[7590-01-P]

**NUCLEAR REGULATORY COMMISSION**

**10 CFR Part 71**

# MAJOR REVISION TO 10 CFR PART 71: COMPATIBILITY WITH ST-1 -- THE IAEA TRANSPORTATION SAFETY STANDARDS -- AND OTHER TRANSPORTATION SAFETY ISSUES, ISSUES PAPER, AND NOTICE OF PUBLIC MEETINGS

AGENCY: Nuclear Regulatory Commission.

ACTION: Request for comment on issues paper, and notice of plans for public meetings.

SUMMARY: The Nuclear Regulatory Commission (NRC) is considering a rulemaking that would revise 10 CFR Part 71 to make it compatible with the International Atomic Energy Agency (IAEA) transportation safety standards as well as codify other requirements. The NRC is seeking early public input on the major issues associated with such a rulemaking. To aid in that process, the NRC is requesting comments on the Part 71 issues paper included in this notice. Specifically, the NRC is interested in public and industry comments related to: (1) quantitative information on the costs and benefits resulting from consideration of the factors described in the issues paper, (2) operational data on radiation exposures (increased or reduced) that might result from implementing the Part 71 changes; (3) whether the presented factors are appropriate; and (4) whether other factors should be considered, including providing quantitative information for these factors. The Commission believes that the stakeholders' comments will help to quantify the potential impact of these changes and will assist the NRC, as the proposed rule is developed, in developing a risk-informed alternative as its preferred option. NRC also intends to conduct four public meetings in July and August of this year to discuss those issues and solicit public comments.

DATES: Submit comments at the public meetings, or in writing by August 31, 2000. Comments received after this date will be considered if it is practicable to do so, but the Commission is able to assure consideration only for comments received on or before this date.

In addition to providing opportunity for written (and electronic) comments, public meetings on the paper will be held as follows (the times and locations will be announced separately):

July 11, 2000 - Chicago, Illinois

July 13, 2000 - San Francisco, California

July 18, 2000 - Atlanta, Georgia

August 10, 2000 - Washington, D.C.

ADDRESSES: Submit comments to: Secretary, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Attention: Rulemaking and Adjudications staff.

Deliver comments to 11555 Rockville Pike, Rockville, Maryland, between 7:30 a.m. and 4:15 p.m. on Federal workdays.

You may also provide comments via the NRC's interactive rulemaking website at http://ruleforum.llnl.gov). This site provides the capability to upload comments as files (any format), if your web browser supports that function. For information about the interactive rulemaking website, contact Ms. Carol Gallagher, (301) 415-5095 (e-mail:CAG@nrc.gov).

Copies of any comments received and documents related to this action may be examined at the NRC Public Document Room, 2120 L Street NW (Lower Level), Washington, D.C. Documents created or received at the NRC after November 1. 1999 are also available electronically at the NRC's Public Electronic Reading Room on the Internet at http://www.nrc.gov/NRC/ADAMS/index.html. From this site, the public can gain entry into the NRC's Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. For more information, contact the NRC Public Document Room (PDR) Reference staff at 1-800-397-4209, 202-634-3273 or email to pdr@nrc.gov.

FOR FURTHER INFORMATION CONTACT: Naiem S. Tanious, telephone: (301) 415-6103; e-mail: nst@nrc.gov, Office of Nuclear Material Safety and Safeguards, USNRC, Washington, D.C. 20555-0001. Specific comments on the public meeting process should be directed to Francis X. Cameron; e-mail fxc@nrc.gov, telephone: (301) 415-1642; Office of the General Counsel, USNRC, Washington, D.C. 20555-0001.

SUPPLEMENTARY INFORMATION:

## I. Background

By international agreement and through Commission direction, the NRC staff is preparing an overall rulemaking effort that addresses the need to make 10 CFR Part 71 regulations, "Packaging and Transportation of Radioactive Material" compatible with the most current revision of the IAEA Safety Standards Series No. ST-1. Part 71 is based, in general, on the safety standards developed by the IAEA. The IAEA has been revising its transportation standards on approximately a 10-year cycle, with the last edition, ST-1, published in December 1996. Further, several additional issues related to other changes to 10 CFR Part 71 are being considered by NRC. These issues include the fissile material exemptions, general license provisions, and the current requirements for double containment of plutonium.

The NRC is supplementing its standard rulemaking process by conducting enhanced public participatory activities including facilitated public meetings before the start of any formal rulemaking process to solicit early and active public input on major issues with revision of 10 CFR Part 71. The NRC will also utilize its rulemaking website to make the issues paper available to the public and to solicit public comments. To facilitate discussion and public comments, the NRC has prepared an issues paper that describes 17 rulemaking issues (IAEA and Non-IAEA-related) to be addressed in revisions to Part 71. These issues are described in more detail in Section III of this notice.

## II. Request for Written and Electronic Comments and Plans for Public Meetings

The NRC is soliciting comments on the items presented in the issues paper in Section III of this notice. Comments may be submitted either in writing or electronically as indicated under the ADDRESSES heading. In addition to providing an opportunity for written comments, the NRC is holding facilitated public meetings at four different geographical locations on the issues discussed in Section III (see the DATES heading of this notice for the dates and locations of these meetings; a separate announcement will be provided on the meeting times and locations).

In addition to inviting public comments on the issues presented in Section III, NRC is soliciting specific comments related to: (1) quantitative information on the costs and benefits resulting from consideration of the factors described in the issues paper, (2) operational data on radiation exposures (increased or reduced) that might result from implementing the Part 71 changes; (3) whether the presented factors are appropriate; and (4) whether other factors should be considered, including providing quantitative information for these factors. The Commission believes that the stakeholders' comments will help to quantify the potential impact of these changes and will assist the NRC, as the proposed rule is developed, in developing a risk-informed alternative as its preferred option.

Based on the comments received in written or electronic form, and at the public meetings, the Commission will then be in a better position to evaluate options for Part 71 rulemaking, to decide on the preferred options, and to proceed with development of a proposed rule.

## III. Issues Paper on Major Revision to 10 CFR Part 71: Compatibility with ST-1 -- the IAEA Transportation Safety Standards -- and Other Transportation Safety Issues

### A. INTRODUCTION

#### 1. Background:

In 1969, the International Atomic Energy Agency (IAEA), recognizing that its international regulations for the safe transportation of radioactive material should be revised from time to time because of scientific and technical advances, and accumulated experience, invited Member States (the U.S. is a Member State) to submit comments and suggest changes to its standards. As a result of this initiative, the IAEA issued revised standards in 1973 (Regulations for the Safe Transport of Radioactive Material, 1973 Edition, Safety Series (SS) No. 6). The IAEA has periodically reviewed its transportation regulations (about every ten years) to ensure that the regulations are kept current. Thus, a review of IAEA regulations was initiated in 1979 and resulted in the publication of revised regulations in 1985 (1985 Edition, SS No. 6).

The U.S. Nuclear Regulatory Commission (NRC) also periodically revises its regulations to make them compatible, to the extent appropriate, with those of the IAEA. On August 5, 1983 (48 FR 35600), the NRC published, in the Federal Register, a final revision to 10 CFR Part 71, "Packaging and Transportation of Radioactive Material." That revision, in combination with a parallel revision of the hazardous materials transportation regulations of the U.S. Department of Transportation (DOT), brought U.S. domestic transport regulations into general accord with the 1973 edition of SS No. 6. The next IAEA revision of the transportation standards in SS No. 6 resulted in a revision to Part 71 that was published on September 28, 1995 (60 FR 50248), to make Part 71 compatible with the 1985 edition of SS No. 6. DOT published its corresponding revision to Title 49 of the Code of Federal Regulations on the same date.

In each case , the NRC coordinated its Part 71 revisions with the DOT. DOT is the U.S. Competent Authority for transportation of hazardous materials. “Radioactive Materials Regulations” is a subset of “Hazardous Materials Regulations” in Title 49. The DOT and the NRC co-regulate transport of radioactive material in the United States and have a Memorandum of Understanding to that effect.

The last revision to the IAEA SS No. 6 was titled Safety Standards Series No. ST-1, referred to hereafter as ST-1, and was published in December 1996.

**2. Scope of Part 71 Rulemaking:**

The Commission has directed the NRC staff to begin rulemaking to revise Part 71 for compatibility with ST-1. The NRC staff compared ST-1 to SS No. 6 to identify changes made in ST-1, and then identified affected sections of Part 71. Based on this comparison, the NRC staff identified eleven Part 71 IAEA-compatibility issues to be addressed through the rulemaking process. These eleven issues are discussed in greater detail in Section B. Six additional issues were identified for incorporation in the rulemaking process, through NRC staff identification and through Commission direction, and are also discussed in further detail in Section B.

