences for such vessels. This is an ultraconservative study since the radioactive materials described in this study will, in all likelihood, be transported in purpose-built ships that incorporate many safety features not found in regular cargo vessels. Follow-on studies can use the information developed in this study in order to estimate the probability of accident occurrences involving purpose-built ships. The accident probabilities developed in this study, for conventional cargo vessels, provide a conservative bounding estimate of the probabilities for accidents involving purpose-built ships. This study estimates the safety of transporting plutonium from Europe to Japan. This includes estimating the probability of a severe transport accident during marine transport over three separate routes."

The study estimated the probabilities of a cargo vessel transport accidents in ports or approach waters to ports were estimated based on Lloyds casualty categories, and were found to range from approximately 1.4 X 10^-6 per vessel movement for a Category 4 accident (collision) while on the route, to as low as approximately 1.3 X 10^-10 per vessel movement for a Category 6 accident (missing vessel, i.e. viewed as a loss by marine perils). The probability of a severe cargo vessel transport accident in a port which might release radioactive material ranged between 10^-9 to 10^-10 per ship movement.

The study also considered the differences between general cargo vessels and purpose built cargo vessels, and concluded that the probability of a serious cargo vessel accident is more probable for general cargo vessels than for purpose-built cargo vessels.
Ammerman [2] performed analytical studies on the collision of a ship into a charter freighter carrying spent fuel casks by varying the mass of the striking ship from one to 10 times the mass of the charter freighter. The impact speed for the striking ship was 5.14 m/s (10 knots) for equal ship mass, up to 15.6 m/s (30 knots) for the case with 10 times the ship mass. It was demonstrated that even "...the most severe ship to ship collisions cannot generate impact or crush forces on a package that are higher than the impact forces from the certification impact test". 

One unique test has been reported by Droste [3] in which a 22 450 kg cask was located next to a 45 m³ LPG tank wagon, partially filled with 10 m³ pressurised liquid propane, both of which were positioned above a fuel pool. At 17 min after fire ignition, the propane tank ruptured, resulting in a boiling liquid expanding vapour explosion. "Although the nonprotected closure lid was exposed to fire, and hit severely by tank wagon fragments, post-test investigations demonstrated that no loss of leak tightness occurred". Droste also reported that the detonation blast wave from a wagon with explosives (21 t equivalent weight of TNT) at a distance of 25 m between the centre of the explosion and the front end of 80–120 t casks was investigated. The accelerations were determined to be "much lower than that in a 9 m drop, and comparable to that in a 1 m drop, indicating that this effect of the explosion is covered by the regulatory test conditions".

Another unique test was described by Lopez et al. [4]. In this test the deflagration from the rupture of a propane/butane tanker with a rail sized cask located close to the tank, such that it is deep within the vapour dome during the discharge phase, was assessed. The study showed that the temperature at the cask seal location and the cask inner wall did not rise in an appreciable manner.

Ballheimer et al. [6] documented the analytical assessment of the potential impact from a detonation blast wave from explosives transported in the same train unit as a spent fuel cask. The calculations showed that the integrity of a robust monolithic cask is preserved after the effect of a 21 t (equivalent weight of TNT) explosive detonation in the regular transport configuration with a distance of 25 m between the centre of the explosion and the front of the cask.

In addition, regulatory bodies have supported analytical evaluations of the adequacy of the regulatory test requirements. For example, the US NRC sponsored what is commonly known as the Modal Study in 1977 [6]. This was followed by a re-evaluation of the Modal Study in 1987 [7], and a further study in 2000 [8].

Another effort that looked at real accidents to assess the adequacy of the regulatory tests intended to represent accident conditions of transport was undertaken by SNL personnel in 2003 for the US Department of Energy (DOE) [9, 10]. This study considered twelve different, severe road and rail accidents that have occurred in the US between 1983 and 1996, and compared the results of this study with the earlier study performed by SNL in 2000 for the US NRC [8]. The accidents considered included bridge collapses, vehicle collisions, a truck trailer accident resulting in an explosion, and railroad derailments. Twenty different technical reports on these accidents were evaluated. The reviewers concluded that no accidents were identified that "were outside the event trees and accident severity already considered in NUREG/CR-6672. In most cases, the accidents were found to be within the regulatory test sequence limits described in 10CFR71"20. The report noted that in one instance the duration of a fire was not reported, but then concluded that "...even if the fire had lasted longer than 30 minutes, it would be within the range of extra-regulatory conditions considered in NUREG/CR-6672."

