NUCLEAR REGULATORY COMMISSION

10 CFR Part 71

RIN 3150–AC41

Compatibility With the International Atomic Energy Agency (IAEA)

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is revising the regulations governing the transportation of radioactive material. The final rule conforms NRC regulations with those of the International Atomic Energy Agency, and codifies criteria for packages used to transport plutonium by air. This action is necessary to ensure that NRC regulations reflect accepted international standards and comply with current legislative requirements.

EFFECTIVE DATE: April 1, 1996. Section 71.52 expires April 1, 1999.

ADDRESSES: Single copies of the regulatory analysis for this rule may be obtained on request from the contact. Copies of the regulatory analysis may be examined and copied, for a fee, in the Commission's Public Document Room, at 2120 L Street (Lower Level), NW., Washington, DC.

FOR FURTHER INFORMATION CONTACT: John R. Cook, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone: (301) 415–8521.

SUPPLEMENTARY INFORMATION:

Background

The U.S. Nuclear Regulatory Commission is revising its regulations, for the safe transportation of radioactive material to make them compatible with those of the International Atomic Energy Agency (IAEA) and to incorporate new criteria for packages used to transport plutonium by air. The revised rule, in combination with a corresponding amendment of Title 49, Code of Federal Regulations, by the U.S. Department of Transportation (DOT), would bring U.S. regulations into general accord with IAEA regulations (Regulations for the Safe Transport of Radioactive Material, 1985 Edition, Safety Series No. 6). The final rule also adopts approval criteria for packages used to transport plutonium by air. These criteria were developed in response to Public Law 94. Except for these revisions, NRC's basic standards for packaging and transportation remain essentially unchanged. These regulations apply to all NRC licensees who transport, or offer for transport, byproduct, source, or special nuclear material, and will help ensure the continued safe transportation of radioactive materials in domestic and international commerce.

In addition, three Petitions for Rulemaking, concerning the transportation of Low Specific Activity (LSA) radioactive material, are denied in this action.

In 1969, the IAEA, recognizing that its international transport regulations should be revised from time to time on the basis of scientific and technical advances, as well as accumulated experience, invited member states to submit comments and suggested changes to the regulations. As a result of this initiative, the IAEA issued revised regulations in 1973 (Regulations for the Safe Transport of Radioactive Material, 1973 Edition, Safety Series No. 6). The IAEA also decided to periodically review its transportation regulations, at intervals of about 10 years, to ensure that the regulations are kept current. As a result, a review of IAEA regulations was initiated, in 1979, that resulted in the publication of revised regulations in 1985 (Regulations for the Safe Transport of Radioactive Material, 1985 Edition, Safety Series No. 6).

On August 5, 1983 (48 FR 35600) 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 DOT, brought U.S. domestic transport regulations at the Federal level into general accord with the 1973 edition of IAEA transport regulations. Some of the revisions that were eventually included in the 1985 IAEA regulations were anticipated by NRC and DOT when they were finalizing their transportation regulations in 1983. These changes were incorporated in Titles 10 and 49 of the Code of Federal Regulations at that time.

On June 8, 1988 (53 FR 21550) NRC published a proposed revision to its regulations in 10 CFR Part 71 in the Federal Register for the purpose of making U.S. transportation regulations compatible with the 1985 edition of the IAEA regulations. In a parallel rulemaking, DOT published a proposed revision to its radioactive material transportation regulations on November 14, 1989 (54 FR 47454). Several corrections to the NRC proposed rule were published in the Federal Register on June 22, 1988 (53 FR 23484). Interested persons were invited to submit written comments and suggestions on the NRC proposal and/or the supporting regulatory analysis by October 6, 1988. The public comment period was subsequently extended to February 9, 1990. On December 8, 1994, the NRC staff provided a briefing on the proposed LSA requirements and the other revisions at the 416th meeting of the Advisory Committee on Reactor Safeguards (ACRS). This meeting also provided industry and the public another opportunity to present their views on the revisions. Based on the public comments, consultations with DOT, and other considerations, the Commission is adopting the proposed rule, with some modifications.

Discussion of Major Changes From Current Requirements

Most of the revisions presented in the proposed rule are being adopted in the final rule. These include additional hypothetical accident test criteria for certain types of packages, an increase in the number of radionuclides with listed A_1 and A_2 values, changes in the currently listed A_1 and A_2 values for some radionuclides, simplification of fissile material transport classes, revised requirements for shipment of LSA materials, and inclusion of criteria for packages used to transport plutonium by air. These changes are discussed in more detail in the following paragraphs.

Additional Accident Test Requirements

IAEA deep-water immersion and dynamic crush tests are adopted in the final rule. The 200 meter (656 ft) deep-water immersion test has been added to the requirements for Type B packages (casks) authorized for irradiated fuel content in excess of 37 PBq (10^6 Ci)(§ 71.61 Special requirement for irradiated nuclear fuel shipments). The purpose of the deep immersion test, which can be satisfied through engineering evaluation or actual physical test (§ 71.41), is to ensure that the cask containment system does not collapse, buckle, nor allow inleakage of water, if submerged at 200 m (656 ft).

A dynamic crush test (§ 71.73(c)(2) *Crush*) has also been added to Type B package requirements, for certain lightweight packages that are minimally vulnerable to damage in the 9 m (30 ft) drop test, but which have a high potential for radiation hazard, if package failure occurs. IAEA regulations require the crush test in place of the 9 m (30 ft) drop test, for these packages. NRC is requiring both the crush test and drop test, for lightweight packages, to ensure that package response to both crush and drop forces is within applicable limits.

These requirements only apply to package designs certified after this final rule becomes effective. Further, this rule does not apply to packages fabricated under previous versions of Part 71; however, previously fabricated packages are subject to multilateral approval, when used for international transport (§ 71.13(b)).

Expansion of Radionuclide List and Changes in Radionuclide Limits

Table A–1, in 10 CFR Part 71, Appendix A, lists the Type A package quantity limits (A_1 and A_2 values) for many radionuclides. The final rule increases the number of radionuclides listed, from 284 to 378. The final rule also adopts the revised A_1 and A_2 values contained in the 1985 edition of the IAEA regulations. As a result, 144 A_1 values previously listed in Table A–1 are being increased, and 73 are being decreased, while 129 A_2 values are being increased, and 95 decreased. In addition, the final rule modifies the method used to determine A_1 and A_2 values for unlisted radionuclides.

Simplification of Fissile Material Classes

The final rule revises the criteria for shipment of fissile material. Specifically, the rule eliminates the three fissile class designations currently used establishes a single set of criteria for all packages of fissile material, uses the transport index as the primary control for the number of fissile packages that may be transported together, and requires special arrangements for fissile packages that do not meet the established criteria.

Inclusion of Criteria for Air Shipment of Plutonium

The final rule amends Part 71 to include approval criteria for packages used to transport plutonium by air (§§ 71.64, 71.74, and 71.88). These criteria were developed as a result of Pub. L. 94–79, which prohibited NRC from licensing the air shipment of plutonium, in any form, until NRC certified to the Congress that a safe container had been developed. The NRC subsequently developed and certified package criteria to Congress and published the criteria in NUREG–0360, Qualification Criteria to Certify a Package for Air Transport of Plutonium, dated January 1978. This final rule incorporates these criteria. There are no corresponding criteria in IAEA regulations.

Modifications From Proposed Rule

The final rule differs from the proposed rule in several significant respects and are described as follows:

1. Package limit for Shipment of LSA and Surface-Containment-Object (SCO) Material. In its 1985 regulations, the IAEA added a limit of 10 mSv/hour (1 rem/hour) at 3 meters for the radiation level from the unshielded contents of LSA and SCO (Surface Contaminated Object) packages not designed to withstand accidents. This radiation level limit controls the external radiation exposures to individuals if an LSA package is severely damaged in a transportation accident.

The IAEA limit considers the loss of package shielding during an accident but it does not consider the possibility that a package's contents might be released and redistributed, causing a reduction in self-shielding of the contents. The reduction in self-shielding could result in potential accident radiation levels that significantly exceed IAEA's 10 mSv/hour (1 rem /hour) at 3 meters limit.

The IAEA dose rate limit provides a significant added degree of protection over the 1973 IAEA regulations (which specify no quantity limit for LSA packages). NRC and DOT did not believe, however that the IAEA limit provided the same level of safety for all types of LSA material, particularly for relatively large quantities of radioactive materials contained in dispersible LSA materials (e.g., resins and other media used in liquid radioactive waste treatment).

In lieu of the radiation level limit, DOT and NRC proposed a $2A_1$ quantity limit for all LSA packages. Although this proposal addressed the accident concern by directly limiting package quantity, it was not compatible with the IAEA provisions. Both agencies received many comments from industry on the proposed $2A_1$ quantity limit that objected to the impacts on occupational dose and shipping costs. Further, after a briefing on the draft final rule on December 8, 1994, the Advisory Committee on Reactor Safeguards (ACRS) issued a letter report, dated December 19, 1994, recommending, inter alia, that the requirements again be reevaluated with the objective of making them equivalent to the IAEA regulations.

After consideration of comments from ACRS and industry, DOT and NRC have agreed to adopt the IAEA LSA provisions. Accordingly, the final rule imposes a limit on the external radiation level at 3 meters from the unshielded contents of LSA–II, LSA–III, or SCO–II packages of 10 mSv/hour (1 rem/hour) (§ 71.10(b)).

2. The final rule delays imposing the LSA package external radiation level limit for 3 years. The effect of imposing the LSA package limit is to reduce the quantity of LSA materials that can be transported in non-Type B, LSA packages. The final rule may increase demand for Type B packages, and there are very few currently available. NRC had proposed a 1 year delay in implementing the new LSA rules. Industry comments expressed the view that 1 year is not an adequate period of time to design a package, have it approved by NRC, and manufacture a reasonable number of Type B waste packages. NRC agrees, and has included a delay of 3 years from the effective date of this rule for implementation of this provision of the final rule (§ 71.52).

3. The proposed rule would have adopted $2A_1$ as the threshold below which licensees are exempt from NRC requirements for packages containing LSA material (except for §§ 71.5, 71.88 and 71.53). Because NRC and DOT are adopting the IAEA LSA package limit, the final rule changes the exemption threshold to 1 rem/h at 3 m (§ 71.10(b)(2)). Thus, designs for packages used to ship LSA or SCO in quantities where the external dose rate exceeds 1 rem/h at 3 m from the unshielded material will be subject to NRC Type B package regulations. Package designs for lesser quantities of LSA or SCO will be self-certified, by package designers, as meeting applicable DOT IP–1, IP–2, IP–3, Type A, or strong tight, package regulations. [Licensees should note that DOT has prescribed, in its final rule, the use of IAEA Industrial Packages (IP–1, IP–2, and IP–3) for LSA and SCO material. For domestic transportation only, DOT also provides for the use of Type A, and strong tight, containers.]

4. For compatibility with IAEA and DOT requirements, a new, ''§ 71.77 Qualification of LSA–III Material,'' has been added to Subpart F. This section prescribes assessment of LSA–III material leaching. (In the proposed rule, § 71.77 contained ''Tests for special form radioactive material.'' Those requirements have been moved to § 71.75 ''Qualification of special form material,'' in the final rule.)

Other Administrative Actions

The final rule corrects numerical errors in §§ 71.20(b)(3) and 71.24(b)(4) of the current rule (§§ (71.20(c)(3) and 71.24(c)(4), respectively, of the proposed rule). These errors, which were not identified at the time the proposed rule was published, resulted when the limit for graphite was expressed as an atomic ratio, instead of a mass ratio. The errors were inadvertently adopted, in Part 71, during a rulemaking in 1983, to make

NRC regulations compatible with 1973 IAEA transportation regulations. IAEA has subsequently corrected these errors in the 1985 edition of its transportation regulations.

Section 71.20(b)(3), as currently written, limits the mass of graphite to ''* * * 150 times the total mass of uranium-235 plus plutonium.'' Section 71.20(c)(3), in the final rule, would be amended to read as follows: ''The total mass of graphite present does not exceed 7.7 times the total mass of uranium-235 plus plutonium.'' Section 71.24(c)(4) would be similarly revised to change the limits on graphite from 150 to 7.7 times the total mass of uranium-235 plus plutonium.

NRC is correcting these errors in this final rule. The affected sections may bear on the criticality safety of fissile materials in transport. In addition, these corrections are expected to have minimal impact because there are no shipping casks currently being used that were designed using the erroneous provisions.

Summary and Resolution of Public Comments

There were 171 letters of comment received on the proposed rule from industry, State, and local governments; environmental organizations; medical facilities; and members of the public. A discussion of general comments is presented below, followed by responses to comments on specific sections of the proposed rule.

One of the most frequent comments noted differences among NRC, DOT, and IAEA definitions and requirements where there were no reasons for the differences. Many of the differences between NRC and DOT requirements resulted from the long period of time between publication of the NRC proposed rule (June 8, 1988) and publication of the DOT proposed rule (November 14, 1989; 54 FR 47454). The two proposed rules were intended to be published on or about the same date but circumstances did not permit concurrent publication. Between publication of the NRC and DOT rules, IAEA published a complete set of minor changes and changes of detail to its regulations. These changes were not contained in the NRC proposed rule, but were introduced in the DOT proposed rule. In addition, a large number of printing errors appeared in the text of the NRC proposed rule. Only the most significant errors were rectified in a correction notice published June 22, 1988 (53 FR 23484). The remaining inconsistencies have been corrected in the final rule.

Another frequently raised comment was in response to NRC's inclusion of new criteria for the air transportation of plutonium. Out of 171 total letters of comment on the proposed rule, 119 of those letters were concerned with the single issue of air transportation of plutonium. In general, these letters requested that NRC codify the NUREG–0360 criteria for the safe air transportation of plutonium, notwithstanding urging by the U.S. Department of Energy (DOE) that NRC withhold codification until it could consider rules being developed by IAEA for the safe air transportation of plutonium. Many of these letters, primarily from residents of Alaska, attributed development of the NUREG–0360 [1] criteria to U.S. Senator Frank Murkowski. However, the criteria in NUREG–0360 were developed by the NRC in response to Public Law 94–79, enacted in 1975. (Senator Murkowski sponsored much more recent legislation on transportation of plutonium by air, identified as Section 5062 of Public Law 100–203, for which regulatory criteria have not been developed.) NRC has relied on the NUREG–0360 criteria for plutonium transportation by air since the criteria were published in 1978. DOE's request that NRC withhold the codification of the NUREG–0360 criteria while NRC considers the IAEA alternative cannot be accommodated because there is no existing IAEA alternative to consider and none is expected for several years. Although the IAEA development process has begun, the process is long and multifaceted. Predictions as to final content of an IAEA alternative cannot be made at this time. It also should be noted that, under Public Law 94–79, the proposed criteria would apply to any U.S. import, export, or domestic plutonium air transport regardless of IAEA regulations. Accordingly, the plutonium air transport criteria are incorporated in the final rule.

Section 71.0 Purpose and Scope

One comment suggested that § 71.0 (a) could be clarified by referring to the need for a Type B package rather than to licensed material in excess of a Type A quantity. Section 71.0 (a)(2) would then read ''Procedures and standards for NRC approval of packaging and shipping procedures for fissile material and for other licensed material required by this Part to be transported in a Type B packaging.''

Although the suggested wording may be a good description of Part 71, Fissile Type A packages are still subject to NRC approval. Therefore a scope based on quantity of radioactive material is better than a scope based on a single type of package.

Section 71.4 Definitions

One comment noted that the term ''licensed material'' is used in Part 71, in several locations, but is not defined in Part 71. In response to this comment, NRC has added the definition of ''licensed material,'' as codified in 10 CFR Part 39, to the definitions in Part 71. The term ''licensed material'' only includes radioactive material licensed by the NRC. One comment noted that in defining the term ''exclusive use,'' the parenthetical note ''* * * also referred to in other regulations as 'sole use' or 'full load' '' is no longer necessary. Those other terms have been almost completely phased out, and IAEA has eliminated the clarifying note. NRC agrees and also has eliminated the clarifying note.

One comment noted that the definition of ''exclusive use'' requires that loading and unloading be performed by personnel having radiological training and resources appropriate for safe handling of the consignment. However, the definition provides no criteria to indicate what that training should be. NRC believes this is an area where the regulation includes a sufficient level of detail to define the intent of the provision. NRC further notes that DOT has established requirements for hazardous material employee training (see 49 CFR Part 172, Subpart H, §§ 172.700–172.704, effective July 2, 1992).

One comment suggested that the term ''transport index'' specify that the number be rounded up ''to the next tenth'' rather than ''to the first decimal place.'' NRC believes that either terminology is adequately clear, and is retaining the original wording for uniformity. This wording has been used satisfactorily over a number of years.

One comment suggested that the ''Natural uranium'' definition should be clarified to indicate that the phrase ''the remainder being uranium-238'' refers strictly to a weight basis, not to a radioactivity basis. NRC has made the clarification.

One comment raised the question whether ''licensee'' and ''licensee of the Commission'' are synonymous, and whether the terms include ''persons

[1] Copies of NUREG–0360 may be purchased from the Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013–7082. Copies are also available from the National Technical Information Service, 5285 Port Royal Road, Springfield, VA 22161. A copy is also available for inspection and copying for a fee in the NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC.

licensed by an Agreement State,'' so that the general licenses of §§ 71.12–71.24 could apply. NRC asserts that the terms ''licensee'' and ''licensee of the Commission'' are synonymous. For uniformity, the NRC has eliminated the longer of the two terms in the final rule. Neither term includes Agreement State licensees. However, Agreement State licensees engaging in activities in non-Agreement States, or in offshore waters, under the reciprocity provisions of 10 CFR Part 150, ''Exemptions and Continued Regulatory Authority in Agreement States and in Offshore Waters under Section 274,'' are subject to the requirements of 10 CFR Part 71. In such instances, the NRC general licenses mentioned above apply to Agreement State licensees.

One comment noted that the term ''specific activity'' should only be used when describing the radioactivity of a radionuclide per unit mass of the element. When describing the radioactivity per unit mass of a material in general, the comment suggested the use of the words ''concentration of radioactivity.'' NRC has been unable to confirm any preferred limited use of the term ''specific activity,'' and, in view of the years of successful international use of the term in its broader sense, plans to continue that broader use.

One comment noted that the NRC and DOT definitions of ''exclusive use'' are not identical, and that the DOT definition appears preferable. In the final rules promulgated by NRC and DOT, the definitions of ''exclusive use'' are identical.

One comment noted a difference in quantities, for DOT's proposed rule ''highway route controlled quantities,'' in 49 CFR 173.403, and for NRC's ''advanced notification of shipment of nuclear waste'' requirements in 10 CFR 71.97. The limits were intended to be the same. As the comment suggested, the error (by NRC) was caused by the rounding of the International System (of units) (SI) and customary units and has been corrected in this final rule.

Section 71.4 Definitions (Dual Unit System—The International System of Units Followed or Preceded by U.S. Standard or Customary Units).