The Part 71 rulemaking and this issues paper are being coordinated with DOT to ensure that consistent regulatory standards are maintained between NRC and DOT radioactive material transportation regulations, and to ensure coordinated publication of the final rules by each agency. Note that on December 28, 1999 (64 FR 72633), DOT published an Advance Notice of Proposed Rule regarding adoption of ST-1 in its regulations.

## B. ISSUES FORMAT

The following format is used in the presentation of the issues that follow. Each issue is assigned a tracking number with a short title, and includes an issue description paragraph and a listing of factors for consideration. The factors for consideration in this document are not meant to be a complete or final listing, but are included to help prompt consideration and discussion of the issue. In July and August 2000, through a series of public meetings and a summary workshop, the public and industry will be requested to 1) comment on and recommend additions, deletions, or modifications to the factors for consideration; 2) propose implementation options for each issue; and 3) provide estimated implementation cost information. Other venues for feedback will be made available through mailings and by internet through the NRC web site.  This public feedback will then be used in developing implementation options for Commission consideration as the Part 71 rulemaking process proceeds.  Comments received that are outside the scope of this rulemaking may be addressed in future rulemaking if warranted.

Factors for consideration that are common to most of the issues are stated here, rather than repeated in each issue.  These include: 1) how should risk considerations (i.e., what can happen, how likely is it, what are the consequences) be factored into rulemaking on applicable issues, 2) costs (i.e., administrative, training, testing) to industry and/or Government agencies in adopting ST-1 requirements (issues 1 -11) or the NRC-initiated changes (issues 12 - 17), and 3) potential problems that may occur as a result of adopting ST-1 requirements, or problems that may occur from partial or non-adoption of the ST-1 requirements resulting in dual standards between domestic (10 CFR 71) and international (ST-1) requirements.  For issues 1 - 11, the "factors for consideration" noted under each issue are generally  written in the context of adopting the ST-1 requirements into Part 71.

In the case of the eleven IAEA-compatibility issues, portions of the Safety Standards Series ST-1 are referenced by the corresponding paragraph number from the original IAEA document.  The full text of the ST-1 references can be found in Appendix A of this issues paper.

## Issue 1. Changing Part 71 to SI units only

### Description

ST-1, Annex II, page 199 states:  "This edition of the Regulations for the Safe Transport of Radioactive Material uses the International System of Units (SI)."  The change to SI units exclusively is evident throughout ST-1.  ST-1 also requires that activity values contained in shipping papers and displayed on package labels be expressed only in SI units (paragraphs 543 and 549).  SS No. 6, 1985 Edition, used SI units as the primary controlling units, with subsidiary units in parentheses; either units were permissible on labels and shipping papers.

The ST-1 requirement regarding only the use of SI units conflicts with the NRC Metrication Policy issued on June 19, 1996 (61 FR 31169).  This policy allows a dual-unit system to be used; SI units with English units in parentheses.  According to the NRC's metrication policy, the following documents should be published in dual units: new regulations, major amendments to existing regulations, regulatory guides, NUREG-series documents, policy statements, information notices, generic letters, bulletins, and all written communications directed to the public.  Documents specific to a licensee, such as inspection reports and docketed material dealing with a particular licensee, will be issued in the system of units employed by the licensee.  Currently, Part 71 utilizes the dual unit scheme in accordance with the NRC Metrication Policy.

### Factors for Consideration

- What changes would licensees and Certificate of Compliance holders have to make to relevant documents if NRC revised 10 CFR Part 71 to require SI units only?
- What risks and safety impacts might occur in shipments because of possible confusion or erroneous conversion between the currently utilized English units and SI units?
- What sort of transition period would be needed to allow for the conversion to exclusive use of SI units?
- What other conforming changes would have to be made to Title 10?

## Issue 2. Radionuclide Exemption Values

### Description

Exempt materials are those which are of such low potential hazard that they may not be required to be shipped in accordance with specific transportation regulations. In ST-1, the IAEA adopted a new approach to specifying these materials by developing radionuclide-specific activity concentration values for exempt materials and activity limits for exempt consignments. These new values are found in ST-1, Tables I and II, and Section IV. Related information is provided in paragraphs 401 through 406 of ST-1. Exempt materials are those that fall below the listed activity concentration values. Exempt consignments are packages or loads that have a total activity less than the listed activity values.

The exempt materials activity concentration values range from 0.1 to 1,000,000 Bq/g, with most radionuclides in the 1 to 100 Bq/g range. This IAEA requirement does not currently exist in Part 71. Appendix A to Part 71 - Determination of $A_1$ and $A_2$, does not contain exemption values for each radionuclide because the exemption for low-level radioactive material as contained in 10 CFR 71.10(a) is 70 Bq/g or less.

Some materials, such as ores containing naturally occurring radionuclides, would be brought into the scope of the regulations for the first time; however, provisions are included in ST-1 that reduce the potential impact on natural materials containing radionuclides at these low levels. The provisions continue to exempt natural material and ores containing naturally occurring radionuclides, that are not intended to be processed for the use of these radionuclides, provided the activity concentration of the material does not exceed 10 times the values [ST-1 paragraph 107(e)]. Additionally, for materials that may appear in the scope of the regulations for the first time, but which have activity concentrations not exceeding 30 times the exempt activity concentrations, provisions exist in ST-1 to allow them to be transported as LSA-I materials that may be transported unpackaged (in bulk).

### Factors for Consideration

- In some cases, would shippers have to expend resources to: 1) identify the radionuclides in a material; 2) measure the activity concentration of each radionuclide; and, 3) apply the method for mixtures of radionuclides when determining the basic radionuclide values for exempt material?
- Should the exemption values apply to domestic as well as export shipments?
- If the exemption values only applied to export shipments, would the resulting standard be practical to implement?

- If DOT specifies the exemption values in its regulations (49 CFR 173), should the NRC incorporate those same exemption values in Part 71, or simply make reference to the exemption values in the DOT regulations?

## Issue 3. Revision of $A_1$ and $A_2$

### Description

The $A_1$ and $A_2$ values specified in Part 71, Appendix A, are basic dose-based values used in several areas of the regulations, including determining the type of package that must be used for transporting radioactive material. For example, the $A_1$ values are the maximum activity of special-form materials allowed in a Type A package, and the $A_2$ values are the maximum activity of non-special-form material allowed in a Type A package. The $A_1$ and $A_2$ values are also used for several other quantitative limits including Type B-package activity release limits, low-specific activity material specifications, and excepted package content limits.

The ST-1 revised $A_1$ and $A_2$ values are primarily based on dosimetric models that use the IAEA's Q system for dose determination. The Q system includes consideration of a broad range of specific exposure pathways consisting of: external photon dose, external beta dose, inhalation dose, skin and ingestion dose because of contamination, and dose from submersion in gaseous isotopes. The main changes in the Q system resulted from making the dosimetric models consistent with those used in International Commission on Radiation Protection (ICRP) Publication 61. The lung model and dose conversion factors were updated to the latest ICRP models and the radionuclide values were recalculated. The Q system reference doses and exposure pathways were not changed.

### Factors for Consideration

- Is there a practical alternative to adoption of the $A_1$ and $A_2$ values?
- Are there specific values that should be modified for domestic use only? What would be the justification for doing so?
- To what extent should the US partial adoption of ICRP 61 be considered for revising the $A_1$ and $A_2$ values?

## Issue 4. Uranium Hexafluoride Package Requirements

### Description

ST-1 introduces detailed requirements for uranium hexaflouride ($UF_6$) packages designed for more than 0.1 kg $UF_6$. NRC certifies Type B and fissile (i.e., enriched uranium) $UF_6$ packages under 10 CFR Part 71. Although most of these issues are under DOT in 49 CFR Part 173, the new ST-1 provisions relevant to 10 CFR Part 71 are summarized as follows (see Appendix A for a listing of the specific ST-1 provisions):

Para 629: Packages shall be packaged and transported in accordance with an international standard, ISO 7195, "Packaging of Uranium Hexafluoride ($UF_6$) for Transport."

ST-1 also allows [para 632(a)] for use of equivalent national standards (e.g., ANSI N14.1); provided that approval by all countries involved in the shipment is obtained (i.e., multilateral approval).

Para 630: ST-1 requires that packages must withstand: (a) a minimum internal pressure test to 2.8 MPa (1.4 MPa for multilateral approval), (b) the "normal conditions of transport" drop test, and (c) the hypothetical accident condition thermal test (except that packages containing grater than 9000 kg are exempt from this test if given multilateral approval).

Para 631: ST-1 prohibits packages from utilizing pressure relief devices.

Para 677(b): ST-1 includes an exception that allows $UF_6$ packages to be evaluated for criticality without considering the in-leakage of water into the containment system. This provision means that a single fissile $UF_6$ package does not have to be subcritical assuming that water leaks into the containment system. This provision only applies when there is no physical contact of the cylinder valve to any other component of the packaging after the hypothetical accident tests, the valve remains leak-tight, and when there is a high degree of quality control in the manufacture, maintenance, and repair of packaging coupled with tests to demonstrate closure of each package before each shipment.