In all cases, these studies showed the adequacy of the existing test requirements for Type B packages. The 1977 Modal Study covered the transport of all types of radioactive material by all transport modes (road, rail, air and water). Based in part on those findings, the NRC's governing

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20 NOTE: The reference in [104] to "the regulatory test sequence limits described in 10CFR71" is to the US equivalent of the tests specified in paras 726 – 730 of TS-R-1.
Commission concluded that “present regulations are adequate to protect the public against unreasonable risk from the transport of radioactive materials…” [7, 8].

A unique study was undertaken by the Bundesananstalt für Materialforschung und-prüfung (BAM) in the late 1990s. A test of a propane tank car in a fire environment was to be undertaken to assess the length of time until the tank would explode. A small flask was located close to the propane tank, and both were simultaneously exposed to a pool fire. Fifteen-minutes into the fire test, the propane tank experienced a boiling liquid expanding vapour explosion (BLEVE), thereby exposing the flask to the explosive effects, tumbling it away from the location. No damage was sustained by the flask. A summary of this test is provided in Reference [RBP022], and a set of coloured pictures taken from that test was provided in working paper at the IAEA-TC1196 meeting which represented a presentation made at the 5th International Conference "Transportation for the Nuclear Industry", Blackpool, UK, 2-4 November 1999 [RBP023]. It was shown that:

- the rupture of the propane tank occurred close to the horizontal centre line of the side furthest from the flask,
- this rupture propelled the tank car directly onto the flask,
- the eccentric impact caused the flask to translate, rotate, and then crash with the lid side into the ground, and
- the integrity and leaktightness of the flask lids were unchanged.

It was concluded that the impact from this highly improbable accident scenario is covered by the regulatory test conditions.

The US National Academies formed the Committee on the Transportation of Radioactive Waste which looked at the safety issues associated with land transport of irradiated nuclear fuel and high-level radioactive waste in the US [11, 12]. A companion paper provides a comprehensive response to that evaluation developed by the US NRC [13]. In summary of these documents, a broad perspective was provided on the safety of the transport of radioactive material when undertaken in packages and using operations that are in compliance with the transport regulations. It can be concluded, in general, considering the various environments that were evaluated, that the existing structural, thermal and immersion hypothetical transport accident condition tests provide an excellent margin of safety for the transport of radioactive material.

A.4.2. Follow-on studies into the adequacy of the 9 m drop accident test scenario

Pope et al. [14] discussed an early test of a highway vehicle carrying a 13.6 t cask that was impacted into a massive barrier at a speed of 45 km/h (28.5 mph). The crushing of the tractor and the front end of the trailer absorbed most of the crash energy, the cask remained attached to the trailer and no damage was detected to the cask. The cask only experienced a deceleration of only about 4 g.

Yoshimura et al. [15] reported on a series of three types of accident-simulating tests that were performed in the late 1970s. The first simulation consisted of two cask carried by trucks impacting twice a massive concrete/steel/earth barrier at high speeds. In the first test, a highway vehicle carrying a 20.5 t cask was impacted into the barrier at a speed of 100 km/h. After inspection and clean-up of the cask, which was determined to be essentially undamaged, the cask was loaded onto a second highway vehicle and then impacted into the same barrier at 135 km/h. Following the second impact, a small of amount of water (~100 cc) leaked from the cask containment system. Two drop tests of a another cask of this same design from 9 m onto an unyielding target showed that the simulated truck crash tests resulted in damage to the contents that were similar to or less than that sustained in the regulatory drop tests.

In the second type of test [15] a 68.2 t railcar carrying a 61.8 t cask was impacted into the same barrier at 130 km/h. The impact resulted in extensive damage to the railcar structure, but the cask body was “undeformed except for minor deformations to the external cooling fins, and there was no leakage".
Yoshimura et al. [15] also reported on a test of a highway vehicle carrying a 2 t fresh mixed oxide fuel package which was impacted into the same barrier at 93 km/h. No damage was detected within the package containment system and it retained its containment integrity. Following this test, the package was exposed to a series of regulatory drop and fire tests, and it still retained its containment integrity.

One series of tests which was performed in Germany that added to the body of knowledge concerning drop testing [16] elaborated on the drops of a modified 18B, FL10, and 6M package designs for plutonium compound transport from a height of 200 m (with a free fall speed of about 55 m/s) onto a surface consisting of 10 cm asphalt on a 30 cm thick layer of broken bricks. These packages were also exposed to other “extended tests”. The authors concluded that “Concerning all types of Pu shipping packagings, which we exposed to the extended testing criteria, the margins of safety lie far beyond the limits as given by the IAEA adopted regulations”.

Droste [3] presented information involving a high-speed impact of a modified design of an ‘18 B’ package, with stiffer spacers. The package impacted an essentially unyielding target at a speed of 129 m s\(^{-1}\) (287 mph). The purpose was to determine the point at which containment would fail. At the speed of impact, loss of containment did occur. This extensive effort has demonstrated that a significant margin of safety for type B packages exists relative to the regulatory drop test.