Ten comments suggested both support for the dual unit system used in both NRC and DOT proposed regulations and potential problems that might result from a dual unit system. Several other comments suggested that NRC and DOT be consistent in the use of units. NRC and DOT intend to use dual units in specifying the regulatory requirements. The introductory language to § 71.4 states that the different units are functionally equivalent and can be used interchangeably for purposes of this part. There are no paperwork requirements in Part 71 (e.g., records, reports) where the mandatory use of units is specified. DOT regulations also specify regulatory requirements in terms of dual units. In 49 CFR 171.10, DOT specifies that the SI units are intended to serve as the standard, but that the customary units (rounded) are included to provide a functionally equivalent limit. The dual unit approaches used by NRC and DOT are compatible.

In addition, DOT specifies, in 49 CFR Part 172, the units that must be used to satisfy the communication standards for shipping papers and package labels. Sections 172.203(d)(4)and 172.403(g)(2) require that shipping papers and package labels be completed either in SI units alone or in SI units and customary units. These requirements also permit, for a period of one year after the effective date of the final rule, the use of customary units on shipping papers and package labels for domestic shipments only.

One comment noted that the double conversion from customary units to SI units, and back to customary units produces specifications that are out of line with standard material sizes. For example, a test with what was a standard 6-inch-diameter mild steel bar, with an edge radius of ¼ inch, was proposed as a test with a 5.91-inch diameter mild steel bar, with an edge radius of 0.236 inch. The converted customary units of length and weight have been returned to their original values in the final rule.

One comment suggested greater consistency of units between the NRC and DOT transportation regulations and the Commission's ''Standards for Protection against Radiation'' in 10 CFR Part 20. Since the NRC and DOT transportation rules were proposed, NRC has revised 10 CFR 20.1004, ''Units of Radiation Dose,'' and 10 CFR 20.1005, ''Units of Radioactivity,'' to permit the use of either customary or SI units, These revisions achieve greater consistency of units among transportation and radiation protection regulations.

One comment noted that differences between IAEA and Part 71 A values (expressed in conventional units) may cause problems in international transport. The curie values in Safety Series #6, Table I are approximate, rounded down from the TBq values after conversion to Ci, whereas the curie values in Table A–1 Part 71 are converted from the TBq values to three significant figures without rounding down. The Part 71 method was used because it yields values that more closely approximate previous Table A–1 values. As noted earlier in this preamble, DOT regulations will require the use of the SI units in shipping papers and labels for international shipments (although conventional units may be used in addition to the SI units). The use of SI units should retain consistency with the IAEA regulations.

One comment suggested that the term ''transport index'' be defined using both customary and SI units, as IAEA has done. The proposed definition was expressed only in customary units. NRC agrees with this suggestion and has adopted the DOT definition of ''transport index'' which includes both customary and SI units.

Section 71.4 Definitions (LSA and SCO in Particular)

Several comments related to clarification of LSA definitions.

Two comments noted the typographical error in the proposed rule in which the ''water with tritium'' concentrations for LSA–II were printed as 27.0 Ci/λ (1 TBq/λ), rather than as 27.0 Ci/l (1 TBq/l). Two other comments noted that the numerical values differed from those in the DOT proposed rule (20 Ci/l and 0.8 TBq/l, respectively). One comment stated a preference for the 27.0 Ci/l limit.

NRC values in the proposed rule were derived from the IAEA and DOT values by rounding up the terabequerel limit and then converting to curies. For consistency, NRC has adopted the IAEA and DOT values in the final rule.

Three comments were concerned with the definition of LSA–I. The first comment noted that material generated from the extraction of uranium or thorium was not classified into any LSA category. The comment recommended an LSA–I classification for this material. Another comment recommended that the term ''contaminated earth'' in LSA–I be expanded to include ''soil, earth, concrete rubble, and other bulk debris.'' A third comment expressed concern that mill tailings exceeding 10^{-6} A_2/g could not be shipped in bulk under the proposed rule. The comment recommended that either mill tailings be specifically included in the definition of LSA–I without an activity or concentration limit, or the specific activity limit for LSA–I be increased to $4x10^{-6}$ A_2/g.

NRC agrees that ore-like materials (materials with highly uniform distribution of small quantities of radionuclides) should be transported as LSA–I material. Accordingly, the definition of LSA–I has been changed from ''contaminated earth * * * '' to ''contaminated earth, mill tailings, concrete rubble and other bulk debris * * *'' Further, NRC believes that mill tailings will meet the proposed 10^{-6} A_2/g specific activity limit, and therefore has not increased the limit.

Two comments suggested that NRC include a definition of the term ''closed transport vehicle'' used in the definition of LSA–I. This term has been removed from the definition of LSA–I because NRC and DOT concluded the use of a vehicle-based term in the definition of a material was inappropriate. ''Closed transport vehicle'' is defined in DOT's rule (49 CFR 173.403(c)).

One comment suggested that LSA–II material definition be expanded to include activated materials, consolidated wastes, and materials intrinsically contained in a relatively insoluble matrix. LSA–II is expected to include primarily unsolidified material in which the radioactive material may or may not be uniformly distributed, including lesser activity resins and filter sludges, other similar materials from reactor operations, similar materials from other fuel cycle operations, scintillation vials, and hospital, biological, and decommissioning wastes. There is, however, no prohibition against activated materials, consolidated wastes, and materials intrinsically contained in a relatively insoluble matrix in group LSA–II, provided the specific activity limit is met. The IAEA established the LSA–III group principally for irradiated reactor parts and other activated, or activated and contaminated, equipment that exceed the limits for the other LSA groups. NRC does not believe it is necessary to expand the LSA–II group definition to include these materials. The NRC believes that to do so might cause confusion with the LSA–III definition.

One comment stated that dewatered material should be defined as a solid for LSA–II. NRC agrees that dewatered resins should be subject to the specific activity for solids under LSA–II and notes that there is no prohibition against dewatered resins in LSA–II.

One comment asked whether the specific activity limits for LSA–II and LSA–III materials were pre- or post-solidification. The specific activity limits apply to materials as prepared for shipment, i.e., post-solidification. However, licensees should note that packaging or shielding material may not be considered in determining either the specific activity or the radiation level at 3 m.

One comment recommended that NRC remove the criterion for leaching that is applicable to LSA–III solids. The criterion limits the loss of radioactive material per package, when the package is placed in water for 7 days, to 0.1 A_2. Another comment stated that the criterion for leaching in the definition of LSA–III needed to be compatible with the leachability index requirements for solidified waste in 10 CFR Parts 60 and 61.

A control on the potential intake of these LSA–III materials is necessary because the radioactivity is not entirely insoluble. Because non-Type A packaging might be used in transporting these materials, a release of 10^{-2}A in an accident is assumed, with a possible bystander uptake of 10^{-3} A_2, under the standard model for determining A_2 values. Because the total body uptake must be limited to 10^{-6} A_2, the package's dispersible radioactive contents (i.e., the leachate liquid), must not exceed 0.1 A_2. For purposes of compatibility with IAEA and DOT requirements, a new § 71.77, ''Qualification of LSA–III Material,'' has been added to Subpart F. This section prescribes testing requirements for assessment of LSA–III material leaching. The hazard from the transportation of these materials is different from that posed by their disposal; therefore, no attempt has been made to achieve compatibility between transportation and disposal leachability limits.

One comment found the proposed rule unclear on the need for three LSA categories and how to classify materials under the criteria, including compacted dry active waste. IAEA developed the three LSA groups to differentiate controls based on the activity, distribution, and form of LSA material. The LSA–I group accommodates very uniformly distributed materials, such as ores. LSA–III accommodates large activated parts or solidified materials. LSA–II accommodates less uniformly distributed materials, such as compacted dry active waste.

One comment described radioactive atoms in activated products as inherently non-dispersible and relatively non-leachable. The comment recommended that activated materials be authorized for shipment as LSA–I, provided other transportation requirements are met. Although activated materials do not pose a dispersibility hazard, these materials are subject to localized concentrations of non-uniformly distributed material. Consequently activated materials are included in groups LSA–III and LSA–II.

One comment suggested changing the definition of SCO from ''* * * not itself radioactive * * *'' to ''* * * not classed as radioactive material under these rules * * *,'' since nothing is free of radioactive material. NRC and DOT have adopted this comment.

Several comments identified a typographical error in the limit for non-fixed contamination from beta and gamma emitters on the accessible surface of SCO–I objects. That value has been changed from 1.08×10^{-5} Ci/cm^2 to 10^{-4} microcurie/cm^2. These comments also noted inconsistencies in the NRC and DOT contamination limits e.g., (1.08×10^{-4} πCi/cm^2 and 10^{-4} microcurie/cm^2, respectively). NRC has adopted the DOT convention for these limits in the final rule.

One comment inquired as to whether it was consistent for NRC not to exempt SCO–I from transportation requirements when facilities with similar contamination levels may be released for unrestricted use according to NRC Regulatory Guide 1.86. Under the final rule, SCO–I group materials are exempt from NRC regulations, except for one § 71.5 requirement that licensees comply with DOT requirements. Further, the SCO–I non-fixed surface contamination limits are greater than, not similar to, the corresponding acceptable surface contamination levels in Table 1 of NRC Regulatory Guide 1.86.

Several comments noted that the term ''inaccessible surface'' used in the SCO–I definition is not defined and that it was not clear how to comply with a limit for surfaces that were inaccessible. This provision provides for the disposal of materials that have contaminated surfaces that are not readily accessible. Examples of inaccessible surfaces include: inner surfaces of pipes, inner surfaces of maintenance equipment for nuclear facilities, and inner surfaces of glove boxes. Compliance can be achieved by sampling a small area of the surface that may be accessible or by a documented estimate of the inaccessible surface contamination.

One comment stated a belief that the implementation of SCO groups would: (a) Further complicate the preparation and shipment process, without an increase in the safety and quality of waste shipments; (b) result in a significant increase in personnel exposure costs, and delays for preparation and disposal of radioactive waste; (c) require substantial initial personnel training; and (d) require extensive revisions of existing procedures and waste shipping computer programs. NRC acknowledges that the introduction of multiple LSA and SCO groups complicates the transportation of LSA materials. The IAEA consensus was that it was appropriate to regulate SCO separately from LSA materials. The purpose of these groups is to recognize the lesser hazard of LSA and SCO relative to other radioactive materials, and to provide relief from shipment requirements that would otherwise apply to these materials, while still assuring safety.

With regard to exposure, it is true that the LSA groups will require some increased material treatment or handling. However, this handling is necessary to eliminate the current practice in which there is no quantity limit on LSA packages. This situation poses a risk to the public during transport. Costs will increase, but not by an amount considered significant for the industry. Training with regard to the LSA groups, or any new provision, will be required. Periodic training of hazardous material employees regarding the safe transportation of hazardous materials is required by DOT regulations (49 CFR Part 172 Subpart H); instruction with regard to the LSA and SCO groups may be included at that time.

Implementing the LSA groups will require revision of procedures and computer codes. These costs are judged to be acceptable in order to achieve compatibility with the IAEA regulations for the safe transport of radioactive materials.

A comment noted that the SCO classification ''appears to be well-meaning,'' but that the proposed criteria (presumably the proposed $2A_1$ limit) ''detract from its potential benefit and utility,'' and that it would be easier and less expensive for both producers and consumers of electricity to enjoy the benefits of new transportation systems without the related restrictions. As stated previously, NRC has adopted the IAEA 10 mSv/h (1 rem/h) at 3 m limit for LSA packages, and believes that a limit is needed to protect the public from the potential for excessive external radiation exposure in the case of a severe transportation accident.

One comment suggested that the rule make clear that not every SCO needs to be surveyed and that a random representative survey is adequate. There is no requirement that each SCO in a package be surveyed. The shipper must be able to demonstrate, however, that the package contents comply with applicable SCO definitions.

One comment objected to the upper limit for removable surface contamination for SCO–II (10^{-2} πCi/cm^2 for beta and gamma emitters) because this limit is a factor of 90 less than current LSA limits, and would require extensive decontamination of reactor outage equipment at each site. The comment stated such decontamination is not warranted because it violates the as low as reasonably achievable (ALARA) principle, and is not justified based on shipping experience. The comment suggested that an SCO–III group be defined for materials exceeding SCO–II, and that Type A packaging be required for such materials.

Apparently, this comment is comparing the SCO–II limit for removable (non-fixed) surface contamination with the current LSA limit that applies to nonradioactive material objects that are externally contaminated with radioactive material that is not readily dispersible. The SCO–II limit for fixed surface contamination is a more appropriate comparison with the current limit for not readily dispersible contamination. The SCO–II fixed contamination limit is 20 times greater than the current LSA limit for not readily dispersible contamination.

Section 71.5 Transportation of Licensed Material

Two comments asked for clarification of the specification ''* * *outside of the confines of its plant or other place of use,'' when describing transportation made subject to DOT regulations. One of those comments suggested that the provision be reworded as ''* * *outside the site of usage, as specified in the NRC license, or where transport is on public highways.'' This wording clarifies the provision and has been included in the final rule. Similar wording has been substituted in § 71.0(c).

A comment asked whether § 71.5(b) means ''that an approval must be obtained when the shipment is covered by local State regulations and those regulations will be followed.'' The purpose of § 71.5(b) is to impose, by NRC authority, pertinent DOT requirements on shipments, by NRC licensees, that are not normally subject to DOT requirements. There is no exemption from the requirement of § 71.5(b) regarding compliance with State or local regulations.

Section 71.10 Exemption for Low Level Materials

A comment noted that the SI unit specification of 74 kBq/kg (0.002μCi/g) for exempted low-level radioactive material in § 71.10(a) is not consistent with the 70 Bq value specified in the DOT proposed rule. The specification in § 71.10(a) has been changed to 70 Bq/g, the value in the DOT's final rule. This exemption is applicable only with respect to transportation, and is not generally applicable to other Commission-regulated activities.

A comment noted that it would be useful to have an exemption for small quantities of radioactive material in § 71.10(a) as well as the exemption for LSA material. The safety rationale developed by IAEA[2] for LSA material does not extend to other radioactive materials. IAEA has been informed that a small quantity exemption may be a useful concept. However, this exemption has not been developed yet.

One comment asked that NRC clarify the use of a reference to § 71.53 in the ''Exemption for low-level materials'' provision of § 71.10(b), a provision that pertains to Type A and LSA packages. In addition to control over excessive radiation, the Commission's responsibility with respect to fissile material is to provide reasonable controls to avoid the occurrence of accidental criticality. The regulatory standards for this are found in §§ 71.55 and 71.59. There are some relatively common types of fissile material packages for which there is no credible risk of criticality in transport, even in the absence of controls. These packages are described in § 71.53, and are exempted from the criticality controls of §§ 71.55 and 71.59, because the controls are unnecessary.

The provisions of § 71.10, ''Exemption for low-level materials,'' provide broad exemptions from 10 CFR Part 71 rules that relinquish to DOT the control of types of shipments that are of low risk both from radiation and criticality standpoints. To ensure that only low criticality risk shipments are included in § 71.10(b), NRC restricts the exemption to Type A and LSA packages that either contain no fissile material or satisfy the fissile material exemptions in § 71.53. It should be noted that the exemption does not relieve licensees from DOT transportation requirements by reason of NRC authority, nor does the exemption relieve licensees from the restrictions on air transportation of plutonium imposed by Congress.

The proposed rule introduced a $2A_1$ quantity limit, for LSA packages not designed to withstand accidents (non-Type B packages), to control potential external radiation exposures. Thirty comments were received requesting that the limit be changed in the final rule. Two comments supported no limit; nine supported the IAEA dose limit of 10 mSv/h (1 rem/h)r at a distance of 3 meters for an unshielded package; 4 supported higher multiples of A_1; and 15 supported the optional use of either the IAEA limit or a higher multiple of A_1. As described previously in this

[2] International Atomic Energy Agency Safety Series #7—''Explanatory Material for the IAEA Regulations for the Safe Transport of Radioactive Material'' (1985 Edition). Available from Bernam-Unipub, 4611–F Assembly Drive, Lanham, MD 20706–4391. Tel. (301) 459–7666.

preamble, NRC and DOT have decided that the best overall response on the LSA issue and these comments is to drop the proposed $2A_1$ quantity limit, and to adopt the IAEA radiation level limit of 10 mSv/h (1 rem/h) at 3 m from the unshielded contents.

One comment suggested that the need for labels on LSA packages should be reconsidered. Package labelling falls under DOT jurisdiction. In its final rule, DOT has retained the exception from package marking and labeling requirements for domestic LSA shipments consigned as exclusive use (see 49 CFR 173.427).

One comment expressed concern over the transition of control of packages for shipping Type B quantities of LSA radioactive material from NRC to DOT. NRC has a centralized package design approval authority, whereas DOT authority allows a shipper to determine acceptable package designs (i.e., self-certify package designs). The comment expressed apprehension about permitting each shipper to review package and shipping restrictions against DOT regulations, a situation that could result in some confusion and different interpretations of the regulations.

In the final rule, the IAEA limit of 1 rem/h at 3 m from the unshielded material contents has been established as the threshold for NRC regulation of LSA or SCO package designs. NRC will review and approve, if adequate, designs for packages that contain quantities of LSA or SCO material that exceed that limit. The review by regulatory authority of package designs for quantities that exceed the IAEA limit is consistent with the approach used by other IAEA member states.

Section 71.13 Previously Approved Package

One comment proposed that the date specified in § 71.13(b)(2) be December 31, 1990, instead of December 31, 1992, to be consistent with IAEA transportation regulations. The original 1985 IAEA transport regulations specified December 31, 1990, as the cutoff date for the routine use of packages manufactured under the 1973 edition of the regulations. That date was subsequently extended for 2 years by one of the periodic updates of IAEA regulations and was properly used in the proposed rule. However, since the proposed date of December 31, 1992, has passed, the final rule has been revised (by eliminating reference to any particular date) to make this provision effective on the date that the final rule becomes effective.

Two comments noted that the preamble to the proposed Part 71 indicated that Type B and fissile packages fabricated before a certain date and not used internationally could continue to be used domestically until the end of their useful lives. The licensee would not need to demonstrate that the packages satisfy the new crush test or deep-immersion test. The comments would take that provision one step further and require the crush and deep-immersion tests only for international use packages.

NRC believes that the international package standards should be used by the United States for both domestic and international shipments, to the extent practicable. However, based on a history of safe use under earlier safety standards, and the absence of unfavorable operational data, NRC will allow the continued use of existing packages in domestic transport until the end of their useful lives. NRC will not allow, however, the continued fabrication of packages to the old designs. This action permits use of existing packages. It does not perpetuate package designs that can be discarded or upgraded to satisfy the new standards.

Another comment suggested grandfathering the existing Type A casks now approved for transporting Type B quantities of LSA radioactive material, until the Type B waste casks required to satisfy the new standards become available. NRC has adopted the suggestion, extending the proposed provisions in § 71.52, ''Exemption for low-specific-activity (LSA) packages,'' to a 3 year period, to give the industry time to design, receive approval, and fabricate new Type B waste packages.

Section 71.22 General license: Fissile Material, Limited Quantity, Controlled Shipment

One comment requested clarification as to whether the Type A limit imposed in § 71.22(c) also applies to § 71.22(d).