**Factors for Consideration**

- NRC practice has been to certify fissile $UF_6$ packages (including the cylinder which is the containment vessel and a protective overpack) that are shown to be leaktight when subject to the hypothetical accident tests and to specify that the cylinder meets ANSI N14.1 (ANSI N14.1 has the domestic pressure test requirement in 630(a), not the regulations). For this reason, it is believed that NRC-certified $UF_6$ packages already comply with the above package performance requirements (para 630 and 677(b)). However, these changes appear to have significant ramifications for non-fissile $UF_6$ packaging that are under the purview of DOT.
- NRC practice has been to reference the ANSI N14.1 standard in the certification, but not to reference the standard in the rule. Although the ISO-7195-2000 standard (in draft) has been drafted taking into account ANSI N14.1, a detailed confirmation of the compatibility of the two standards has not been performed. NRC has representation on the ANSI N14.1 revision panel.

## Issue 5. Introduction of Criticality Safety Index (CSI) Requirements

**Description**

For fissile material packages, ST-1 defines a new term, "criticality safety index" (CSI) (paragraph 218), that applies in addition to the traditional package transport index (TI). In current domestic regulations and in the previous IAEA regulations, the overall package TI was determined based upon the more limiting of a "TI based upon criticality considerations" and a "TI based on package radiation levels." Both NRC and DOT regulations define and rely on the

TI to determine appropriate safety requirements.

The CSI is determined in the same manner as the current TI "based upon criticality considerations," but it now must be displayed on shipments of fissile material (paras 544-545) using a new "fissile material" label. A package TI is still determined in the same way as the "TI based on package radiation levels" and continues to be displayed on the traditional "radioactive material" label.

**Factors for Consideration**

- Under the new approach, it is believed that some shipments of fissile material packages might be made more efficiently (equivalent safety but more packages allowed in a single shipment), due to avoiding the situation where separation distance requirements (radiological safety) restrict package accumulation (criticality safety), or vice versa.
- Are any issues envisioned in the use of two TI values for shipments?

## Issue 6. Type C Packages and Low Dispersible Material

**Description**

IAEA has adopted the concept of a new category of package, the Type C package (paragraphs 230, 667-670, 730, 734-737) that could withstand severe accident conditions in air transport without loss of containment or significant increase in external radiation levels. At the same time, ST-1 introduced a new category of material, Low Dispersible Material (LDM), which due to its limited radiation hazard and low dispersibility could continue to be transported by aircraft in Type B packages. U.S. regulations have no Type C package or LDM category, but do have specific requirements for the air transport of plutonium. These specific NRC requirements for the air transportation of plutonium (10 CFR 71.64 and 71.74) continue to apply, and will **not** be addressed in this rulemaking.

The Type C requirements apply to packages destined for air transport that contain a total activity above the following thresholds: for special form material - 3,000 $A_1$ or 100,000 $A_2$, whichever is lesser, and for all other radioactive material - 3,000 $A_2$ . Below these thresholds, Type B packages would be permitted to be used in air transport.

The Type C package performance requirements are significantly more stringent than those for Type B packages. For example, a 90 m/s impact test is required instead of the 9 m-drop test. A 60-minute fire test is required instead of the 30-minute Type B requirement. Other additional tests, such as a puncture/tearing test are also imposed. These tests are more stringent and are expected to result in package designs that will survive more severe aircraft accidents than Type B package designs.

The LDM specification was added to account for materials (package contents) that have inherently limited dispersibility, solubility, and external radiation levels. The test requirements for LDM are a subset of the Type C package requirements (90 m/s impact and 60 minute thermal test) with an added solubility test, and must be performed on the material without packaging. Specific acceptance criteria are established for evaluating the performance of the material during and after the tests (less than 100 $A_2$ in gaseous or particulate form of less than 100 micrometer aerodynamic equivalent diameter and less than 100 $A_2$ in solution). These stringent performance and acceptance requirements are intended to ensure that these materials can continue to be transported safely in Type B packages aboard aircraft.

**Factors for Consideration**

- What would be the impact on air transport of currently certified Type B packages if the activity content is limited to the activity content thresholds specified above ?
- What tests and analyses would be a practical method for demonstrating compliance with the type C package standards ?

## Issue 7. Deep Immersion Test

**Description**

The IAEA performance requirement for deep water immersion contained in ST-1 (para. 657 and 730) is an expansion of the requirement contained in SS No. 6. Previously, the deep immersion test was only required for packages of irradiated fuel exceeding 37 PBq (1,000,000 Ci). The ST-1 requirements apply to all Type B(U) and B(M) packages containing more than $10^5 A_2$ and to Type C packages.

10 CFR 71.61 requires a deep immersion test for packages of irradiated nuclear fuel with activity greater than $10^6$ Ci. Currently, 10 CFR 71.61 is more conservative than SS No. 6, with respect to irradiated fuel package design requirements because it requires that a package for irradiated nuclear fuel must be designed such that its undamaged containment system can withstand an external water pressure of 2 MPa for a period of not less than one hour *without collapse, buckling, or in leakage of water*. The conservatism lies in the test criteria of no collapse, buckling, or in leakage as compared to the *"no rupture"* criteria found in SS No. 6 and ST-1.

To be consistent with ST-1, the NRC would have to revise 10 CFR Part 71.61 to apply to all packages with activity greater than $10^5 A_2$ and adopt the ST-1 test criteria.

**Factors for Consideration**

- How should the differences in the acceptance standards be addressed?
- What would be the impact on availability of packages and shipping costs if all packages with an activity greater than $10^5 A_2$ are required to pass the immersion test requirements?
- Would US origin package designs have to be specially reviewed and certified before shippers could export them in accordance with international regulations if ST-1 requirements were not adopted?

## Issue 8. Grandfathering Previously Approved Packages

**Description**:

Historically, IAEA, DOT, and NRC regulations have included transitional arrangements or “grandfathering” provisions whenever the regulations have undergone major revision. The purpose of grandfathering is to minimize the costs and impacts of implementing changes in the regulations. Package designs and packagings compliant with the existing regulations do not become “unsafe” when the regulations are amended (unless a significant safety issue is corrected in the revision).

Grandfathering typically includes provisions that allow for: 1) continued use of existing package designs and packagings already fabricated, although some additional requirements may be imposed, 2) completion of packagings in the process of being fabricated or that may be fabricated within a given time period after the regulatory change; and 3) limited modifications to package designs and packagings without the need to demonstrate full compliance with the revised regulations, provided that the modifications do not significantly affect the safety of the package.

A major change in ST-1 is that “grandfathering” should be limited to only those package designs that have been certified under the last two major revisions of the regulations. Packages approved under an earlier revision would either be removed from service or be required to be re-certified under the revised regulations that result from this rulemaking.

As revised in 1996, IAEA regulations in ST-1 only recognize the “grandfathering” of package designs certified under the 1973 and 1985 editions of IAEA regulations (SS No. 6 ). Package designs approved under the 1967 edition of SS No. 6 would be required to be re-certified, removed from service, or shipped via exemption (i.e., special arrangement). If this approach to “grandfathering” is adopted in DOT and NRC regulations, package designs approved to earlier versions of DOT and NRC regulations (i.e., those based on 1967 IAEA regulations) would be required to be re-certified, removed from service, or shipped via exemption .

**Factors for Consideration**

- Should the “grandfathering “ of previously approved packages be limited to those approved under the last two major revisions of the regulations? If not, on what basis should the “grandfathering “ of previously approved packages be allowed?
- How long should “grandfathered” packages be allowed to be fabricated or used?
- What type and magnitude of package design changes should be allowed for “grandfathered” packages, before re-certification to the current set of regulations is required?
- IAEA has initiated a process to review and update ST-1 on a two-year frequency and does this new process raise any issues on the grandfathering limitations to the last two major revisions?

## Issue 9 . Changes to Various Definitions

### Description

The NRC is contemplating changes to various definitions in Part 71 to provide internal consistency and improve correlation with ST-1. 10 CFR 71.4 includes defined terms used throughout Part 71. These terms require clear definition so that they can be used to accurately communicate requirements to licensees. The NRC would add the following definitions from ST-1: 1) *Confinement system* (paragraph 209), 2) *Criticality safety index* (paragraph 218; reference issue 5), 3) *Low dispersible radioactive material* (paragraph 225; reference issue 6), and 4) *Quality assurance* (paragraph 232). Additionally, the NRC would propose to revise the definition of “package” in 10 CFR 71.4 to be consistent with ST-1. For reference, the ST-1 definitions are contained in Appendix A and provided below.

Para. 209. “*Confinement System* shall mean the assembly of fissile material and packaging components specified by the designer and agreed to by the competent authority as intended to preserve criticality safety.”