Droste, et. al. [RBP031] reported on drop tests of a ductile cast iron (DCI) cask with intentionally imposed, big artificial flaws to demonstrate the ability of such flawed structure to withstand drops from up to 14 m. One purpose of the tests was to evaluate the safety of such designs against brittle fracture. In these tests, the casks dropped were 1:2.5 scale models, and the drops were made onto rails (i.e. “round mild steel bars” with a span width of 1830 mm fixed upon the steel plate of the BAM unyielding target) rather than directly onto the regulatory unyielding target. The intent of the rails was to impose locally high stresses in the cask body. The worst case drop resulted in exposing the weakened cask body to impact forces that “exceeded the Type B package mechanical test condition by 650%”. The authors concluded that:

“Even in that case no brittle fracture failure occurred on that DCI cask. For the German competent authority DCI is a suitable material for Type B packages, proven on the state of science and technology. As demonstrated, casks of this material can have a remarkable safety margin beyond the design limits.”

Ammerman et al. [17] also reported on an analysis which considered various cask designs impacting a rigid (essentially unyielding) target at speeds from 13 to 54 m/s (30–20 mph). The study demonstrated the viability of the concept of graceful failure and that the spent fuel transportation casks have a large margin of safety against release of radioactive material. One of the cask designs was dropped onto an unyielding target from heights of 9, 20 and 36 m, with impact velocities of 13.4, 20 and 27 m/s (i.e. 30, 45 and 60 mph respectively). These tests demonstrated that, even for the largest drop height studied, “the package remained leak-tight, and there was very little plastic deformation in the closure bolts”, and further illustrated a large margin of safety in the design.

A.4.3. Follow-on studies into the adequacy of the 1 m puncture probe accident test scenario

Yoshimura [15] documented a 186 t locomotive impacted a 22.7 t cask at a speed of 130 km/h. It was reported that leak testing “of the cask after impact indicated a small leak in the head seal”, but “had the cask contained cooling water as it was designed to do, this leakage would have caused essentially no risk to the public”. This test was related to the puncture test because the two main frame members of the locomotive parallel to the railroad tracks (and perpendicular to the cask) provided the main impacts to the cask and resulted in the main deformations to the cask heat transfer fins.
Tso et al. [181] summarised the results of a test where a large rail locomotive pulling three passenger coaches, travelling at 100 mph (161 km/h), impacted a Magnox spent fuel flask. The flask was projected "... [s]ome 55 m forwards with the locomotive coming to rest 100 m from the point of impact. ... The flask internal pressure, which had been set at 0.69 MPa before the test was measured immediately after the crash to be 0.688 MPa ...". The flask was essentially undamaged other than some external scarring, and it was reported that the small loss of water (estimated to be ~500 cc) from the flask would, in a real event, have been radiologically insignificant. This test was also related closely to the puncture test because the impact scenario was structured to have the "draw hook" of the locomotive strike the flask directly on the joint between the flask body and flask lid.

The TN 24 family of casks are required, in some countries to be able to withstand impact of a high velocity aircraft (see Malesys [19]). Analyses followed by tests on scale models of the casks were performed for impacts of a mass of 14 600 kg (representing and F16 military fighter) travelling at a speed at the moment of impact of 150 m/s for casks to be used in Belgium and 20 500 kg (representing an F18 military fighter) travelling at a speed at the moment of impact of 215 m/s for casks to be used in Switzerland. In all cases, "a very low leak rate was maintained after the test, several orders of magnitude better than required". These two tests are also related to the puncture test because the main rigid element of these aircraft is the turbine shaft, when upon impact, acts as a puncture probe.

In a similar set of tests, Droste [3] indicated that steel pipe projectiles, with a mass of 1000 kg, were propelled to a velocity of 300 m/s onto different cask specimens. The impacts of the projectiles resulted in some deformation of the outer parts of the cask body or lid system, but a quantifiable leak tightness still existed. Analyses of other aircraft impacts into casks were performed. It was concluded that such impacts do not do any damage to massive cask structures.

A.4.4. Follow-on studies into the adequacy of the dynamic crush accident test scenario

Itoh et al. [20] describes a simulated roof slab of a storage building that was dropped from heights of 5 and 17.1 m onto a large cask. No leakage from the package lid was detected and no strain larger than the yield strain in the package body was measured.

A paper by Feldman [21] summarizes the results of crush testing five different package designs in recent years. It provides information on 26 crush tests and notes that "In all cases, the deformation of the outer drum created by the crush test was significantly greater than the deformation damage caused by the 9 m drop test. The crush test is a highly effective means for testing structural soundness of smaller non-dense type B shipping package designs."