The requirements of §§ 71.22(a) through 71.22(e) are cumulative, each imposing additional requirements on the use of the general license. The radioactivity limit and mass limits of § 71.22(c) apply to packages, whereas the mass and mass ratio limits of § 71.22(d) apply to shipments.

A comment noted an error, in § 71.22(d)(3), which changed the intent of the section. The commenter suggests that the phrase ''exceeds unity'' at the end of § 71.22(d)(3) be replaced by the phrase ''does not exceed unity.'' NRC agrees and has made that change.

Section 71.24 General License: Fissile Material, Limited Moderator, Controlled Shipment

One commenter asked if the statement in § 71.24(b), ''* * * a quality assurance program approved by the Commission as satisfying the provisions of Subpart H of this part,'' is any different from ''* * * a quality assurance program approved by the Commission.'' The two statements are different in that the first is more specific and provides more detail. There are several different quality assurance programs, in different licensing areas, approved by the Commission. Specifying that the program must satisfy Subpart H makes it clear as to the type of quality assurance program is required.

One commenter recommended inserting ''by weight'' after ''1 percent'' in § 71.24(c)(6). NRC agrees and has made this change in § 71.24(c)(7), as well.

With respect to a general license for a package containing fissile contents, one commenter requested clarification of what is meant by ''no uranium-233'' in § 71.24(c)(6). For a general license under § 71.24(c)(6), a package containing fissile contents must have no detectable U–233. The method for making this determination can be decided by the licensee. For example, the licensee can make this determination by performing an assay or by knowing the history of the material.

Subpart D—Application for Package Approval

One comment suggested changing the title of Subpart D to ''Application for Type B Package Approval'' for clarity. Because NRC also approves Type A packages for fissile material, the title of Subpart D continues to refer to ''Package Approval.''

Section 71.38 Renewal

One comment suggested that NRC provide some administrative acknowledgment when a timely application for renewal of a certificate of compliance has been received to provide proof that timely renewal is in effect. The Commission does not believe that proof of timely renewal is particularly important and that providing an acknowledgment to each registered user of a package would be too burdensome for the benefit gained.

Section 71.43 General Standards for All Packages

Four comments suggested the addition of IAEA regulations relating to packaging of liquids and gases to Part 71, including those pertaining to the special free drop and penetration tests for liquids and gases. The NRC approves only Type B and fissile material packages. The NRC also notes that fissile material packages must be evaluated for hypothetical accident conditions more severe than the tests for liquids. Furthermore, there are currently no NRC-licensed packages designed for gaseous fissile materials and NRC does not anticipate any future applications for such packages. These additional provisions would complicate regulations that are presently adequate. IAEA standards on absorbent material and double containment have been selectively included in DOT regulations.

Eight comments disagreed with the NRC view that § 71.43(f) should continue to restrict to ''no significant increase'' any change in external surface radiation levels, as a result of subjecting a package to the defined normal conditions of transport. The comments argued that the 20 percent increase specified in IAEA regulations is a safe, reasonable, and practical number that could not reasonably be lower, and that specifying a value in the rule provides the package design engineer and the NRC review engineer a measurable goal that is consistent both with IAEA and with engineering practice.

Type B and fissile material packages can be readily designed so that normal conditions of transport result in no significant increase in dose rates, and that a twenty percent increase in dose rates because of normal handling is excessive. In addition, if a package were designed so that the external dose rate could increase 20 percent during normal handling, the package could exceed the dose rate limits in § 71.47 during transport, and would be an item of non-compliance. NRC and DOT have therefore decided to not adopt the IAEA ''20 percent increase'' provision, and to retain the current ''no significant increase'' provision.

Four comments suggest the addition of the special provisions of IAEA regulations pertaining to the transportation of radioactive material by the air mode. NRC has determined that special requirements for transport of packages by air should be excluded from Part 71 because these provisions are properly incorporated in the carrier restrictions imposed by the Department of Transportation.

Two comments suggested that the phrase ''Account must be taken of the behavior of materials under irradiation'' be clarified and quantified, perhaps in a regulatory guide, or deleted from Part 71. Although there is no regulatory guidance now available relating this requirement to transportation packages, it is clear that any effects of irradiation on materials used in the package must be taken into account. These effects could be the accelerated aging or embrittlement of elastomers or elastics and may result in requiring a frequent change of gaskets, for example.

One comment suggested the performance requirement of § 71.43(f) be changed to include a numerical sensitivity for the requirement that there be ''no loss or dispersal of radioactive contents'' as a result of subjecting a package to the specified normal conditions of transport. The equivalent paragraph in the IAEA regulations for Type A packages is paragraph 537, and does not contain a numerical sensitivity. Paragraph 548, of IAEA Safety Series #6, is the equivalent of 10 CFR 71.51, for Type B package leaktight sensitivity. Both those provisions require Type B packages to be leaktight to a sensitivity of 10^{-6} A_2/h.

Three comments noted that IAEA no longer prohibits continuous venting of packages in its 1985 edition and urged the NRC to allow the practice domestically for Type B packages. The commenters argued that although NRC took a strong position, in the preamble to the proposed rule, that continuous package venting is ''poor engineering practice,'' NRC did not explain why. The commenters noted that DOT regulations do not prohibit continuous venting for Type A packages, leaving the acceptability of continuous venting to be decided by performance requirements. The commenters stated that in some cases it would make good sense to allow continuous venting to provide pressure equalization and discharge of organically generated hydrogen gas.

NRC is continuing its ban on continuous venting of Type B packages for the following reasons:

1. Venting of a package containment system during normal conditions of transport defeats the purpose of the containment system;

2. It is practical to design packages that do not rely on venting, to relieve pressure under normal conditions of transport;

3. The use of a vent does not necessarily prevent the generation of potentially flammable or explosive gas mixtures; and

4. The reliability of filters under temperature extremes, varied operating conditions, and sustained service has not been established.

Two comments stated that Mo–99/Tc–99m radiopharmaceutical generators are open to the atmosphere to allow changes in ambient pressure and that the generators do not vent radioactive material. The comments recommended that the prohibition against venting be limited to venting radioactive material only and that NRC continue current practices.

NRC believes these comments arise from concern over the reduction in the A_2 quantity for Mo–99 from 20 curies to 13.5 curies in the proposed rule. NRC recognizes that the shipment of Mo–99/Tc–99m generators is a special case, and is retaining the 20 curie A_2 value for Mo–99, to permit the continuation of current practices.

Section 71.47 External Radiation Standards for All Packages

NRC used the term ''accessible external surface'' in its proposed rule for determining radiation levels on package surfaces, whereas DOT used the term ''external surface'' in its proposed rule. Four comments argued that the NRC and DOT regulations for radiation level limits on package surfaces should be identical. Most believed that a limit on accessible surfaces was the more reasonable standard.

DOT has indicated that it is considering a petition for rulemaking to add the word ''accessible'' to its radiation level regulations and will consider that complex issue in a separate action. Pending completion of the DOT separate action, NRC has deleted the word ''accessible'' from this section of the final rule but does not intend to alter its practices regarding this provision.

One comment stated that this paragraph tends to be confusing in that it establishes a limit of 2 mSv/h (200 mrem/h) for package surface radiation levels, yet § 71.47(b)(2) seems to state that packages transported on a flatbed trailer can exceed 2 mSv/h (200 mrem/h), provided the radiation level at the planar edges of the trailer is less than or equal to 2 mSv/h (200 mrem/h).

Section 71.47 establishes a generally applicable 2 mSv/h (200 mrem/h) Package surface radiation-level limit. The section further establishes that, if a package is shipped as exclusive use, the radiation level may exceed 2 mSv/h (200 mrem/h), provided the applicable provisions of paragraphs (a) (with repect to Transport Index) through (d) are met. Paragraph (b)(2) restricts the radiation level at any point on the vertical planes projected by the outer edges of a flat-bed style vehicle to 2 mSv/h (200 mrem/h) (the same limit imposed in paragraph (a) for the outer surfaces of closed transport vehicles). Thus, provided packages are shipped as exclusive use, external radiation levels may exceed 2 mSv/h (200 mrem/h) at the surface of packages on flatbed trailers, but not at the outer-edge planes of the vehicle.

Section 71.51 Additional Requirements for Type B Packages

One comment suggested that the clarifying provision following paragraphs 548(a) and (b) of IAEA regulations be added to Part 71 for consistency. The clarifying provision pertains to allowable releases of radioactive material from a package containing a mixture of radionuclides. This is the case, for example, with spent nuclear fuel casks. That clarifying provision has been added.

Section 71.52 Exemption for LSA Packages

Twelve comments expressed concern that the proposed Part 71 affords only a 1-year delay in applying the new LSA rules. NRC established the 1-year delay to give the industry an opportunity to design and build the Type B waste casks that would be required under the new rules. The comments uniformly argued that 1 year was not a sufficient period of time to design a waste cask, to have it reviewed and approved by NRC, and to fabricate an adequate number of casks, to approved designs, that satisfy the needs of the new LSA rule. The commenters differed in how long they thought that process would take, varying over 2, 3, and 5 year periods. NRC agrees with the thrust of this comment and has established the exemption period at 3 years. Thus existing packagings may be used for 3 years and new packagings may be fabricated from existing designs for 3 years.

A consequence of establishing the IAEA LSA/SCO package limit as the delineator between NRC and DOT regulation of LSA and SCO packaging [see § 71.10(b)(2)] is that, after the 3 year exemption period, LSA will be shipped either in DOT authorized packagings, or in NRC certified Type B packagings. Accordingly, NRC is discontinuing the practice of certifying Type A LSA packages. NRC has therefore not adopted a proposed exemption (§ 71.52(a)) that only would have applied to NRC certification of new Type A LSA package designs.

One comment stated that the demand for waste casks would rise until 1993 and then fall again because few of the low-level radioactive waste disposal site compacts will permit disposal access. Vendors will hesitate to invest in casks that will not be used after 1993 and waste will need to be stored onsite.

NRC is unwilling to accept this proposition and believes that as long as NRC specifies the requirements for transportation of waste, given adequate time, industry will continue to develop disposal options.

One comment argues that the specific reference to § 71.43(f) should be deleted because it is included in the broader reference to §§ 71.41–71.47.

Section 71.52 exempts exclusive use LSA and SCO packages from the additional requirements for Type B packages for a period of 3 years from the effective date of the final rule. These LSA packages are still subject to other requirements that apply to all packages. The referral to these other package requirements includes §§ 71.41–71.47, plus a specific reference to. An argument could also be made for deleting the entire reference because those requirements apply regardless of the reference in this section. However, NRC chose to include the reference in § 71.52 as a reminder that the exemption is only from § 71.51, not from all packaging requirements. NRC believes the reference to § 71.43(f) (normal conditions of transport tests) is important and has decided that it will be retained.

One comment suggested that SCO be included within the scope of § 71.52, and that the $2A_1$ limit be included in the section for clarity. NRC agrees with the comment and has made the clarifications, substituting the IAEA LSA limit for $2A_1$.

Section 71.53 Fissile Material Exemptions

One comment suggested spelling out the word ''liter'' instead of using ''l'' as the abbreviation. Considering the typing errors caused by the use of that abbreviation, the final rule spells out the word ''liter'' wherever it appears.

Section 71.55 General Requirements for Fissile Material Packages

One comment suggested that by adding the word ''full'' to the water reflection criterion of § 71.55(b)(3), the NRC has added more cost with no apparent benefit ''* * * since transport limits already take this consideration into account.'' The latter part of this comment probably refers to the ''transport index'' controls that limit the number of packages which can be transported and stored together, but do not consider the safety of an individual package in isolation. Addition of the word ''full'' in § 71.55(b)(3) is a matter of clarification. NRC has always required ''full'' reflection wherever reflection is required. IAEA regulations required ''full'' reflection in the 1973 edition, and go a step further in the 1985 edition, to define ''full'' as ''water 20-cm thick (or its equivalent).'' NRC has retained the word ''full,'' in § 71.55(b)(3), and has added the word ''full,'' in § 71.55(e)(3), for consistency.

A commenter agrees that the proposed Part 71 begins to simplify the system of shipping fissile material but that most of the difficulties still exist. The commenter advocates development of ''a system of performance-oriented packaging,'' to reduce the current complexity of the ''design-oriented package choices.'' NRC agrees that there are a number of radiation control design requirements that apply to the fissile material packages as well as to packages of other radioactive material. However, NRC views the criticality control provisions as performance-oriented rather than design-oriented. NRC must specify the conditions against which the package must be designed. Without the environmental tests and package objectives, there would be no level of protection against which to design packages.

Section 71.61 Special Requirement for Irradiated Nuclear Fuel Shipments

One comment recommended that the rule clarify that the deep immersion test is to be applied to an otherwise undamaged package. This important detail is implied, but not specifically stated. The Commission agrees and has made that clarification.

In the final rule, this section has been modified to require that the external pressure test be applied directly to the containment system of a package. NRC does not believe the external structure should play a part in helping the containment system of a package withstand an external pressure test and has chosen to ignore its existence in specifying the requirement.

A comment recommended that the word ''rupture,'' as used in this requirement, be defined as a gross structural collapse and not just an inleakage of water. Although the word ''rupture'' in the proposed rule did mean gross structural collapse, NRC has since decided that the term ''rupture'' cannot be determined by engineering analysis. NRC has decided to change the acceptance criteria for the deep immersion test from ''rupture'' to ''collapse, buckling, or inleakage of water.''

A comment stated that this requirement should include the 1-hour time specification included in the IAEA requirement to avoid later misinterpretation of the test. The NRC agrees that adding the 1-hour test specification would help prevent confusion between IAEA and domestic regulations, and has included the time specification.

A comment noted that the term ''at least'' is used two times in the proposed requirement, thereby creating an opportunity for misinterpretation. Although the term is used in the IAEA text, the NRC agrees with the commenter that it serves no useful purpose and has deleted the term.

A comment stated that the deep-water immersion test should be clarified to ensure that an engineering evaluation is an acceptable alternative to a physical test because an actual 200-m test would be costly and difficult. NRC believes it is clear that an engineering evaluation is acceptable because the equivalent external gauge pressure is specified in the text of the requirement. The provisions of § 71.41(a) are intended to allow the use of engineering evaluations when they are reasonably applied.

The remaining three comments relating to this section all deal with transition periods and special provisions for casks for which there will be no further fabrication and that are not used internationally. The earlier portion of this preamble dealing with the provisions of § 71.13 presents the NRC view on these matters.

Section 71.63 Special Requirements for Plutonium Shipments

Four comments argued that the extension of this provision to radionuclides other than plutonium is unjustified and that the provision, even without the extension to other radionuclides, differs from IAEA rules and is inconsistent with the principles of IAEA rules. Two of the commenters argued further that the existing provisions, if examined in the light of current regulatory analyses, probably could not be justified.

NRC recognizes that some requirements have been added to the regulations over the years strictly on the basis of prudent judgment. Because the basis for current rules is not a part of this rulemaking action, NRC will simply refrain from extending the present rule to other radionuclides.

One commenter argued that the rule should be rewritten using multiples of the A_2 values, not only to define radionuclides subject to the rule, but also to define the level of activity at which the extra requirements come into effect. Because the extension to other radionuclides is being withdrawn, the inclusion of A values does not appear to improve the requirement.

Section 71.71 Normal Conditions of Transport

Three comments noted that the provision of IAEA's paragraph 528 requiring consideration of a temperature range from −40 °C to +70 °C for the components of the packaging is not reflected in Part 71. NRC omitted this provision because NRC does not want to limit the high end temperature consideration to 70 °C because that would imply that +70 °C is the highest temperature that has to be considered for package design. This does not take into account the considerably higher temperatures resulting from decay heat in certain Type B packages.

Three comments noted that 10 CFR 71.71(c)(4) prescribes an increased external pressure specification of 140 kPa absolute but IAEA regulations do not have that exact requirement. NRC believes there is a need for an external pressure test for normal conditions to ensure that a package filled at low pressure or high altitude will withstand an external pressure increase. The additional pressure test has been retained.

Three comments observed that § 71.71(c)(7) states that the free drop test be conducted between 1.5 and 2.5 hours after the conclusion of the water spray test but the same requirement is not included in the IAEA regulations. The IAEA rules, however, do include restrictions, in paragraph 620, on the timing of the mechanical tests after the water spray test. NRC has retained the water spray test as is and believes the NRC test meets the intent of the IAEA test.

One comment noted that with the deletion of the fissile classes, the corner drop test, which was required only for Fissile Class II packages, is proposed to be applied to all fissile packages. The commenter argued that for a large and heavy package, such as a spent fuel shipping cask, ''it is considered highly implausible for a package to undergo a one-foot corner drop as a normal condition of transport. Only a free drop with the package in its normal orientation should be specified as a normal condition of transport for large and heavy packages, therefore saving valuable analysis effort and time.''

NRC agrees with the comment and has deleted the corner drop test for fiberboard, wood, or fissile material rectangular packages weighing more than 50 kg (110 lb), and for fissile material cylindrical packages weighing more than 100 kg (220 lb). For these packages, NRC does not believe that the corner drop tests are significant in developing a safe fissile material package.

Section 71.73 Hypothetical Accident Conditions

One comment stated that reversing the order of the two immersion tests in §§ 71.73 (c)(5) and (c)(6) would restore the order of the tests, which must be run consecutively, and would therefore clarify the text. NRC agrees and has made the change.

One comment recommended that the temperature extremes specified for the initial test conditions in § 71.73(b) be given a reasonable tolerance because ambient air temperatures cannot be controlled. NRC agrees that temperatures, as with other required parameters of the test conditions, cannot be accurately controlled. NRC's position, however, is not to establish tolerances, but to require that the effects of test conditions different from those specified be analyzed as part of the overall evaluation. Every analysis would then be normalized to the same set of specifications.

One comment recommended that the word ''single,'' in the second line of the thermal test in § 71.73(c)(4), should be ''simple''. NRC agrees and has made that change.

Two comments asked that NRC include some information as to how the effects of solar radiation should be treated. One comment stated, ''The solar insolation can be a significant factor and should be consistently evaluated.'' Others have argued that the effects of solar insolation are insignificant compared with the thermal effects of the fire test and should be ignored.

NRC adopts the view of the thermal experts who participated in developing the IAEA regulations. Those experts thought the effects of solar radiation may be neglected before and during the thermal test but that such effects should be considered in the subsequent evaluation of the package response.

One comment recommended the development of guidance on how designers should interpret the revised thermal test requirement. Although there is guidance provided in the IAEA's companion documents to its transportation regulations (IAEA Safety Series No. 7, ''Explanatory Material for the IAEA Regulations for the Safe Transport of Radioactive Material—1985 Edition,'' and IAEA Safety Series No. 37, ''Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material—1985 Edition''), further guidance may be necessary. If so, it is the industry that can best propose guidance, based on its capabilities. If coordinated under the auspices of the American National Standards Institute (ANSI), Committee N–14, with NRC representation, there is a good chance that a consensus standard could be developed that could be endorsed by NRC as a satisfactory means to satisfy regulatory requirements.

One comment stated that packages that are subjected to the crush test should not also be subjected to the 30-foot free drop test, as required in the proposed rule. Instead, consistent with IAEA, the crush test should be in lieu of the 30-foot free drop test.