Para. 218. “*Criticality safety index* (CSI) assigned to a package, overpack or freight container containing fissile material shall mean a number which is used to provide control over the accumulation of packages, overpacks or freight containers containing material.”

Para. 225. “*Low dispersible radioactive material* shall mean either a solid radioactive material or a solid radioactive material in a sealed capsule, that has limited dispersibility and is not in powdered form.”

Para. 232. “*Quality assurance* shall mean a systematic programme of controls qand inspections applied by an organization or body involved in the transport of radioactive material which is aimed at providing adequate confidence that the standard of safety prescribed in these Regulations is achieved in practice.”

### Factors for Consideration

- Do the definitions conflict with existing programs, or introduce other issues or concerns?
- Are there other definitions of terms that are recommended for incorporation in Part 71?

## Issue 10. Crush Test for Fissile Material Package Design

### Description

Under requirements for packages containing fissile material, ST-1 682(b) requires tests specified in paragraphs 719-724 followed by whichever of the following is the more limiting: the drop test onto a bar as identified in paragraph 727(b) and, either the crush test listed in paragraph 727(c) for packages having a mass not greater than 500 kg and an overall density not greater than 1000 $kg/m^3$ based on external dimensions, or the nine meter drop test listed in paragraph 727(a) for all other packages; or the water immersion test of paragraph 729.

SS No.6 and Part 71 presently require the crush test for fissile material packages having a mass not greater than 500 kg and an overall density not greater than 1000 kg/$m^3$ based on external dimensions, and radioactive contents greater than 1000 $A_2$ not as special from radioactive material. Under ST-1, the crush test is no longer limited to fissile material packages containing an activity greater than 1000 $A_2$ because ST-1 has extended the crush test requirement to include fissile material package designs regardless of the activity of the contents. This was done in recognition that the crush environment was a potential accident force that should be protected against for both radiological safety purposes (packages containing more than 1000 $A_2$ in normal form) and criticality safety purposes (fissile material package designs).

To be consistent with ST-1, the NRC would have to revise 10 CFR Part 71 wording to recognize removal of the 1000 $A_2$ activity limit with respect to the crush test requirement for fissile material package designs. However, full compliance with ST-1 requirements for fissile material packages would also require changes to the hypothetical accident conditions test sequencing of 10 CFR 71.73 and would require performance of the nine-meter free drop test or the crush test, but not both as presently required by §71.73.

**Factors for Consideration**

- How should the differences in the test sequencing and required tests be addressed? Would the test sequencing requirements be applied to Type B packages as well?
- What would be the impact on availability of packages and shipping costs due to elimination of the 1000 $A_2$ activity limit for fissile material packages having a mass not greater than 500 kg and an overall density not greater than 1000 kg/$m^3$ based on external dimensions?
- If Part 71 is changed to only eliminate the 1000 $A_2$ activity limit for fissile material packages, but all other tests and the testing sequence remains unchanged, what implications would this have for US origin packages for export?

## Issue 11. Fissile Material Package Design for Transport by Aircraft

**Issue Description**

For shipment of fissile material by air, ST-1 requires that packages with quantities greater than excepted amounts (that would include all the NRC certified packages) require an additional criticality evaluation. Specifically, the requirements are:

Para 680(a): Packages must remain subcritical, assuming 20 centimeters water reflection but not inleakage (i.e., moderation) when subjected to the tests for Type C packages (see Issue 6). The specification of no water ingress is given as the objective of this requirement is protection from criticality events resulting from mechanical or physical rearrangement of the geometry of the package (i.e., fast criticality).

Para 680(b) This provision states that if a package takes credit for “special features,” this package can only be presented for air transport if it is shown that these features remain effective even under the Type C test conditions followed by a water immersion test. “Special features” are specified in ST-1 Para 677, and include features that provide moderator exclusion.

The application of the paragraph 680 requirement to fissile-by-air packages is in addition to the normal condition tests (and possibly accident tests) that the package already must meet. Thus:

- A Type IF or AF package by air must: 1) withstand incident-free conditions of transport with respect to release, shielding, and maintaining subcriticality (single package and array of packages), 2) withstand accident condition tests with respect to maintaining subcriticality (single package and array of packages), and 3) comply with para 680 with respect to maintaining subcriticality (single package).
- A Type BF package by air must: 1) withstand incident-free conditions of transport and Type B tests with respect to release, shielding, and maintaining subcriticality (single package and array of packages); and 2) comply with para 680 with respect to maintaining subcriticality (single package).
- A Type C fissile material package must withstand: incident-free conditions of transport (single package and array of packages), Type B tests (single package and array of packages), and Type C tests (single package) with respect to release, shielding, and maintaining subcriticality.

**Factors for Consideration**

- Certain factors need to be considered in determining the practical impacts of domestic adoption of ST-1 paragraph 680. First, all uranium can be shipped in non-Type C package (IF, AF) due to its $A_1$ and $A_2$ values. The paragraph 680(a) requirements appear to be readily satisfied by low-enriched uranium, because low enriched uranium (less than approximately 5% enrichment) would typically require moderation (e.g., by water) to achieve nuclear criticality, but the test specifies no water ingress. Secondly, there are statutory restrictions on air transport of plutonium in the U.S. Finally, packaging for air transportation may follow International Civil Aviation Organization Technical Instructions that are also being revised for compatibility with ST-1.

## Issue 12: Special Package Approvals

**Description**

The transport of large objects that are too large for certified packagings and cannot satisfy the packaging requirements was not considered in the development of Part 71. However, as decommissioning activities increase, the need to transport large objects is rising. For example, in 1997, Portland General Electric Company (PGE) requested approval of the Trojan Reactor Vessel Package (TRVP) (including internals) for transport to the disposal facility operated by US Ecology on the Hanford Nuclear Reservation near Richland, Washington. The TRVP contained approximately 74 petabequerels (2 million curies) in the form of activated metal and 5.7 terabequerels (155 curies) in the form of internal surface contamination; was filled with low-density concrete; and weighed approximately 900 metric tons (1000 tons).

The Commission approved the Trojan shipment under exemptions issued through 10 CFR Part 71.8. Also, the U.S. Department of Transportation's (DOT's) regulations that govern radioactive material shipments do not recognize packages approved via NRC exemption, so DOT also had to consider and issue an exemption for the Trojan shipment.

Because it is the Commission's policy to avoid the use of exemptions for recurring licensing actions, the NRC staff is considering adding regulatory provisions to Part 71 to address special package approvals. If adopted, these provisions would provide a mechanism for review of special packages under the regulations without the need for exemptions.

**Factors for Consideration**

- Should Part 71 be revised to address reactor vessels specifically or to address large objects in general?
- Should NRC consider adopting an analogue of IAEA's special arrangement provision modified to address packaging?
- What (additional) determinations should be included in an application for a special package approval?
- Should the risk-informed basis used specifically for the Trojan approval be adopted for other special package approvals?

## Issue 13. Expansion of Part 71 Quality Assurance Requirements to Holders of, and Applicants for, a Certificate of Compliance

**Description**

The NRC has observed problems with the performance of 10 CFR Part 72 Certificate of Compliance (CoC) holders in implementing the Part 72 quality assurance (QA) requirements. Problems have occurred in design, design control, fabrication, and corrective action areas. Although CoCs are legally binding documents, certificate holders or applicants for a CoC and their contractors and subcontractors have not clearly been brought within the scope of Part 72 requirements. Therefore, because the terms "certificate holder" and "applicant for a certificate of compliance" do not appear in the Part 72, Subpart G regulations, the NRC has not had a clear basis to cite these persons for violations of Part 72 requirements in the same way it treats licensees.

The NRC Enforcement Policy[1] and its implementing program were established to support the NRC's overall safety mission in protecting public health and safety and the environment. Consistent with this purpose, enforcement actions are used as a deterrent to emphasize the importance of compliance with requirements and to encourage prompt identification and comprehensive correction of the violations. Enforcement sanctions consist of Notices of Violation (NOVs), civil penalties, and orders of various types. In addition to formal enforcement actions, the NRC also uses related administrative actions such as Notices of Nonconformance

[1] NUREG-1600, Revision 1, "General Statement of Policy and Procedures for NRC Enforcement Actions," May 1998 (at 63 FR 26630; May 13, 1998).

(NONs), Confirmatory Action Letters, and Demands for Information to supplement its enforcement program. The NRC expects licensees, certificate holders, and applicants for a CoC to adhere to any obligations and commitments that result from these actions and will not hesitate to issue appropriate orders to ensure that these obligations and commitments are met. The nature and extent of the enforcement action are intended to reflect the seriousness of the violation involved. An NOV is a written notice setting forth one or more violations of a legally binding requirement.