A paper by Smith [22] elaborates on the early tests that were considered in deciding to implement the dynamic crush test, and then shows the results of tests on a US DOE package designed as "general purpose fissile material package (GPFP)". The comparison of damage from the dynamic crush test to this GPFP with the damage that occurred with the earlier package designs demonstrates that – with proper consideration for the dynamic crush environment, a reasonable design solution can be obtained.

Feldman [21] notes at the end of his paper that "Further regulatory guidance could alleviate the need to perform the crush test in a wide range of orientations and crush plate CG alignments".

Bjorkman and Pope [23] addressed a discussion that occurred at an IAEA consultants services meeting in September 2008 where valid and invalid package positioning for the dynamic crush test was addressed. It was recommended that

"A valid crush test is one wherein:
(a) at the instant of the plate’s first contact with the package, a direct load path (crush path) is established between the plate at the point of contact and the unyielding surface, where this load path must be capable of statically supporting 500 kg; and
(b) *the drop location cannot produce a condition where the first action of the plate is to tear the package.*

### A.4.5. Follow-on studies into the adequacy of the accident-simulating thermal test

The basis and requirements for the thermal test were re-evaluated during the efforts leading to the 1985 Edition of the Transport Regulations through deliberations by international panels of experts convened in the late 1970s and early 1980s. A recommendation for clarifying text for the thermal test emanated from the meeting of the IAEA Technical Committee on Transport Package Test Standards (TC-406) [24]. As already noted, this meeting was preceded by earlier technical committee, consultants services and advisory group meetings as well as meetings of the Standing Advisory Group on the Safe Transport of Radioactive Material (SAGSTRAM).

During the TC-406 meeting, one working group addressed the thermal test issue in detail. Revised text was provided that was intended to "provide a clearly defined test procedure which is not confused by a mixture of testing and calculational procedures, and provide a realistic representation of convective heating for calculational methods, which may result in an increase in the thermal input of 10 to 15 percent relative to the existing specification...". It was further noted in the TC-406 report that, as a result of the inclusion "of realistic convection in the procedure for demonstrating compliance may require that some existing packages be re-evaluated, and that the applicability of oven test procedures given in Safety Series No. 37 be reassessed and possibly redefined."

Additional documentation on the thermal test requirement is summarized here. First, the following are results of studies where packages were exposed to fully-engulfing thermal environments exceeding those specified in the Regulations, either for durations longer than 30 minutes and/or at temperatures greater than 800 °C.

Droste [3] reported that a type '18B' Pu nitrate package that had previously been exposed to a drop test onto a concrete/gravel target from 185 m (see Section 6.3.2) was then exposed to a fully engulfing, long duration fire. Results showed that "a loss of tightness would not have occurred before a fire period of 75 min".

Yoshimura et al. [25] summarized the test of an obsolete 67 t spent nuclear fuel cask, located in its 68 t transport railcar, the combination of which had been previously impacted at high speed into a massive barrier (see Section 6.3.2), was exposed to an open pool fire test for 125 min. The average flame temperatures during the test ranged from 980 °C to 1200 °C, well above the 800 °C specified in the transport regulations. The cask "survived at least 90 min of fire exposure without apparent failure".

Itoh et al. [20] describes the testing of a 104 t cask which was unexpectedly exposed to a 30 min, fully engulfing thermal test where the temperatures exceeded 1000 °C. The resulting temperatures of the O ring seal and the cavity pressure of the cask "were smaller than those in the safety analysis for both the maximum allowable material temperature, and the design pressure".

Lopez et al. [4] summarizes thermal analyses of four generic truck and rail casks to assess the cask’s behaviour to very long duration, fully engulfing fires were performed for various positions in the regulatory 800 °C (1475 °F) fire, and an extra regulatory 1000 °C (1832 °F) fire environment. For the higher thermal test environment, i.e. the 1000 °C (1832 °F) fire, the study showed that it would require exposure times ranging from 1 to more than 11 h to reach various critical temperatures in generic casks, depending upon the design of the cask.

Malesys [19] describes how an FS 47 packaging for carriage of Pu oxide powder was exposed to long-duration, fully engulfing fire. The test was conducted with a duration of 3 h and 30 min, at which time it was determined that the leaktightness of the packaging was still better than 1028 Pa m3/s. In other words, the fire test did not alter the leaktightness of the package. He also describes an analysis of packages from the TN 12 and the TN 28 VT families of casks performed
against fully engulfing, long duration fires. The results showed that these designs provide for safety margins "larger than 3 when compared with the IAEA Transport regulatory fire test (800°C/30 min)."

