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TRANSPORT OF RADIOACTIVE MATERIAL

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TRANSPORT DE MATIERES RADIOACTIVES

INHOUD

Dit nummer bevat teksten van de uiteenzettingen gedaan op 16 oktober 1996 ter gelegenheid van van een vergadering van de Belgische Vereniging voor Stralingsbescherming, de Belgian Nuclear Society and de Nederlandse Vereniging voor Stralingshygiëne

VERVOER VAN RADIOACTIEVE STOFFEN

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TRANSPORT OF RADIOACTIVE MATERIAL
Introduction

The revision of the IAEA Regulations for the safe transport of radioactive material has been completed.

The changes with respect to the current regulations has been presented the 16th of October 1996 at a meeting jointly organised by the Belgan Radiation Protection Association, the Belgian Nuclear Society and the Dutch Radiation Protection Society at the Nuclear Research Center in Mol. Some "hot topics" were also discussed.

This meeting was incorporated in a training course on the transport of radioactive material, which was organised by the IAEA and the European Commission, in collaboration with SCK!CEN and the Radiological Protection Office of the Belgian Ministry of Social Afffairs, Public health and Environment.

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WHAT IS NEW IN THE 1996 EDITION OF THE IAEA TRANSPORT REGULATION

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INTRODUCTION

Since the Regulations were first published in 1961, they have been comprehensively revised at intervals of about ten years. It has been argued that this provides a reasonable balance between the need for regulatory stability and the need to keep abreast of scientific and technical developments. Transport needs change as new materials emerge from new programmes of work such as decommissioning. In addition, the publication of new standards, such as those in radiation protection, can lead to a need for consequential changes in the Regulations.

In December 1996 the International atomic Energy Agency published the 1996 Edition of the *Regulations for the Safe Transport of Radioactive Material* (formerly Safety Series No. 6) in a document called ST-1. That document supersedes all editions of the Regulations issued under Safety Series No. 6 (SS6).

For convenience, the requirements to be met for the transport of specified types of consignments are included in an abbreviated form as Schedules in the publication, ST-1. Those Schedules reproduce many of the provisions of the Regulations, but do not contain any additional requirements. They simply provide a summary of the main provisions applicable to that type of consignment and they replace the information previously published under Safety Series No. 80.

In support of the 1996 Edition the IAEA has prepared a companion document called ST-2, *Advisory Material for the Regulations for the Safe Transport of Radioactive Material*. That companion document contains both the advisory and explanatory text that supersedes all editions of the Safety Series Nos. 7 and 37. An important part of the revision process was to review the content of the explanatory and advisory material to ensure that it was kept up-to-date with changes in the Regulations and current knowledge. In addition to the text supporting individual paragraphs of the Regulations there are several appendices which have been extensively updated:

- i) the Q system for the calculation and application of A_1 and A_2 values;
- ii) radionuclide data;
- iii) quality assurance in the safe transport of radioactive materials;
- iv) acceleration values and calculation methods for package tie-down forces;

v) guidelines for safe design of shipping packages against brittle fracture; and
vi) criticality safety assessments

The decision was also taken to delete the appendix that dealt with radiation protection programmes for exclusive use vessels because such programmes are now needed as a general requirement for the transport of radioactive material. Other appendices were either left largely unchanged or were amalgamated into the main body of the text of ST-2 or were incorporated into other appendices.

MAIN CHANGES TO THE REGULATIONS

Basic Safety Standards changes

One of the major topics considered in the revision process leading to ST-1 was the incorporation of the *International Basic Safety Standards for Protection Against Ionizing Radiation and for the Safety of Radiation Sources*, Safety Series No. 115, published in 1996. This document is often referred to as the BSS. These Standards were revised to reflect the consensus on the recommendations of the International Commission on Radiation Protection published in 1990. The transport Regulations call upon the Basic Safety Standards in the General Provisions for Radiation Protection and compliance with the Regulations is deemed to satisfy the principles of the BSS. These changes are covered in detail by the paper presented by Mr. L. Baekelandt t this meeting.

Basis for new definition of radioactive material

A cornerstone of the Regulations in the past was the definition of radioactive material as being any material having a specific activity greater than 70 Bq/g. Unfortunately, this concentration had no radiological protection basis. The BSS established some radiological criteria for exemption and consistency of the transport regulations with the BSS was seen to be important. Following assessment of the suitability for application to transport, the BSS exemption values were adopted into the Regulations. The issue of exemption is the subject of another paper by Mr. L. Baekelandt presented at this meeting.

Plutonium (Pu) in air shipments

The introduction of a new package type - the Type C Package and a new material type - low dispersible material (LDM), were two fundamental changes to the Regulations that traditionally have been written to be independent of the mode of transport used. The need for Type C packages was driven largely by the legislation passed in the United States of America prohibiting the US Nuclear Regulatory Commission from licensing the air shipment of plutonium until a safe container had been designed. The criteria for the safe design of a container for the air transport of plutonium were published in a document called NUREG-0360. NUREG-0360 seeks almost absolute protection irrespective of the probability of an accident. In response the international community wanted to develop a consensus standard for packages carrying large quantities of radioactive material by air. There was recognition that air accidents are more severe than for surface modes (road, rail and sea) especially for impacts. It was also appreciated that the air mode was a very safe mode of transport in terms of the number of accidents for a given distance travelled, but a risk only approach would lead to relaxed not more stringent requirements for air transport. Therefore it was decided to seek a package standard that would achieve a level of protection that was comparable to that for surface transport. These developments are addressed in a separate paper by the author and presented at this meeting.

Uranium hexafluoride (UF_6) - other hazards, pressure, temperature

The decision to draft regulations for a specific material represents a significant change in philosophy in the Regulations which have been framed to apply generally to all radioactive materials. The impetus for the change was driven more by a desire to ensure physical and chemical toxicity safety in the event of an accident involving a fire, rather than concerns about radiological or criticality safety. This change reflects the importance of UF_6 within the fuel cycle, the very large quantities being shipped and the peculiar physical and chemical properties of the material. The new provisions for packages containing UF_6 are restricted to package performance standards and administrative requirements. Otherwise the Regulations draw on the International Organization for Standardization document ISO 7195:1993(E) for more detailed technical specifications. Mr. G. Sert presents this subject in his paper delivered at this meeting.

OTHER KEY AMENDMENTS

Creating two package indices

The purpose of the Transport Index (TI) has been to limit the hazards associated with the accumulation of radioactive material packages. These hazards are two, namely radiation dose rates and inadvertent criticality. Since the beginning of the IAEA Regulations the TI has been used for both purposes and this arrangement has worked reasonably well. However, it did give rise to situations which were overly conservative and also resulted in several other difficulties and ambiguities. Part of the problem lay in the fact that for some applications there was a need to know how the TI was derived (radiation or criticality) in order to properly apply it.

In ST-1 these problems were solved by treating the two hazards separately and giving each its own index. The consequence is that the TI has been much simplified, in that it now only applies for radiation protection purposes. It continues to be based on the radiation level at 1 meter from the package surface. The fairly complex Table XI of SS6 is replaced with the simpler Table IX of ST-1.

As its name implies the Criticality Safety Index (CSI) limits the accumulation of packages solely based on criticality considerations. Its method of calculation is the same as that previously used when the TI was being determined for criticality safety purposes. The corresponding limitations of Table XI in SS6 are found in Table X of ST-1 which provides the CSI limits for freight containers and conveyances.

The introduction of the separate CSI necessitated a new label for all packages containing fissile material unless containing only fissile excepted quantities. This label must be affixed adjacent to the radioactive category labels, and must not cover any required markings. Similarly the CSI also must be included on the transport documents for fissile consignments other than those which are fissile excepted.

Separation of the two indices will allow shipments to be controlled on the basis of the specific value of concern. Previously, for example, a TI based on criticality would result in a value which was so high that the segregation tables of mode-related regulations caused problems of stowage on board aircraft or vessels in relation to trim and available space. It could also cause problems when ensuring adequate separation distance for additional personnel required on conveyances for the purpose of fulfilling physical protection requirements. Now, fissile packages with low radiation levels will not have to be segregated from persons solely on the basis of a high TI. In addition, simplification of several tables will hopefully introduce clarity which should enhance compliance with the Regulations.

New UN Numbers

It has long been recognised that many of the existing UN numbers for Class 7 material were not as useful as they could be. For example, the most common number, UN 2982 does not provide any more information about the material being transported other than the fact that it is radioactive. Since the primary purpose of displaying UN numbers on packages and on conveyances is to provide a key into emergency response procedures in a language-independent way, it was decided to revamp the numbering system so that it might lead to more useful response procedures.

In a similar way, it is clear that the Schedules which were formerly in Safety Series No. 80 , are more useful if they have a direct relationship with UN numbers. In this case, any one UN number would lead to only one schedule where the user will find most of the references and regulations needed to make that particular consignment. This is indeed the case in the new numbering system developed for ST-1, where a UN number is assigned to each of the Schedules.

The Schedules are now appended to the Regulations instead of being in a separate document and therefore SS80 is superseded by ST-1. The existing UN numbers for uranium hexafluoride were retained because of its importance as a commercial substance and its significant subsidiary risk. However, a separate set of numbers was assigned for packages containing fissile material. Some UN numbers became redundant, but as none of the deleted numbers are re-used, the transition process will not be a problem.

UN number changes also required changes to their associated proper shipping names. All names which did not begin with "RADIOACTIVE MATERIAL..." had those words added to ensure that any alphabetical list of Class 7 Names/Descriptions will have all of the Class 7 entries in one block. In addition, the letters "N.O.S." were removed from all proper shipping names because they also became redundant. The UN Recommendations for the Transport of Dangerous Goods (the Orange Book) has already been republished incorporating the new UN numbers.

It is to be hoped that the new UN numbers will facilitate more specific emergency response procedures, better harmonise the transport of Class 7 materials with those of other dangerous goods and help with compliance checks and controls through a numerical link with the Schedules.

Criticality safety

The provisions for fissile material were carefully reviewed during the revision process. As a result, the paragraphs dealing with the requirements for packages containing fissile material have been extensively re-arranged in an attempt to improve clarity. There are also some technical changes which are described below.

Definitions of terms with a bearing on fissile material

For the purposes of ST-1, ^{238}Pu is no longer considered fissile. Although it can be made to support a fast neutron chain reaction this is not a credible circumstance in transport.

A definition for a confinement system has been introduced to ST-1. This was introduced to deal with the short comings of using the term containment system which is only partially relevant to criticality safety. A confinement system is the assembly of fissile material and packaging components such as spacers and neutron poisons that are intended to preserve criticality safety.

Also, the definition of LSA-I is altered in ST-1 and now allows fissile material to be present in excepted quantities. The definition in SS6 appeared to exclude fissile material, even in excepted quantities and this anomaly was rectified.

Fissile exceptions

A mass limit per consignment has been introduced into ST-1 for three exception categories which hitherto had implicitly relied upon packaging to maintain subcriticality. These categories are:

- i) packages not containing more than 15 g of fissile material
- ii) packages not containing more than 5 g of fissile material in any 10 litre volume of material; and
- iii) fissile material in a homogenous hydrogenous solution or mixture where the ratio of fissile nuclides to hydrogen is less than 5% by mass .

Reliance on packaging for fissile exceptions is erroneous especially in accident conditions where neither the geometry nor the degree of moderation can be guaranteed. The mass limit per consignment was introduced to dissuade a consignor from circumventing the Regulations by transporting a very large number of packages each containing just less than the excepted quantity limit. Under ST-1 only one type of fissile exception per consignment is allowed, which is more cautious than the SS6 fissile exceptions which applied to packages meeting one of the six exception categories.

The single package

A slight relaxation of the Regulations has been permitted when considering the single package in isolation following an accident. If, following the accident tests, the confinement system remains within the packaging, reflection by 20 cm thick water of the damaged package is allowed. Previously it was required to consider the more reactive condition between reflection of the package and reflection of the containment system.

It is not possible to demonstrate subcriticality for all designs of package used to carry enriched UF_6 if water is allowed to leak into them. A large number of such packages exist those designs are excepted from the requirement for multiple high standard water barriers if it can be demonstrated that the valve has not been impacted by any component of the packaging during the mechanical tests and where in addition the valve remains leaktight following the thermal test.

The decision to introduce a more robust package for the air transport of radioactive material led to a consequential decision to provide equivalent protection against accidental criticality in air transport for all fissile material. ST-1 places a new requirement on packages containing fissile material if designed to be transported by air.

Individual packages whether Type IP-2F, Type IP-3F, Type AF, Type B(M)F, Type B(U)F or Type CF will have to be designed to be subcritical under conditions consistent with the tests for Type C packages when reflected by at least 20 cm of water (although water in-leakage need not be assumed) if they are to be used for in air transport. This precludes extremely large criticality events that can occur when initiated by a rapid change in geometry. Worst case assumptions regarding geometric changes and rearrangement of components will need to be made, unless other assumptions can be supported after a specimen has been subjected to the enhanced test requirements.

In addition, package designs which relied on multiple high standard water barriers in the demonstration of subcriticality of a single package in isolation will no longer qualify for transport by air unless those barriers can be shown to withstand the test requirements for Type C packages. This precludes a criticality event being caused by water inleakage alone. In particular it deals with the concern that the failure of such barriers in an air crash in just a single package might lead to a criticality accident.

No additional requirements have been placed on arrays of packages in air transport on the basis that high speed aircraft crashes will lead to the packages being dispersed. The existing requirement for arrays of packages to be assessed for accident conditions of transport in the damaged condition associated with surface modes should cover the situation of lower speed impacts which may lead to conditions of compression.