However, when the NRC has identified a failure to comply with Part 72 QA requirements by certificate holders or applicants for a CoC, it has issued an NON rather than an NOV. Although an NON and an NOV appear to be similar, the Commission prefers the issuance of an NOV because: (1) the issuance of an NOV effectively conveys to both the person violating the requirement and the public that a violation of a legally binding requirement has occurred; (2) the use of graduated severity levels associated with an NOV allows the NRC to effectively convey to both the person violating the requirement and the public a clearer perspective on the safety and regulatory significance of the violation; and (3) violation of a regulation reflects the NRC's conclusion that potential risk to public health and safety could exist. Therefore, the NRC believed that limiting the available enforcement sanctions to administrative actions was insufficient to address the performance problems observed in industry.

In response to this problem, the NRC staff submitted a rulemaking plan to revise Part 72 to the Commission in SECY-97-214.[2] In a Staff Requirements Memorandum (SRM) to SECY-97-214, the Commission approved the staff's rulemaking plan and directed the staff to also consider whether conforming changes to the quality assurance (QA) regulations in Part 71 would be necessary, because of dual purpose cask designs. Dual purpose cask designs are intended for both the storage of spent fuel under Part 72 and the transportation of spent fuel under Part 71. In a memorandum from the EDO to the Commission, dated December 3, 1997, the NRC staff indicated that expansion of the Part 71 QA provisions to include certificate holders and applicants for a Certificate of Compliance (CoC) would be made as part of the rulemaking to conform Part 71 to IAEA standard ST-1.

The Commission recently issued a final rule expanding QA regulations in Part 72, Subpart G, to specifically include certificate holders and applicants for a CoC. Consequently, the NRC is now considering similarly expanding the QA regulations in Part 71, Subpart H, to specifically include certificate holders and applicants for a CoC. The NRC believes that this change is necessary to ensure consistency between the QA provisions of Parts 71 and 72, particularly in light of NRC approval of dual purpose cask designs. As with the Part 72 final rule, this issue would provide explicit notice to certificate holders and applicants for a CoC of their QA responsibilities; and would provide the NRC staff with additional enforcement sanctions — should violations of the Part 71 QA requirements occur.

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[2] SECY-97-214, "Changes to 10 CFR Part 72, Expand Applicability to Include Certificate Holders and Applicants and Their Contractors and Subcontractors," dated September 24, 1997. This rulemaking plan expanded the applicability of the QA provision of Part 72, Subpart G, to specifically include Part 72 certificate holders and applicants for a Certificate of Compliance.

## Factors for Consideration

- Should consistency be maintained between the QA provisions of Parts 71 and 72, in light of the existence of dual purpose cask designs?

## Issue 14. Adoption of ASME Code

## Description

The NRC staff proposes that the ASME (American Society of Mechanical Engineers) Code, Section III, Division 3, be incorporated by reference in 10 CFR Part 71 via rulemaking. This rule will ensure implementation of the ASME Code in cask fabrication, including all QA aspects of the code, such as the presence of an authorized nuclear inspector (ANI) during the fabrication to ensure that the code requirements are met, and stamping of components after fabrication is complete. This approach would be similar to how the ASME Code is endorsed for power reactors under 10 CFR 50.55(a) and would make the fabrication process for transportation cask containments commensurate with that used for nuclear power plant components,.

NRC inspections of vendors'/fabricators' shops (for fabrication of spent fuel storage canisters and transportation casks) have identified, over the past several years, quality control (QC) and quality assurance (QA) problems in these fabricated systems. A major reason for these problems is that these fabricators/vendors do not fully use a code for QA in the fabrication process of these systems. These QA problems have in some instances continued in spite of repeated adverse NRC and licensee findings.

The NRC staff intends to incorporate two recent developments. First, ASME issued a consensus code in May 1997 entitled: "Containment Systems and Transport Packages for Spent Fuel and High Level Radioactive Waste," ASME B&PV Code Section III, Division 3, that would require stamping of components constructed to it (i.e., the transportation cask's containment). Second, Public Law 104-113 "National Technology Transfer and Advancement Act" was enacted in 1996 to require that Federal agencies use consensus standards (e.g., the ASME B&PV Code), except when there are justified reasons for not doing so. These two developments support efforts to initiate rulemaking in this area.

## Factors for Consideration

- Can other regulatory vehicles for NRC endorsement of Code be used or should this only be done by rulemaking?
- Are there other voluntary consensus standards that should be considered in addition to, or in lieu of, ASME code?

## Issue 15. Adoption of Changes, Tests, and Experiments Authority

## Description

The Commission recently approved a final rule to expand the provisions of

10 CFR 72.48, “Changes, Tests, and Experiments,” to include Part 72 certificate holders (October 4, 1999; 64 FR 53582). 10 CFR Part 72 Certificate holders are allowed to make changes to a spent fuel storage cask design or conduct tests and experiments, without prior NRC review and approval, if certain requirements are met. However, Part 71 contains no similar provisions to permit a certificate holder to change the design of a Part 71 transportation package. The NRC has issued Certificates of Compliance (CoC) under Parts 71 and 72 for dual purpose casks [packages] (i.e., containers intended for both the storage and transportation of spent fuel). This has created the situation where a 10 CFR Part 72 certificate holder is authorized to change a storage design feature of a dual-purpose storage/transportation cask without obtaining NRC prior approval; however, the 10 CFR Part 71 certificate holder is **not** authorized to modify transportation package design without obtaining NRC prior approval, even when the same physical component and change is involved.

In SECY-99-130 [3] and SECY-99-054.[4] the staff indicated that comments had been received on the proposed rule that requested that authority similar to 10 CFR 72.48 be created in Part 71, particularly with respect to dual purpose casks. Staff indicated that this issue would be addressed in the subsequent rulemaking to conform Part 71 with IAEA standard ST-1. The Commission adopted the staff’s recommendations in a Staff Requirements Memorandum (SRM) dated June 22, 1999.

In SECY-99-054 staff recommended that a similar authority to 10 CFR 72.48 be created for spent fuel transportation packages intended for domestic use only. Staff also recommended that this authority be limited to Part 50 and 72 licensees shipping spent fuel and the Part 71 certificate holder. Furthermore, other supporting changes to Part 71 would be required to ensure consistency with the process contained in 10 CFR 72.48. These changes would include using common terminology such as “changes to the cask design, as described in the final safety analysis report” (FSAR) and a process for requesting amendments to a CoC. Requirements for periodically updating a transportation package FSAR would also be required to ensure an accurate “licensing” basis is available for evaluating future proposed changes, and requirements for package users to have a copy of the FSAR, and the updated FSAR.

The current IAEA standard ST-1 does not contain any equivalent provisions for changing a transportation package’s design, without prior review by the competent authority.

**Factors for Consideration**

- Should this change authority apply to spent fuel packages involved in domestic commerce only?
- Should this change authority be expanded to include all types of transportation packages, licensees, or users?
- Should the change authority apply to all domestic transportation packages?
- Should the change authority apply to dual purpose spent fuel packages?

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[3] SECY-99-130, “Final Rule — Revisions to Requirements of 10 CFR Parts 50 and 72 Concerning Changes, Tests, and Experiments, “dated May 12, 1999.

[4] SECY-99-054, “Plans for Final Rule — Revisions to Requirements of 10 CFR Parts 50, 52, and 72 Concerning Changes, Tests, and Experiments,” dated February 22, 1999.

## Issue 16. Fissile Material Exemptions and General License Provisions

### Discussion

The NRC published an emergency final rule on February 10, 1997 (62 FR 5907), amending Part 71 regulations that deal with shipments of exempt quantities of fissile material and shipments of fissile material under a general license. An NRC licensee had identified that a shipment of waste material (beryllium oxide containing a low concentration of high-enriched uranium) that met the fissile exemption provisions of 10 CFR 71.53 had the potential for an accidental criticality in certain specific circumstances. Packages shipped under the provisions of 10 CFR 71.53 were considered inherently safe for criticality-safety purposes. These regulations assumed that only ordinary water ($H_2O$) could be present as a moderating material. The regulations did not contemplate the presence of special moderating materials (e.g., beryllium, graphite, or deuterium). Because of this criticality safety issue, the NRC published a rule that was immediately effective with no opportunity for pre-promulgation public comment. The NRC did solicit comments after the rule was effective. All public comments supported the need for the emergency final rule when the shipments contained special moderators (moderators other than water); however, the commenters stated that the rule had gone too far for water moderated shipments, that it was excessively restrictive and costly to licensees, and that further rulemaking was necessary.

Based on these comments, NRC staff contracted with Oak Ridge National Laboratory (ORNL) to thoroughly review fissile material exemptions and general license provisions. ORNL performed computer model calculations of $k_{eff}$ (k-effective) for various combinations of fissile material and moderating material - including beryllium, carbon, deuterium, silicon-dioxide, and water - to verify the accuracy of minimum critical mass values. These minimum critical mass values were then applied to the regulatory structure contained in Part 71, and revised mass limits for both the general license and exemption provisions to Part 71 were determined. Also, ORNL researched the historical bases for the fissile material exemption and general license regulations in Part 71 and discussed the impact of the emergency final rule's restrictions on NRC licensees. The ORNL study was issued as NUREG/CR-5342 in July 1998 (available via the following NRC website: http://www.nrc.gov/NRC/NUREGS/CR5342/index.html). The ORNL study confirmed that the emergency rule was needed to provide safe transportation of packages with special moderators that are shipped under the general license and fissile material exemptions, but may be excessive for water-moderated shipments.