NRC believes that the crush test and the free drop test impart different types of loadings onto the package. Having sufficient crush resistance for the crush test does not ensure the adequacy of the package under the inertial loadings that occur during the 30-foot drop tests. NRC believes that it is important for packages to have resistance to impact and that the crush test should not be a substitute for the impact test.

One comment stated that a crush scenario is not likely during ''dedicated'' shipments because heavy loads are not placed above the shipment at any time during transport. The comment questioned the applicability of the test for dedicated shipments, and requested that at least an engineering evaluation be allowed as an alternative to a physical test. NRC has made it clear (see § 71.41) that appropriate analyses may be used to demonstrate the ability of a package to meet crush test conditions.

Section 71.75 Qualifications of Special Form Radioactive Material

One comment indicates that changes in § 71.75(a) from the current rule have changed the concept of special form from being a provision for special properties of the radioactive material contents of the package to being a provision for special properties of the package—a change from qualifying a ''special form source'' to qualifying a ''special form package.''

NRC regrets the confusion, but intended no substantive change to the concept of special form. Special form criteria in this final rule have been brought closer to those of DOT, but still without any basic changes.

One comment noted that the reference in § 71.75(e) [§ 71.75(d), in the final rule], to a standard of the International Standard Organization (ISO) is vague and should be made more specific.

Although the ISO standard could be written in all its detail in Part 71, rather than simply referenced there, most comments over the years have encouraged NRC to have less repetition and more simple references to other requirements.

Section 71.83 Assumptions as to Unknown Properties

One comment pointed out an error in line 7 of § 71.83, where the proposed rule referred to ''known properties'', where it should have referred to ''unknown properties.'' That error has been corrected.

Section 71.85 Preliminary Determinations

One comment recommended that the term ''durable'' in the context of ''durably mark the packaging,'' as in § 71.85, be defined in terms of the conditions that the markings on the packaging must be able to withstand. When developing its regulations, NRC must decide at what level of detail they are to be written. Sometimes that level of detail is changed as a result of experience if a widespread misuse of a standard becomes known because of a lack of detail. NRC is not aware of any problem with the term ''durably,'' even though it has been used since 1968 in the preliminary determinations section. In the absence of a significant problem, NRC prefers to leave the term as is.

Section 71.87 Routine Determinations

One comment recommended that NRC's Table V ''Removable External Radioactive Contamination Wipe Limits,'' be used by DOT in place of its Table 11. NRC notes that the only significant difference between the two tables is that the term ''low toxicity alpha emitters'' is replaced by its definition in the NRC table. The NRC final rule simply refers to the DOT requirement (49 CFR 173.443) for maximum permissible contamination limits.

Section 71.88 Air Transport of Plutonium

One comment recommended that the forward tie-down specification of 9 g detailed in § 71.88(c)(2) be reduced to 1.5 g for plutonium packages transported on a Boeing 747 aircraft. The reason for this recommendation has to do with the 14 CFR 25.561 regulatory requirement of the Federal Aviation Administration (FAA), that the supporting structure of an airplane must be designed to restrain, up to specified inertial forces, including 9–g in the forward direction, ''* * * each item of mass that could injure an occupant if it came loose in a minor crash landing.'' NRC, in prescribing tie-down requirements for plutonium packages in aircraft, took note of the supporting structure requirements of the FAA and required a 9-g tie-down system for the package on the main deck of the aircraft. The Boeing 747 cargo aircraft, however, with no passengers and the cockpit located above the main deck, is not subject to the requirements of 14 CFR 25.561 because there are no occupants to injure if ''* * * the package came loose in a minor crash landing.'' Thus, the Boeing 747 ''Weight and Balance Manual,'' DG–13700, shows a load factor of 1.5 g in the forward direction.

The purpose of the NRC tie-down requirement was not to protect occupants of the aircraft from cargo that has come loose in a minor crash landing. Therefore, the comparison with the FAA supporting structure requirement is not germane. The purpose of the NRC requirement was to protect the plutonium package from the uncontrolled potential for damage inherent in having the package unrestrained in a crash landing.

Paragraph (c) of § 71.88 proposed a requirement that the licensee make special arrangements with the carrier on where to place the plutonium cargo in the aircraft, how to tie it down, and what restrictions are to be placed on other cargo. Recognizing that these restrictions would be more appropriately placed directly on the carrier rather than through the shipper, the DOT has placed these restrictions in its air carrier regulations (§ 175.704 of 49 CFR Part 175, ''Carriage By Aircraft.'') These regulations are now referenced in § 71.88.

Section 71.95 Reports

All three public comments on this section were directed at the newly proposed provisions of paragraph (c), which require a 30-day report of ''* * * instances in which the conditions of approval in the certificate of compliance were not observed in making a shipment.''

One comment requested clarification whether § 71.95(c) applies to shippers or receivers.

The scope of Part 71 (§ 71.0(c)) makes the regulation applicable only to shippers of radioactive material. Therefore, § 71.95(c) applies only to shippers of radioactive material. However, shipment deficiency may be detected by the receiver of the shipment. If the receiver reports that deficiency to the shipper, the shipper is obligated to report it to NRC. Further, note that 10 CFR Part 21, ''Reporting of Defects and Noncompliance'', is applicable to receiving facilities.

The other two comments dealt with the substance of the event that would prompt the report. One suggested the regulation be more specific on conditions that would require a report. The second comment suggested that the report include the consequences of the deficient shipment such as radioactive contamination, a loosened sealing cap, etc.

Although both of these suggestions have merit, neither has been incorporated in the final rule. The purpose of the requirement is to provide feedback to NRC on quality assurance program effectiveness by an indication of the number and type of packaging and other mistakes and on the safety significance of those mistakes by an indication of the mistake consequences. NRC believes the reporting requirement should retain its broad scope. A large number of reports is not expected. NRC also believes that individual follow-up is the only reasonable way to uncover any procedural deficiency that might cause mistakes.

One comment questioned whether this type of report is important enough to be required within 30 days. NRC judges that the timing is about right, and expects the staff's review of submitted reports to be completed within a similar time frame.

Section 71.97 Advance Notification of Shipment of Irradiated Reactor Fuel and Nuclear Waste

Of the five comments submitted on this notification requirement, two suggested changing the value for the number of curies in § 71.97(b)(3)(iii), so it corresponds to the same limit in the regulations of DOT and IAEA. That change has been made.

The other three comments stated that this requirement was not clearly expressed. The requirement has been reorganized in the final rule, and consists of the following parts:

1. Paragraph (a) provides a broad general requirement that licensees prenotify governors of States of any shipments of radioactive material going to, through, or across the boundary of the State;

2. Paragraph (b) limits the prenotification requirement to certain types of shipments. All the conditions of paragraph (b) must be satisfied for the prenotification requirement to apply. The licensed material must be required to be in a Type B package, limiting the requirement to shipments of relatively high potential hazard. The shipment must be destined to a disposal site or to a collection point for transport to a disposal site, further limiting the requirement to waste material. The quantity of radioactive waste in a single package must exceed the limits specified in the DOT regulations for highway-route controlled quantities. Lastly, for irradiated fuel, the quantity contained in a single package must be less than that subject to the similar advance notification requirement of 10 CFR 73.37(f).

3. Paragraphs (c), (d), (e) and (f) contain the details for timing, information in the notification, revisions, and cancellation.

One comment noted that from the wording in § 71.97(a), a reader would expect to find exceptions in § 71.97(b). The comment notes that the provision does not contain exceptions. NRC agrees with this comment and has revised § 71.97(a) for clarity.

One comment questioned the value of proposed § 71.97(b)(4) [§ 71.97(b) in the final rule] which required that ''* * * the quantity of irradiated fuel is less than that subject to advance notification requirements of § 73.37(f) of this chapter.'' Paragraph 73.37(f) refers to a separate part of the Commission's regulations, 10 CFR Part 73, ''Physical Protection of Plants and Materials,'' and imposes an advance notification requirement for irradiated fuel shipments similar to the one under discussion. The scope of Part 73 (see § 73.1(b)(5)) limits its applicability regarding shipments of irradiated reactor fuel to ''* * * quantities that in a single shipment both exceed 100 grams in net weight of irradiated fuel, exclusive of cladding or other structural or packaging material, and have a total radiation dose rate in excess of 100 rems per hour at a distance of 3 feet from any accessible surface without intervening shielding.'' If the quantity of irradiated fuel in a shipment exceeded the quantity specified in § 73.1(b)(5), the notification would be made under § 73.37(f). If not, the notification would be made under § 71.97. The proposed provision in § 71.97(b)(4) was intended to prevent duplicate notifications for some shipments.

The final comment on § 71.97 included a clear rewrite of § 71.97(b) that has been used in its entirety in the final rule.

Comments on Appendix A

Five comments supported the inclusion of new radionuclides in Table A–1 of Appendix A as useful and justified. Five other comments pointed out errors and inconsistencies between NRC and DOT for the A_1/A_2 values in Table A–1. These inconsistencies have been corrected in the NRC and DOT final rules.

Three comments recommended a grandfathering provision for the continued authority to transport molybdenum (Mo) 99/technetium (Tc) 99m generators, in Type A packages, with radioactivity between the current A_2 value of 20 Ci and the new A_2 value of 13.5 Ci for Mo-99. The lower A_2 value is the result of a new dosimetric model, for beta-emitting radionuclides, to address skin contamination. In the preamble to the NRC proposed rule, the NRC noted, with respect to the changes in the A_1 and A_2 values:

> Based on our most current knowledge of radioactive material shipments in the United States, the economic impacts of these changes are not likely to be large. However, any situations where a potential exists for significant economic impacts as a result of changes in the A_1 or A_2 values should be brought to the NRC's attention in public comments.

NRC agrees that this is a situation where health care in the United States could be significantly impacted as a result of forcing the larger quantity Mo-99/Tc-99m generators now transported in Type A packages into Type B packages. In view of the favorable experience over the years with these generators, NRC and DOT will allow the continued domestic transportation of generators that contain up to 20 Ci of radioactive material in Type A packages.

Two similar proposals to grandfather the transportation of carbon-14, phosphorus-32, sulfur-35, and iodine-125 at existing levels were not as persuasive and have not been adopted. The decrease in A_1 and A_2 values would apparently force many shipments out of the ''limited quantity'' category, where they are excepted from specification packaging, shipping papers and certification, and marking and labeling requirements, and into the ''Type A'' category.

Although there are clearly more packaging and communication requirements associated with the ''Type A'' category than with the ''limited quantity'' category, NRC does not view that change as creating the same economic impact as a change from the ''Type A'' to the ''Type B'' category.

One comment suggested that the radionuclides einsteinium-253 and einsteinium-254 be added to Table A–1 because shipment of those transuranics are increasing in number and the default values are not expected to be adequate. NRC has added those radionuclides and will also propose them for addition to the IAEA regulations. Until they are included in IAEA Safety Series No. 6, however, multilateral approval is required for international shipments. This limitation is identified by footnote in Table A–1.

One comment objected to having to obtain NRC approval of A_1/A_2 values that are not in Table A–1. In addition to NRC approval, international shipments require multilateral approval of A values that are not included in the IAEA regulations by each country through or into which the consignment is to be transported. The development of A values may not be a simple matter, requiring consideration of daughter radionuclides and differing radioactive emissions. Although a competent health physicist or nuclear engineer should not have too much difficulty determining an A value, NRC must assure that a system exists to protect against faulty determinations. Use of the conservative A values from Table A–2 does not require regulatory approval.

One commenter questioned the unlimited values, for A_1 and A_2 in Table A–1, for uranium-235 enriched less than 5 percent. The comment argued that U–235 is a fissile material and the unlimited values may not be appropriate. The A_1/A_2 values are for radiological, not fissile, considerations. The A_1/A_2 values set the maximum quantity of radioactive material that can be shipped in a Type A package (except for LSA); other package characteristics, such as heat generation, weight, criticality, external radiation, etc., can further limit the quantity of radioactive material in that Type A package. Limitations with respect to fissile characteristics, for example, are addressed in §§ 71.53, 71.55, and 71.59. NRC has decided to add a clarifying note, currently in the IAEA regulations, to the A_1/A_2 Table in Appendix A of Part 71. The Appendix A note reads ''Where values of A_1 and A_2 are unlimited, it is for radiation control purposes only. For nuclear criticality safety, some materials are subject to controls placed on fissile material.''

Finally, one comment suggested that we eliminate the specific activity column from Table A–1. The comment argues that ''Specific activity information is not required or explained in the regulations, and it is difficult to keep the information accurate.''

Although the NRC is in basic agreement with the comment and would have no problem in eliminating the specific activity data from Part 71 if there were a good source of comparable data available for the times it is needed to implement the transportation regulations. NRC is not familiar with any good substitute source. Though IAEA Safety Series No. 37, ''Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material (1985 Edition),'' third edition, published in June 1987, includes a table of half-lives and specific-activities, there is no indication yet of a system of periodic reviews that would keep that information up to date.

Comments on Draft Regulatory Analysis

Ten persons commented on the impacts associated with the proposed changes to limit the content of LSA/SCO packages to $2A_1$. The main thrust of these comments is that the impacts are much greater than presented. In part in response to these comments, NRC has adopted in the final rule the IAEA LSA/SCO package limit of 10 mSv/h (1 rem/h) at 3 m, in lieu of the proposed $2A_1$ limit.

Because the NRC data base for determining the additional shipments expected to be caused by the proposed rule dated back to 1980, and because a clear preference was developing in the public comments for the IAEA radiation level limit rather than the $2A_1$ limit, NRC repeated its analysis using more recent data. An NRC contractor gathered 1989 data from the 3 shallow land burial facilities for all waste shipments of resins, evaporator bottoms, and filter media. The contractor analyzed the characteristics of those 4600 Type A cask shipments and found that approximately 150 of those shipments would have exceeded the IAEA limit. NRC assumes that each shipment exceeding the limit is split into 2 shipments due to the smaller capacity of Type B packaging. Thus 150 additional shipments are caused by the LSA limit.

The impacts of preparing additional packages of LSA waste for shipment and receiving those additional shipments at the burial ground were absent from the draft regulatory analysis. One comment advised the NRC of the results of an exposure study which concluded that the extent of the collective exposure for preparation and receipt of waste casks was approximately 0.5 person-rem per shipment. The NRC noted that half of the 0.5 person-rem per shipment factor multiplied by the 4600 waste cask shipments per year from the new data base corresponds fairly well to a large portion of the 1726 person-rem collective exposure reported for all light water reactors for 1986 under the category ''waste processing'' by Barbara G. Brooks, NRC, and D. Hagemeyer, SAIC in NUREG–0713, Vol. 8, dated August 1989 (this version was current at the time the contractor prepared the regulatory analysis). On the basis of this data, NRC has accepted the 0.5 man-rem per shipment number as a reasonable estimate. Multiplying that 0.5 man-rem per shipment conversion factor by the 150 additional shipments which the limit of 1 rem per hour at 3 meters would cause, the effect of the limit would be 75 person-rem per year.

Because the IAEA LSA provisions permit a greater quantity of LSA/SCO material to be shipped in a package, fewer packages and shipments are needed to transport a given quantity of material. The estimated burden on industry from the final rule is therefore less than that for the proposed rule. The NRC draft regulatory analysis dated November, 1987 developed industry costs resulting from a $2A_1$ limit on LSA shipments of $1.7 million per year. These costs consist of package costs and shipment costs resulting from an estimated 311 additional cask shipments per year. Through the same simple modeling used in the older analysis, the new NRC regulatory analysis shows increased dollar costs associated with the 150 additional LSA/SCO shipments of $1.0 million per year. These estimates include differential package costs and differential shipping and handling costs, annualizing and summing each component. These estimates do not include cost components recognized but not quantified in the public comments as training, procedure revisions, computer program changes and upgrades, insurance premiums, and disposal costs.

There were no significant comments related to the projected number of non-radiological deaths and injuries associated with the increased shipments caused by the new standards.

Agreement State Compatibility

Section 274d.(2) of the Atomic Energy Act of 1954, as amended, requires that before entering into an agreement with any State, the Commission shall make a determination that the State's program is compatible with the Commission's program. Section 274g authorizes and directs the Commission to cooperate with the States in the formulation of standards to assure that State and Commission programs will be coordinated and compatible. The basic objective of NRC's State Agreements Program has been to achieve uniformity among the various programs to the maximum extent practicable recognizing that the States must be allowed some flexibility to accommodate local conditions. Under this Program, procedures have established criteria for better defining compatibility, and for determining the degree to which States regulations must show uniformity with Commission regulations. In practice, the Commission's regulations are categorized as Division 1–4 Rules according to the degree of State regulation uniformity required, as summarized in the following table:

Division	Agreement State regulation uniformity
1	Agreement States are expected to adopt, essentially verbatim, the regulation to provide consistency between Federal and State requirements.

Division	Agreement State regulation uniformity
2	Agreement States have the flexibility to adopt similar or more stringent requirements based on their radiation protection experience, professional judgements, and community values.
3	Agreement States should adopt the requirement, but there is no degree of uniformity between NRC and Agreement States required.
4	Agreement States should not adopt the requirement since these are regulatory functions reserved to NRC.

The final rule does not affect the current compatibility categorization of Part 71 regulations. The following table lists the Part 71 Sections and corresponding rule categorization (Division 1–4):

Division	Section	Title
1	71.4	Definitions.
1	71.5	Transportation of Licensed Material.
1	71.10	Exemption for Low-Level Materials.
1	Appendix A	Determination of A1 and A2.
2	71.12	General License: NRC-Approved Package.
2	71.13	Previously Approved Package.
2	71.14	General License: DOT Specification Container.
2	71.16	General License: Use of Foreign Approved Package.
2	71.81	Applicability of Operating Controls and Procedures.
2	71.85	Preliminary Determinations.
2	71.87	Routine Determinations.
2	71.88	Air Transport of Plutonium.
2	71.89	Opening Instructions.
2	71.97	Advance Notification of Shipment of Irradiated Reactor Fuel and Nuclear Waste.
3	71.0	Purpose and Scope.
3	71.1	Communications.
3	71.2	Interpretations.
3	71.3	Requirement for License.
3	71.7	Completeness and Accuracy of Information.
3	71.8	Specific Exemptions.
3	71.9	Exemption of Physicians.
3	71.91	Records.
3	71.93	Inspections and Tests.
3	71.95	Reports.
3	71.99	Violations.
3	71.101	Quality Assurance Requirements.
3	71.103	Quality Assurance Organization.
3	71.105	Quality Assurance Program.
3	71.107	Package Design Control.
3	71.109	Procurement Document Control.
3	71.111	Instructions, Procedures, and Drawings.
3	71.113	Document Control.
3	71.115	Control of Purchased Material, Equipment, and Services.
3	71.117	Identification and Control of Materials, Parts, and Components.
3	71.119	Control of Special Process.
3	71.121	Internal Inspection.
3	71.123	Test Control.
3	71.125	Control of Measuring and Test Equipment.
3	71.127	Handling, Storage, and Shipping Control.
3	71.129	Inspection, Test and Operating Status.
3	71.131	Nonconforming Materials, Parts, or Components.
3	71.133	Corrective Action.
3	71.135	Quality Assurance Records.
3	71.137	Audits.
4	71.6	Information Collection Requirements: OMB Approval.
4	71.18	General License: Fissile Material, Limited Quantity per Package.
4	71.20	General license: Fissile Material, Limited Moderator per Package.
4	71.22	General License: Fissile Material, Limited Quantity, Controlled Shipment.
4	71.24	General License: Fissile Material, Limited Moderator, Controlled Shipment.
4	71.31	Contents of Application.
4	71.33	Package Description.
4	71.35	Package Evaluation.
4	71.37	Quality Assurance.
4	71.38	Renewal of a Certificate of Compliance or Quality Assurance Program Approval.
4	71.39	Requirement for Additional Information.
4	71.41	Demonstration of Compliance.
4	71.43	General Standards for all Packages.
4	71.45	Lifting and Tie-down Standards for all Packages.
4	71.47	External Radiation Standards for all Packages.
4	71.51	Additional Requirements for Type B Packages.