Burnup credit

There has been a trend towards a higher initial enrichment, from around 3% in the early 1980's to above 4% in current PWR operation. The cautious assumption that the fuel is fresh is becoming increasingly restrictive. Taking credit for irradiation or burnup is one the methods of increasing the payloads of flasks that were designed for fuel of lower initial enrichment. ST-1 requires that the isotopic compositions used in the criticality assessment are either those providing the maximum neutron multiplication consistent with the irradiation history or a demonstrably conservative estimate of the actual neutron multiplication. To reinforce this, emphasis is on conservatism in the operational phase of loading a package, where the criticality assessment is not based on the maximum neutron multiplication, a pre-shipment measurement is now required in order to confirm that the assumed composition for assessment purposes is conservative compared to the actual isotopic composition and distribution.

Package design and performance testing

The requirement for packages containing fissile material to be subjected to a 0.3 m on each corner has been removed in order to introduce a measure of consistency in package testing. The tests to withstand normal conditions of transport were considered to be sufficient and for larger packages, the additional eight corner drops (for cuboid packages), linked to the other free drops for normal and accident conditions were often regarded as a nuisance.

ST-1 has introduced an additional fissile related criterion such that a crush test is required for all packages having a mass not greater than 500 kg and a density not greater than 1000 kg/m^3. This test requirement is applied to all those packages containing fissile material that are considered most vulnerable to crushing. Previously, the test was applied to only to fissile packages if they contained more than 1000 A_2 not in special form. This quantity threshold was a radiological criterion that was unrelated to criticality safety. For non-fissile or fissile excepted packages, the threshold value of 1000 A_2 not as special form continues to act as the radiological trigger for the crush test to be required.

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REQUIREMENTS FOR UF_6 IN THE 1996 EDITION OF SAFETY SERIES N° 6

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1. THE UF6 MATERIAL

The Uranium Hexafluoride (UF6) has been selected as the chemical form the most easy to be enriched in U235 since the fluorine component constitutes only one isotope.
It is also a solid cristal at ambient temperature that can be easily transformed to gas at a moderate temperature (triple point at 64°C).
Therefore UF6 was chosen for enrichment by gazeous diffusion, first, and then by ultra centrifugation.

It exists in three isotopic composition

- the natural composition, containing 0,72% of U235 ; this is the feed material before enrichment
- the enriched compositions, containing between 1% and almost 100 % of U235
- the depleted compositions, containing typically 0,15 to 0,35% of U235, also called tail material, by product of the enrichment process.

Several kinds of hazards may be encountered with UF6 :

The natural radioactivity of uranium mainly U238, U235, U234 and daughter products such as Pa 234, Th 230, Ra 226, Rn 222 etc...

- the fissile properties of uranium may create a criticality risk for specific conditions of enrichment and concentration of uranium and hydrogen or other moderating elements,

- the chemical properties of uranium and fluorine causes UF6 to be a highly toxic material ; HF and UO2F2, the by products of the reaction of UF6 with water vapour are also of the same range of chemical toxicity ; HF is a corrosive and toxic gas

- the physical properties of UF6 with a low temperature triple point, and a large fusion and liquid thermal expansion, creating a rupture hazard when the containment is submitted to high temperature environments, either by heat expansion or by pressure increase (see figures 1 and 2)

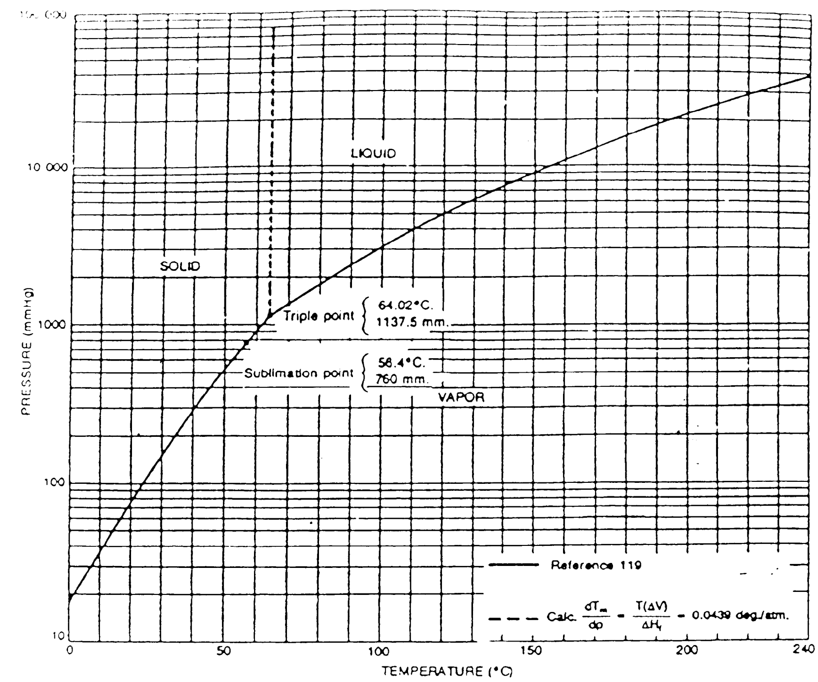

FIG.1 : Phase diagram of UF6 [8]

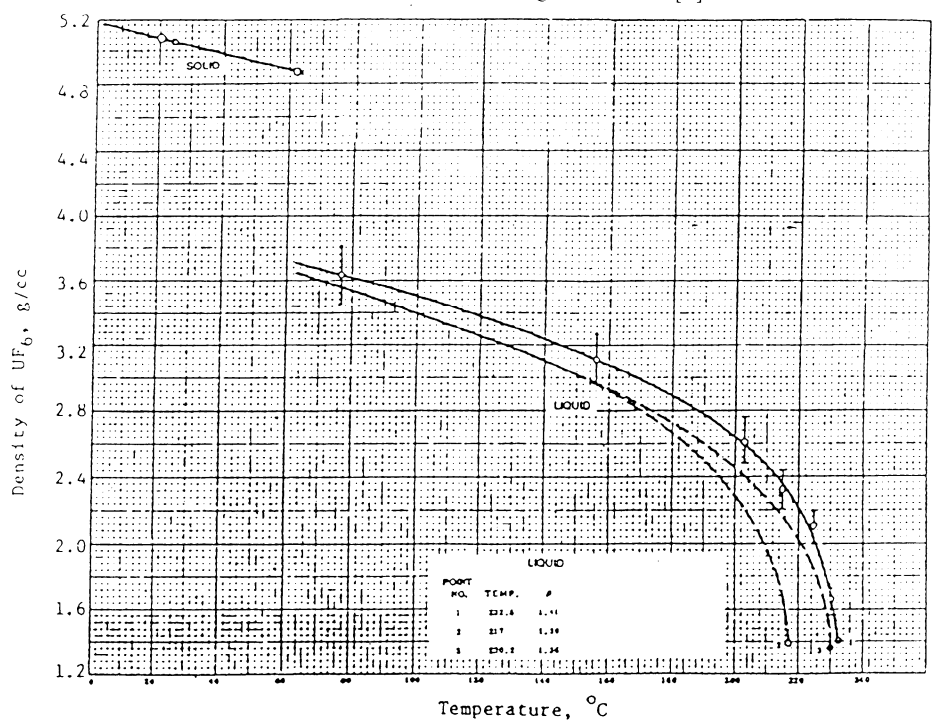

FIG.2 : Density of solid and liquid UF6 [8]

Non fissile or fissile excepted uranium hexafluoride has been given the UN NR, 2978 (Table VIII of new IAEA SS.6 [1]).

2. PAST REGULATIONS

Until and including the 1985 edition of the IAEA regulations on the safe transport of radioactive materials, the requirements were

- for enriched, fissile UF6, the material shall remain subcritical even in accident conditions. Therefore enriched UF6 has been transported in cylinders (ANSI Standard N14.1 - 1982, Type 30B) that remain safe under impact and fire conditions, thanks to their impact and fire protection by an overpack.

- for natural or depleted UF6, the material was considered as a LSA II material and transported in Industrial Packages type 2 (ANSI Standard N.14.1 1982, type 48 Y....), that need not necessarily remain safe in accident conditions, consistently with their radiotoxicity but regardless of their chemical toxicity. For this chemical hazard, the road and rail transport regulations required that the containers used for UF6, be tested at a pressure of 10 bar. This pressure test was more severe than the 4 bar pressure test required for other chemical substances such as hydrogen chloride. It guaranteed safe plant operations such as liquid phase filling, heating up to 120°C for filling etc... but this increased mechanical resistance did not bring any guarantee as concerns the behaviour in a fire.

However, after the sinking of the Montlouis Ship with a cargo freight of 30 cylinders of 48 Y type filled with natural UF6, despite the severe sea conditions that caused multiple shocks on the cylinders, none of them failed. In practice they were designed to withstand a pressure test at 28 bar. Therefore they were much stronger than other tanks used for chemical materials of similar toxicity and their increased resistance helped their survival. But the question of the survivability to a fire was raised again, which lead the French delegation to propose in 1986 to SAGSTRAM that a guidance should be developed under the leadership of IAEA, on provisions to be taken with respect to chemical hazards specific to the transport of UF6.

3 THE REQUIREMENT OF THE 1996 EDITION OF SAFETY SERIES 6

Different kinds of requirements have been specified ; they are associated to operational safety, over pressure safety, fire safety. They are applicable to all packagings containing more than 0.1 kg of UF6, since studies have shown there was no significant chemical hazard for the release of quantity of UF6 below this value.
The Consultants Service Meeting 1-3 october 1990 adopted an exemption level of 0.1 kg for packages designed to contain UF6. The 0.1 kg is well below the toxic limit of 10 to 40 kg (based on the work of C. RINGOT and J. HAMARD[6] and of A. BIAGGIO and J. LOPEZ-VIETRI[7]) and will permit the shipment of sample tubes containing UF6. These sample tubes are constructed of plastic or other non pressure containing material, and thus, cannot explode. The sample tubes contain less than 0,065 kg UF6.

3.1. Solid State requirements (SS6 paragraph 419)

The mechanical stresses occurring during transport due to shocks, vibrations, tie-down forces etc... should not be combined in routine transport with any mechanical stress caused by any internal vapor pressure of UF6 : *"The uranium hexafluoride shall be in a solid form and the internal pressure of the package shall be below atmospheric pressure when presented to transport.*

This requirement means that after filling with hot UF6, the package shall be allowed for sufficient cooling time for complete solidification before shipment. Typical minimum cooling times in ambient air are 3 or 4 days.

3.2. Filling requirement (SS 6 paragraph 419)

This operational requirement mainly concern the safety of the UF6 cylinder when used in the enrichment facilities ; it prevents the rupture of cylinder due to the overpressure that could be caused either by overheating or overfilling.
"The mass of uranium hexafluoride in a package shall not exceed a value that would lead to an ullage smaller than 5% at the maximum temperature of the package as specified for the plant systems where the package shall be used".

3.3. Packaging design, manufacturing, inspections and transport requirements (SS 6 paragraph 629)

"Except as allowed in paragraph 632, uranium hexafluoride shall be packaged and transported in accordance with the provisions of the International Organization for Standardization document ISO 7195 : "Packaging of uranium hexafluoride (UF6) for transport...."
The reference of the today (1996) applicable standard is ISO 7195 : 1993 (E)[5].
This standard mainly specifies for different cylinder capacities the conceptual dimension, the material grades, the manufacturing process, inspections including in service routine or periodic ones, performance tests including leak tightness test, hydraulic test, cleaning, marking, certification, assembling of valve protectors, etc...

Paragraph 632 allows not to meet the requirements of ISO 7195, subject to multilateral approval (paragraph 805.a).
Therefore if the requirements of ISO 7195 are not met, the packages cannot be unilaterally approved.

3.4. Pressure resistance (paragraphs 630.a and 718)

630.a : The packagings "*shall be designed to withstand without leakage and without unacceptable stress as specified in ISO 7195, the structural test as specified in paragraph 718.*

718 : "*Specimens that comprise or simulate packagings (...) shall be tested hydraulically at an internal pressure of at least 1.4 MPa but when the pressure is less than 2.8 MPa, the design shall require multilateral approval".*

As a consequence some thin walled cylinders might not be unilaterally approved (48H-48HX-48G-48OM).

3.5. Free drop test (paragraph 630.b and 722)

The requirement of no loss or dispersion of UF6 caused by the free drop test is not a new one since it was already applicable under IAEA-SS6-1985 to the package classified as IP2.

3.6. Fire test (paragraphs 630.c and 728)

630.c : "*Each package (...) shall be designed so that it would (...) withstand without rupture of the containment system the test specified in paragraph 728*."
The paragraph 728 refers to the thermal test representative of hypothetical accident conditions (800°C during 30 minutes).

This requirement is quite new since it has never been included either in standards as ANSI or ISO, or in IAEA SS 6 edition 1985.
Note that, the initial condition of the specimen for this test is not clearly specified in the edition 1996 of IAEA SS 6.
However it is sure that the specimen need not to be previously submitted to the mechanical tests (9m and 1m) representative for the accident conditions of transport.

3.7. Relaxation of previous requirements (paragraph 632)

The paragraph 632 allows the relaxation of the requirements from
- paragraph 629 : conformity with ISO 7195
- paragraph 630.a : design to withstand the structural test (hydrostatic pressure)
- paragraph 630.c : design to withstand the thermal test, only for the packages designed to contain more than 9000 kg of UF6.

For any of these relaxations, the approval of the competent authority is required and paragraph 805 precises that the approval then is multilateral.

If a national competent authority approves packages not meeting the new requirements, as allowed by paragraph 632, these packages will be used without any restriction inside this country, but to be used in international transport, they should be approved by the competent authorities of all the countries where the transport is due to take place. This approval may then be impossible if it can not be demonstrated that their safety is equivalent to the one of the packages that meet all the new requirements.

3.8. Approval (paragraphs 802 and 805 and 828)

The design of packages containing 0.1 kg or more of UF6 is required to be approved
- from the 1.1.2001 for packages not meeting the requirements 629 or 630.a or c and on a multilateral basis
- from the 1.1.2004 for packages meeting the requirements 629 and 630.a and c on a unilateral basis.