Sert [1] reports on the thermal behaviour of spent fuel, fresh MOX fuel, vitrified waste and plutonium oxide powder packages to fully engulfing fires with various durations (up to ~21 h) and with fire temperatures (400, 600, 800 and 1000°C) were analysed. The results show that "with respect to the risk of activity release due to the IAEA thermal test conditions, the packages considered in this study present significant safety margins on fire duration, from 1.5 to 13 when compared with the regulatory requirement".

Yoshimura et al. [25] provides information on a 2 t mixed oxide fresh fuel cask located in its transport trailer that had previously been tested in a 90 km/h (56 mph) head-on impact (see Section 6.3.2) was exposed to an open pool fire test for 72 min. It was concluded that "the containment integrity of the cask was not impaired by the preceding three impact test or the subsequent 72 min fire test".

Having looked at data for fully engulfing fires, the following provides results from studies where packages were exposed to thermal environments other than fully-engulfing.

Yoshimura et al. [25] and Hamann, et. al. [26] summarized the testing of a 45 t cask that was exposed multiple times to the effects of a hydrocarbon fuel torch fire, where each exposure occurred for a duration of 30 min. These results of the tests demonstrated 'that exposure to the torch environment does not significantly threaten the containment system' of the cask.

Aritomi, et. al. [RBP027] performed open pool testing and analysis of BWR and PWR fresh fuel containers, for durations of 30 minutes. The behaviour of the package contents and the package itself in the fire environments were both evaluated. It was concluded that the "shipping containers of both the BWR and PWR types both protect their contents and verify the safety of the new fuel assembly in transport, even though they may suffer some damage in the actual size fire of 30 min duration".

Ammerman [2] evaluated experimentally simulated maritime engine room, pool and cargo fires. It was demonstrated that because the IMDG code prevents the transportation of flammable materials and radioactive material (RAM) in the same hold, "a pool fire is very unlikely to occur in the same hold as a RAM package". In addition, "the bulkhead acts as a very good radiation shield and limits the heat flux to a RAM package in an adjacent hold". Although combustible cargo may be transported in the same hold as a RAM package, "the heat flux from a combustible cargo fire beside a package is less than for the certification fire".

Yamamoto, et. al. [RBP030] undertook an analytical study of the thermal environment that would exist in the engine room of a purpose-built maritime vessel, where packages of plutonium were assumed to be located in adjoining cargo holds of the ship. It was concluded that the construction of "purpose-built ships provides excellent protection from an engine room fire thermal event for sea transport of plutonium oxide powder in packages; and it was estimated that under the worst conditions envisioned in the study that the "seal area [of the packages] will remain within its temperature operating range" for a two hour fire.

Lopez et al. [27] the NCI-21PF UF6 package was analytically assessed for the regulatory fire and for more realistic conditions where the regulatory fire was offset from the package, located with the edge of the pool fire 1, 5 and 10 m from the side of the package and 1 m from the end of the package. The times required to reach the same temperatures as in a fully engulfing regulatory fire ranged from 70 to 150 min for the offset fires. Assessment of the probabilities of exceeding regulatory conditions in accidents reveals only a limited number of circumstances under which regulatory conditions may be exceeded, and that the likelihood of UF6 being dispersed by fire is small.
Lopez et al. [4] reported on an analysis of a rail cask placed on the ground at different locations relative to the centre of a 10 h fire. Three different pool fire shapes were simulated, and the distance of the cask centre from the centre of the pool was increased in increments of 2 m (i.e. 2, 4 and 6 m). In the worst cases calculated, assuming seal failure at 350 °C, the temperature of concern was exceeded in 1–2 h. In addition, it was concluded that the fuel rods would probably not reach their temperature of concern regardless of fire duration when the cask was located just a few metres away from the fire.

In addition, Lopez et al. [4] analysed five scenarios where it was assumed that a train carrying a steel–lead–steel spent nuclear fuel cask derailed and the car carrying the cask overturned in such a way that the cask ended up lying on the ground still attached to the railcar by the tiedowns. The five scenarios used combinations of the pool fire lying under or next to the cask, and with different prevailing wind conditions. In some cases, it was found that the rail car bed provided significant protection, and for the pool under the cask, lead melt was predicted to start after 33–41 min. These results are similar to those obtained by Pope, et. al. [28] and Hamann, et. al. [26] where a detailed assessment was performed of the pool fire test of a 67 t cask inside a 68 t transport railcar (which had significant steel structure enveloping the cask). Because of the intervening structure, the test was not necessarily a fully-engulfing event; the structure perturbed the behaviour of the fire and acted in part as a radiation shield. The test was performed until evidence of cask failure was detected, and it was determined the cask/railcar system was exposed to roughly six times the thermal input that would have resulted from the 30-minute, 800°C thermal test required by the Regulations. The assessment included summarizing both pre-test predictions and post-test modelling; the latter performed with a view to obtaining a better understanding of the parameters involved in such fire exposures. It was concluded that "the computer analysis of this test configuration proved to be difficult, not because of the cask model, but because of the definition of the boundary conditions." The effects of variations with time of the absorptivity of the cask and the reduction of heat input to the cask from intervening structure were addressed in these analyses. The paper concluded that the "present regulatory criteria appear to properly account for the wide range of environments a package might be exposed to in a severe accident."