NUREG/CR-5342 identified 16 recommended actions for additional rulemaking. Additionally, the Commission’s SRM on SECY-96-268 approving the emergency final rule directed the staff to issue guidance for instances where fissile materials may be mixed in the same shipping container with different moderators. The staff indicated that this issue would be addressed in a forthcoming rulemaking (memorandum from the EDO to the Commission, dated September 8, 1998). On October 27, 1999, the NRC published Federal Register Notice 64 FR 57769 responding to public comments on the emergency final rule, and also requesting information on the cost impact of the final rule from the public, industry, and the DOE, because the NRC staff had not been successful in obtaining this information. The requirements for the fissile material general licenses are provided in 10 CFR 71.18, 71.20, 71.22, and 71.24, and the fissile material exemptions are provided in 71.53.

IAEA standard ST-1 contains language on fissile exemptions and restrictions on the use of special moderators. However, ST-1 does not presently contain provisions on general licenses for shipment of fissile material; previous version did contain general license conditions.

**Factors for Consideration**

- Should all, or only some, of the 16 sub-issues (i.e., the recommendations contained in NUREG/CR-5342) be included in this rulemaking on this issue?
- Should additional issues or alternative approaches on the fissile exemptions or general license provisions be included in this rulemaking?
- Is there available cost data that may help to understand the cost impact of the implemented emergency rule; or help to better understand the possible cost impact of the ORNL recommendations?

## Issue 17. Double Containment of Plutonium (PRM-71-12)

**Description**

The NRC received a Petition for Rulemaking from International Energy Consultants, Inc. (IEC), dated September 25, 1997. The petition was docketed as PRM-71-12 and was published for public comment on February 19, 1998. The comment period was extended to July 31, 1998. The petitioner requested that regulations in 10 CFR 71.63 be eliminated. The petitioner argued that the double containment requirement in 71.63(b) was not consistent with the basis for other packaging standards (i.e., the Q-value system for identifying the $A_1$ and $A_2$ values for each nuclide). The petitioner also argued that the use of double containment for shipments of plutonium imposed unnecessary costs (i.e., fabrication of shipping packages and a weight penalty). As an option, the petitioner requested that 71.63 be entirely eliminated.

In 1974, the Atomic Energy Commission (AEC) issued 10 CFR 71.63 which imposed special requirements on the shipment of plutonium in excess of 0.74 terabecquerels (20 curies). These requirements specify that plutonium must be in solid form [71.63(a)] and that packages used to ship plutonium must provide a separate inner containment (i.e., the "double containment" requirement) [71.63(b)]. In adopting these requirements, the AEC specifically excluded plutonium in the form of reactor fuel elements, metal or metal alloys, and other plutonium-bearing solids that the Commission determines, on a case-by-case basis, do not require double containment. These regulations have remained essentially unchanged since 1974, except for the addition in 1998 of vitrified high-level waste in sealed canisters to the list of exempt forms of plutonium. Double containment is in addition to Type B packaging standards and is not required for any other nuclides that are listed in Part 71. Additionally, IAEA standard ST-1 does not contain a double containment requirement for any nuclide.

The AEC issued this regulation at a time when wide-spread reprocessing of commercial spent fuel was anticipated. The AEC expected increases in the quantities of plutonium to be shipped and the number of shipments of plutonium. In addition, the specific activity of the plutonium was expected to increase with increased burnup, resulting in higher gamma and neutron radiation levels, greater heat generation, and greater pressure generation potential from plutonium nitrate solutions in shipping containers. Because of these expected changes and because of the susceptibility of liquids to leakage, the AEC believed that safety would be significantly enhanced if the basic form for shipments of plutonium were changed from liquid to solid, and if the solid form of plutonium were required to be shipped in a package providing double containment of the contents.

The AEC indicated that “The arguments for requiring a solid form of plutonium for shipment are largely subjective, in that there is no hard evidence on which to base statistical probabilities or to assess quantitatively the incremental increase in safety which is expected.” [5] The AEC also indicated that the double containment provision compensates for the fact that the plutonium may not be in a "nonrespirable" form. Notwithstanding these rationales, some of the underlying assumptions for this rule were altered in 1979 when the U.S. government decided that reprocessing of civilian spent fuel and reuse of plutonium was not desirable. Consequently, the expected plutonium reprocessing economy and wide-spread shipments never materialized.

With respect to PRM-71-12, eight public comments were received on the petition; of those, three supported the petition and five opposed the petition. The supporting comments essentially stated that the IAEA’s Q-System accurately reflects the dangers of nuclides, including plutonium, and that elimination of 10 CFR 71.63(a) and (b) would make the regulations more performance based, reduce costs and personnel exposures, and be consistent with the IAEA standards.

The five opposing comments essentially stated that plutonium is very dangerous, especially in liquid form, and therefore additional regulatory requirements are warranted, that existing regulations are not overly burdensome, especially in light of the total expected transportation cost, that TRUPACT-II package meets 71.63(b) requirement, that a commenter (i.e., the Western Governors Association) has worked for over 10 years to ensure a safe transportation system for WIPP, including educating the public about the TRUPACT-II package, and that any change now would erode public confidence and be detrimental to the entire transportation system for WIPP shipments, and that additional personnel exposure due to double containment is insignificant.

**Factors for Consideration**

- Should NRC change any of the special requirements for the transportation of plutonium?
- Should the double containment requirement in 71.63(b) be eliminated?
- Should both the solid form and the double containment requirements of 71.63(a) and (b) be eliminated?
- Is consistency with IAEA standard ST-1 important on this issue?

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[5] SECY-R-74-5, dated July 6, 1973

## Appendix A
## Paragraphs Referenced from IAEA ST-1

Appendix A contains the full text of specific paragraphs from ST-1 referenced in the eleven IAEA-compatibility issues. Paragraphs are listed numerically in ascending order, with the corresponding issue identified in bold text at the end of the reference.

**107(e).** The Regulations do not apply to:

(e) natural material and ores containing naturally occurring radionuclides which are not intended to be processed for use of these radionuclides provided the activity concentration of the material does not exceed 10 times the values specified in paras 401-406. **(Issue 2)**

**209.** *Confinement system* shall mean the assembly of *fissile material* and *packaging* components specified by the designer and agreed to by the *competent authority* as intended to preserve criticality safety. **(Issue 9)**

**218.** *Criticality safety index (CSI)* assigned to a *package*, *overpack* or *freight container* containing *fissile material* shall mean a number which is used to provide control over the accumulation of *packages*, *overpacks* or *freight containers* containing *fissile material*. **(Issue 9)**

**225.** *Low dispersible radioactive material* shall mean either a solid *radioactive material* or a solid *radioactive material* in a sealed capsule, that has limited dispersibility and is not in powder form. **(Issue 9)**

**230.** *Package* shall mean the *packaging* with its *radioactive contents* as presented for transport. The types of *packages* covered by these Regulations, which are subject to the activity limits and material restrictions of Section IV and meet the corresponding requirements, are:

(a) *Excepted package*;
(b) *Industrial package Type 1 (Type IP-1)*;
(c) *Industrial package Type 2 (Type IP-2)*;
(d) *Industrial package Type 3 (Type IP-3)*;
(e) *Type A package*;
(f) *Type B(U) package*;
(g) *Type B(M) package*;
(h) *Type C package*.