The identification mark assigned to the approval certificate shall include the following types codes :
- H(U) for a unilateral approval
- H(M) for a multilateral approval

3.9. Marking (paragraph 538)

When the design of a package containing UF6 is approved under paragraph 805, *each package (...) shall be legibly and durably marked on the outside of the packaging with :*

(a) The identification mark allocated to that design by the competent authority,

(b) A serial number to uniquely identify each packaging which conforms to that design

3.10. Transitional arrangements (paragraph 815)

815. (...), Industrial packages (...), Type IP-2 (...) that did not require approval of design by the competent authority and which meet the requirements of the 1985 or 1985 (As Amended 1990) Editions of these Regulations may continue to be used subject to the mandatory programme of quality assurance in accordance with the requirements of para.310 and the activity limits and material restrictions of Section IV. Any packaging modified, unless to improve safety, or manufactured after 31 December 2003, shall meet this Edition of the Regulations in full. Packages prepared for transport not later than 31 December, 2003 under the 1985 or 1985 (As Amended 1990) Editions of these Regulations may continue in transport. Packages prepared for transport after this date shall meet this Edition of the Regulations in full.

In this paragraph, it is required to comply with the requirement of filling limitation of paragraph 419 as soon as IAEA SS6 (1996) is applicable.

There is no other restriction for packages prepared for transport not later that 31.12.2003 under the 1985 or 1990 Edition of SS6, except the necessity of a multilateral approval for packages not meeting the requirements of para. 629 or 630.a or 630.c.

After 31.12.2003, modifications and manufacturing of packagings, and preparation of packages for transport, shall meet the 1996 edition of the regulations in full.

In addition the design of packages meeting the requirements of para. 629 and 630.a and 630.c shall be uniterally approved.

It may be noted that if unilateral approval of packages may be issued from 31.12.2003 it will facilitate international transport.

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TYPE C PACKAGES AND LOW DISPERSIBLE MATERIALS

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INTRODUCTION

The introduction of a new package type - the Type C Package and a new material type - low dispersible material (LDM), were two fundamental changes to the 1996 Edition of the IAEA Transport Regulations that traditionally have been written to be independent of the mode of transport used.

TYPE C PACKAGES

The Regulations introduce a new, "Type C", package specification which, when adopted by the International Civil Aviation Authority (ICAO), will apply to the transport of large quantities of radioactive material by air. The introduction of the Type C package places a content limit on Type B(U) and Type B(M) packages travelling by air. The content limits may be expressed as:

i) 3000 A_1 or 100,000 A_2 whichever is the lower for material as special form radioactive material; and
ii) 3000 A_2 for all other forms of radioactive material.

The choice of 3000 A_2 for non special form was linked mainly to an older version of the Regulations which defined a large source as being 3000 A_1 or A_2. This quantity is retained in the Regulations as the threshold quantity for which shipment approval of Type B(M) packages is required. Also a study undertaken in France suggested that at typical impact speeds for aircraft crashes the release fraction might be as high as 3 x 10^{-2} for a Type B package. This release fraction in combination with the assumption that a release of 100 A_2 represents a significant hazard gave a further basis for the 3000 A_2 content limit. For special form radioactive material the same content limit is used, but there is recognition that the properties of the special form material may be impaired in an aircraft accident so a cap of 100,000 A_2 was placed on the content limit.

Another new feature of ST-1 is a specification for "low dispersible radioactive material" which, if met, would allow those materials to be carried in large quantity by air in a Type B(U) or Type B(M) package. Type B(U) and Type B(M) packages will continue to be available for transport, by modes other than air, for quantities of material that are limited by their design and specified in their certificates of design approval.

Test criteria

Type C packages are required to undergo the following cumulative test sequences, after which they must be shown to retain their radioactive contents and shielding properties within defined limits. Like Type B(U) and Type B(M) packages, they must restrict loss of radioactive contents to no more than A_2 in a week and restrict the radiation level at 1 m from the package surface to not more than 10 mSv/h:

Sequence 1 (to be carried out in the order given on the same specimen)

i) The specimen is dropped from 9 metres onto an effectively unyielding target so as to suffer maximum damage (the impact speed is 13.3 metre/second).

ii) The specimen is subjected to dynamic crush by placing it on the same unyielding target, in the worst orientation, and dropping a 500 kg plate onto it from a height of 9 metres.

iii) The specimen is subjected to a puncture/tearing test by dropping it from 3 metres onto a conical probe.

iv) The specimen is subjected to an engulfing 800C thermal test for a period of one hour.

This test sequence recognises that it is possible for a long duration (1 hour) fire to occur following a low speed impact with subsequent crushing and puncture/tearing. As for Type B package testing, the fire specification demands that the average flame temperature must be 800 C for the duration of the fire. This means that peak temperatures (which are often reported in accounts of fires) may be much higher. Also the size of the pool of flammable liquid is specified to ensure that the specimen is fully engulfed, but not so large as to cause oxygen starvation around the specimen. Finally the height at which the specimen is suspended above the surface of the flammable liquid must be maintained to ensure the maximum heat input into the package. Thus the fire specification is very severe and it is difficult to envisage a higher heat input to the package being achieved in a real accident.

Sequence 2 (this may be carried out on a separate specimen from sequence1)

i) The specimen is subjected to an impact at a speed of 90 metre/second onto an essentially unyielding target so as to suffer maximum damage.

A separate specimen may be used for this test because exposure to long duration fires in the aftermath of a high speed impact is highly unlikely. The fuel on board the aircraft will be dispersed in the crash and will not form pools to supply long lasting fires.

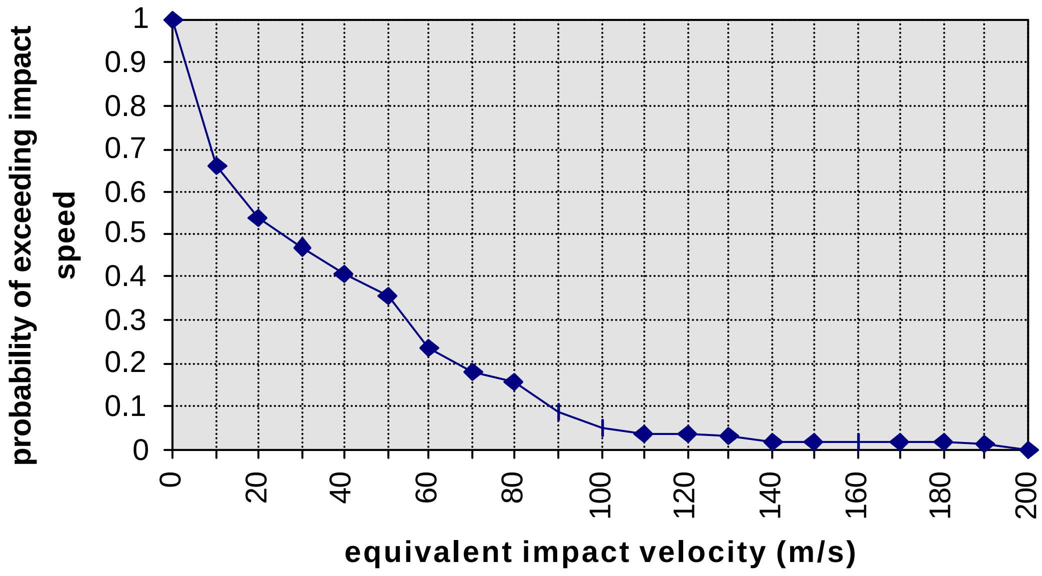
The enhanced impact speed

The impact speed of 90 metres per second was derived from data collected for analysis by the Lawrence Livermore National Laboratory in the USA. Data on jet aircraft crashes were collected for the period 1952-1989. In total over 700 accidents were recorded but of these 220 were deemed to be irrelevant because the incident happened during refurbishment or maintenance or were acts of sabotage. Of the relevant accidents, only 104 were sufficiently well documented to enable a calculation of the equivalent impact speed in terms of a velocity at which a specimen would hit an unyielding target. The equivalent impact velocity takes account of both the angle of impact (è) and the hardness of the target (H_S):

Some representative values of H_S for different impact surfaces are as follow:

1.0	theoretical unyielding target
0.89	runways and concrete surfaces
0.78	soft rock
0.67	farmland
0.54	water or swamp

The figure below shows a plot of the probability vs. equivalent impact speed. The curve is very flat in the range above 100 metres per second implying that there is very little added protection to be gained by increasing the impact velocity of this test. It is acknowledged that Type B(U) and Type B(M) packages are designed to withstand nearly all conceivable accidents associated with surface mode transportation, but not all accidents. Similarly, it was agreed that an equivalent level of protection would be achieved for Type C packages by specifying an impact speed of 90 metres per second.



Additionally, Type C packages must be shown to meet the same containment and shielding criteria as those above when subjected to burial and must be shown not to rupture when immersed in 200 metre deep water. The requirement to demonstrate the ability to withstand burial is done by assessment assuming the package to be undamaged, buried in dry soil at 38 C in a steady state condition. The burial test was introduced because packages involved in high speed crashes may be covered by debris or buried in soil. If packages whose contents generate heat become buried, an increase in package temperature and internal pressure may result.

The tests for Type C packages are considerably more onerous than those applying to Type B(U) and Type B(M) packages and, for a given radioactive content, the packages may be expected to be considerably more robust and heavier.

LDM CONSIDERATIONS

ST-1 also introduces a new specification for "low dispersible radioactive material". This specification recognises that certain radioactive materials, which by their nature are solids of limited dispersibility, limited solubility and emit a low external radiation field (not exceeding 10 mSv/h at 3 metres from the unshielded material), may be safely carried by air in a Type B(U) or Type B(M) package.

The specification for low dispersible radioactive material requires that separate specimens of the material should be subjected to the 90 metre/second impact test and the 800C thermal test for a period of one hour, followed in each case by a leaching test. The pass criteria are that no more than 100 A_2 of material shall be released in either case. It has been shown that for a reference distance of 100 metres, for a large fraction of atmospheric dispersion conditions, a release of 100 A_2 would lead to an effective dose to an individual below 50 mSv.

Low dispersible radioactive material will be subject to multilateral approval and certification by the competent regulatory authority of each country involved in any air transport operation, as also will be the package, Type B(U) or Type B(M), in which it is to be carried.

CONCLUSIONS

The new provisions for Type C packages and LDM will come into effect for international air shipment when the International Civil Aviation Organization implements them in their Technical Instructions. The earliest implementation date is 1 January 2001 and, as no transitional arrangements are foreseen, the new provisions will have immediate effect. This means that package design approvals for Type B(U) and Type B(M) packages will have to be amended to reflect the content limit for transport by air. If the contents can be demonstrated to meet the requirements for LDM both the design of the LDM and the design of the Type B(U) or Type B(M) package will be subject to multilateral approval.

An example of where the impact of the new provisions will be felt is in the air shipment of large cobalt sources. These sources are used for medical therapy, in sterilisation plants and some food irradiation facilities. Sometimes they are needed in remote areas and air transport is the most convenient and secure means of supply. The sources are far larger than the 3000 A_1 content limit for a Type B(U) or Type B(M) package travelling by air, and the high external dose rate precludes the sources from qualifying as LDM. Thus large cobalt sources are candidate materials for Type C packages which have not yet been designed nor shown to be commercially viable.

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

RADIOLOGICAL PROTECTION ASPECTS

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Introduction

The IAEA Basic Safety Standards[1] have been revised to reflect the 1990 recommendations of the ICRP[2]. These recommendations introduce lower dose limits for occupationally exposed workers and for members of the general public. In addition, they recognize that individuals can be exposed to radiation from more than one source and prescribe that dose constraints be applied to each practice involving radioactive material. A new principle is introduced by requiring that account be taken in the planning stage of the effect of mishaps and accidents (potential exposures).

It is obvious that the IAEA Transport Regulations[3] should continue to be in conformity with the Basic Safety Standards.
One of the major topics being considered in the revision process was the incorporation of these new Basic Safety Standards.
Several meetings have been convened by the IAEA to evaluate the impact of the ICRP recommendations and the revised Basic Safety Standards on the transport regulations and to propose amendments to incorporate these fundamental standards into the ongoing revision of the transport regulations.

This paper deals with this evaluation and the resulting changes of the transport regulations: the radiation protection programme for the transport of radioactive material; the dose assessment programmes; segregation distances; surface contamination limits; the revised Q System and derivation of A_1 and A_2 values and the classification of Low Specific Activity (LSA) material ans Surface Contaminated Objects (SCO). The issues of exclusion and exemption will be dealt with in a separate paper.

Radiation Protection Programme

The transport regulations call upon the new BSS as a general provision for radiological protection.

One of the major changes is the emphasis on the requirement that a Radiation Protection Programme shall be established for the transport of radioactive material. The nature and extent of the measures to be employed in the programme shall be related to the magnitude and likelihood of radiation exposures. Programme documents shall be available, on request, for inspection by the relevant competent authority.
The purpose of such a Radiation Protection Programme (RPP) is to:

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

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

a) scope of the programme;
b) roles and responsibilities for the implementation of the programme;
c) dose assessment, as required by para. 305;
d) dose limits, constraints and optimization, as indicated in para. 302;
e) segregation distances, as indicated in paras 306 and 307, and further explained in the revised Advisory and Explanatory Material;
f) emergency response, as indicated in paras 308 and 309;
g) training, as required in para 303; and
h) quality assurance, as required in para 310.

The scope of the Radiation Protection Programme (RPP) should include all the aspects of transport as defined in para. 106 of the Regulations, i. e. design of packages, preparation for transport, loading, stowing, unloading. It is however recognized that in some cases certain aspects of the RPP for transport may be covered in RPPs at the consigning, receiving or storage-in-transit sites.

Since the magnitude and extent of measures to be employed in the RPPs will depend on the magnitude and likelihood of exposures involved, a graded approach should be followed.

Both the package type and the package category need to be considered. For routine transport the external radiation is important and the package category provides a classification for this; under accident conditions however, it is the package type (excepted, industrial, Type A, Type B or Type C) that is important.