Lopez [4] also studied fire environments that could occur in rail accidents involving typical casks exposed to a fire jet impinging on three locations of a cask. Duration of exposure before potential spent fuel rod burst, and the temperatures were predicted to be 2 h and 20 min if the fire jet has a temperature of y1200 °C (2192 °F), 1 h and 30 min for ~1400 °C (2552 °F) and 1 h for ~1600 °C (2912 °F).

In order to address concerns that a spent nuclear fuel cask could become involved in an accident in a tunnel that resulted in a fire, the US NRC undertook a study of a severe tunnel fires and fires involving highway underpasses in the USA [29 – 36].

Considering different casks involved in the tunnel fire scenarios, the results of these studies indicated, for the various tunnel fires examined, that:

- "The results of this evaluation also strongly indicate that neither spent nuclear fuel (SNF) particles nor fission products would be released from a spent fuel transportation package carrying intact spent fuel involved in a severe tunnel fire such as the Baltimore tunnel fire" [31].
- "For the Howard Street Tunnel fire, the peak calculated temperatures within the tunnel were approximately 1,000 °C (1,800 °F) within the flaming regions, and on average approximately 500 °C (900 °F) when averaged over a length of the tunnel equal to three to four rail car lengths. Because of the insulation provided by the thick brick walls of the tunnel, the calculated temperatures within a few car lengths of the fire were relatively uniform, consistent with what one would expect to find in an oven or furnace. The peak wall surface temperature reached
about 800 °C (1,500 °F) where the flames were directly impinging, and on average 400 °C (750 °F) over the length of three to four rail cars. The steel temperature of the rail cars would be expected to be similar to the surrounding gas temperature because of the long exposure time and the high thermal conductivity of steel.” [32]

- “USNRC staff evaluated the radiological consequences of the package response to the Caldecott Tunnel fire. The results of this evaluation strongly indicate that neither spent nuclear fuel (SNF) particles nor fission products would be released from a spent fuel transportation package carrying intact spent fuel involved in a severe tunnel fire such as the Caldecott Tunnel fire.” [33]

- The assessment of materials involved in the Newhall Pass tunnel fire [34], indicated that temperatures experienced by these materials ranged from 560 °C to as high as 884 °C, with great variability as a function of location in the tunnel.

Easton and Bajwa [35] further elaborated on these studies, which included an analysis of how rail car components might react to the tunnel fire environments [36]. Specifically, they considered the potential for a release of radioactive material from the transport casks. In summary, the suite of studies considered various cask designs, ranging in mass from 24 to 126 t, and actual tunnel fires that have occurred in the USA. The study found that, for the tunnel fire events, a release of radioactive material from any of the casks analysed would be unlikely, and that any potential release would be very small, less than an A2 for the radionuclides of greatest concern.

In addition to the above noted tunnel fire studies, NRC also considered the environment that existed in a severe transport accident which occurred in 2007, in California (known as the MacArthur Maze Accident and Fire) [37]. This accident involved a tractor trailer carrying gasoline which impacted an overpass support column and burst into flames. The subsequent fire, which burned for over 2 h, led to the collapse of the overpass. An analysis was made concerning how a radioactive material package, including one carrying spent nuclear fuel, could have behaved if it had been involved in this accident. Bajwa, Easton, et. al. [37, 38] described the assessment where the effects of the fire on the overpass structure were used to estimate the temperatures in the fire. The study illustrated that the thermal input from the fire was very non-uniform; it was estimated that the temperatures experienced by various elements of the overpass ranged from 850 °C (1562 °F) to 1000 °C (1832 °F) at the hottest locations; however, at other locations on the affected structure, it is estimated that the temperatures were much cooler – ranging from as low as 500 °C (932 °F) to 750 °C (1382 °F).

The US National Academies study that looked at the safety issues associated with land transport of irradiated nuclear fuel and high-level radioactive waste in the US [11 – 13] identified concerns with the potential of tunnel fires occurring concurrently with a cask in a tunnel. To address this one concern, the study recommended that "transportation planners and managers undertake detailed surveys of transportation routes to identify potential hazards that could lead to or exacerbate extreme accidents involving long duration, fully engulfing fires, and also take steps to avoid or mitigate such hazards”. By imposing this one operational constraint on surface transport of these radioactive materials, it was determined that the existing structural, thermal and immersion hypothetical transport accident condition tests provide an excellent margin of safety for the transport of radioactive material. It further noted that studies of accidents involving severe fires have been completed. These studies have confirmed that spent fuel packages would be expected to withstand very severe fires without the release of any fission products from the spent fuel.