Division	Section	Title
4	71.52	Exemption for Low-Specific-Activity (LSA) Packages.
4	71.53	Fissile Material Exemptions.
4	71.55	General Requirements for Fissile Material Packages.
4	71.59	Standards for Arrays of fissile Material Packages.
4	71.61	Special Requirement for Irradiated Nuclear Fuel Shipments.
4	71.63	Special Requirements for Plutonium Shipments.
4	71.64	Special Requirements for Plutonium Air Shipments.
4	71.65	Additional Requirements.
4	71.71	Normal Conditions of Transport.
4	71.73	Hypothetical Accident Conditions.
4	71.74	Accident Conditions for Air Transport of Plutonium.
4	71.75	Qualification of Special Form Radioactive Material.
4	71.77	Qualification of LSA–III Material.
4	71.83	Assumptions as to Unknown Properties.
4	71.100	Criminal Penalties.

Petitions for Rulemaking

Three petitions for rulemaking were filed with the NRC in connection with the rules for transporting LSA radioactive material. The substance of each of the three petitions was essentially the same, to request that NRC exempt LSA materials from its requirements in Part 71.

The petitioners were the Energy Research and Development Administration (now the U.S. Department of Energy) in its letter dated July 23, 1975 (PRM–71–1); ANSI Committee N14, in its letter dated March 10, 1976 (PRM–71–2); and Chem-Nuclear Systems, Inc., in its letter dated November 22, 1976 (PRM–71–4). At the time these petitions were filed, DOT regulated carriers and shippers of small quantities of all radioactive materials (including LSA materials) through provisions in its regulations in 49 CFR Parts 170–189, whereas NRC regulated shippers of fissile material and of larger quantities of other radioactive materials (including LSA materials) through its regulations in Part 71 and its licensing program. All three petitioners argued that the control NRC was exerting over transportation of LSA materials created an inconsistency between NRC regulations and those of the IAEA and should be discontinued. A proposed rule that would have provided the exemption for LSA materials requested in the petitions was published by NRC for public comment on August 17, 1979 (44 FR 48234). Before finalization of that rule, however, a deficiency in the new LSA requirements, as proposed, was recognized so that the entire LSA proposal, including the exemption, was withdrawn. In the interim, the corresponding deficiency in the LSA requirements in the IAEA regulations was recognized and corrected. That correction is discussed under the ‘‘major modifications from proposed rule’’ section of this preamble. This correction is implemented in both DOT regulations and NRC regulations.

The exemption requested in the three petitions has been superseded by the changes in LSA requirements. The LSA requirements imposed in NRC regulations are an integral part of the NRC/DOT regulatory scheme for LSA materials. This scheme is based on IAEA regulations. There is an exemption provided for LSA materials in § 71.10 that clearly defines the level where NRC regulations impose additional packaging requirements. For the above reasons, NRC has denied the petitions.

Administrative Correction

At about the same time the Notice of Proposed Rulemaking regarding compatibility with IAEA transportation regulations was published for public comment on June 8, 1988 (53 FR 21550), a separate notice of final rulemaking was issued, by NRC, affecting the retention periods for records (53 FR 19240, May 27, 1988). Included in that separate notice were changes to the transportation regulations in Part 71, specifically to §§ 71.105, ‘‘Quality assurance program,’’ and 71.135, ‘‘Quality assurance records.’’ Because the two rules were being processed at the same time by different organizations, NRC's internal controls failed to recognize that the new quality assurance provisions needed to be incorporated in the June 8, 1988, notice of proposed rulemaking. No written comments were filed with respect to the quality assurance sections proposed, although two phone calls were received advising NRC of its error. The quality assurance changes that were made effective by the final rule, published on May 27, 1988, are included in this final rule.

Finding of No Significant Environmental Impact: Availability

The Commission has determined, under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR Part 51, that this rule is not a major Federal action significantly affecting the quality of the human environment, and therefore an environmental impact statement (EIS) is not required.

The Commission's ‘‘Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes,’’ NUREG–0170,[3] dated December 1977, is NRC's generic EIS, covering all types of radioactive material transportation by all modes (road, rail, air, and water). From the Commission's latest survey of radioactive material shipments and their characteristics, ‘‘Transport of Radioactive Material in the United States,’’ SAND 84–7174, April 1985, it can be concluded that current radioactive material shipments are not so different from those evaluated in NUREG–0170 as to invalidate the results or conclusions of that EIS. Environmental impacts associated with this rulemaking are evaluated in ‘‘Regulatory Analysis of Changes to 10 CFR Part 71—NRC Regulations on Packaging and Transportation of Radioactive Material,’’ dated April 1995.

NUREG–0170 established the non-accident related radiation exposures associated with transportation of radioactive material in the United States as 98 person-Sv (9800 person-rem) which, based on the conservative linear

[3] Copies of NUREG–0170 may be purchased from the Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013–7082. Copies are also available from the National Technical Information Service, 5285 Port Royal Road, Springfield, VA 22161. A copy is also available for inspection and copying for a fee in the NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC.

radiation dose hypothesis, resulted in a maximum of 1.7 genetic effects and 1.2 latent cancer effects per year. More than half this impact resulted from shipment of medical-use radioactive materials. Accident related impacts were established at a maximum of one genetic effect and one latent cancer fatality for 200 years of transporting radioactive materials. The principal nonradiological impacts were found to be two injuries per year, and less than one accidental death per 4 years. In contrast, non-accident related radiation exposures associated with this rulemaking would be increased by 0.75 person-Sv/y (75.0 person-rem/y), whereas accident related impacts would be decreased by approximately 0.006 person-Sv/y (0.6 person-rem/y). Nonradiological traffic injuries would be increased by 0.06 per year and nonradiological traffic deaths by 0.003 per year (less than 1 accidental death per 330 years). These impacts are judged to be insignificant compared with the baseline impacts established in NUREG–0170.

The environmental assessment and finding of no significant impact on which this determination is based are available, for inspection, at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC. Single copies of the environmental assessment and finding of no significant impact are also available from the contact listed under the Addresses heading.

Paperwork Reduction Act Statement

This final rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). These requirements were approved by the Office of Management and Budget, Approval Number 3150–0008.

The public reporting burden for this collection of information is estimated to average 7 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch (T–6F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB–10202, (3150–0008), Office of Management and Budget, Washington, D.C. 20503.

Regulatory Analysis

The NRC has prepared a regulatory analysis on this final regulation. The analysis examines the costs and benefits of the alternatives considered by NRC. Interested persons may examine a copy of the regulatory analysis at the NRC Public Document Room at 2120 L Street NW. (Lower Level), Washington, DC. Single copies of the analysis may be obtained from the contact listed under the Addresses heading.

Regulatory Flexibility Act Certification

In accordance with the Regulatory Flexibility Act of 1980 (5 U.S.C. 605(b)), the Commission certifies that this rule does not have a significant economic impact on a substantial number of small entities. This final rule affects NRC licensees, including operators of nuclear power plants, who transport or deliver to a carrier, for transport, relatively large quantities of radioactive material, in a single package. These companies do not generally fall within the scope of the definition of ''small entities'' set forth in the Regulatory Flexibility Act or the size standards adopted by the NRC (10 CFR 2.810).

Backfit Analysis

The Commission has determined that the backfit rule does not apply to the Part 71 final rule because the final rule is not a backfit under 10 CFR Part 50.109. However, NRC analyzed the accident-resistant packaging requirement for the specified LSA shipments and found that there is an increase in overall protection to be derived from the requirement and that direct and indirect costs of implementation are justified in view of this increased protection.

The factors normally considered in a backfit analysis are evaluated in the ''Regulatory Analysis of Changes to 10 CFR Part 71—NRC Regulations on Packaging and Transportation of Radioactive Material,'' dated April 1995. That evaluation shows very small changes in accident risks as a result of the adoption of the revision, but some reduction in maximum consequences given an accident. The evaluation shows broad improvement in NRC regulatory consistency with IAEA, at an initial cost of $1.375 million to industry, and continual annual costs to industry of $1.0 million (See Table S.1 of Regulatory Analysis). NRC costs are estimated at $0.463 million.

The continuing costs are associated with the addition of new limits on the quantity of LSA radioactive material allowed in a single transportation package. Internationally, a new limit is considered to be a necessary safety requirement to limit the consequences of a severe transportation accident involving LSA material.

The one-time costs are chiefly associated with industry upgrading of its package safety analyses to include the proposed new accident crush and immersion tests and with NRC review of those new analyses. The estimated costs are overstated because of the assumption that all licensees using packages approved under earlier regulatory standards would take immediate steps to upgrade the package analyses so the package approvals would reflect approval, under the latest revised standards. Although that is a prudent assumption, absent any reasonable basis for predicting actual licensee reaction, there is little reason licensees would take any immediate action to upgrade their package approvals. Both domestic and international regulations are based on the responsible agency's confidence that packages built to a design approved under earlier standards are adequately safe for continued use, although new package construction to that design would be limited, and international use requires approval by all countries through which the package is to be transported. In actual practice, some package approvals would never be upgraded. Those that would be upgraded would be done over a period of several years as guidance and experience in upgrading become available.

Although the regulatory analysis shows a small reduction in accident risks from the amendments to this rule and some reduction in maximum consequences given an accident, the primary benefit of this rulemaking is to achieve consistency in radioactive material transportation regulations between the United States and the rest of the world. This consistency would not only facilitate the free movement of radioactive materials between countries for medical, research, industrial, and nuclear fuel cycle purposes, but it would also contribute to safety by concentrating the efforts of the world's experts on a single set of safety standards and guidance (those of the IAEA) from which individual countries could develop their domestic regulations. In addition, the accident experience of every country that bases its domestic regulations on those of the IAEA could be applied to every other country with consistent regulations to improve its safety program.

In summary, the effort to make U.S. regulations compatible with those of the IAEA provides major benefits including a substantial increase in the overall protection of the public health and safety, and it is associated with short-term and relatively minor costs that are justified in view of this increased protection. This effort is associated with ongoing costs, but the new limit is considered to be a justified safety requirement, to limit the consequences of a severe transportation accident involving LSA material.

List of Subjects in 10 CFR Part 71

Criminal penalties, Hazardous materials transportation, Nuclear materials, Packaging and containers, Reporting and recordkeeping requirements.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C. 552 and 553, 10 CFR part 71 is revised to read as follows:

PART 71—PACKAGING AND TRANSPORTATION OF RADIOACTIVE MATERIAL

Authority: Secs. 53, 57, 62, 63, 81, 161, 182, 183, 68 Stat. 930, 932, 933, 935, 948, 953, 954, as amended, sec. 1701, 106 Stat. 2951, 2952, 2953 (42 U.S.C. 2073, 2077, 2092, 2093, 2111, 2201, 2232, 2233, 2297f); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246 (42 U.S.C. 5841, 5842, 5846).

Section 71.97 also issued under sec. 301, Pub. L. 96–295, 94 Stat. 789–790.

Subpart A—General Provisions

§ 71.0 Purpose and scope.

(a) This part establishes—

(1) Requirements for packaging, preparation for shipment, and transportation of licensed material; and

(2) Procedures and standards for NRC approval of packaging and shipping procedures for fissile material and for a quantity of other licensed material in excess of a Type A quantity.

(b) The packaging and transport of licensed material are also subject to other parts of this chapter (e.g., 10 CFR parts 20, 21, 30, 40, 70, and 73) and to the regulations of other agencies (e.g., the U.S. Department of Transportation (DOT) and the U.S. Postal Service [1]) having jurisdiction over means of transport. The requirements of this part are in addition to, and not in substitution for, other requirements.

(c) The regulations in this part apply to any licensee authorized by specific or general license issued by the Commission to receive, possess, use, or transfer licensed material, if the licensee delivers that material to a carrier for transport, transports the material outside the site of usage as specified in the NRC license, or transports that material on public highways. No provision of this part authorizes possession of licensed material.

(d) Exemptions from the requirement for license in § 71.3 are specified in § 71.10. General licenses for which no NRC package approval is required are issued in §§ 71.14 through 71.24. The general license in § 71.12 requires that an NRC certificate of compliance or other package approval be issued for the package to be used under the general license. Application for package

[1] Postal Service Manual (Domestic Mail Manual), section 124.3, which is incorporated by reference at 39 CFR 111.1.

approval must be completed in accordance with subpart D of this part, demonstrating that the design of the package to be used satisfies the package approval standards contained in subpart E of this part, as related to the tests of subpart F of this part. The transport of licensed material or delivery of licensed material to a carrier for transport is subject to the operating controls and procedures requirements of subpart G of this part, to the quality assurance requirements of subpart H of this part, and to the general provisions of subpart A of this part, including DOT regulations referenced in § 71.5.

(e) The regulations in this part apply to any person required to obtain a certificate of compliance or an approved compliance plan pursuant to part 76 of this chapter if the person delivers radioactive material to a common or contract carrier for transport or transports the material outside the confines of the person's plant or other authorized place of use.

§ 71.1 Communications and records.

(a) All communications concerning the regulations in this part should be addressed to the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, or may be delivered in person, at the Commission offices, at 11545 Rockville Pike, Rockville, Maryland.

(b) Each record required by this part must be legible throughout the retention period specified by each Commission regulation. The record may be the original or a reproduced copy or a microform provided that the copy or microform is authenticated by authorized personnel and that the microform is capable of producing a clear copy throughout the required retention period. The record may also be stored in electronic media with the capability for producing legible, accurate, and complete records during the required retention period. Records such as letters, drawings, specifications, must include all pertinent information such as stamps, initials, and signatures. The licensee shall maintain adequate safeguards against tampering with and loss of records.

§ 71.2 Interpretations.

Except as specifically authorized by the Commission in writing, no interpretation of the meaning of the regulations in this part by any officer or employee of the Commission, other than a written interpretation by the General Counsel, will be recognized to be binding upon the Commission.

§ 71.3 Requirement for license.

Except as authorized in a general license or a specific license issued by the Commission, or as exempted in this part, no licensee may—

(a) Deliver licensed material to a carrier for transport; or

(b) Transport licensed material.

§ 71.4 Definitions.

The following terms are as defined here for the purpose of this part. To ensure compatibility with international transportation standards, all limits in this part are given in terms of dual units: The International System of Units (SI) followed or preceded by U.S. standard or customary units. The U.S. customary units are not exact equivalents, but are rounded to a convenient value, providing a functionally equivalent unit. For the purpose of this part, either unit may be used.

A_1 means the maximum activity of special form radioactive material permitted in a Type A package. A_2 means the maximum activity of radioactive material, other than special form, LSA and SCO material, permitted in a Type A package. These values are either listed in Appendix A of this part, Table A–1, or may be derived in accordance with the procedure prescribed in Appendix A of this part.

Carrier means a person engaged in the transportation of passengers or property by land or water as a common, contract, or private carrier, or by civil aircraft.

Certificate holder means a person who has been issued a certificate of compliance or other package approval by the Commission.

Close reflection by water means immediate contact by water of sufficient thickness for maximum reflection of neutrons.

Containment system means the assembly of components of the packaging intended to retain the radioactive material during transport.

Conveyance means:

(1) *For transport by public highway or rail* any transport vehicle or large freight container;

(2) *For transport by water* any vessel, or any hold, compartment, or defined deck area of a vessel including any transport vehicle on board the vessel; and

(3) *For transport by aircraft* any aircraft.

Exclusive use means the sole use by a single consignor of a conveyance for which all initial, intermediate, and final loading and unloading are carried out in accordance with the direction of the consignor or consignee. The consignor and the carrier must ensure that any loading or unloading is performed by personnel having radiological training and resources appropriate for safe handling of the consignment. The consignor must issue specific instructions, in writing, for maintenance of exclusive use shipment controls, and include them with the shipping paper information provided to the carrier by the consignor.

Fissile material means plutonium-238, plutonium-239, plutonium-241, uranium-233, uranium-235, or any combination of these radionuclides. Unirradiated natural uranium and depleted uranium, and natural uranium or depleted uranium that has been irradiated in thermal reactors only are not included in this definition. Certain exclusions from fissile material controls are provided in § 71.53.

Licensed material means by-product, source, or special nuclear material received, possessed, used, or transferred under a general or specific license issued by the Commission pursuant to the regulations in this chapter.

Low Specific Activity (LSA) material means radioactive material with limited specific activity that satisfies the descriptions and limits set forth below. Shielding materials surrounding the LSA material may not be considered in determining the estimated average specific activity of the package contents. LSA material must be in one of three groups:

(1) *LSA–I.*

(i) Ores containing only naturally occurring radionuclides (e.g., uranium, thorium) and uranium or thorium concentrates of such ores; or

(ii) Solid unirradiated natural uranium or depleted uranium or natural thorium or their solid or liquid compounds or mixtures; or

(iii) Radioactive material, other than fissile material, for which the A_2 value is unlimited; or

(iv) Mill tailings, contaminated earth, concrete, rubble, other debris, and activated material in which the radioactive material is essentially uniformly distributed, and the average specific activity does not exceed 10^{-6} A_2/g.

(2) LSA–II.

(i) Water with tritium concentration up to 0.8 TBq/liter (20.0 Ci/liter); or

(ii) Material in which the radioactive material is essentially uniformly distributed, and the average specific activity does not exceed 10^{-4} A_2/g for solids and gases, and 10^{-5} A_2/g for liquids.

(3) *LSA–III.* Solids (e.g., consolidated wastes, activated materials) in which:

(i) The radioactive material is essentially uniformly distributed throughout a solid or a collection of solid objects, or is essentially uniformly distributed in a solid compact binding agent (such as concrete, bitumen, ceramic, etc.);

(ii) The radioactive material is relatively insoluble, or it is intrinsically contained in a relatively insoluble material, so that, even under loss of packaging, the loss of radioactive material per package by leaching, when placed in water for 7 days, would not exceed 0.1 A_2; and

(iii) The average specific activity of the solid does not exceed 2 x 10^{-3} A_2/g.

Low toxicity alpha emitters means natural uranium, depleted uranium, natural thorium; uranium-235, uranium-238, thorium-232, thorium-228 or thorium-230 when contained in ores or physical or chemical concentrates or tailings; or alpha emitters with a half-life of less than 10 days.