Excepted, industrial, and Type A packages are not required to withstand accidents. Those aspects of the RPP related to accident conditions of transport will need to consider the possibility of leakage from these package types as the result of relatively minor transport or handling accidents. In contrast, Type B and Type C packages can be expected to withstand all but very severe accidents.

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

The radiation protection programme will best be established through the cooperative effort of consignors, carriers and consignees engaged in the transport of radioactive material. Consignors and consignees should normally have an appropriate radiation protection programme as part of fixed facility operations.

The role and responsibilities of the different parties and individuals involved in the implementation of the RPP should clearly be identified and described. Overlapping of responsibilities should be avoided.

Depending on the magnitude and likelihood of radiation exposures, the overall responsibility for the establishment and the implementation of the RPP may be attributed to a health physics or safety officer recognized by the competent authorities (called "qualified expert" in the European Basic Safety Standards for Radiation Protection)[4] .

Dose assessment programmes

Dose assessment programmes for occupational exposures arising from transport operations are prescribed on the basis of likely annual doses.
For occupational exposures which are unlikely to exceed 1 mSv/y, no special actions such as special work patterns, detailed monitoring, nor individual record keeping are required. Workplace monitoring is required for exposures expected to be in the range of 1 to 6 mSv/y. Individual monitoring is required for exposures likely to exceed 6 mSv/y.
A similar categorisation is prescribed in the 1986 edition of the Regulations, but based on the "old" dose limit of 50 mSv/year compared to the "new" dose limit of 20 mSv/year.
To assist operators in estimating the exposure of their workforce, advice has been provided that correlates exposure with the number of packages handled and the radiation at 1 m from the packages. It is believed to be unlikely that carriers handling less than a total sum of transport indices of 300 per year will exceed the 1 mSv threshold. Workers engaged by consignors and consignees can be expected to be covered by radiation protection programmes implemented at the fixed site. Carriage, on the other hand, is a transient operation for which the classification of work areas can be difficult to apply.

Segregation distances

The basis for the calculation of segregation distances has not been changed :

- for workers in regularly occupied working areas : 5 mSv/year;
- for members of the public, in areas where the public has regular access : 1 mSv/year to the individuals of the critical group;
- for undeveloped photographic film : 0.1mSv per consignment.

Transport Index

In the current regulations the transport index is a number that is assigned to a package, an overpack, a tank, a freight container or unpackaged LSA-I material ans SCO-I and which is used to provide control over both nuclear criticality safety and radiation safety.
In the 1996 edition two separate package indexes have been introduced. The transport index (TI) for radiation protection is unchanged and continues to be based on the radiation level at 1 m of the package [TI=1 corresponds to a radiation level of 10 mSv/h (1 mrem) at 1 m distance].
A new criticality safety index is based on the allowable number of packages that can be transported together. Separation of the two indexes will allow shipments to be controlled on the basis of the specific values of concern. The changes are believed to introduce clarity which should enhance compliance with the regulations.

Table XI with the TI limits for freight containers and conveyances has been split into two tables, a new Table IX dealing with the transport index (for radiation control) and a new Table X dealing with the criticality safety index. It is believed that this change will make the regulations more user-friendly.

Surface Contamination

In the current regulations, a distinction is made between the limits for non-fixed surface contamination for

excepted packages and those for other packages.
In the revised edition, the reduction by a factor of ten for excepted packages has been eliminated. This decision was based on an assessment of doses from surface contamination, using exposure scenarios and pathways relevant to transport, and realistic data. The assessment led to the conclusion that the current non-fixed surface contamination limits for excepted packages are unnecessarily restrictive and that there is no justification on radiological protection grounds, to keep different surface contamination limits for excepted packages. The relaxed requirement, coupled with the constant need to keep contamination levels as low as reasonably achievable, is believed to ensure an adequate level of safety.
Hence, the rather complex Table III has disappeared and has been replaced by a more simple paragraph 508:

"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 the following limits :

(a) 4 Bq/cm^2 for beta and gamma emitters and low toxicity alpha emitters, and
(b) 0.4 Bq/cm^2 for all other alpha emitters.

These limits are applicable when averaged over any area of 300 cm^2 of any part of the surface.

Derivation of A_1/A_2 values and the Q-system

The Q-system is a dose-based set of models which is used to derive the A_1 and A_2 values in the regulations. These values are activity quantities, calculated for each radionuclide, that set the limits on contents for type A packages (A_1 for special form and A_2 for other forms) and are used for specifying other activity and activity concentration limits (e.g. for LSA-II and LSA-III).
The fundamental assumptions in the Q-system constrain the detriment to an individual in the event of serious damage to a single type A package by restricting the dose to the order of 50 mSv.
The impact of the new BSS is limited in extent following the decision to place the Q-system in the domain of potential exposures. Potential exposures are not expected to be delivered with certainty, and can result from an accident or events of a probabilistic nature. Since potential exposures are not subject to the dose limits applying to practices (20 mSv/year, in general, for occupationally exposed workers), the reference dose of 50 mSv can continue to be used in the context of the Q-system.
However, the derivation of the A_1 and A_2 values has been scientifically reexamined, using the latest radiological data (such as the complete spectral emissions from radionuclides, new radiation weighting factors and tissue weighting factors) and the latest metabolic models.

Under the Q-system a series of exposure routes are considered, each of which might lead to radiation exposure, either external or internal, to persons in the vicinity of a type A package involved in a severe transport accident.

These dosimetric routes (schematically illustrated in figure 1) lead to five contents limit values:
- Q_A for external photon dose;
- Q_B for external beta dose;
- Q_C for inhalation dose;
- Q_D for skin and ingestion dose due to contamination transfer;
- Q_E for submersion dose.

Contents limits for special form alpha emitters and tritium are considered separately. Consideration is also given to the physical form of the contents where physical characteristics may be more restrictive than radiological considerations.

As in the 1985 edition of the Regulations, type A package contents limits are determined for individual radionuclides. The A_1 value for special form material is the lesser of the two values Q_A and Q_B. The A_2 value for non-special form material is the least of A_1 and the remaining Q values.

The derivation of individual Q values is based on the following (unchanged) radiological criteria:

(a) the effective or committed effective dose[(1)] to a person exposed in the vicinity of a transport package following an accident does not exceed a reference dose of 50 mSv;

(b) the dose or committed equivalent dose to an individual organ (including the skin), of a person involved in the accident does not exceed 500 mSv, or in the special case of the lens of the eye 150 mSv;

(c) a person is unlikely to remain at 1 m from the damaged package for more than 30 minutes.

Q_A is determined by consideration of the external dose due to gamma or X rays, assuming complete loss of shielding in the accident. The consequent dose rate at a distance of 1 m from the unshielded radioactive material is limited to 100 mSv/h.
In the earlier Q system, the Q_A value was calculated using the mean photon energy per disintegration taken from ICRP Publication 38[5]. The conversion to effective dose per unit exposure free-in-air was approximated as 6.7 mSv per roentgen for photon energies between 50 keV and 5 MeV.
In the revised Q system, the complete X and gamma emission spectrum for the radionuclides, as given in ICRP 38, was used. Furthermore, the energy dependent relationship between effective dose and exposure free-in-air was taken from ICRP 51[6] for an isotropic radiation geometry.

Q_B is determined by consideration of the beta dose to the skin. The shielding of the package is again assumed to be completely lost, but the concept of a residual shielding factor for beta emitters (associated with package debris for instance) from the 1985 edition has been retained.
In the revised Q system, the complete beta spectrum (instead of the weighted beta energy) for the radionuclides, as given in ICRP 38, was used, together with recent data from Cross[7] on the skin dose rate per unit activity of a monoenergetic electron emitter.
The positron annihilation radiation has been included in the derivation of Q_A.

Q_C is determined by consideration of the inhalation dose due to the activity released from a damaged type A package.
As in the earlier Q system, an intake fraction of 10^{-6} was assumed, being a combination of a range of respirable aerosol release fraction of 10^{-2} to 10^{-3} of the package contents and a range of uptake factors of 10^{-4} to 10^{-3} of the released material. Instead of the Annual Limits on Intake, the effective dose coefficient for inhalation (Sv per Bq)[1] is used. As in the earlier Q system, the most restrictive chemical form has been assumed. For uranium, the Q_C values are now presented in terms of the absorption types assigned for the major chemical forms of uranium. This more detailed evaluation of Q_C was undertaken because of sensitivity of the dose per unit intake to the absorption type and the fact that the chemical form of uranium in transport is generally known. This procedure leads to different values for all uranium isotopes except U-235 and U-238.

Q_D is determined by consideration of the beta dose to the skin and the ingestion dose resulting from handling a damaged package. The model assumes that 1 % of the package contents is spread uniformly over an area of 1 m^2; handling of the debris is assumed to result in contamination of the hands to 10 % of this level, and the person is assumed not to wear gloves and to have his hands washed within a period of 5 hours. This person

1 defined as the summation of the tissue equivalent dose, each multiplied by the appropriate tissue weighting factors, as given in ICRP publication no. 60.

is further assumed to ingest all the contamination from 10 cm^2 of skin over a 24 hour period.
As for Q_B, the complete beta spectra were used together with recent data from Cross[7] on the skin dose rate per unit activity of a monoenergetic electron emitter.

Q_E is determined by consideration of the submersion dose following their release in an accident. A 100 % release of the package into a store room of 3 x 10 x 10 m^3 and with four air changes per hour is assumed. In the earlier Q system the Derived Air Concentrations were used, while in the revised Q system the effective dose coefficients for submersion in a semi-infinite cloud were used, as tabulated in the USEPA Federal Guidance Report No. 12 [8].

As in the earlier Q system, the decay products with half lives less than ten days are assumed to be in equilibrium with the parent nuclide, but the daughter's contribution is now summed with that of the parent. In the past, the Q values were calculated for daughter and parent and the most restrictive was assigned to the parent.
A special case is the calculation of Q_E for Rn-222, where not only the submersion dose is calculated, but also account must be taken for the lung dose due to the daughter-products Po-218, Pb-214, At-218, Bi-214 and Po-214. Therefore, radon and its progeny is treated as a noble gas.

For alpha emitters(2) it is in general not appropriate to calculate Q_A and Q_B values for special form material, because of the relatively weak beta and gamma emission. Although there is no real dosimetric justification, the arbitrary contents limit of 10 000 Q_C has been kept.

The arbitrary cut-off of 40 TBq for A_1 and A_2 to account for possible effects of bremsstrahlung radiation was also kept.

A comparison between the new and old A_1 and A_2 values for a selected number of radionuclides is given in table 1.
The changes in the A_1 and A_2 values are rather limited, except for the alpha-emitters, where higher values are found. This is mainly due to the use of new tissue weighting factors and metabolic models recommended by the ICRP.

Classification of LSA material and SCO

It is also worth mentioning that some problems have been identified with respect to the classification of low specific material and surface contaminated objects and the package requirements for these materials and objects.
There were concerns about the underlying radiological model and about the fact that the requirements for the different categories of industrial packages do no fit within the scheme of withstanding routine conditions of transport (excepted packages), normal conditions of transport, including minor mishaps (type A packages), accidents (type B and type C packages). It was recognised that no credit can be taken of the non-combustible nature of many IP-2 and IP-3 packages and that no account can be taken of the great strength inherent in some IP-2 and IP-3 packages of consolidated wastes.
Time in the review process was limited and work on the rationalisation (radiological basis) of this categorisation could only be started. The in-depth discussion of this issue has been deferred to the next revision.

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2 A radionuclide is defined as an alpha emitter if in greater than 0.1 % of its decays it emits alpha particles or it decays to an alpha emitter.

With the exception of the introduction of a new category of LSA-I material as a consequence of the introduction of radionuclide specific exemption values, no changes were introduced neither in the classification of low specific activity material and surface contaminated objects nor in the package requirements for these materials and objects.

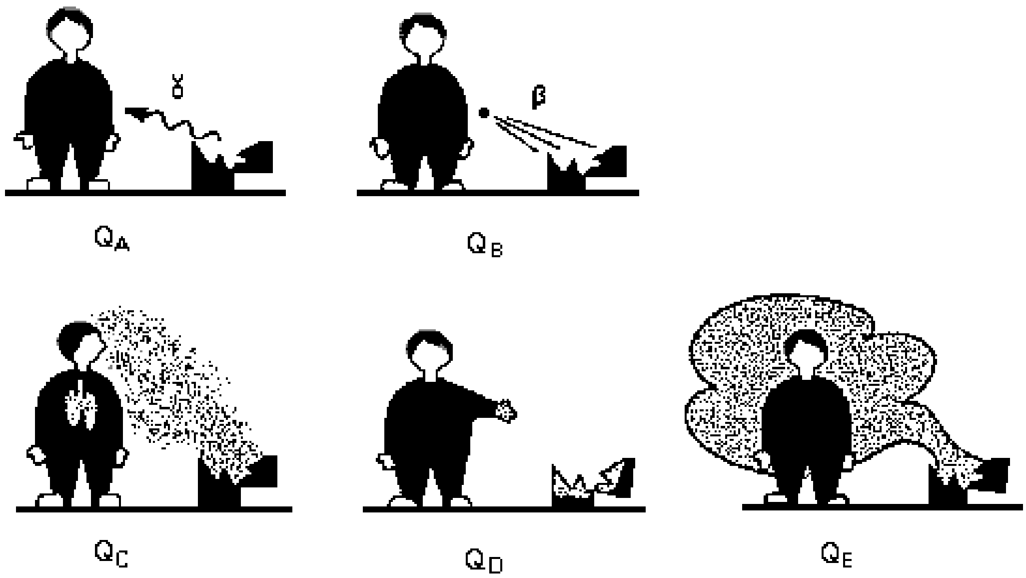

FIG. 1. SCHEMATIC REPRESENTATION OF EXPOSURE PATHWAYS EMPLOYED IN THE Q SYSTEM.