A.4.6. Follow-on Studies into the adequacy of the enhanced 200 m water immersion test

Itoh et al. [20] describes tests where casks larger than 100 t for the transport of spent fuel and high level waste were exposed not only to pressures resulting from the immersion of the package to the depths of 15 and 200 m as required by the regulations, but also to a pressure
equivalent to an immersion depth of 3000 m (30 MPa). Two cycles at this pressure, for 60 min each, were applied. No leakage was detected at any sealing boundary during the immersion test, no reduction in the sealing characteristics was observed and the stress intensity of the package main body was very small, indicating that the package would retain its integrity against immersion to the water pressure equivalent to a head of 3000 m.

Malesys [19] summarized a test of an FS 47 packaging for carriage of Pu oxide powder which was tested to deep immersion environments. He concluded that “When the pressure reached 930 bars, the first significant plastic deformations appeared, but the vessel was still leaktight. The test was further pursued, and leaktightness was lost for a pressure of 3430 bars (equivalent to a head of water of tens of thousands of meters)”. He further reported that analysis and scale model tests of packages from the TN 12 and the TN 28 VT family of casks were performed against a deep immersion. This assessment showed that these casks can withstand an immersion with a head of water of thousands of metres, which are many times higher than the 200 m immersion depth required by the transport regulations. Finally, he reported that a scale model of the FS 65 packaging for carriage of fresh MOX fuel was tested to deep immersion environments. The tests demonstrated that the package design can withstand the 200 m regulatory water immersion test “with a safety margin ranging from 2 to 9, depending upon the type of basket which is fitted in the packaging”.

The requirement that a certain packages needed to be able to withstand immersion to a depth of 200 m without rupture was established in the 1985 Edition of the Regulations in order to provide the capability to recover a package lost at sea if it were located on the continental shelf. Recovery at greater depths was not deemed to be necessary since long-term, slow corrosion of the package and/or its contents at greater depths had been determined to pose essentially not risk to the public or environment. Corrosion of packages and/or their contents located at depths of 200 m or less was deemed to be of concern. In a study reported by Pierce, et. al. [39] in 2006 of plutonium transport by sea between Europe and Japan considered potential accidents including the potential sinking of the vessel or loss of a package from a transporting vessel into the ocean. The study considered corrosion for generic transport packagings to estimate the time it would take to breach a typical containment boundary. “This study was done to determine an estimate of the length of time available to recover the packaging if it is not feasible to abandon the material on the sea floor for reasons of safety, security, environment or politics”. The study showed that, depending upon the design of the package and the locations most vulnerable to corrosion, cask breach could occur from as early as 3 months to as long as 24 years after immersion. Thus, for packages submerged at depths of 200 m or less, where the packagings are designed to not rupture, it was concluded that sufficient time is available to perform recovery if needed.

Tsumune et. al. [40] reported on simulations of radionuclide concentrations for the hypothetical release from a submerged transport package of fresh mixed oxide fuel in the Japan Sea and the global ocean. Simulated concentrations were quite small compared to the background concentration already existing by the fallout, and the effective doses of radiation exposure to the public for both cases were much less than the effective dose limit (1 mSv/year) by ICRP recommendations.

References for A.4


{RP145}

[29] Nuclear Regulatory Commission, *NRC releases evaluation of effects of Baltimore tunnel fire on rail transportation of spent nuclear fuel*, No. 03-039, Nuclear Regulatory Commission, Washington, DC. (March 27, 2003). {RP144}


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Documents not included in preceding list


"Transportation for the Nuclear Industry" Blackpool, UK, 2-4 November 1999 (1999); also provided as a handout to all members of IAEA-TC-1196 meeting, (2000). {RBP023}


APPENDIX 5 - Detailed Technical Basis for Each Criticality Safety Requirement

Text yet to be developed
APPENDIX 6 - CONTRIBUTORS TO THE DOCUMENT

IAEA Technical Meeting TM-38950, 11 to 15 October 2010

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C. Getrey  France  M. Kirose  WNTI
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IAEA Consultants Services Meeting CS-05, 7 to 11 February 2011

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IAEA Technical Meeting TM-41001, 14 to 18 March 2011

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### TRANSSC 25, Working Group 3

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### IAEA Consultants Services Meeting CS-45203, 3 to 7 December 2012