Maximum normal operating pressure means the maximum gauge pressure that would develop in the containment system in a period of 1 year under the heat condition specified in § 71.71(c)(1), in the absence of venting, external cooling by an ancillary system, or operational controls during transport.

Natural thorium means thorium with the naturally occurring distribution of thorium isotopes (essentially 100 weight percent thorium-232).

Normal form radioactive material means radioactive material that has not been demonstrated to qualify as ''special form radioactive material.''

Optimum interspersed hydrogenous moderation means the presence of hydrogenous material between packages to such an extent that the maximum nuclear reactivity results.

Package means the packaging together with its radioactive contents as presented for transport.

(1) *Fissile material package* means a fissile material packaging together with its fissile material contents.

(2) *Type B package* means a Type B packaging together with its radioactive contents. On approval, a Type B package design is designated by NRC as B(U) unless the package has a maximum normal operating pressure of more than 700 kPa (100 lb/in^2) gauge or a pressure relief device that would allow the release of radioactive material to the environment under the tests specified in § 71.73 (hypothetical accident conditions), in which case it will receive a designation B(M). B(U) refers to the need for unilateral approval of international shipments; B(M) refers to the need for multilateral approval of international shipments. There is no distinction made in how packages with these designations may be used in domestic transportation. To determine their distinction for international transportation, see DOT regulations in 49 CFR Part 173. A Type B package approved before September 6, 1983, was designated only as Type B. Limitations on its use are specified in § 71.13.

Packaging means the assembly of components necessary to ensure compliance with the packaging requirements of this part. It may consist of one or more receptacles, absorbent materials, spacing structures, thermal insulation, radiation shielding, and devices for cooling or absorbing mechanical shocks. The vehicle, tie-down system, and auxiliary equipment may be designated as part of the packaging.

Special form radioactive material means radioactive material that satisfies the following conditions:

(1) It is either a single solid piece or is contained in a sealed capsule that can be opened only by destroying the capsule;

(2) The piece or capsule has at least one dimension not less than 5 mm (0.2 in); and

(3) It satisfies the requirements of § 71.75. A special form encapsulation designed in accordance with the requirements of § 71.4 in effect on June 30, 1983, (see 10 CFR part 71, revised as of January 1, 1983), and constructed before July 1, 1985, and a special form encapsulation designed in accordance with the requirements of § 71.4 in effect on March 31, 1996, (see 10 CFR part 71, revised as of January 1, 1983), and constructed before April 1, 1998, may continue to be used. Any other special form encapsulation must meet the specifications of this definition.

Specific activity of a radionuclide means the radioactivity of the radionuclide per unit mass of that nuclide. The specific activity of a material in which the radionuclide is essentially uniformly distributed is the radioactivity per unit mass of the material.

State means a State of the United States, the District of Columbia, the Commonwealth of Puerto Rico, the Virgin Islands, Guam, American Samoa, and the Commonwealth of the Northern Mariana Islands.

Surface Contaminated Object (SCO) means a solid object that is not itself classed as radioactive material, but which has radioactive material distributed on any of its surfaces. SCO must be in one of two groups with surface activity not exceeding the following limits:

(1) SCO–I: A solid object on which:

(i) The non-fixed contamination on the accessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 4 Bq/cm^2 (10^{-4} microcurie/cm^2) for beta and gamma and low toxicity alpha emitters, or 0.4 Bq/cm^2 (10^{-5} microcurie/cm^2) for all other alpha emitters;

(ii) The fixed contamination on the accessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 4x10^4 Bq/cm^2 (1.0 microcurie/cm^2) for beta and gamma and low toxicity alpha emitters, or 4x10^3 Bq/cm^2 (0.1 microcurie/cm^2) for all other alpha emitters; and

(iii) The non-fixed contamination plus the fixed contamination on the inaccessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 4x10^4 Bq/cm^2 (1 microcurie/cm^2) for beta and gamma and low toxicity alpha emitters, or 4x10^3 Bq/cm^2 (0.1 microcurie/cm^2) for all other alpha emitters.

(2) SCO–II: A solid object on which the limits for SCO–I are exceeded and on which:

(i) The non-fixed contamination on the accessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 400 Bq/cm^2 (10^{-2} microcurie/cm^2) for beta and gamma and low toxicity alpha emitters or 40 Bq/cm^2 (10^{-3} microcurie/cm^2) for all other alpha emitters;

(ii) The fixed contamination on the accessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 8×10^5 Bq/cm^2 (20 microcuries/cm^2) for beta and gamma and low toxicity alpha emitters, or 8×10^4 Bq/cm^2 (2 microcuries/cm^2) for all other alpha emitters; and

(iii) The non-fixed contamination plus the fixed contamination on the inaccessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 8×10^5 Bq/cm^2 (20 microcuries/cm^2) for beta and gamma and low toxicity alpha emitters, or 8×10^4 Bq/cm^2 (2 microcuries/cm^2) for all other alpha emitters.

Transport index means the dimensionless number (rounded up to the next tenth) placed on the label of a package, to designate the degree of control to be exercised by the carrier during transportation. The transport index is determined as follows:

(1) For non-fissile material packages, the number determined by multiplying the maximum radiation level in millisievert (mSv) per hour at one meter (3.3 ft) from the external surface of the package by 100 (equivalent to the

maximum radiation level in millirem per hour at one meter (3.3 ft)); or

(2) For fissile material packages, the number determined by multiplying the maximum radiation level in millisievert per hour at one meter (3.3 ft) from the external surface of the package by 100 (equivalent to the maximum radiation level in millirem per hour at one meter (3.3 ft)), or, for criticality control purposes, the number obtained as described in § 71.59, whichever is larger.

Type A quantity means a quantity of radioactive material, the aggregate radioactivity of which does not exceed A_1 for special form radioactive material, or A_2, for normal form radioactive material, where A_1 and A_2 are given in Table A–1 of this part, or may be determined by procedures described in Appendix A of this part.

Type B quantity means a quantity of radioactive material greater than a Type A quantity.

Uranium—natural, depleted, enriched

(1) *Natural uranium* means uranium with the naturally occurring distribution of uranium isotopes (approximately 0.711 weight percent uranium-235, and the remainder by weight essentially uranium-238).

(2) *Depleted uranium* means uranium containing less uranium-235 than the naturally occurring distribution of uranium isotopes.

(3) *Enriched uranium* means uranium containing more uranium-235 than the naturally occurring distribution of uranium isotopes.

§ 71.5 Transportation of licensed material.

(a) Each licensee who transports licensed material outside the site of usage, as specified in the NRC license, or where transport is on public highways, or who delivers licensed material to a carrier for transport, shall comply with the applicable requirements of the DOT regulations in 49 CFR parts 170 through 189 appropriate to the mode of transport.

(1) The licensee shall particularly note DOT regulations in the following areas:

(i) Packaging—49 CFR part 173: Subparts A and B and I.

(ii) Marking and labeling—49 CFR part 172: Subpart D, §§ 172.400 through 172.407, §§ 172.436 through 172.440, and subpart E.

(iii) Placarding—49 CFR part 172: Subpart F, especially §§ 172.500 through 172.519, 172.556, and appendices B and C.

(iv) Accident reporting—49 CFR part 171: §§ 171.15 and 171.16.

(v) Shipping papers and emergency information—49 CFR part 172: Subparts C and G.

(vi) Hazardous material employee training—49 CFR part 172: Subpart H.

(vii) Hazardous material shipper/carrier registration—49 CFR part 107: Subpart G.

(2) The licensee shall also note DOT regulations pertaining to the following modes of transportation:

(i) Rail—49 CFR part 174: Subparts A through D and K.

(ii) Air—49 CFR part 175.

(iii) Vessel—49 CFR part 176: Subparts A through F and M.

(iv) Public Highway—49 CFR part 177 and parts 390 through 397.

(b) If DOT regulations are not applicable to a shipment of licensed material, the licensee shall conform to the standards and requirements of the DOT specified in paragraph (a) of this section to the same extent as if the shipment or transportation were subject to DOT regulations. A request for modification, waiver, or exemption from those requirements, and any notification referred to in those requirements, must be filed with, or made to, the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

Subpart B—Exemptions

§ 71.6 Information collection requirements: OMB approval.

(a) The Nuclear Regulatory Commission has submitted the information collection requirements contained in this part to the Office of Management and Budget (OMB) for approval, as required by the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 *et seq.*). OMB has approved the information collection requirements contained in this part, under control number 3150–0008.

(b) The approved information collection requirements contained in this part appear in §§ 71.5, 71.6a, 71.7, 71.12, 71.13, 71.31, 71.33, 71.35, 71.37, 71.38, 71.39, 71.47, 71.85, 71.87, 71.89, 71.91, 71.93, 71.95, 71.97, 71.101, 71.103, 71.105, 71.107, 71.109, 71.111, 71.113, 71.115, 71.117, 71.119, 71.121, 71.123, 71.125, 71.127, 71.129, 71.131, 71.133, 71.135, and 71.137.

§ 71.7 Completeness and accuracy of information.

(a) Information provided to the Commission by an applicant for a license, or by a licensee, or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee must be complete and accurate in all material respects.

(b) Each applicant or licensee shall notify the Commission of information identified by the applicant or licensee as having, for the regulated activity, a significant implication for public health and safety or common defense and security. An applicant or licensee violates this requirement only if the applicant or licensee fails to notify the Commission of information that the applicant or licensee has identified as having a significant implication for public health and safety or common defense and security. Notification must be provided to the Administrator of the appropriate Regional Office within two working days of identifying the information. This requirement is not applicable to information that is already required to be provided to the Commission by other reporting or updating requirements.

§ 71.8 Specific exemptions.

On application of any interested person or on its own initiative, the Commission may grant any exemption from the requirements of the regulations in this part that it determines is authorized by law and will not endanger life or property nor the common defense and security.

§ 71.9 Exemption of physicians.

Any physician licensed by a State to dispense drugs in the practice of medicine is exempt from § 71.5 with respect to transport by the physician of licensed material for use in the practice of medicine. However, any physician operating under this exemption must be licensed under 10 CFR part 35 or the equivalent Agreement State regulations.

§ 71.10 Exemption for low-level materials.

(a) A licensee is exempt from all requirements of this part with respect to shipment or carriage of a package containing radioactive material having a specific activity not greater than 70 Bq/g (0.002 μCi/g).

(b) A licensee is exempt from all requirements of this part, other than § 71.5 and § 71.88, with respect to shipment or carriage of the following packages, provided the packages contain no fissile material, or the fissile material exemption standards of § 71.53 are satisfied:

(1) A package containing no more than a Type A quantity of radioactive material;

(2) A package in which the only radioactive material is low specific activity (LSA) material or surface contaminated objects (SCO), provided the external radiation level at 3 m from the unshielded material or objects does not exceed 10 mSv/h (1 rem/h); or

(3) A package transported within locations within the United States which contains only americium or plutonium in special form with an aggregate radioactivity not to exceed 20 curies.

(c) A licensee is exempt from all requirements of this part, other than §§ 71.5 and 71.88, with respect to shipment or carriage of low-specific-activity (LSA) material in group LSA–I, or surface contaminated objects (SCOs) in group SCO–I.

§ 71.11 [Reserved]

Subpart C—General Licenses

§ 71.12 General license: NRC-approved package.

(a) A general license is hereby issued to any licensee of the Commission to transport, or to deliver to a carrier for transport, licensed material in a package for which a license, certificate of compliance, or other approval has been issued by the NRC.

(b) This general license applies only to a licensee who has a quality assurance program approved by the Commission as satisfying the provisions of subpart H of this part.

(c) This general license applies only to a licensee who—

(1) Has a copy of the certificate of compliance, or other approval of the package, and has the drawings and other documents referenced in the approval relating to the use and maintenance of the packaging and to the actions to be taken before shipment;

(2) Complies with the terms and conditions of the license, certificate, or other approval, as applicable, and the applicable requirements of subparts A, G, and H of this part; and

(3) Submits in writing to the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, before the licensee's first use of the package, the licensee's name and license number and the package identification number specified in the package approval.

(d) This general license applies only when the package approval authorizes use of the package under this general license.

(e) For a Type B or fissile material package, the design of which was approved by NRC before April 1, 1996, the general license is subject to the additional restrictions of § 71.13.

§ 71.13 Previously approved package.

(a) A Type B package previously approved by NRC but not designated as B(U) or B(M) in the identification number of the NRC Certificate of Compliance, may be used under the general license of § 71.12 with the following additional conditions:

(1) Fabrication of the packaging was satisfactorily completed by August 31, 1986, as demonstrated by application of its model number in accordance with § 71.85(c);

(2) A package used for a shipment to a location outside the United States is subject to multilateral approval, as defined in DOT regulations at 49 CFR 173.403; and

(3) A serial number that uniquely identifies each packaging which conforms to the approved design is assigned to, and legibly and durably marked on, the outside of each packaging.

(b) A Type B(U) package, a Type B(M) package, a low specific activity (LSA) material package or a fissile material package, previously approved by the NRC but without the designation ''-85'' in the identification number of the NRC Certificate of Compliance, may be used under the general license of § 71.12 with the following additional conditions:

(1) Fabrication of the package is satisfactorily completed by April 1, 1999 as demonstrated by application of its model number in accordance with § 71.85(c);

(2) A package used for a shipment to a location outside the United States is subject to multilateral approval as defined in DOT regulations at 49 CFR 173.403; and

(3) A serial number which uniquely identifies each packaging which conforms to the approved design is assigned to and legibly and durably marked on the outside of each packaging.

(c) NRC will approve modifications to the design and authorized contents of a Type B package, or a fissile material package, previously approved by NRC, provided—

(1) The modifications of a Type B package are not significant with respect to the design, operating characteristics, or safe performance of the containment system, when the package is subjected to the tests specified in §§ 71.71 and 71.73;

(2) The modifications of a fissile material package are not significant, with respect to the prevention of criticality, when the package is subjected to the tests specified in §§ 71.71 and 71.73; and

(3) The modifications to the package satisfy the requirements of this part.

(d) NRC will revise the package identification number to designate previously approved package designs as B(U), B(M), AF, BF, or A as appropriate, and with the identification number suffix ''-85'' after receipt of an application demonstrating that the design meets the requirements of this part.

§ 71.14 General license: DOT specification container.

(a) A general license is issued to any licensee of the Commission to transport, or to deliver to a carrier for transport, licensed material in a specification container for fissile material or for a Type B quantity of radioactive material as specified in DOT regulations at 49 CFR parts 173 and 178.

(b) This general license applies only to a licensee who has a quality assurance program approved by the Commission as satisfying the provisions of subpart H of this part.

(c) This general license applies only to a licensee who—

(1) Has a copy of the specification; and

(2) Complies with the terms and conditions of the specification and the applicable requirements of subparts A, G, and H of this part.

(d) This general license is subject to the limitation that the specification container may not be used for a shipment to a location outside the United States, except by multilateral approval, as defined in DOT regulations at 49 CFR 173.403.

§ 71.16 General License: Use of foreign approved package.

(a) A general license is issued to any licensee of the Commission to transport, or to deliver to a carrier for transport, licensed material in a package the design of which has been approved in a foreign national competent authority certificate that has been revalidated by DOT as meeting the applicable requirements of 49 CFR 171.12.

(b) Except as otherwise provided in this section, the general license applies only to a licensee who has a quality assurance program approved by the Commission as satisfying the applicable provisions of subpart H of this part.

(c) This general license applies only to shipments made to or from locations outside the United States.

(d) This general license applies only to a licensee who—

(1) Has a copy of the applicable certificate, the revalidation, and the drawings and other documents referenced in the certificate, relating to the use and maintenance of the packaging and to the actions to be taken before shipment; and

(2) Complies with the terms and conditions of the certificate and revalidation, and with the applicable requirements of subparts A, G, and H of

this part. With respect to the quality assurance provisions of subpart H of this part, the licensee is exempt from design, construction, and fabrication considerations.

§ 71.18 General license: Fissile material, limited quantity per package.

(a) A general license is issued to any licensee of the Commission to transport fissile material, or to deliver fissile material to a carrier for transport, without complying with the package standards of subparts E and F of this part, if the material is shipped in accordance with this section.

(b) The general license applies only to a licensee who has a quality assurance program approved by the Commission as satisfying the provisions of subpart H of this part.

(c) This general license applies only when a package contains no more than a Type A quantity of radioactive material, including only one of the following:

(1) Up to 40 g of uranium-235;

(2) Up to 30 g of uranium-233;

(3) Up to 25 g of the fissile radionuclides of plutonium, except that for encapsulated plutonium-beryllium neutron sources in special form, an A_1 quantity of plutonium may be present; or

(4) A combination of fissile radionuclides in which the sum of the ratios of the amount of each radionuclide to the corresponding maximum amounts in paragraphs (c)(1), (2), and (3) of this section does not exceed unity.

(d) (1) This general license applies only when, except as specified below for encapsulated plutonium-beryllium sources, a package containing more than 15 g of fissile radionuclides is labeled with a transport index not less than the number given by the following equation, where the package contains x grams of uranium-235, y grams of uranium-233, and z grams of the fissile radionuclides of plutonium:

Minimum Transport Index =
$(0.40x+0.67y+z)\ (1-15\ ).x+y+z$

(2) For a package in which the only fissile material is in the form of encapsulated plutonium-beryllium neutron sources in special form, the transport index based on criticality considerations may be taken as 0.026 times the number of grams of the fissile radionuclides of plutonium in excess of 15 g. In all cases, the transport index must be rounded up to one decimal place and may not exceed 10.0.

§ 71.20 General license: Fissile material, limited moderator per package.

(a) A general license is issued to any licensee of the Commission to transport fissile material, or to deliver fissile material to a carrier for transport, without complying with the package standards of subparts E and F of this part if the material is shipped in accordance with this section.

(b) The general license applies only to a licensee who has a quality assurance program approved by the Commission as satisfying the provisions of subpart H of this part.

(c) This general license applies only when—

(1) The package contains no more than a Type A quantity of radioactive material;

(2) Neither beryllium nor hydrogenous material enriched in deuterium is present;

(3) The total mass of graphite present does not exceed 7.7 times the total mass of uranium-235 plus plutonium;

(4) Substances having a higher hydrogen density than water (e.g., certain hydrocarbon oils), are not present, except that polyethylene may be used for packing or wrapping;

(5) Uranium-233 is not present, and the amount of plutonium does not exceed 1 percent of the amount of uranium-235;

(6) The amount of uranium-235 is limited as follows:

(i) If the fissile radionuclides are not uniformly distributed, the maximum amount of uranium-235 per package may not exceed the value given in Table I of this part; or

(ii) If the fissile radionuclides are distributed uniformly (i.e., cannot form a lattice arrangement within the packaging), the maximum amount of uranium-235 per package may not exceed the value given in Table II of this part; and

(7) The transport index of each package, based on criticality considerations, is taken as 10 times the number of grams of uranium-235 in the package divided by the maximum allowable number of grams per package in accordance with Table I or Table II of this part, as applicable.