Table 1 - Comparison of old and new A_1 and A_2 values

radionuclide	A_1 (in TBq)		A_2 (in TBq)	
	1985 Edition	1996 Edition	1985 Edition	1996 Edition
H-3	40	40	40	40
C-14	40	40	2	3
Co-60	0.4	0.4	0.4	0.4
Tc-99m	8	10	8	4
I-125	20	20	2	3
Cs-137	2	2	0.5	0.6
Ir-192	1	1	0.5	0.6
Ra-226	0.3	0.2	0.02	0.003
natural Th	unlimited	unlimited	unlimited	unlimited
natural U	unlimited	unlimited	unlimited	unlimited
Pu-239	2	10	0.000 2	0.001
Am-241	2	10	0.000 2	0.001

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

EXCLUSIONS AND EXEMPTIONS IN THE 1996 EDITION OF THE IAEA REGULATIONS FOR THE SAFE TRANSPORT OF RADIOACTIVE MATERIAL

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Introduction

Exclusions and exemptions (total and partial) have always been part of the transport regulations[1], but these provisions were dispersed over various sections, for instance:

- the regulations do not apply to shipments within regulated facilities, neither to radioactive material that is an integral part of the means of transport (such as ballast in aircraft or ships), nor to material having a specific activity greater than 70 Bq/g;
- excepted packages are exempted from certain requirements of the regulations.

Some of the exclusions have been kept in the revised regulations and have been grouped in para. 107:

The Regulations do not apply to:
- *(a) radioactive material that is an integral part of the means of transport;*
- *(b) radioactive material moved within an establishment which is subject to appropriate safety regulations in force in the establishment and where the movement does not involve public roads or railways;*
- *(c) radioactive material implanted or incorporated into a person or live animal for diagnosis or treatment;*
- *(d) radioactive material in consumer products, which have received regulatory approval, following their sale to the end user;*
- *(e) natural material and ores containing naturally occurring radionuclides which are not intended to be processed for use of these radionuclides provided the activty concentration of the material does not exceed ten times the appropriate exemption values.*

The exclusions in subparagraphs (a) and (b) are not new in the transport regulations. The exclusion in subparagraph (c) has been extended to animals.
The exclusion in subparagraph (d) applies for instance to smoke detectors containing relatively small amounts of radioactive materials; such articles are also excluded from the BSS system of notification and authorisation.
The rationale behind subparagraph (e) will be explained later in relation with the departure from the single exemption level of 70 Bq/g to a radionuclide specific approach in the definition of radioactive material.

Principles for exemption

In 1988 an international (IAEA, NEA) consensus was reached on the general principles for exemption from radiological protection measures, and published as IAEA Safety Series No. 89[2].
The exemption of a practice or a source from regulatory control (notification, registration, licensing) must be seen in relation to the basic radiological protection principles: justification of a practice, optimisation of protection, individual risk and dose limits.
A "practice" is defined as "a set of co-ordinated and continuing activities involving radiation exposure which are aimed at a given purpose, or the combination of similar such sets".
The "source" is then defined as "the physical entity (e.g. radioactive material, nuclear installation) whose use, manipulation, operation, decommissioning and/or disposal are constituents of the coordinated set of activities defined as practice".

From a radiological protection standpoint, there are two basic criteria for determining whether or not a practice can be a candidate for an exemption :

- the individual risks must be trivial, i.e. sufficiently low as not to warrant regulatory concern;
- the radiological protection must be optimised, taking the cost of regulatory control into account.

A schematic view of the exemption procedure is given in figure 1.

An individual dose is likely to be regarded as trivial if it is of the order of some tens of microsieverts per year. Because an individual may be exposed to radiation from several exempt practices, it is reasonable to apportion a fraction to each exempt practice. This could lead to individual doses to the critical group of the order of 10 mSv in a year from a single practice.

In the optimisation assessment, the relevant quantity is the collective dose commitment per year of practice. A generic study of the available options should be made and the conclusion reached that exemption is the option that optimizes protection. If this generic study indicates that the collective dose commitment from one year of the unregulated practice will be less than about 1 man.Sv, it may be concluded that the total detriment is low enough to permit exemption without more detailed examination of other options.

In its 1990 recommendations, the International Commission on Radiological Protection (ICRP) recognizes "that the exemption of sources is an important component of the regulatory functions"[3]. The Commission reiterates the two basic criteria for exempting a source or an environmental situation from regulatory control. One is that the source gives rise to small individual doses and small collective doses in both normal and accident conditions. The other is that no reasonable control procedures can achieve significant reductions in individual and collective doses.

These principles have been endorsed and made more explicit in the revised IAEA Basic Safety Standards[4] as follows :

"(a) the radiation risks to individuals caused by the exempted practice or source be sufficiently low as to be of no regulatory concern;

(b) the collective radiological impact of the exempted practice or source be sufficiently low as not to warrant regulatory control under the prevailing circumstances; and

(c) the exempted practices and sources be inherently safe, with no appreciable likelihood of scenarios that could lead to a failure to meet the criteria in (a) and (b)."

Application of Exemption Principles

Unregulated practices give rise to small individual doses (10 or some tens of microSv per year), which are not measurable in practice. Therefore, the exemption criteria in terms of dose must be converted to more practical and measurable quantities, such as radioactivity concentration levels in Bq/g or total activity in Bq.
During the revision of the Basic Safety Standards an international consensus was reached on the various scenarios, models and exposure pathways to be considered for the derivation of exemption levels in terms of total activity and activity concentration[5]. The radiological criteria mentioned above were used for normal conditions; in addition to those, for accident scenarios a reference level of 1 mSv was used, on the basis that the probability of occurrence is less than unity, and a reference dose to the skin of 50 mSv was taken into account.
Such exemption levels were derived for the most common radionuclides. The exemption values for some selected radionuclides are given in table 1.
They range several orders of magnitude[4][5] : from 1 to 1 000 000 Bq/g in terms of activity concentration, from 1 000 to 1 000 000 000 Bq in terms of total activity.

The benefit of harmonization between the BSS and the transport regulations is obvious. Such harmonization would avoid problems at interfaces and legal and procedural complications. It was recognized that the single exemption level of 70 Bq/g has no dose basis and that it is unlikely that this level satisfies the general dose criterion of 10 mSv/y for all radionuclides.
A careful examination of the underlying scenarios and models used to derive the exemption levels for the BSS led to the conclusion that they had not been demonstrated to be appropriate for transport purposes. Therefore, a set of transport-specific scenarios were developed which reflected various exposure situations (exposure times, distances, source geometries and shielding factors):

- a postman or courier delivers a package containing radioactive material to a laboratory or a hospital after having carried it during his delivery round (200 hrs/year);
- a driver transports bulk material or packages in a truck or van (400 hrs/year);
- a person loads bulk material or packages in a truck or van (200 hrs/year);
- a member of the public travelling in an aircraft is exposed to radioactive material being transported in the hold of the aircraft (200 hrs/year).

The accident scenarios were analysed following the Q system exposure pathways (only for scenarios leading to exemtion levels in terms of total activity).

Based on these scenarios both activity concentration and total activity (per consignment) values were calculated which would result in an annual dose of 10 mSv (or 1 mSv in the case of an accident). One of the most restrictive scenarios is the exposure of a truck driver transporting 20 m^3 of bulk material for a total duration of 400 hours per year.

It was shown that the single exemption value of 70 Bq/g was not compatible with the dose criteria. For some radionuclides (e.g. Co-60, Ra-226, Th-232, U-238) it results in doses of the order of 1 mSv/year or more (see figure 2).

These transport derived values were generally more restrictive than the BSS values, but generally did not differ more than one order of magnitude (see figure 3). Taking into account the obvious advantages of having the same set of values to be applied to both fixed installations and transport operations, it was decided to adopt the BSS derived values also for transport purposes[6].

For mixtures of radionuclides, the "ratio rule" must be applied so that the sum of the activities (or activity concentrations for each radionuclide divided by the applicable exemption value is less than or equal to 1.

It must be noted that, in the case of decay chains, the values explicitly refer to the parent nuclide. The radiological impact of the daughter nuclides has been taken into account in the calculations. The single value of 70 Bq/g in the current regulations is not unambiguous in this respect and leads to divergent interpretations.

The scope of the Regulations includes those natural materials or ores which form part of the nuclear fuel cycle or which will be processed in order to use their radioactive properties. The Regulations do not apply to other ores which may contain naturally occurring radionuclides, but whose usefulness does not lie in the fissile, fertile or radioactive properties of those nuclides, as long as the concentration of the radionuclides has not been artificially enhanced. Were this not the case the Regulations would have to be applied to enormous quantities of material that present a very low hazard.
However, there are ores where the activity concentration is much higher than the exemption values. Since the regular transport of these ores may be of radiological concern, a need was felt to put a limit for the activity concentration, above which radiological protection measures need to be considered. A factor of 10 above the exemption values was chosen as a compromise between the radiological protection concerns and the practical inconvenience of regulating large quantities of material with low level activity concentration.

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

Table 1 - Exemption values for some selected radionuclides

Radionuclide	**Activity Concentration (Bq/g)**	**Total Activity (Bq)**
H-3	1 000 000	1 000 000 000
C-14	10 000	10 000 000
P-32	1 000	100 000
S-35	100 000	100 000 000
Cl-36	10 000	1 000 000
K-40	100	1 000 000
Co-60	10	100 000
Kr-85	100 000	10 000
Sr-89	1 000	1 000 000
Sr-90(*)	100	10 000
Mo-99	100	1 000 000
Tc-99m	100	10 000 000
I-125	1 000	1 000 000
I-131	100	1 000 000
Cs-137(*)	10	10 000
Ir-192	10	10 000
Au-198	100	1 000 000
Tl-201	100	1 000 000
Ra-226(*)	10	10 000
Th-nat(*)	1	1 000
U-nat(*)	1	1 000
Pu-239	1	10 000
Am-241	1	10 000

(*) parent nuclides and their progeny included in secular equilibrium :

Sr-90	Y-90
Cs-137	Ba-137m
Ra-226	Rn-222, Po-218, Pb-214, Bi-214, Po-214, Pb-210, Bi-210, Po-210
Th-nat	Ra-228, Ac-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212,Tl-208 (0.36), Po-212 (0.64)
U-nat	Th-234, Pa-234m, U-234, Th-230, Ra-226, Rn-222, Po-218, Pb-214, Bi-214, Po-214, Pb-210, Bi-210, Po-210

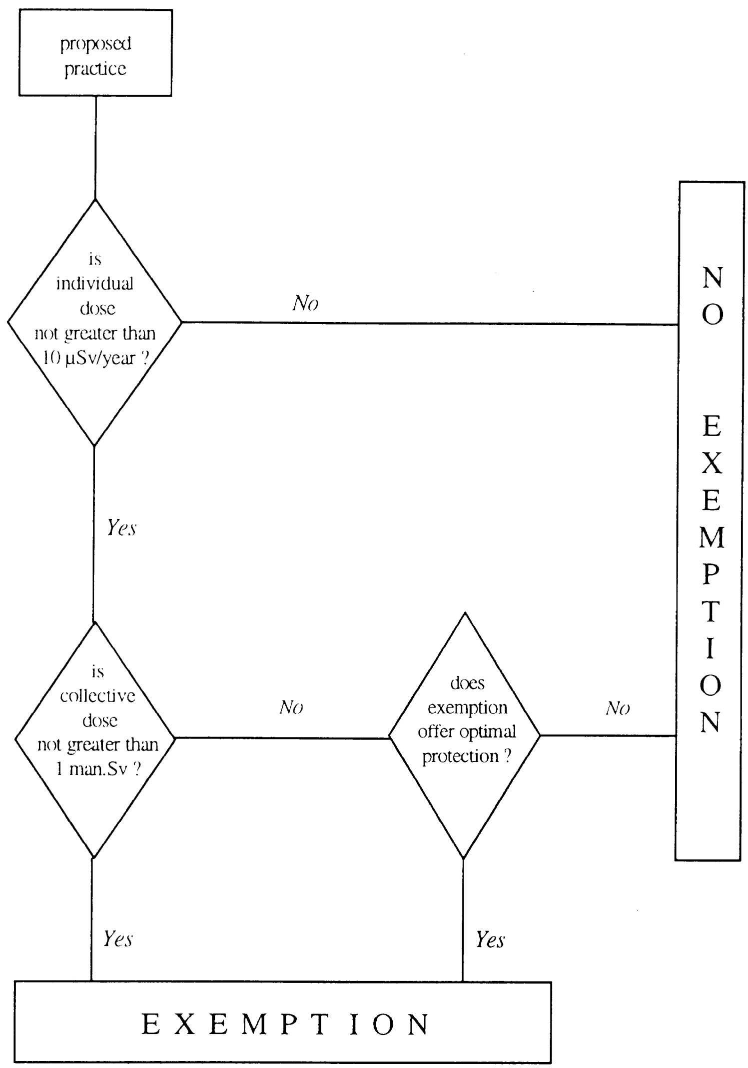


Figure 1 - Schematic view of exemption procedure

Figure 2 - Radiological Impact (µSv/year)
of material with activity concentration of 70 Bq/g