*Packages* containing *fissile material* or uranium hexafluoride are subject to additional requirements. **(Issue 6)**

**232.** *Quality assurance* shall mean a systematic programme of controls and inspections applied by any organization or body involved in the transport of *radioactive material* which is aimed at providing adequate confidence that the standard of safety prescribed in these Regulations is achieved in practice. **(Issue 9)**

**401.** The following basic values for individual radionuclides are given in Table I:

(a) $A_1$ and $A_2$ in TBq;
(b) activity concentration for exempt material in Bq/g; and
(c) activity limits for exempt consignments in Bq. (Issue 2)

**402.** For individual radionuclides which are not listed in Table I the determination of the basic radionuclide values referred to in para. 401 shall require *competent authority* approval or, for international transport, *multilateral approval*. Where the chemical form of each radionuclide is known, it is permissible to use the $A_2$ value related to its solubility class as recommended by the International Commission on Radiological Protection, if the chemical forms under both normal and accident conditions of transport are taken into consideration. Alternatively, the radionuclide values in Table II may be used without obtaining *competent authority* approval. **(Issue 2)**

**403.** In the calculations of $A_1$ and $A_2$ for a radionuclide not in Table I, a single radioactive decay chain in which the radionuclides are present in their naturally occurring proportions, and in which no daughter nuclide has a half-life either longer than 10 days or longer than that of the parent nuclide, shall be considered as a single radionuclide; and the activity to be taken into account and the $A_1$ or $A_2$ value to be applied shall be those corresponding to the parent nuclide of that chain. In the case of radioactive decay chains in which any daughter nuclide has a half-life either longer than 10 days or greater than that of the parent nuclide, the parent and such daughter nuclides shall be considered as mixtures of different nuclides. **(Issue 2)**

**404.** For mixtures of radionuclides, the determination of the basic radionuclide values referred to in para. 401 may be determined as follows:

$$X_m = \frac{1}{\sum_i \frac{f(i)}{X(i)}}$$

where,

f(i) is the fraction of activity or activity concentration of radionuclide i in the mixture;
X(i) is the appropriate value of $A_1$ or $A_2$, or the activity concentration for exempt material or the activity limit for an exempt consignment as appropriate for the radionuclide i; and
$X_m$ is the derived value of $A_1$ or $A_2$, or the activity concentration for exempt material or the activity limit for an exempt consignment in the case of a mixture. **(Issue 2)**

**405.** When the identity of each radionuclide is known but the individual activities of some of the radionuclides are not known, the radionuclides may be grouped and the lowest radionuclide value, as appropriate, for the radionuclides in each group may be used in applying the formulas in paras 404 and 414. Groups may be based on the total alpha activity and the total beta/gamma activity when these are known, using the lowest radionuclide values for the alpha emitters or beta/gamma emitters, respectively. **(Issue 2)**

**406.** For individual radionuclides or for mixtures of radionuclides for which relevant data are not available, the values shown in Table II shall be used. **(Issue 2)**

**543.** Each label conforming to the models in Fig. 2, Fig. 3 and Fig. 4 shall be completed with the following information:

(a) Contents:
   (i) Except for *LSA-I material*, the name(s) of the radionuclide(s) as taken from Table I, using the symbols prescribed therein. For mixtures of radionuclides, the most restrictive nuclides must be listed to the extent the space on the line permits. The group of *LSA* or *SCO* shall be shown following the name(s) of the radionuclide(s). The terms "LSA-II","LSA-III", "SCO-I" and "SCO-II" shall be used for this purpose.
   (ii) For *LSA-I material*, the term "LSA-I" is all that is necessary; the name of the radionuclide is not necessary.

(b) Activity: The maximum activity of the *radioactive contents* during transport expressed in units of becquerels (Bq) with the appropriate SI prefix (see Annex II). For *fissile material*, the mass of *fissile material* in units of grams (g), or multiples thereof, may be used in place of activity.

(c) For *overpacks* and *freight containers* the "contents" and "activity" entries on the label shall bear the information required in subparas 543(a) and 543(b), respectively, totalled together for the entire contents of the *overpack* or *freight container* except that on labels for *overpacks* or *freight containers* containing mixed loads of *packages* containing different radionuclides, such entries may read "See Transport Documents".

(d) *Transport index*: See paras 526 and 527. (No *transport index* entry is required for category I-WHITE.) **(Issue 1)**

**544.** Each label conforming to the model in Fig. 5 shall be completed with the *criticality safety index (CSI)* as stated in the certificate of approval for *special arrangement* or the certificate of approval for the *package design* issued by the *competent authority*. **(Issue 5)**

**545.** For *overpacks* and *freight containers*, the *criticality safety index (CSI)* on the label shall bear the information required in para. 544 totalled together for the fissile contents of the *overpack* or *freight container*. **(Issue 5)**

**549.** The *consignor* shall include in the transport documents with each *consignment* the following information, as applicable in the order given:

(a) The proper shipping name, as specified in Table VIII;
(b) The United Nations Class number "7";
(c) The United Nations number assigned to the material as specified in Table VIII, preceded by the letters "UN";
(d) The name or symbol of each radionuclide or, for mixtures of radionuclides, an appropriate general description or a list of the most restrictive nuclides;
(e) A description of the physical and chemical form of the material, or a notation that the material is *special form radioactive material* or *low dispersible radioactive material*. A generic chemical description is acceptable for chemical form;
(f) The maximum activity of the *radioactive contents* during transport expressed in units of becquerels (Bq) with an appropriate SI prefix (see Annex II). For *fissile material*, the mass of *fissile material* in units of grams (g), or appropriate multiples thereof, may be used in place of activity.
(g) The category of the *package*, i.e. I-WHITE, II-YELLOW, III-YELLOW;
(h) The *transport index* (categories II-YELLOW and III-YELLOW only);
(i) For *consignments* including *fissile material* other than *consignments* excepted under para. 672, the *criticality safety index*;

(j) The identification mark for each *competent authority* approval certificate (*special form radioactive material*, *low dispersible radioactive material*, *special arrangement*, *package design*, or *shipment*) applicable to the *consignment*;
(k) For *consignments* of *packages* in an *overpack* or *freight container*, a detailed statement of the contents of each *package* within the *overpack* or *freight container* and, where appropriate, of each *overpack* or *freight container* in the *consignment*. If *packages* are to be removed from the *overpack* or *freight container* at a point of intermediate unloading, appropriate transport documents shall be made available;
(l) Where a *consignment* is required to be shipped under *exclusive use*, the statement "EXCLUSIVE USE SHIPMENT"; and
(m) For *LSA-II*, *LSA-III*, *SCO-I* and *SCO-II*, the total activity of the *consignment* as a multiple of $A_2$. **(Issue 1)**

**629.** Except as allowed in para. 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 ($UF_6$) for transport" [1], and the requirements of paras 630-631. The *package* shall also meet the requirements prescribed elsewhere in these Regulations which pertain to the radioactive and fissile properties of the material. **(Issue 4)**

**630.** Each *package* designed to contain 0.1 kg or more of uranium hexafluoride shall be designed so that it would meet the following requirements:

(a) withstand without leakage and without unacceptable stress, as specified in the International Organization for Standardization document ISO 7195 [10], the structural test as specified in para. 718;
(b) withstand without loss or dispersal of the uranium hexafluoride the test specified in para. 722; and
(c) withstand without rupture of the *containment system* the test specified in para. 728. **(Issue 4)**

**631.** *Packages* designed to contain 0.1 kg or more of uranium hexafluoride shall not be provided with pressure relief devices. **(Issue 4)**

**632.** Subject to the approval of the *competent authority*, *packages* designed to contain 0.1 kg or more of uranium hexafluoride may be transported if:

(a) the *packages* are designed to requirements other than those given in ISO 7195 [10] and paras 630-631 but, notwithstanding, the requirements of paras 630-631 are met as far as practicable. **(Issue 4)**

**657.** A *package* for *radioactive contents* with activity greater than $10^5$ $A_2$ shall be so designed that if it were subjected to the enhanced water immersion test specified in para. 730, there would be no rupture of the *containment system*. **(Issue 7)**

**667.** *Type C packages* shall be designed to meet the requirements specified in paras 606-619, and of paras 634-647, except as specified in para. 646(a), and of the requirements specified in paras 651-654, paras 658-664, and, in addition, of paras 668-670. **(Issue 6)**

**668.** A *package* shall be capable of meeting the assessment criteria prescribed for tests in paras 656(b) and 660 after burial in an environment defined by a thermal conductivity of 0.33 W/m.K and a temperature of 38°C in the steady state. Initial conditions for the assessment shall assume that any thermal insulation of the *package* remains intact, the *package* is at the *maximum normal operating pressure* and the ambient temperature is 38°C. **(Issue 6)**

**669.** A *package* shall be so designed that, if it were at the *maximum normal operating pressure* and subjected to:

(a) the tests specified in paras 719-724, it would restrict the loss of *radioactive contents* to not more than $10^{-6}$ $A_2$ per hour; and
(b) the test sequences in para. 734, it would meet the following requirements:
  (i) retain sufficient shielding to ensure that the *radiation level* at 1 m from the surface of the *package* would not exceed 10 mSv/h with the maximum *radioactive contents* which the *package* is designed to contain; and
  (ii) restrict the accumulated loss of *radioactive contents* in a period of 1 week to not more than 10 $A_2$ for krypton-85 and not more than $A_2$ for all other radionuclides.