TABLE I.—PERMISSIBLE MASS OF URANIUM-235 PER FISSILE MATERIAL PACKAGE, APPLICABLE TO § 71.20(c)(6)(i)

[Nonuniform distribution]

Uranium enrichment in weight percent of uranium-235 not exceeding	Permissible maximum grams of uranium-235 per package
24	40
20	42
15	45
11	48
10	51
9.5	52
9	54
8.5	55
8	57
7.5	59
7	60
6.5	62
6	65
5.5	68
5	72
4.5	76
4	80
3.5	88
3	100
2.5	120
2	164
1.5	272
1.35	320
1	680
0.92	1,200

TABLE II.—PERMISSIBLE MASS OF URANIUM-235 PER FISSILE MATERIAL PACKAGE, APPLICABLE TO § 71.20(c)(6)(ii)

[Uniform Distribution]

Uranium enrichment in weight percent of uranium-235 not exceeding	Permissible maximum grams of uranium-235 per package
4	84
3.5	92
3	112
2.5	148
2	240
1.5	560
1.35	800

§ 71.22 General license: Fissile material, limited quantity, controlled shipment.

(a) A general license is issued to any licensee of the Commission to transport fissile material, or to deliver fissile material to a carrier for transport, without complying with the package standards of Subparts E and F of this part, if limited material is shipped in accordance with this section.

(b) The general license applies only to a licensee who has a quality assurance program approved by the Commission as satisfying the provisions of Subpart H of this part.

(c) This general license applies only when a package contains no more than a Type A quantity of radioactive material and no more than 400 g total of the fissile radionuclides of plutonium encapsulated as plutonium-beryllium neutron sources in special form.

(d) This general license applies only when the fissile radionuclides in the shipment exceed none of the following:

(1) 500 g of uranium-235;

(2) 300 g total of uranium-233, and the fissile radionuclides of plutonium;

(3) A total quantity of uranium-233, uranium-235, and the fissile radionuclides of plutonium so that the sum of the ratios of the quantity of each radionuclide to the quantity specified in paragraphs (d)(1) and (d)(2) of this section does not exceed unity; or

(4) 2500 g total of the fissile radionuclides of plutonium encapsulated as plutonium-beryllium neutron sources in special form.

(e) This general license applies only when shipment of these packages is made under procedures specifically authorized by DOT, in accordance with 49 CFR part 173 of its regulations, to prevent loading, transport, or storage of these packages with other fissile material shipments.

§ 71.24 General license: Fissile material, limited moderator, controlled shipment.

(a) A general license is issued to any licensee of the Commission to transport fissile material, or to deliver fissile material to a carrier for transport, without complying with the package standards of subparts E and F of this part, if limited material is shipped in accordance with this section.

(b) The general license applies only to a licensee who has a quality assurance program approved by the Commission as satisfying the provisions of subpart H of this part.

(c) This general license applies only when—

(1) No package contains more than a Type A quantity of radioactive material;

(2) The packaging does not incorporate lead shielding exceeding 5 cm in thickness, tungsten shielding, or uranium shielding;

(3) Neither beryllium nor hydrogenous material enriched in deuterium is present;

(4) The total mass of graphite present does not exceed 7.7 times the total mass of uranium-235 and plutonium;

(5) Substances having a higher hydrogen density than water (e.g., certain hydrocarbon oils), are not present, except that polyethylene may be used for packing or wrapping;

(6) For fissile contents containing no uranium-233 and less than 1 percent by weight total plutonium, if the fissile radionuclides are—

(i) Not uniformly distributed, the maximum amount of uranium-235 per consignment does not exceed the value given in Table III of this part; or

(ii) Distributed uniformly and cannot form a lattice arrangement within the packaging, the maximum amount of uranium-235 per shipment does not exceed the value given in Table IV of this part;

(7) For fissile contents containing uranium-233 or more than 1 percent by weight plutonium, the total mass of fissile material per shipment is limited so that the sum of the number of grams of uranium-235 divided by 400, the number of grams of plutonium divided by 225, and the number of grams of uranium-233 divided by 250, does not exceed unity, as expressed in the formula:

$$\frac{\text{grams uranium} - 235}{400\text{ g}} + \frac{\text{grams plutonium}}{225\text{ g}} + \frac{\text{grams uranium} - 233}{250\text{ g}} \leq 1;$$

(8) The transport must be direct to the consignee without any intermediate transit storage; and

(9) Shipment of these packages is made under procedures specifically authorized by DOT in accordance with 49 CFR part 173 of its regulations to prevent loading, transport, or storage of these packages with other fissile material shipments.

TABLE III.—PERMISSIBLE MASS OF URANIUM-235 PER FISSILE MATERIAL SHIPMENT APPLICABLE TO § 71.24(c)(6)(i)

[Nonuniform distribution]

Uranium enrichment in weight percent of uranium-235 not exceeding	Permissible maximum grams of uranium-235 per consignment
20	520
15	560
11	600
10	640
9.5	655
9	675

TABLE III.—PERMISSIBLE MASS OF URANIUM-235 PER FISSILE MATERIAL SHIPMENT APPLICABLE TO § 71.24(c)(6)(i)—Continued

[Nonuniform distribution]

Uranium enrichment in weight percent of uranium-235 not exceeding	Permissible maximum grams of uranium-235 per consignment
8.5	690
8	710
7.5	730
7	750
6.5	780
6	810
5.5	850
5	900
4.5	950
4	1,000
3.5	1,100
3	1,250
2.5	1,500
2	2,050
1.5	3,400
1.35	4,000
1	8,500
0.92	15,000

TABLE IV.—PERMISSIBLE MASS OF URANIUM-235 PER FISSILE MATERIAL SHIPMENT APPLICABLE TO § 71.24(c)(6)(ii)

[Uniform distribution]

Uranium enrichment in weight percent of uranium-235 not exceeding	Permissible maximum grams of uranium-235 per consignment
4	1,050
3.5	1,150
3	1,400
2.5	1,800
2	3,000
1.5	7,000
1.35	10,000

Subpart D—Application for Package Approval

§ 71.31 Contents of application.

(a) An application for an approval under this part must include, for each proposed packaging design, the following information:

(1) A package description as required by § 71.33;

(2) A package evaluation as required by § 71.35; and

(3) A quality assurance program description, as required by § 71.37, or a reference to a previously approved quality assurance program.

(b) Except as provided in § 71.13, an application for modification of a package design, whether for modification of the packaging or authorized contents, must include sufficient information to demonstrate that the proposed design satisfies the package standards in effect at the time the application is filed.

(c) The applicant shall identify any established codes and standards proposed for use in package design, fabrication, assembly, testing, maintenance, and use. In the absence of any codes and standards, the applicant shall describe and justify the basis and rationale used to formulate the package quality assurance program.

§ 71.33 Package description.

The application must include a description of the proposed package in sufficient detail to identify the package accurately and provide a sufficient basis for evaluation of the package. The description must include—

(a) With respect to the packaging—

(1) Classification as Type B(U), Type B(M), or fissile material packaging;

(2) Gross weight;

(3) Model number;

(4) Identification of the containment system;

(5) Specific materials of construction, weights, dimensions, and fabrication methods of—

(i) Receptacles;

(ii) Materials specifically used as nonfissile neutron absorbers or moderators;

(iii) Internal and external structures supporting or protecting receptacles;

(iv) Valves, sampling ports, lifting devices, and tie-down devices; and

(v) Structural and mechanical means for the transfer and dissipation of heat; and

(6) Identification and volumes of any receptacles containing coolant.

(b) With respect to the contents of the package—

(1) Identification and maximum radioactivity of radioactive constituents;

(2) Identification and maximum quantities of fissile constituents;

(3) Chemical and physical form;

(4) Extent of reflection, the amount and identity of nonfissile materials used as neutron absorbers or moderators, and the atomic ratio of moderator to fissile constituents;

(5) Maximum normal operating pressure;

(6) Maximum weight;

(7) Maximum amount of decay heat; and

(8) Identification and volumes of any coolants.

§ 71.35 Package evaluation.

The application must include the following:

(a) A demonstration that the package satisfies the standards specified in subparts E and F of this part;

(b) For a fissile material package, the allowable number of packages that may be transported in the same vehicle in accordance with § 71.59; and

(c) For a fissile material shipment, any proposed special controls and precautions for transport, loading, unloading, and handling and any proposed special controls in case of an accident or delay.

§ 71.37 Quality assurance.

(a) The applicant shall describe the quality assurance program (see Subpart H of this part) for the design, fabrication, assembly, testing, maintenance, repair, modification, and use of the proposed package.

(b) The applicant shall identify any specific provisions of the quality assurance program that are applicable to the particular package design under consideration, including a description of the leak testing procedures.

§ 71.38 Renewal of a certificate of compliance or quality assurance program approval.

(a) Except as provided in paragraph (b) of this section, each Certificate of Compliance or Quality Assurance Program Approval expires at the end of the day, in the month and year stated in the approval.

(b) In any case in which a person, not less than 30 days before the expiration of an existing Certificate of Compliance or Quality Assurance Program Approval issued pursuant to the part, has filed an application in proper form for renewal of either of those approvals, the existing Certificate of Compliance or Quality Assurance Program Approval for which the renewal application was filed shall not be deemed to have expired until final action on the application for renewal has been taken by the Commission.

(c) In applying for renewal of an existing Certificate of Compliance or Quality Assurance Program Approval, an applicant may be required to submit a consolidated application that incorporates all changes to its program that, are incorporated by reference in the existing approval or certificate, into as few referenceable documents as reasonably achievable.

§ 71.39 Requirement for additional information.

The Commission may at any time require additional information in order to enable it to determine whether a license, certificate of compliance, or other approval should be granted, renewed, denied, modified, suspended, or revoked.

Subpart E—Package Approval Standards

§ 71.41 Demonstration of compliance.

(a) The effects on a package of the tests specified in § 71.71 (''Normal conditions of transport''), and the tests specified in § 71.73 (''Hypothetical accident conditions''), and § 71.61 (Special requirement for irradiated nuclear fuel shipments''), must be evaluated by subjecting a specimen or scale model to a specific test, or by another method of demonstration acceptable to the Commission, as appropriate for the particular feature being considered.

(b) Taking into account the type of vehicle, the method of securing or attaching the package, and the controls to be exercised by the shipper, the Commission may permit the shipment to be evaluated together with the transporting vehicle.

(c) Environmental and test conditions different from those specified in §§ 71.71 and 71.73 may be approved by the Commission if the controls proposed to be exercised by the shipper are demonstrated to be adequate to provide equivalent safety of the shipment.

§ 71.43 General standards for all packages.

(a) The smallest overall dimension of a package may not be less than 10 cm (4 in).

(b) The outside of a package must incorporate a feature, such as a seal, that is not readily breakable and that, while intact, would be evidence that the package has not been opened by unauthorized persons.

(c) Each package must include a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package.

(d) A package must be made of materials and construction that assure that there will be no significant chemical, galvanic, or other reaction among the packaging components, among package contents, or between the packaging components and the package contents, including possible reaction resulting from inleakage of water, to the maximum credible extent. Account must be taken of the behavior of materials under irradiation.

(e) A package valve or other device, the failure of which would allow radioactive contents to escape, must be protected against unauthorized operation and, except for a pressure relief device, must be provided with an enclosure to retain any leakage.

(f) A package must be designed, constructed, and prepared for shipment so that under the tests specified in § 71.71 (''Normal conditions of transport'') there would be no loss or dispersal of radioactive contents, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging.

(g) A package must be designed, constructed, and prepared for transport so that in still air at 38°C (100°F) and in the shade, no accessible surface of a package would have a temperature exceeding 50°C (122°F) in a nonexclusive use shipment, or 85°C (185°F) in an exclusive use shipment.

(h) A package may not incorporate a feature intended to allow continuous venting during transport.

§ 71.45 Lifting and tie-down standards for all packages.

(a) Any lifting attachment that is a structural part of a package must be designed with a minimum safety factor of three against yielding when used to lift the package in the intended manner, and it must be designed so that failure of any lifting device under excessive load would not impair the ability of the package to meet other requirements of this subpart. Any other structural part of the package that could be used to lift the package must be capable of being rendered inoperable for lifting the package during transport, or must be designed with strength equivalent to that required for lifting attachments.

(b) Tie-down devices:

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

(2) Any other structural part of the package that could be used to tie down the package must be capable of being rendered inoperable for tying down the package during transport, or must be designed with strength equivalent to that required for tie-down devices.

(3) Each tie-down device that is a structural part of a package must be designed so that failure of the device under excessive load would not impair the ability of the package to meet other requirements of this part.

§ 71.47 External radiation standards for all packages.

(a) Except as provided in paragraph (b) of this section, each package of radioactive materials offered for transportation must be designed and prepared for shipment so that under conditions normally incident to transportation the radiation level does not exceed 2 mSv/h (200 mrem/h) at any point on the external surface of the package, and the transport index does not exceed 10.

(b) A package that exceeds the radiation level limits specified in paragraph (a) of this section must be transported by exclusive use shipment only, and the radiation levels for such shipment must not exceed the following during transportation:

(1) 2 mSv/h (200 mrem/h) on the external surface of the package, unless the following conditions are met, in which case the limit is 10 mSv/h (1000 mrem/h):

(i) The shipment is made in a closed transport vehicle;

(ii) The package is secured within the vehicle so that its position remains fixed during transportation; and

(iii) There are no loading or unloading operations between the beginning and end of the transportation;

(2) 2 mSv/h (200 mrem/h) at any point on the outer surface of the vehicle, including the top and underside of the vehicle; or in the case of a flat-bed style vehicle, at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the load or enclosure, if used, and on the lower external surface of the vehicle; and

(3) 0.1 mSv/h (10 mrem/h) at any point 2 meters (80 in) from the outer lateral surfaces of the vehicle (excluding the top and underside of the vehicle); or in the case of a flat-bed style vehicle, at any point 2 meters (6.6 feet) from the vertical planes projected by the outer edges of the vehicle (excluding the top and underside of the vehicle); and

(4) 0.02 mSv/h (2 mrem/h) in any normally occupied space, except that this provision does not apply to private carriers, if exposed personnel under their control wear radiation dosimetry devices in conformance with 10 CFR 20.1502.

(c) For shipments made under the provisions of paragraph (b) of this section, the shipper shall provide specific written instructions to the carrier for maintenance of the exclusive use shipment controls. The instructions must be included with the shipping paper information.

(d) The written instructions required for exclusive use shipments must be sufficient so that, when followed, they will cause the carrier to avoid actions that will unnecessarily delay delivery or unnecessarily result in increased radiation levels or radiation exposures to transport workers or members of the general public.

§ 71.51 Additional requirements for Type B packages.

(a) Except as provided in § 71.52, a Type B package, in addition to satisfying the requirements of §§ 71.41 through 71.47, must be designed, constructed, and prepared for shipment so that under the tests specified in:

(1) Section 71.71 (''Normal conditions of transport''), there would be no loss or dispersal of radioactive contents—as demonstrated to a sensitivity of 10^{-6} A_2 per hour, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging; and

(2) Section 71.73 (''Hypothetical accident conditions''), there would be no escape of krypton-85 exceeding 10 A_2 in 1 week, no escape of other radioactive material exceeding a total amount A_2 in 1 week, and no external radiation dose rate exceeding 10 mSv/h (1 rem/h) at 1 m (40 in) from the external surface of the package.

(b) Where mixtures of different radionuclides are present, the provisions of appendix A, paragraph IV of this part shall apply, except that for Krypton-85, an effective A_2 value equal to 10 A_2 may be used.

(c) Compliance with the permitted activity release limits of paragraph (a) of this section may not depend on filters or on a mechanical cooling system.

§ 71.52 Exemption for low-specific-activity (LSA) packages.

A package need not satisfy the requirements of § 71.51 if it contains only LSA or SCO material, and is transported as exclusive use, but is subject to §§ 71.41 through 71.47, including § 71.43(f). This section expires April 1, 1999.

§ 71.53 Fissile material exemptions.

The following packages are exempt from fissile material classification and from the fissile material standards of § 71.55 and § 71.59, but are subject to all other requirements of this part:

(a) A package containing not more than 15 g of fissile material. If material is transported in bulk, the quantity limitation applies to the conveyance;

(b) A package containing homogeneous hydrogenous solutions or mixtures where:

(1) The minimum ratio of the number of hydrogen atoms to the number of atoms of fissile radionuclides (H/X) is 5200;

(2) The maximum concentration of fissile radionuclides is 5 g/liter; and

(3) The maximum mass of fissile radionuclides in the package is 800 g, with an exception for a mixture where the total mass of plutonium and uranium-233 exceeds 1 percent of the mass of uranium-235, the limit is 500 g. If the material is transported in bulk, other than by aircraft, the quantity limitations apply to the conveyance;

(c) A package containing uranium enriched in uranium-235 to a maximum of 1 percent by weight, and with a total plutonium and uranium-233 content of up to 1 percent of the mass of uranium-235, if the fissile radionuclides are distributed homogeneously throughout the package contents and do not form a lattice arrangement within the package;

(d) A package containing any fissile material if it does not contain more than 5 g of fissile radionuclides in any 10 liter volume, and if the material is packaged so as to maintain this limit of fissile radionuclide concentration during normal transport;

(e) A package containing not more than 1 kg of plutonium of which not more than 20 percent by mass may consist of plutonium-239, plutonium-241, or any combination of those radionuclides; or

(f) A package containing liquid solutions of uranyl nitrate enriched in uranium-235 to a maximum of 2 percent by weight, with total plutonium and uranium-233 not more than 0.1 percent of the mass of uranium-235 and with a minimum nitrogen-to-uranium atomic ratio (N/U) of 2.

§ 71.55 General requirements for fissile material packages.

(a) A package used for the shipment of fissile material must be designed and constructed in accordance with §§ 71.41 through 71.47. When required by the total amount of radioactive material, a package used for the shipment of fissile material must also be designed and constructed in accordance with § 71.51.

(b) Except as provided in paragraph (c) of this section, a package used for the shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system, or liquid contents were to leak out of the containment system so that, under the following conditions, maximum reactivity of the fissile material would be attained:

(1) The most reactive credible configuration consistent with the chemical and physical form of the material;

(2) Moderation by water to the most reactive credible extent; and

(3) Close full reflection of the containment system by water on all sides, or such greater reflection of the containment system as may additionally be provided by the surrounding material of the packaging.

(c) The Commission may approve exceptions to the requirements of paragraph (b) of this section if the package incorporates special design features that ensure that no single packaging error would permit leakage, and if appropriate measures are taken before each shipment to ensure that the containment system does not leak.

(d) A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in § 71.71 (''Normal conditions of transport'')—

(1) The contents would be subcritical;

(2) The geometric form of the package contents would not be substantially altered;

(3) There would be no leakage of water into the containment system unless, in the evaluation of undamaged packages under § 71.59(b)(1), it has been assumed that moderation is present to such an extent as to cause maximum reactivity consistent with the chemical and physical form of the material; and

(4) There will be no substantial reduction in the effectiveness of the packaging, including:

(i) No more than 5 percent reduction in the total effective volume of the packaging on which nuclear safety is assessed;

(ii) No more than 5 percent reduction in the effective spacing between the fissile contents and the outer surface of the packaging; and

(iii) No occurrence of an aperture in the outer surface of the packaging large enough to permit the entry of a 10 cm (4 in) cube.