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| Chapter 10 | B. Dekker, WNTI; R. Pope, USA |
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| Appendix 4 | R. Pope, USA |
| Appendix 5 | D. Menntedahl |
| Appendix 6 | R. Pope, USA |
ENDNOTES

i  NOTE: only the graded approach is so far addressed here. The experts working on this document have, as yet, been unsuccessful in identifying where the principle of defence in depth has been specifically addressed in the development of the regulations. This section may also need to also address:
(a) Safety Culture (which is not mentioned in the Regulations), and
(b) Management systems (which was only mentioned once in the 2009 edition, but now in the 2012 edition, is addressed more completely – e.g. paras 102, 105, 228, 306, and in multiple places in Section VIII). Para 228.2 of the draft replacement document for TS-G-1.1 discusses briefly why management system has been introduced.}

ii  NOTE: Text yet to be developed and added here.

iii  NOTE: In assembling this version of the TecBasDoc pursuant to the TRANSSC guidance, the administrative controls have not been addressed in this chapter, though such was included in Appendix 16 of the March 2012 version of the draft. Where, if at all, should administrative controls be addressed? The paragraphs following this endnote introduce briefly the concept of administrative controls.

iv  NOTE: additional effort for the background and the technical bases for both TI and CSI is needed here.

v  NOTE: The text below on CSI needs to be coordinated with what Dennis Mennerdahl's text on CSI in Chapter 11.

vi  NOTE: This section needs to be developed to address UN Numbers, noting that
1. the assignment of the UN Number to the radioactive material being shipped according to requirements in the 2012 edition of the Regulations (para. 401 and Table 1); and
2. the use of the UN Numbers is then specified in Section V of the Regulations. The history and purpose of the UN Numbers should then be elaborated, including pointing out that the UN Numbers were first introduced in the 1985 edition of the regulations, and that one primary use of them is to guide emergency responders on the hazards posed by the shipment and the means for properly responding – consistent with the approach taken by modal authorities in developing emergency response guides.

vii  NOTE: RB Pope has been unsuccessful in obtaining a copy of this report, but it is referred to in the AG-365.2 report; and that is the basis for the following statements.
viii NOTE: need to find text for the technical basis for the activity limits for LSA material, in general.

ix NOTE: the technical basis for imposing this leaching test on LSA-III materials has not yet been found in the available documents leading to the 1985 edition of the regulations; however, the test itself appears to be derived from the special form test that already existed in the regulations (i.e. para. 736 of the 1973 edition).

x NOTE: Need to locate the TECDOC on the LSA/SCO CRP and summarize the findings.

xi Need to look in the archives for documentation from the revision panels, associated CSMs and TMs, and possibly papers from PATRAM or in the Journal leading up to the 1996 edition of the regulations. Need to determine the basis for the actual thresholds for moving to Type C packages.

xii NOTE: the mid-1964 Panel Meeting [RBP036] did not address this issue of the bending test for special form material.

xiii NOTE: the basis for the leaching test and the alternative to it needs to be researched.

xiv NOTE: nothing has as yet been found on documentation discussing the addition of the 1.7 m drop (which occurred between the 1967 and the 1973 editions of the Regulations) and the change to testing the complete package for gaseous contents.; the reports of the relevant revision panels and associated technical committees and/or consultants services meetings would need to be available to further establish technical basis here.

xv NOTE: the actual technical basis for applying Type C requirements, and the later logic used to impose these requirements on all fissile material packages, could not be fully established by documents that were available to Mr. Pope when drafting the text in early 2012; some the reports of the revision panels and associated technical committee and/or consultants services meetings have been obtained from archives, but still need to be searched to further establish technical basis here. A further search for meeting reports leading to the decision to add Type C package requirements may ultimately be required.

xvi NOTE: one page of CB024 is upside down, it is suggested that this file be replaced with file RBP016, which has the upside down page corrected.
NOTE: in an email from Ben Dekker to Chris Bajwa and Ron Pope, dated 9 September 2013, he noted the following:

“I have been in touch with Henk Selling, Mathieu ter Morshuizen and Hans van Halem. However the two meeting reports

International Atomic Energy Agency, Revision Panel Meeting Report,
Vienna (25-29 September 1995).

International Atomic Energy Agency, TRANSSAC Meeting Report,
Vienna (26 February – 1 March 1996).

could not be traced in the NL archives.

I also have approached Rick Rawl. You have heard his response during PATRAM to one of your papers, saying, approach the national CAs to look in their files.

I did this for the NL side already, as mentioned above.”

When the document is complete, the acronym list will need to be checked for completeness.

NOTE: need to search later Agency meeting documents to locate where the decision was made that the release limit would be $10^{-6}$ $A_2/h$ for both Type B(M) and Type B(U) packages.

NOTE: WP4 of AG-406 is not as yet available.