Where mixtures of different radionuclides are present, the provisions of paras 404-406 shall apply except that for krypton-85 an effective $A_2$(i) value equal to 10 $A_2$ may be used. For case (a) above, the assessment shall take into account the external *contamination* limits of para. 508. **(Issue 6)**

**670.** A *package* shall be so designed that there will be no rupture of the *containment system* following performance of the enhanced water immersion test specified in para. 730. **(Issue 6)**

**677.** For a *package* in isolation, it shall be assumed that water can leak into or out of all void spaces of the *package*, including those within the *containment system*. However, if the *design* incorporates special features to prevent such leakage of water into or out of certain void spaces, even as a result of error, absence of leakage may be assumed in respect of those void spaces. Special features shall include the following:

(a) Multiple high standard water barriers, each of which would remain watertight if the *package* were subject to the tests prescribed in para. 682(b), a high degree of quality control in the manufacture, maintenance and repair of *packagings* and tests to demonstrate the closure of each *package* before each *shipment*; or
(b) For *packages* containing uranium hexafluoride only:
  (i) *packages* where, following the tests prescribed in para. 682(b), there is no physical contact between the valve and any other component of the *packaging* other than at its original point of attachment and where, in addition, following the test prescribed in para. 728 the valves remain leaktight; and
  (ii) a high degree of quality control in the manufacture, maintenance and repair of *packagings* coupled with tests to demonstrate closure of each *package* before each *shipment.* **(Issue 4 and issue 11)**

**680.** For *packages* to be transported by air:

(a) the *package* shall be subcritical under conditions consistent with the tests prescribed in para. 734 assuming reflection by at least 20cm of water but no water inleakage; and

(b) allowance shall not be made for special features of para. 677 unless, following the tests specified in para. 734 and, subsequently, para. 733, leakage of water into or out of the void spaces is prevented. **(Issue 11)**

**682.** A number "N" shall be derived, such that two times "N" shall be subcritical for the arrangement and *package* conditions that provide the maximum neutron multiplication consistent with the following:

(a) Hydrogenous moderation between *packages*, and the *package* arrangement reflected on all sides by at least 20 cm of water; and
(b) The tests specified in paras 719-724 followed by whichever of the following is the more limiting:
  (i) the tests specified in para. 727(b) and, either para. 727(c) for *packages* having a mass not greater than 500 kg and an overall density not greater than 1000 kg/$m^3$ based on the external dimensions, or para. 727(a) for all other *packages*; followed by the test specified in para. 728 and completed by the tests specified in paras 731-733; or
  (ii) the test specified in para. 729; and
(c ) Where any part of the *fissile material* escapes from the *containment system* following the tests specified in para. 682(b), it shall be assumed that *fissile material* escapes from each *package* in the array and all of the *fissile material* shall be arranged in the configuration and moderation that results in the maximum neutron multiplication with close reflection by at least 20 cm of water. **(Issue 10)**

**719.** The tests are: the water spray test, the free drop test, the stacking test and the penetration test. Specimens of the *package* shall be subjected to the free drop test, the stacking test and the penetration test, preceded in each case by the water spray test. One specimen may be used for all the tests, provided that the requirements of para. 720 are fulfilled. **(Issue 10)**

**720.** The time interval between the conclusion of the water spray test and the succeeding test shall be such that the water has soaked in to the maximum extent, without appreciable drying of the exterior of the specimen. In the absence of any evidence to the contrary, this interval shall be taken to be two hours if the water spray is applied from four directions simultaneously. No time interval shall elapse, however, if the water spray is applied from each of the four directions consecutively. **(Issue 10)**

**721.** Water spray test: The specimen shall be subjected to a water spray test that simulates exposure to rainfall of approximately 5 cm per hour for at least one hour. (Issue 10)

**722.** Free drop test: The specimen shall drop onto the target so as to suffer maximum damage in respect of the safety features to be tested.

(a) The height of drop measured from the lowest point of the specimen to the upper surface of the target shall be not less than the distance specified in Table XIII for the applicable mass. The target shall be as defined in para. 717.
(b) For rectangular fibreboard or wood *packages* not exceeding a mass of 50 kg, a separate specimen shall be subjected to a free drop onto each corner from a height of 0.3 m.
(c) For cylindrical fibreboard *packages* not exceeding a mass of 100 kg, a separate

specimen shall be subjected to a free drop onto each of the quarters of each rim from a height of 0.3 m. **(Issue 10)**

**723.** Stacking test: Unless the shape of the *packaging* effectively prevents stacking, the specimen shall be subjected, for a period of 24 h, to a compressive load equal to the greater of the following:

(a) The equivalent of 5 times the mass of the actual *package*; and
(b) The equivalent of 13 kPa multiplied by the vertically projected area of the *package*.

The load shall be applied uniformly to two opposite sides of the specimen, one of which shall be the base on which the *package* would typically rest. **(Issue 10)**

**724.** Penetration test: The specimen shall be placed on a rigid, flat, horizontal surface which will not move significantly while the test is being carried out.

(a) A bar of 3.2 cm in diameter with a hemispherical end and a mass of 6 kg shall be dropped and directed to fall, with its longitudinal axis vertical, onto the centre of the weakest part of the specimen, so that, if it penetrates sufficiently far, it will hit the *containment system*. The bar shall not be significantly deformed by the test performance.
(b) The height of drop of the bar measured from its lower end to the intended point of impact on the upper surface of the specimen shall be 1 m. **(Issue 10)**

**727.** Mechanical test: The mechanical test consists of three different drop tests. Each specimen shall be subjected to the applicable drops as specified in para. 656 or para. 682. The order in which the specimen is subjected to the drops shall be such that, on completion of the mechanical test, the specimen shall have suffered such damage as will lead to the maximum damage in the thermal test which follows.

(a) For drop I, the specimen shall drop onto the target so as to suffer the maximum damage, and the height of the drop measured from the lowest point of the specimen to the upper surface of the target shall be 9 m. The target shall be as defined in para. 717.
(b) For drop II, the specimen shall drop so as to suffer the maximum damage onto a bar rigidly mounted perpendicularly on the target. The height of the drop measured from the intended point of impact of the specimen to the upper surface of the bar shall be 1 m. The bar shall be of solid mild steel of circular section, (15.0 ± 0.5) cm in diameter and 20 cm long unless a longer bar would cause greater damage, in which case a bar of sufficient length to cause maximum damage shall be used. The upper end of the bar shall be flat and horizontal with its edges rounded off to a radius of not more than 6 mm. The target on which the bar is mounted shall be as described in para. 717.
(c) For drop III, the specimen shall be subjected to a dynamic crush test by positioning the specimen on the target so as to suffer maximum damage by the drop of a 500 kg mass from 9 m onto the specimen. The mass shall consist of a solid mild steel plate 1 m by 1 m and shall fall in a horizontal attitude. The height of the drop shall be measured from the underside of the plate to the highest point of the specimen. The target on which the specimen rests shall be as defined in para. 717. **(Issue 10)**

**729.** Water immersion test: The specimen shall be immersed under a head of water of at least

15 m for a period of not less than eight hours in the attitude which will lead to maximum damage. For demonstration purposes, an external gauge pressure of at least 150 kPa shall be considered to meet these conditions. **(Issue 10)**

**730.** Enhanced water immersion test: The specimen shall be immersed under a head of water of at least 200 m for a period of not less than one hour. For demonstration purposes, an external gauge pressure of at least 2 MPa shall be considered to meet these conditions. **(Issue 7)**

**734.** Specimens shall be subjected to the effects of each of the following test sequences in the orders specified:

(a) the tests specified in paras 727(a), 727(c), 735 and 736; and
(b) the test specified in para. 737.

Separate specimens are allowed to be used for each of the sequences (a) and (b). **(Issue 6)**

**735.** Puncture/tearing test: The specimen shall be subjected to the damaging effects of a solid probe made of mild steel. The orientation of the probe to the surface of the specimen shall be as to cause maximum damage at the conclusion of the test sequence specified in para. 734(a).

(a) The specimen, representing a *package* having a mass less than 250 kg, shall be placed on a target and subjected to a probe having a mass of 250 kg falling from a height of 3 m above the intended impact point. For this test the probe shall be a 20 cm diameter cylindrical bar with the striking end forming a frustum of a right circular cone with the following dimensions: 30 cm height and 2.5 cm in diameter at the top. The target on which the specimen is placed shall be as specified in para. 717.
(b) For *packages* having a mass of 250 kg or more, the base of the probe shall be placed on a target and the specimen dropped onto the probe. The height of the drop , measured from the point of impact with the specimen to the upper surface of the probe shall be 3 m. For this test the probe shall have the same properties and dimensions as specified in (a) above, except that the length and mass of the probe shall be such as to incur maximum damage to the specimen. The target on which the base of the probe is placed shall be as specified in para. 717. **(Issue 6)**

**736.** Enhanced thermal test: The conditions for this test shall be as specified in para. 728, except that the exposure to the thermal environment shall be for a period of 60 minutes. **(Issue 6)**

**737.** Impact test: The specimen shall be subject to an impact on a target at a velocity of not less than 90 m/s, at such an orientation as to suffer maximum damage. The target shall be as defined in para. 717. **(Issue 6)**

Dated at Rockville, Maryland, this ----- day of ------, 2000.

For the Nuclear Regulatory Commission.

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William F. Kane, Director
Office of Nuclear Material Safety
and Safeguards.