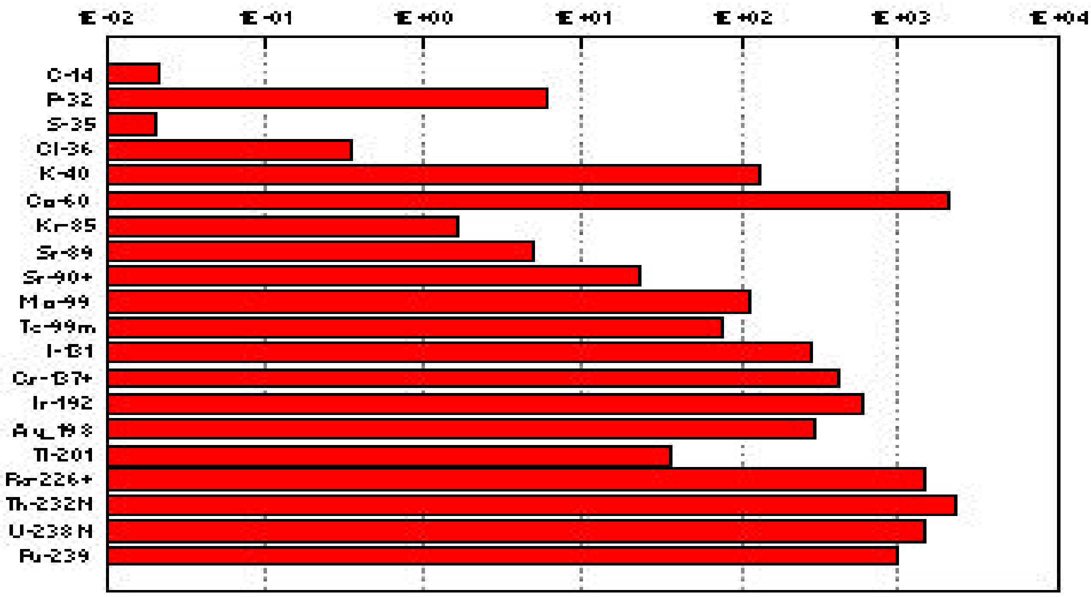
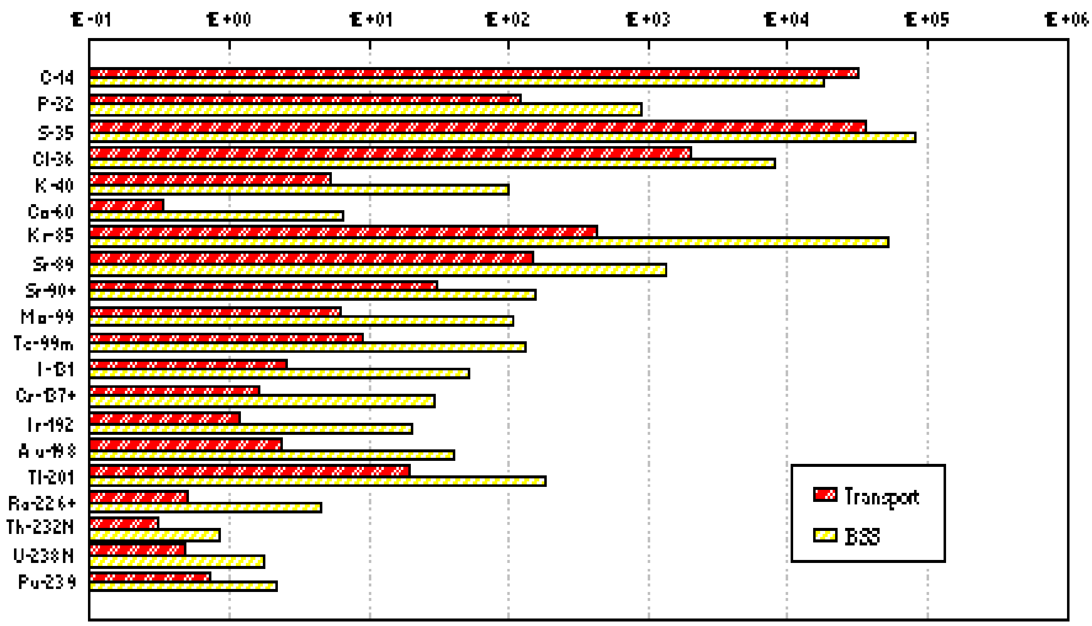


Figure 3 - Exemption values in terms of activity concentration (Bq/g)



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TRANSPORT OF PLUTONIUM OXIDE AND OF MOX FUEL

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1. Introduction

The safety of radioactive material transportation relies upon three fundamental bases : strict regulations, suitable equipment and competent operators. A common approach for all the transports of radioactive material and particularly in the nuclear fuel cycle is their compliance with strict regulations and performance under Quality Assurance.
This statement can be taken up by every sector of the nuclear fuel cycle. But it is particularly true for the transportation of such sensitive materials as plutonium dioxide and mixed oxide fuel.

Such transports are regularly performed in Western Europe by Transnubel and its partners. Some of their main features will be described in this paper.

2. Background

Belgonucleaire has been active in the field of plutonium for more than 30 years. It is normal then that its daughter company Transnubel has gained a unique experience in the transportation of plutonium bearing material.

This experience has been instrumental in dealing with countries as diversified in their regulations as Belgium, France, Switzerland, Germany, United Kingdom.

Transnubel works on a regular basis with partners having a similar experience, such as Transnucléaire in France, NCS in Germany and BNFL in the United Kingdom.

For obvious safety and security reasons, the transport of plutonium is a highly specialized activity. It made its first steps with the initial development of the nuclear energy, the operation of the European pilot reprocessing plant Eurochemic in Dessel and reached its industrial status with the development of plutonium separation in the modern reprocessing plants.

3. Vehicles for the transport of plutonium bearing material.

The transport of plutonium bearing material is performed using specially designed dedicated vehicles. A set of stringent physical protection regulations have been developed in the European countries where such transports take place to guarantee that appropriate security measures are always applied throughout the whole operation, in order to prevent diversion of fissile material.

For continental Europe, several security vehicles are available : 5 in France, 1 in Belgium and 1 in Germany. The design of those vehicles are not identical but they all achieve a common objective in that they provide the security features required for the protection of the transported material.

4. Transport of plutonium dioxide.

Due to the particular characteristics of the plutonium arising from oxide fuel reprocessing it was necessary to adopt specific equipment and techniques in order to :

- provide adequate gamma and neutron shielding as well as heat dissipation,
- allow automatic handling to reduce personnel exposure,
- take into account the physical protection requirements applicable to the transport of such material.

After extensive tests and a long development work, a system meeting all the above requirements has been put into service by Cogema at the end of 1983 to be used as well within France as between France and Belgium and has been operating to the general satisfaction since then.

It includes for each unit the following features :

- a special ISO container licensed by the security authorities of several countries such as France and Belgium,
- an internal rack capable of accomodating 10 FS 47 packagings,
- a series of ten FS 47 packagings.

Packagings

FS 47 packagings

The transport of plutonium dioxide has reached an industrial stage with the introduction of the FS 47 packaging. Its handling at the reprocessing plant and at the fuel manufacturing plants among which Belgonucleaire/Dessel, did not face any difficulty and this system has become standard.

The plutonium oxide powder separated at La Hague is conditioned in stainless steel cans with a swaged lid, each containing approximately 3 kg of material.

These cans are stacked in a welded stainless steel cartridge box which is in turn placed in a cylindrical container closed by a leaktight plug.

For transportation, this set of protective containers is placed in the FS 47 packaging, which is approximately 2 m high and 75 cm in diameter. The wheight of the FS 47 packaging is in the range of 1.5 t and its capacity is around 19 kg of plutonium oxide (see table 1).

Ten FS 47 packagings are grouped in a rack which is then loaded in a specially designed ISO type container providing additional physical protection.

The FS 47 design has been licensed in France in 1983 as a type B / F packaging and has been further licensed or validated in other European countries among them Belgium, as well as in Japan.

The safety margin offered by this packaging is largely in excess of the IAEA regulatory requirements. It has been demonstrated, for instance, that it can withstand extreme external pressures up to 1000 bar and that the seals of the containment envelope would not be affected by a 1000°C fire for a period of one and a half hour.

Besides road transportation, which is performed on a regular basis, the FS 47 system has also been used for the return shipment by sea of plutonium to Japan.

TNB 145 packaging

For smaller quantities of plutonium, TNB 145 packagings can be used too. This design was licensed in Belgium in the end of the 1970's and is validated in several countries. It is used for the transportation of various types of materials including fissile material such as uranium and plutonium oxide.

Various sizes of such drum-like packagings are available, depending on the size and quantities of the material to be transported. The maximum allowed quantity of plutonium oxide is 4.5 kg.

5. Transport of fresh MOX fuel.

The transport of fresh MOX fuel assemblies is performed using packagings specially designed for such purpose.

At present two designs are available :
- TNB 176 (known in France as FS 69)
- Siemens/Biblis

The TNB 176/FS 69 packagings are used for the transport of fuel assemblies in Belgium, France, Switzerland. The Siemens/Biblis packagings are used for the transport of MOX fuel assemblies to German power stations.

The handling of MOX fuel containers, their unloading at power stations and the handling of MOX fuel assemblies within the power stations have been performed in a wide variety of circumstances. Several power stations in Belgium, France, Germany, Switzerland have been licensed for the use of MOX fuel. For instance in Belgium, Tihange 2 and Doel 3, in Switzerland Beznau and Gösgen. In France 16 power stations are licensed. In Germany, the power stations of Brokdorf, Phillipsburg, Unterweser, among others are licensed. Transport of MOX fuel to several of those are performed on a regular basis. Tihange and Doel are among them.

That kind of transport between fuel fabrication plants and power stations is now well established. Larger quantities will be involved in the coming years with the gradual increase of the Melox plant production.

Because of the presence of plutonium in the fuel, two issues had to be solved :
- technical difficulties inherent to the presence of plutonium (need of an appropriate neutron shielding and of an adequate thermal dissipation as some plutonium isotopes are heat producers) with the subsequent need to design type B/F packagings,
- the special security measures which are far more restrictive for plutonium fuel than for enriched uranium fuel.

In the same manner as for plutonium oxide, stringent security rules apply to the transport of MOX fuel assemblies. Depending on the type of security vehicle used for the transport, up to 8 MOX fuel assemblies can be transported at a time. Detailed procedures have been set up and approved by the Competent Authorities of the various European countries concerned.

Packagings

TNB 176/FS 69 packaging (see table 1)

The TNB 176/FS 69 packaging used for the transport of MOX fuel assemblies shares some features of the well known RCC packaging used for the transportation of uranium oxide fuel assemblies. It has a capacity of two assemblies which are loaded and tied down in the vertical position before being tilted horizontally as it is the case for the RCC's. But the TNB176/FS69 design differs substantially from the RCC design since it is equipped with a neutron shielding and since it allows the dissipation of 1.2 kW, the thermal power resulting from the presence of plutonium. Additionally, it is designed to take into account the requirements of B / F packagings.

Siemens/Biblis packaging (see table 1)

As for the TNB 176/FS 69 packaging, the Siemens/Biblis is based on the RCC design with additional features such as neutron shielding and shock absorbing. Actually it consists of a Type A packaging similar to the RCC design, placed into an overpack providing the neutron shielding and the shock absorbing characteristics.

6. Quality Assurance

In the introduction, it has been pointed out that transport activities should be performed under Quality Assurance.
IAEA Regulations becoming operative in 1990 made Quality Assurance compulsory for those activities. The minimum criteria applicable to Quality Assurance systems are defined in the appendix IV of the IAEA Safety Series N°37. It covers the design, the fabrication, the use and the maintenance of transport equipment such as packagings and vehicles, and the performance of the transports themselves. Specialized transport organizations such as Transnubel had not waited for such requirements to become compulsory but had established already long before Quality Assurance systems covering all aspects of their activities, not only in the field of transport itself but also for engineering studies and manufacturing of equipment related to transportation.

These systems meet the requirements of the main Quality Assurance standards such as IAEA 50 CQA and ISO 9000. More and more of the important transport companies are now qualified according to the ISO 9000 Standard.

Over the last years, the widespread application of Quality Assurance principles throughout all steps of nuclear material transportation has further improved its already high standard of safety and reliability.

Table 1

PACKAGINGS FOR PuO_2
AND MOX FUEL ASSEMBLIES

	Packaging	
	FS 47	**TNB 176 /FS 69 Siemens/Biblis**
Content	**PUO_2**	**MOX fuel**
Capacity	**19 kg**	**2 assemblies**
Typical Activity (PBq)	**7.5**	**11**
Heat dissipation (max) (kW)	**0.3**	**1.2**
Weight (t)	**1.5**	**5 / 6.6**

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THE TRANSPORT OF DEPLETED URANIUM BETWEEN AMSTERDAM, GENEVA AND HAMBURG

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THE TRANSPORT OF VITRIFIED RESIDUES

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Presentation

I. Background: reprocessing Belgian spent fuel
II. Vitrified residues (including QA / QC)
III. Transportation (including QA / QC)
IV. Experience
V. Conclusion

I. Background: reprocessing Belgian spent fuel

- Reprocessing contracts with COGEMA

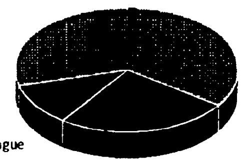
- Return of vitrified residues from La Hague to Dessel starting in 1997

- Interim storage in a specially erected building on Belgoprocess site in Dessel (building 136)

Spent fuel storage status
January 1996

- 1150 tonnes stored at reactor sites

- 450 tormes already reprocessed at La Hague



Current Reprocessing Contracts

Reprocessing period	Contract quantity	Already reprocessed (1st Jan 96)
1980 - 1985	140 tonnes	140 tonnes
1990- 2000	530 tonnes	310 tonnes
2001- 2010	225 tonnes	
2001- 2015	options up to 120 t/y	

Current governmental policy

- Use as MOX the plutonium from 1978 reprocessing contract
- Keep new post-2000 reproc. contract on hold
- Re-examine back-end routes in preparation for a new debate in Parliament around 1998
- In the meantime: assess the technical & economic feasibility of the once-through option

II. Vitrified residues
(including QA / QC)

- Vitrification experience:

facility	glass canisters produced 31.1.96
AVM Marcoule	2 416
R7 La Hague	2 480
T7 La Hague	1320
WVP Sellafield	1046
Pamela Dessel	2 200

Glass canister
main characteristics (1)

height (with lid)	1.34 m
external diameter	43 cm
thickness	5 mm
vitrified residue per canister	~ 150 liter
glass weight per canister	~ 412 kg
material	stainless steel

The vitrified residues contained in one canister correspond to the reprocessing of approx. 1.5 tonnes of spent fuel.

Glass canister main characteristics (2)

percent waste in residue	~ 14%
main radionuclides	90-Sr, 106-Ru, 137-Cs, 144-Ce, 154-Eu, 241-Am, 244-Cm
activity content α + β,γ	~ 1 E15 Bq
maximum decay heat	< 1.46 kW
maximum quantityof Pu	< 0.11 kgPu
maximum quantity of U	< 4.50 kg U
maximum surface contamination β	< 3.7 Bq/ sq. cm

Vitrified residues specifications

- Vitrified residues specifications approved by safety authorities in:
  - France
  - Japan
  - Gerrnany
  - Belgium
  - Switzerland
  - the Netherlands

QA / QC procedures for vitrified residues

- Complete QA / QC procedures
- 3 control levels
  - COGEMA (internal)
  - Bureau Veritas (on behalf of baseload customers)
  - ANDRA (on behalf of French safety authorities)
- In addition
  - regular inspections of COGEMA and ANDRA by French safety authorities

Ensuring quality and safety
through stringent QA / QC

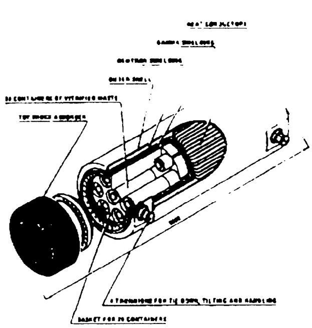
- At the time of glass production
  - quality of glass components
  - process control
  - internal inspection by a COGEMA body independent from operations
- Before shipment
  - final control of each canister
  - customers' representatives witness the operations

III. Transportation
(including QA / QC procedures)

- For Synatom - COGEMA contracts prior to 2000: only 15 transports of vitrified residues from La Hague to Dessel
- Equivalent to 6 years of nuclear electricity production in Belgium

✓ Nuclear waste volumes are very small

TN **28 VT transport flask**



- Compliance with IAEA regulations type B(U)F
- Conformity with transport regulations:
  - international: IAEA Safety Series n°6, RID, etc
  - European: ADR
  - French: RTMDR, RTMDF
  - Belgian: Royal Decree of 28.2.1963

TN 28 VT flask
main characteristics (I)

Transport flask TN 28 VT based on spent fuel transportation casks

height (with lid)	6.6 m
external diameter	2.4 m
thickness	26 cm
glass canisters per flask	28
total weight with load	112 tonnes
material	mainly carbon steel, resins

TN 28 VT flask
main characteristics (2)

maximum heat production	< 41 kW per flask
maximum surface temperature	108°C
activity content α + β,γ	~ 1 E16 Bq
maximum allowed dose γ+n at surface	< 2 mSv / h(*)
maximum allowed dose γ+n at 1 m	< 0.5 mSv / h(*)
maximum non fixed contamination α	< 0.4 Bq/sq. cm
maximum non fixed contamination β,γ	< 4 Bq/sq. cm

(*) in practice, much lower levels are observed, by orders of magnitude

QA / QC procedures for transportation

- specific QA / QC for transportation, involving:
  - COGEMA
    - service pour la protection radiologique
    - Service des transports
    - Service de la gestion des contrats
    - Service désentreposage de résidus vitrifiés La Hague
  - Belgoprocess
  - Transnucléaire
  - Transnubel
  - ONDRAF/NIRAS
  - SNCF
  - SNCB
  - Transport Lemaréchal
  - Société de Manutention et de Levage
  - Controlatom

Quality plan for transportation

- 74 steps, fully documented, covered by:
  - contracts
  - specifications
  - procedures
  - agreement certificates

- Examples:
  - granting of French & Belgian authorisations
  - export & import licenses
  - numerous radiological controls (at La Hague, Valognes, Mol, Dessel)
  - attendance of official controllers
  - etc.

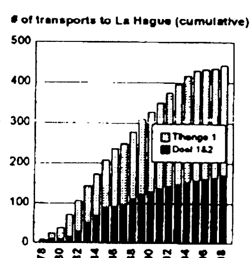
Transport documentation file

- 51 pages in total. Examples:
  - consignor's certificate
  - glass canisters loading plan
  - flask turnround inspection & maintenance check-list
  - flask radiological survey
  - road vehicle radiological survey
  - wagon radiological survey
  - certificate of transfer of responsibilities
  - etc.

IV. Experience

- On-going production of glass canisters at La Hague
- Vitrified residues safely transported to Japan & Germany
- Experience with more than 4500 spent fuel shipments from all COGEMA customers to La Hague
- Two test transports to Dessel in 1995 & 1996

Spent fuel transportation pre-2000 contracts



- approx. 440 transports of spent fuel from Belgium to La Hague effected

- approx.15 return transports of vitrified residues to be effected

Return to Belgium
technical aspects

- 28 canisters per transport in a flask TN 28 VT
- Road / rail / road
- Flask very similar to those used
  for spent fuel transportation (TN 12)

V. Conclusion

- product quality: OK
- transportation equipment: OK
- transportation operations: OK
- tests: OK
- storage building: OK

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THE TRANSPORT OF RADIOACTIVE MATERIAL IN EVERYDAY LIFE PRACTICAL ASPECTS

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TABLE DES MATIERES

INTRODUCTION

At the end of a cycle of courses or lectures on the various aspects concerning the transport of radioactive material, we are all quite convinced that this sector of the nuclear industry, as others, is among the safest there is and one can be pleased about this state of affairs. In the event of an accident, a range of measures have been developed to prevent - in most cases - and always to minimize, irradiation of all the people involved and the dispersion of radioactive matter.

The basic principles underlying radioprotection of the population and of workers appear to be perfectly applied in practice but we all know that perfection does not exist and that the regulations are highly complex. One can, therefore, quite rightly ask the following questions : How is it applied ? Are all the recommendations and statutory provisions applied with the same degree of perfection ? What does this cost ? Are some recommendations too cumbersome to be applied in detail ? Is there a temptation, in some poorer sectors, to circumvent the laws ? Or possibly to misunderstand or misinterpret them. Do all the countries apply and monitor the regulations with the same attention to detail and the same degree of professionalism ?

Finally, is there a wealthy sector (the nuclear fuel cycle in Western countries) where there are the means and the political will to develop a safety policy and is there a not so wealthy sector where the situation is not so easy, and where safety has a price and where those responsible for safety have to make efforts to be heard ?
To make this paper as clear and as lively as possible, we are going to illustrate actual conditions thanks to the use of examples taken from daily experience. We shall, on the basis of our own experience, offer you the results of our thinking, which we hope will see the beginning of a lively and useful debate.

Before embarking on these examples, we feel that it is vital, in order to avoid any misunderstanding, to adopt the following precautions :

- What will be said is not to be taken as criticism of anybody whomsoever but is intended to be a series of observations of the real situation as we see it. The aim of our approach is to strive for better understanding between all those responsible for the safety of transport. The greater the understanding, the easier collaboration will be and the greater the level of safety. In this respect, I would like to emphasize the excellent relations which exist in Belgium between the “Controllers” and the “Controlled”. We benefit, each and every day, from an excellent quality of dialogue. I do not know whether this is the case everywhere.

- There can, of course, be <u>no question</u> on the basis of these examples of <u>justifying shortcomings</u> in the field of safety or of trying to <u>endorse such failings</u> but only of emphasizing the difficulties or of making constructive remarks with the aim of searching for a remedy or of finding solutions in those areas where we detect zones of particularly complex regulations, thus not so clear-cut and more likely to give rise to misinterpretations or deviation.

- In the nuclear industry, and looking beyond techniques and regulations, it is the adoption of a "safety culture" which gives a meaning to all the various activities.

Real professionalism involves technology, regulations and a safety culture.

2. EXCEPTED PACKAGES AND THE RISK OF LOSS OF RADIOACTIVE SOURCES

<u>Definition of an "excepted package</u>" :

An excepted package may contain radioactive material alone or radioactive material which is incorporated in various equipment, within the limits mentioned in the table below :

	Radioactive matter	**The equipment**
Solids in a special form	10^{-3} A_1	A_1
Solids in another form	10^{-3} A_2	A_2
Liquids	10^{-4} A_2	10^{-1} A_2
Gas in a special form	10^{-3} A_1	10^{-2} A_1
Gas in another form	10^{-3} A_2	10^{-2} A_2

The above represents fairly significant levels of activity and some radioactive sources, used in gauges (incorporated in equipment) are transported under this designation without encountering any problems with the dose rate on contact.

<u>Example :</u>

1. A level gauge (equipment) containing a source of ^{241} Am (A_1 = 2 TBq - 50 Ci incorporated in an equipment) in a special form with an activity of 4 GBq (0.1 Ci)
2. A chromatograph containing a source of ^{63} Ni (A_2 = 30 TBq - 800 Ci), with an activity of 500 MBq or a ^{226} Ra source (A_2 = 210^{-2} TBq - 0.5 Ci) (not in a special form).

Since 1963, Belgian regulations require a transport approval for much more restrictive activities.

It is only since 1993 that Euratom regulations call for a transfer statement between Member States for activities comparable to the levels requiring Belgium transport approval.

Example : 5 KBq for ^{241} Am
500 KBq for ^{63} Ni

Before the Euratom directive, a great many sources where shipped "in cognito" and/or arrived in facilities which were not yet authorized.

Nevertheless, it is sometimes the case today that sources from non-European countries are either shipped by unauthorized transport companies when they were ordered by operators not yet fully aware of the required authorizations.

This situation may become worrying because if one is not aware of the existence of a source, how can one ensure follow-up and decommissioning at the end of its life ? This may lead to potential “Goianas” or to problems similar to those mentioned in Chapter 3.

We would also like to point out that with regard to transport by post, parcel limits are ten times more restrictive but continue to remain fairly high. Within the context of internal transport inside a country, the package does not have to carry a special mention, and thus no documents are necessary. In the case of international transport, the consignment has to show the name and address of the shipper, together with a label stating “radioactive matter” but there is no specification as to where the label is to be affixed or to the size of the label.

In Belgium, transport by post is prohibited, which is in our view an excellent thing. It has to be noted that there have been occasions of internal transport (without any mention on the parcel), as well as international transport by a foreign shipper and a label which was not particularly “legible” !

To conclude :

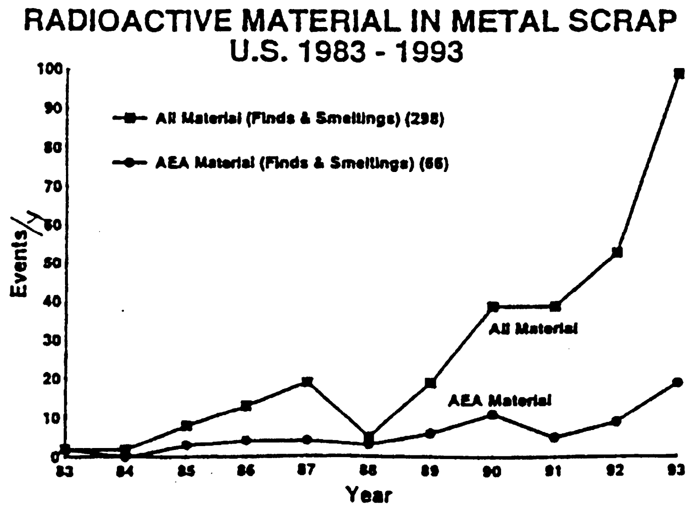
In our view, the Euratom limits and Belgian regulations ought to be applied internationally. This would prevent transfers and shipment of fairly high and uncontrolled levels of activities and transport by post would be totally excluded.

3. CONTAMINATION OF METAL SCRAP

3.1 Position of the problem

Since the early 80s, the number of radioactive sources which have been detected in metal scrap has increased quite considerably, and has now become worrisome.

The following diagram clearly illustrates the problem.



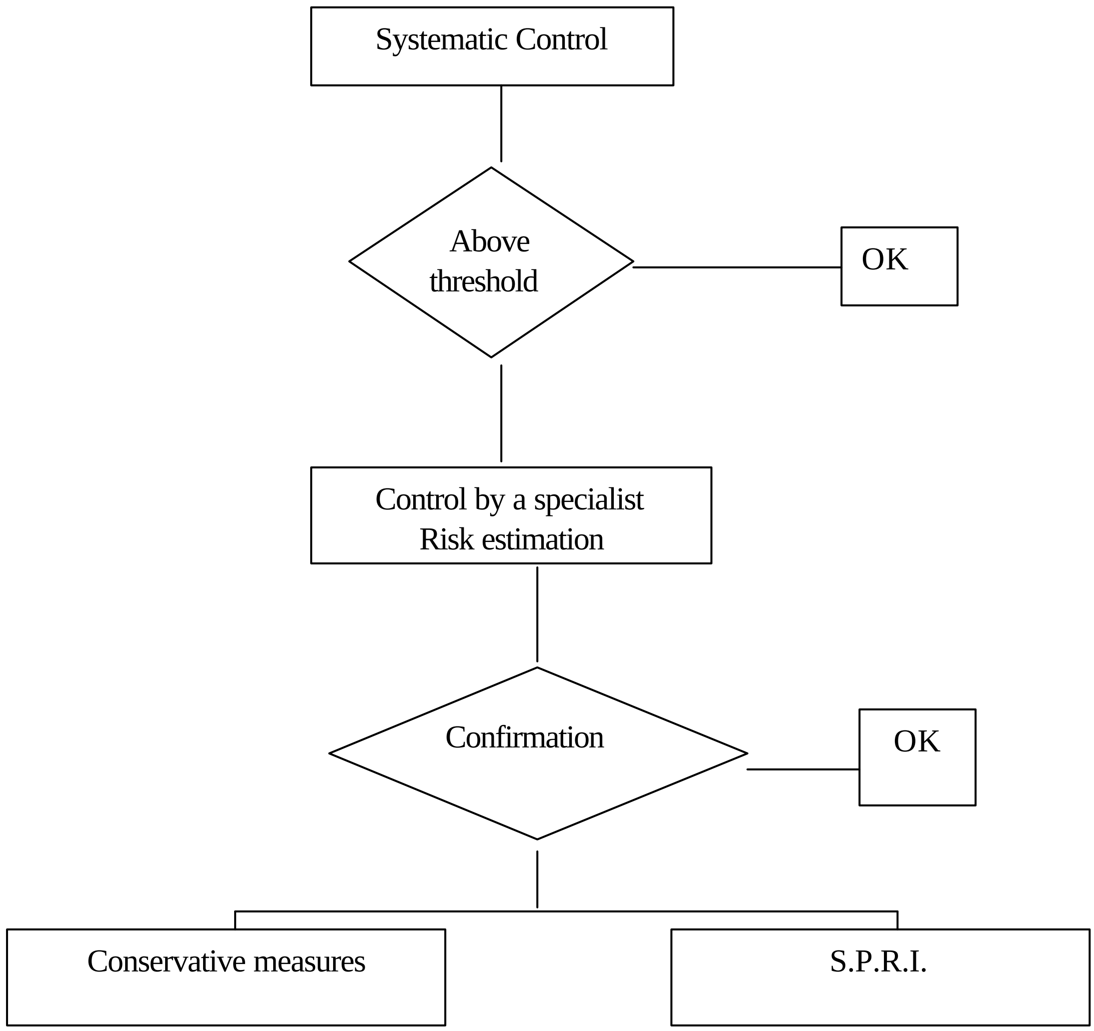
The steel industry has stressed that it will have to face increases which are exponential to the number of incidences linked to metal scrap contamination. The industry also points out that some sources, still in their lead shielding, are able to avoid detection by most of detection systems, even the most sophisticated ones.

Accidental melt-down of one of them can have dramatic financial consequences for the industry and radiological consequences, which are at least as great for the workers and/or the environment.

Quite frequently, the real origins of the metal scrap is not clearly known, if not completely unknown. It has become increasingly evident that in most cases the problems are due to sources from former East-bloc countries where safety rules (inspections) are not as effective as ours.

Indeed, radioactive material is shipped by boat, train, truck .... in a totally illegal manner.

DECISION-MAKING DIAGRAM



Moreover, once they have been detected, a number of problems linked to the transport of radioactive material arise, with most of them being quite specific. It is in this respect that they are interesting to examine because the problem is both national and international.

It is most important to know how to react ! And it is very tempting for the person who detects the "contaminated" vehicle to get rid of the problem by discretely sending it back to the shipper, who in turn will get rid of the consignment. The safety culture adopted in the scrap metal industry is probably far from that adopted in the nuclear sector and we regularly encounter situations which can be considered as potential "Goianas".

3.2 The problem from the point of view of transport of radioactive material.

IF : A is the owner of the metal scrap (non-nuclear)
B is the transport company (non-nuclear)

C is the user (steelworks) (non-nuclear)

In the event of detection, the scrap is physically at C's site, i.e. the steelworks, and the tendency would be to "discretely" send the scrap back to its owner, A. This would result in an unhealthy and dangerous situation because the real owner is not always known.

Since there has been detection, this means that radioactive waste is involved, with unauthorized circulation (transfer) being prohibited (above all if the shipper is foreign). The waste may only circulate after transport approval has been received. The authorities must, therefore, be advised as rapidly as possible and the appropriate action must be taken.

The vehicle is, therefore, immobilized and concrete action has to be taken *in situ* by the specialized workforce because the situation may become complex, with all aspects of radioprotection being of paramount importance (workers, environment).

For the sake of reflection, the following decision-making diagram is suggested :

However, from the point of view of the regulations, we are faced with a situation which is difficult to manage.

- There was, from the outset, a very major shortcoming;
- the types of radioactive sources responsible for detection, can vary greatly;
- the transport regulations for radioactive material are, in this instance, not easily complied with;
- the analysis of the radioactive contents may be complex and expensive. Financial considerations are important and responsibilities are not well-defined;
- counter-measures are more to do with emergency measures than routine operations;
- the emotional factor and the "media" must not be under-estimated.

Whether this concerns scrap which is transported by barge, by goods wagon or by truck :

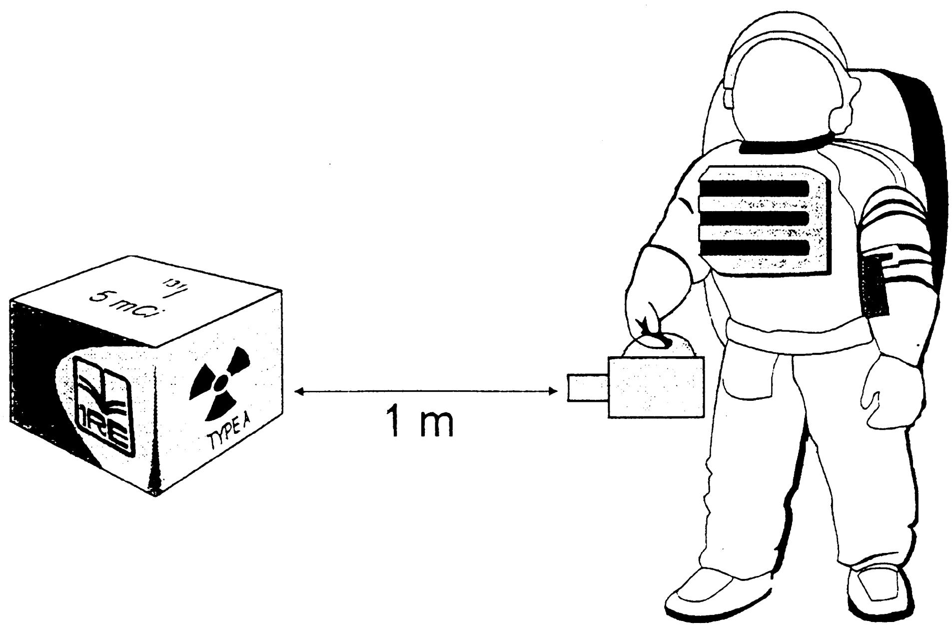
- the authorities are virtually unable to give transport approval without knowledge of the contents;
- it is difficult to envisage the necessary on-site action, which must not aggravate the situation or the risk;
- it is necessary to react quickly and to intervene immediately;
- there is no packaging, there is no possibility of confinement;
- it is difficult to label the package because it is not properly defined.

It is, thus, more a question of integrating the action to be taken with emergency problems and to reflect on matters beyond the normal framework of transport regulations, with this giving rise to the usual difficulties.

And finally, who will pay ?

4. TRANSPORT INDEX (T.I.)

4.1. Definition



4.2. Practical determination

- In some cases, the transport index is calculated on the basis of the activity present in the package and which is not systematically measured. This is the case of ^{99m}Tc generators which are produced and packaged in their hundreds by remote-control.

  Systematic determination is impossible and occasionally a loading error occurs which can result in the real index being different from that printed on the package.

  A physical inspection will not detect this error because the vehicles are loaded with a great many generators (several tens of generators) and the unit weight of each generator is quite high (50 kg).

- In the case of a large-scale consignment, it is quite frequent to “forget” the multiplication factor which depends on the surface area involved (up to 10 for more than 20 sq.m.).

- With regard to a package containing fissile material, it will be necessary to take account of the criticality TI and to retain the larger of the two TI values (measured and calculated index), with the latter being mentioned on the package. This was often forgotten.

Indeed, it was not rare to see transport labels on a container of more than 20 sq.m. containing an assortment of waste fuel with a TI of less than 1.

5. IRRADIATION OF THE DRIVER

5.1. Position of the Problem

The ADR specifies that the intensity of the radiation at the driver's place cannot exceed 0.02 mSv/h **unless** they (the driver and the conveyer) are equipped with individual radiation monitoring devices.

Belgian regulations are more restrictive. The Belgian authorities, in its transport approvals for transport companies, imposes **a maximum** of 0.02 mSv/h for the driver and the obligation to wear a dosimeter as soon as 0.0025 mSv/h is exceeded in the driver's cab.

5.2. Conclusion

To comply with Belgian regulations, additional shielding is generally necessary. This is placed behind the driver's cab and is made of 1- to 2-cm thick lead plates of more than 2 sq. m. in size.

In our view, although radioprotection is ensured, conventional safety is still not guaranteed because this extra shielding is not always installed in such a way that it will actually resist in the case of an accident. In this case, is the risk of impact not greater than the prevented radiological risk ?

6. PHYSICAL CONTROL OF THE TRANSPORT COMPANY AND THE SHIPPER

6.1. Belgian directives

A. The Shipper's physical control department is responsible for :

* monitoring the packaging of radioactive or fissile material and the loading inside the facility of these packages in/on the vehicle;
* measuring the dose rate on the package's outer surface;
* establishing the transport index;
* affixing the appropriate labels on each package.

B. <u>The Transport Company's physical inspection department is responsible for:</u>

* monitoring the lashing of the package in/onto the vehicle;
* inspecting contamination of the vehicle;
* measuring the dose rate on the outside (surface area, at 1 m and 2 m away from the outer surface) of the vehicle and the driver's cab;
* affixing the appropriate labels on the vehicle.

6.2. Position of the Problem

In Belgium, each company handling radioactive materials must appoint an officer in charge of physical inspection.
Should the company not have such a competent person, an agreement must be entered into with a competent control agency.

As described above, regarding the transport of radioactive materials, the responsibilities of the shipper and the transport company are determined as follows :

A. <u>Physical inspection by the Shipper</u>

* Determination of the type of package, correct loading of the material in the package;
* Control of the contamination on the outer surface of the package;
* Dose rate measurements on the outer surface of the package;
* Determination of the transport index;
* Labelling of the package.

B. <u>Physical inspection by the Transport Company</u>

* Control of the package's lashings in or on the truck;
* Control of the contamination of the truck;
* Dose rate measurements of the truck :
    - on the outer surface,
    - at 2m from the outer surface,
    - in the driver's cab.
* Labelling of the truck.

6.3. Comments

The missions described at the international and national level are clear but in practice there continue to present difficulties.

In major nuclear facilities, a physical inspection is carried out on a permanent basis - i.e. for each consignment - and most of the time, the various elements of the inspection are based on the fail-safe principle and are checked by both parties.

In smaller nuclear facilities and when the transport companies are too small, physical inspection is not really very present at the time the vehicle leaves the site. How can one be sure in the case of the transport of nuclear waste, for instance, that all the provisions have been properly complied with ....?

6.3.1 Physical inspection - shipper

A. <u>Type of package, loading in the package</u>

* The type of package will be determined by :
  - A1/A2 value isotope,
  - special form of radioisotope,
  - activity isotope, chemical and physical form,
  - fissile or not fissile.
* In the case of a type B or type F package, a copy of the package approval shall be present during transport.
* If the isotope is in a special form, the special form certificate shall be present during transport.
* Special requirements, if present in the package approval, regarding the loading of the material in the package shall be met.
* Before loading of the package(s) in or onto the truck, the following items shall be checked :
  - Is the truck technically OK to receive that type of package(s) ?
  - Does the truck have sufficient means of lashing ?
  - Does the truck/driver have the necessary paperwork (e.g. licence, ADR certificate, ...)?

B. <u>Contamination of packages</u>

* Contamination on the package's outer surface may not exceed the following values :
  - Bêta, gamma and low toxicity Alpha emitter : 4 Bq/cm²
  - Alpha emitters other than low toxicity emitter : 0,4 Bq/cm²

C. <u>Dose rate measurements of packages (not for excepted packages)</u>

* The dose rate shall not exceed the following values :
  - 0.1 mSv/h at 1 m from the outer surface (except when transported under exclusive use);
  - 2 mSv/h on the outer surface.

D. Transport Index

* The transport index is the value of the dose rate at 1 m of the package's surface expressed in mRem/h.
* For fissile packages, the transport index is determined by the criticality analysis; this is to be mentioned on the package approval certificate.

! ***Special attention should be given to large containers***

E. Package labelling

* Determination of the type of label (7A, 7B or 7C) depending on the dose rate measurement and the transport index. Two labels per package on opposite sides.
* When there is a subsidiary risk - additional labels (e.g. UF6 - radioactive + corrosion labels).
* Depending on the different means of transport (e.g. air or sea), additional labels may be required.

6.3.2. Physical inspection - Transport Company

A. Inspection of lashing in/on the truck

* The truck needs special provision for adequate lashing of the package(s).
* The lashing is the responsibility of the transport company.
* Lashing equipment has to be determined to resist the following forces during road transportation:

  - longitudinal : 2 g
  - lateral : 1 g
  - vertical : 3 g down, 2g up.

B. Contamination of the truck

* The contamination shall not exceed the following values :
  - Beta, gamma and low toxicity alpha emitters : 4 Bq/cm^2
  - Alpha emitters other than low toxicity : 0.4 Bq/cm^2.

* It is very important to measure the contamination of the truck's loading area after each unloading operation for immediate determination of the reason for any possible contamination.

C. Dose rates of the truck

* The dose rates shall not exceed the following values :

- dose rate at the outer surface	:	2 mSv/h
- dose rate at 2 m of the outer surface	:	0.1 mSv/h
- dose rate in the driver's cab	:	0.02 mSv/h

D. Labelling of the truck

* 3 type 7D labels (one on each side wall and one on the rear);
  These labels must be at least 25 x 25 cm square (10 x 10 cm in the case of small vehicles).
* In case of a subsidiary risk, 3 additional labels (e.g. UF_6 - radioactive + corrosion).
* The labels are to be removed after unloading.

7. CONCLUSIONS

* Transport regulations for radioactive material is comprehensive and well thought out. It is, however, very complex. Since perfection does not exist in this world, there are some slightly shady areas and this is normal. We have tried to point these out, together with the risks involved.

* What is important is that the workforce adopts a safety culture.

* Beware, inspection is the very last barrier !

This is a plea for effective, independent inspection which is up to the task at hand.

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