(e) A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in § 71.73 (''Hypothetical accident conditions''), the package would be subcritical. For this determination, it must be assumed that:

(1) The fissile material is in the most reactive credible configuration consistent with the damaged condition of the package and the chemical and physical form of the contents;

(2) Water moderation occurs to the most reactive credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents; and

(3) There is full reflection by water on all sides, as close as is consistent with the damaged condition of the package.

§ 71.57 [Reserved]

§ 71.59 Standards for arrays of fissile material packages.

(a) A fissile material package must be controlled by either the shipper or the carrier during transport to assure that an array of such packages remains subcritical. To enable this control, the designer of a fissile material package shall derive a number ''N'' based on all the following conditions being satisfied, assuming packages are stacked together in any arrangement and with close full reflection on all sides of the stack by water:

(1) Five times ''N'' undamaged packages with nothing between the packages would be subcritical;

(2) Two times ''N'' damaged packages, if each package were subjected to the tests specified in § 71.73 (''Hypothetical accident conditions'') would be subcritical with optimum interspersed hydrogenous moderation; and

(3) The value of ''N'' cannot be less than 0.5.

(b) The transport index based on nuclear criticality control must be obtained by dividing the number 50 by the value of ''N'' derived using the procedures specified in paragraph (a) of this section. The value of the transport index for nuclear criticality control may be zero provided that an unlimited number of packages is subcritical such that the value of ''N'' is effectively equal to infinity under the procedures specified in paragraph (a) of this section. Any transport index greater than zero must be rounded up to the first decimal place.

(c) Where a fissile material package is assigned a nuclear criticality control transport index—

(1) Not in excess of 10, that package may be shipped by any carrier, and that carrier provides adequate criticality control by limiting the sum of the transport indexes to 50 in a non-exclusive use vehicle, and to 100 in an exclusive use vehicle.

(2) In excess of 10, that package may only be shipped by exclusive use

vehicle or other shipper controlled system specified by DOT for fissile material packages. The shipper provides adequate criticality control by limiting the sum of the transport indexes to 100 in an exclusive use vehicle.

§ 71.61 Special requirement for irradiated nuclear fuel shipments.

A package for irradiated nuclear fuel with activity greater than 37 PBq (10^6 Ci) must be so designed that its undamaged containment system can withstand an external water pressure of 2 MPa (290 psi) for a period of not less than one hour without collapse, buckling, or inleakage of water.

§ 71.63 Special requirements for plutonium shipments.

(a) Plutonium in excess of 20 Ci (0.74 TBq) per package must be shipped as a solid.

(b) Plutonium in excess of 20 Ci (0.74 TBq) per package must be packaged in a separate inner container placed within outer packaging that meets the requirements of subparts E and F of this part for packaging of material in normal form. If the entire package is subjected to the tests specified in § 71.71 (''Normal conditions of transport''), the separate inner container must not release plutonium as demonstrated to a sensitivity of 10^{-6} A_2/h. If the entire package is subjected to the tests specified in § 71.73 (''Hypothetical accident conditions''), the separate inner container must restrict the loss of plutonium to not more than A_2 in 1 week. Solid plutonium in the following forms is exempt from the requirements of this paragraph:

(1) Reactor fuel elements;

(2) Metal or metal alloy; and

(3) Other plutonium bearing solids that the Commission determines should be exempt from the requirements of this section.

§ 71.64 Special requirements for plutonium air shipments.

(a) A package for the shipment of plutonium by air subject to § 71.88(a)(4), in addition to satisfying the requirements of §§ 71.41 through 71.63, as applicable, must be designed, constructed, and prepared for shipment so that under the tests specified in—

(1) Section 71.74 (''Accident conditions for air transport of plutonium'')—

(i) The containment vessel would not be ruptured in its post-tested condition, and the package must provide a sufficient degree of containment to restrict accumulated loss of plutonium contents to not more than an A_2 quantity in a period of 1 week;

(ii) The external radiation level would not exceed 10 mSv/h (1 rem/h) at a distance of 1 m (40 in) from the surface of the package in its post-tested condition in air; and

(iii) A single package and an array of packages are demonstrated to be subcritical in accordance with this part, except that the damaged condition of the package must be considered to be that which results from the plutonium accident tests in § 71.74, rather than the hypothetical accident tests in § 71.73; and

(2) Section 71.74(c), there would be no detectable leakage of water into the containment vessel of the package.

(b) With respect to the package requirements of paragraph (a), there must be a demonstration or analytical assessment showing that—

(1) The results of the physical testing for package qualification would not be adversely affected to a significant extent by—

(i) The presence, during the tests, of the actual contents that will be transported in the package; and

(ii) Ambient water temperatures ranging from 0.6°C (+33°F) to 38°C (+100°F) for those qualification tests involving water, and ambient atmospheric temperatures ranging from −40°C (−40°F) to +54°C (+130°F) for the other qualification tests.

(2) The ability of the package to meet the acceptance standards prescribed for the accident condition sequential tests would not be adversely affected if one or more tests in the sequence were deleted.

§ 71.65 Additional requirements.

The Commission may, by rule, regulation, or order, impose requirements on any licensee, in addition to those established in this part, as it deems necessary or appropriate to protect public health or to minimize danger to life or property.

Subpart F—Package, Special Form, and LSA–III Tests [2]

§ 71.71 Normal conditions of transport.

(a) *Evaluation*. Evaluation of each package design under normal conditions of transport must include a determination of the effect on that design of the conditions and tests specified in this section. Separate specimens may be used for the free drop test, the compression test, and the penetration test, if each specimen is subjected to the water spray test before being subjected to any of the other tests.

[2] The package standards related to the tests in this subpart are contained in subpart E of this part.

(b) *Initial conditions.* With respect to the initial conditions for the tests in this section, the demonstration of compliance with the requirements of this part must be based on the ambient temperature preceding and following the tests remaining constant at that value between −29°C (−20°F) and +38°C (+100°F) which is most unfavorable for the feature under consideration. The initial internal pressure within the containment system must be considered to be the maximum normal operating pressure, unless a lower internal pressure consistent with the ambient temperature considered to precede and follow the tests is more unfavorable.

(c) *Conditions and tests.*

(1) *Heat.* An ambient temperature of 38°C (100°F) in still air, and insolation according to the following table:

INSOLATION DATA

Form and location of surface	Total insolation for a 12-hour period (g cal/cm²)
Flat surfaces transported horizontally:	
Base	None
Other surfaces	800
Flat surfaces not transported horizontally.	200
Curved surfaces	400

(2) *Cold.* An ambient temperature of −40°C (−40°F) in still air and shade.

(3) *Reduced external pressure.* An external pressure of 25 kPa (3.5 lbf/in²) absolute.

(4) *Increased external pressure.* An external pressure of 140 kPa (20 lbf/in²) absolute.

(5) *Vibration.* Vibration normally incident to transport.

(6) *Water spray.* A water spray that simulates exposure to rainfall of approximately 5 cm/h (2 in/h) for at least 1 hour.

(7) *Free drop.* Between 1.5 and 2.5 hours after the conclusion of the water spray test, a free drop through the distance specified below onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected.

CRITERIA FOR FREE DROP TEST (WEIGHT/DISTANCE)

Package weight		Free drop distance	
Kilograms	(Pounds)	Meters	(Feet)
Less than 5,000.	(Less than 11,000).	1.2	(4)

CRITERIA FOR FREE DROP TEST (WEIGHT/DISTANCE)—Continued

Package weight		Free drop distance	
Kilograms	(Pounds)	Meters	(Feet)
5,000 to 10,000.	(11,000 to 22,000).	0.9	(3)
10,000 to 15,000.	(22,000 to 33,100).	0.6	(2)
More than 15,000.	(More than 33,100).	0.3	(1)

(8) *Corner drop.* A free drop onto each corner of the package in succession, or in the case of a cylindrical package onto each quarter of each rim, from a height of 0.3 m (1 ft) onto a flat, essentially unyielding, horizontal surface. This test applies only to fiberboard, wood, or fissile material rectangular packages not exceeding 50 kg (110 lbs) and fiberboard, wood, or fissile material cylindrical packages not exceeding 100 kg (220 lbs).

(9) *Compression.* For packages weighing up to 5000 kg (11,000 lbs), the package must be subjected, for a period of 24 hours, to a compressive load applied uniformly to the top and bottom of the package in the position in which the package would normally be transported. The compressive load must be the greater of the following:

(i) The equivalent of 5 times the weight of the package; or

(ii) The equivalent of 13 kPa (2 lbf/in^2) multiplied by the vertically projected area of the package.

(10) *Penetration.* Impact of the hemispherical end of a vertical steel cylinder of 3.2 cm (1.25 in) diameter and 6 kg (13 lbs) mass, dropped from a height of 1 m (40 in) onto the exposed surface of the package that is expected to be most vulnerable to puncture. The long axis of the cylinder must be perpendicular to the package surface.

§ 71.73 Hypothetical accident conditions.

(a) *Test procedures.* Evaluation for hypothetical accident conditions is to be based on sequential application of the tests specified in this section, in the order indicated, to determine their cumulative effect on a package or array of packages. An undamaged specimen may be used for the water immersion tests specified in paragraph (c)(6) of this section.

(b) *Test conditions.* With respect to the initial conditions for the tests, except for the water immersion tests, to demonstrate compliance with the requirements of this part during testing, the ambient air temperature before and after the tests must remain constant at that value between −29°C (−20°F) and +38°C (+100°F) which is most unfavorable for the feature under consideration. The initial internal pressure within the containment system must be the maximum normal operating pressure, unless a lower internal pressure, consistent with the ambient temperature assumed to precede and follow the tests, is more unfavorable.

(c) *Tests.* Tests for hypothetical accident conditions must be conducted as follows:

(1) *Free Drop.* A free drop of the specimen through a distance of 9 m (30 ft) onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected.

(2) *Crush.* Subjection of the specimen to a dynamic crush test by positioning the specimen on a flat, essentially unyielding, horizontal surface so as to suffer maximum damage by the drop of a 500 kg (1100 pound) mass from 9 m (30 ft) onto the specimen. The mass must consist of a solid mild steel plate 1 m (40 in) by 1 m and must fall in a horizontal attitude. The crush test is required only when the specimen has a mass not greater than 500 kg (1100 lbs), an overall density not greater than 1000 kg/m^3 (62.4 lbs/ft^3) based on external dimensions, and radioactive contents greater than 1000 A_2 not as special form radioactive material.

(3) *Puncture.* A free drop of the specimen through a distance of 1 m (40 in) in a position for which maximum damage is expected, onto the upper end of a solid, vertical, cylindrical, mild steel bar mounted on an essentially unyielding, horizontal surface. The bar must be 15 cm (6 in) in diameter, with the top horizontal and its edge rounded to a radius of not more than 6 mm (0.25 in), and of a length as to cause maximum damage to the package, but not less than 20 cm (8 in) long. The long axis of the bar must be vertical.

(4) *Thermal.* Exposure of the specimen fully engulfed, except for a simple support system, in a hydrocarbon fuel/air fire of sufficient extent, and in sufficiently quiescent ambient conditions, to provide an average emissivity coefficient of at least 0.9, with an average flame temperature of at least 800°C (1475°F) for a period of 30 minutes, or any other thermal test that provides the equivalent total heat input to the package and which provides a time averaged environmental temperature of 800°C. The fuel source must extend horizontally at least 1 m (40 in), but may not extend more than 3 m (10 ft), beyond any external surface of the specimen, and the specimen must be positioned 1 m (40 in) above the surface of the fuel source. For purposes of calculation, the surface absorptivity coefficient must be either that value which the package may be expected to possess if exposed to the fire specified or 0.8, whichever is greater; and the convective coefficient must be that value which may be demonstrated to exist if the package were exposed to the fire specified. Artificial cooling may not be applied after cessation of external heat input, and any combustion of materials of construction, must be allowed to proceed until it terminates naturally.

(5) *Immersion—fissile material.* For fissile material subject to § 71.55, in those cases where water inleakage has not been assumed for criticality analysis, immersion under a head of water of at least 0.9 m (3 ft) in the attitude for which maximum leakage is expected.

(6) *Immersion—all packages.* A separate, undamaged specimen must be subjected to water pressure equivalent to immersion under a head of water of at least 15 m (50 ft). For test purposes, an external pressure of water of 150 kPa (21.7 lbf/in^2) gauge is considered to meet these conditions.

§ 71.74 Accident conditions for air transport of plutonium.

(a) *Test conditions—Sequence of tests.* A package must be physically tested to the following conditions in the order indicated to determine their cumulative effect.

(1) Impact at a velocity of not less than 129 m/sec (422 ft/sec) at a right angle onto a flat, essentially unyielding, horizontal surface, in the orientation (e.g., side, end, corner) expected to result in maximum damage at the conclusion of the test sequence.

(2) A static compressive load of 31,800 kg (70,000 lbs) applied in the orientation expected to result in maximum damage at the conclusion of the test sequence. The force on the package must be developed between a flat steel surface and a 5 cm (2 in) wide, straight, solid, steel bar. The length of the bar must be at least as long as the diameter of the package, and the longitudinal axis of the bar must be parallel to the plane of the flat surface. The load must be applied to the bar in a manner that prevents any members or devices used to support the bar from contacting the package.

(3) Packages weighing less than 227 kg (500 lbs) must be placed on a flat, essentially unyielding, horizontal surface, and subjected to a weight of 227 kg (500 lbs) falling from a height of 3 m (10 ft) and striking in the position expected to result in maximum damage at the conclusion of the test sequence.

The end of the weight contacting the package must be a solid probe made of mild steel. The probe must be the shape of the frustum of a right circular cone, 30 cm (12 in) long, 20 cm (8 in) in diameter at the base, and 2.5 cm (1 in) in diameter at the end. The longitudinal axis of the probe must be perpendicular to the horizontal surface. For packages weighing 227 kg (500 lbs) or more, the base of the probe must be placed on a flat, essentially unyielding horizontal surface, and the package dropped from a height of 3 m (10 ft) onto the probe, striking in the position expected to result in maximum damage at the conclusion of the test sequence.

(4) The package must be firmly restrained and supported such that its longitudinal axis is inclined approximately 45° to the horizontal. The area of the package that made first contact with the impact surface in paragraph (a)(1) of this section must be in the lowermost position. The package must be struck at approximately the center of its vertical projection by the end of a structural steel angle section falling from a height of at least 46 m (150 ft). The angle section must be at least 1.8 m (6 ft) in length with equal legs at least 13 cm (5 in) long and 1.3 cm (0.5 in) thick. The angle section must be guided in such a way as to fall end-on, without tumbling. The package must be rotated approximately 90° about its longitudinal axis and struck by the steel angle section falling as before.

(5) The package must be exposed to luminous flames from a pool fire of JP–4 or JP–5 aviation fuel for a period of at least 60 minutes. The luminous flames must extend an average of at least 0.9 m (3 ft) and no more than 3 m (10 ft) beyond the package in all horizontal directions. The position and orientation of the package in relation to the fuel must be that which is expected to result in maximum damage at the conclusion of the test sequence. An alternate method of thermal testing may be substituted for this fire test, provided that the alternate test is not of shorter duration and would not result in a lower heating rate to the package. At the conclusion of the thermal test, the package must be allowed to cool naturally or must be cooled by water sprinkling, whichever is expected to result in maximum damage at the conclusion of the test sequence.

(6) Immersion under at least 0.9 m (3 ft) of water.

(b) *Individual free-fall impact test.*

(1) An undamaged package must be physically subjected to an impact at a velocity not less than the calculated terminal free-fall velocity, at mean sea level, at a right angle onto a flat, essentially unyielding, horizontal surface, in the orientation (e.g., side, end, corner) expected to result in maximum damage.

(2) This test is not required if the calculated terminal free-fall velocity of the package is less than 129 m/sec (422 ft/sec), or if a velocity not less than either 129 m/sec (422 ft/sec) or the calculated terminal free-fall velocity of the package is used in the sequential test of paragraph (a)(1) of this section.

(c) *Individual deep submersion test.* An undamaged package must be physically submerged and physically subjected to an external water pressure of at least 4 MPa (600 lbs/in ^{2}).

§ 71.75 Qualification of special form radioactive material.

(a) Special form radioactive materials must meet the test requirements of paragraph (b) of this section. Each solid radioactive material or capsule specimen to be tested must be manufactured or fabricated so that it is representative of the actual solid material or capsule that will be transported, with the proposed radioactive content duplicated as closely as practicable. Any differences between the material to be transported and the test material, such as the use of non-radioactive contents, must be taken into account in determining whether the test requirements have been met. In addition:

(1) A different specimen may be used for each of the tests;

(2) The specimen may not break or shatter when subjected to the impact, percussion, or bending tests;

(3) The specimen may not melt or disperse when subjected to the heat test;

(4) After each test, leaktightness or indispersibility of the specimen must be determined by a method no less sensitive than the leaching assessment procedure prescribed in paragraph (c) of this section. For a capsule resistant to corrosion by water, and which has an internal void volume greater than 0.1 milliliter, an alternative to the leaching assessment is a demonstration of leaktightness of $\times 10^{-4}$ torr-liter/s ($1.3\times\times 10^{-4}$ atm-cm^3/s) based on air at 25°C (77°F) and one atmosphere differential pressure for solid radioactive content, or $\times 10^{-6}$ torr-liter/s ($1.3\times\times 10^{-6}$ atm $-$ cm^3/s) for liquid or gaseous radioactive content; and

(5) A specimen that comprises or simulates radioactive material contained in a sealed capsule need not be subjected to the leaktightness procedure specified in this section, provided it is alternatively subjected to any of the tests prescribed in ISO/TR4826–1979(E), ‘‘Sealed radioactive sources leak test methods’’ which is available from the American National Standards Institute, 1430 Broadway, New York, N.Y. 10018.

(b) *Test methods.*

(1) *Impact Test.* The specimen must fall onto the target from a height of 9 m (30 ft) or greater in the orientation expected to result in maximum damage. The target must be a flat, horizontal surface of such mass and rigidity that any increase in its resistance to displacement or deformation, on impact by the specimen, would not significantly increase the damage to the specimen.

(2) *Percussion Test.*

(i) The specimen must be placed on a sheet of lead that is supported by a smooth solid surface, and struck by the flat face of a steel billet so as to produce an impact equivalent to that resulting from a free drop of 1.4 kg (3 lbs) through 1 m (40 in);

(ii) The flat face of the billet must be 25 millimeters (mm) (1 inch) in diameter with the edges rounded off to a radius of 3 mm $\pm$ 0.3 mm(.12 in $\pm$ 0.012 in);

(iii) The lead must be hardness number 3.5 to 4.5 on the Vickers scale and thickness 25 mm (1 in) or greater, and must cover an area greater than that covered by the specimen;

(iv) A fresh surface of lead must be used for each impact; and

(v) The billet must strike the specimen so as to cause maximum damage.

(3) *Bending test.*

(i) This test applies only to long, slender sources with a length of 10 cm (4 inches) or greater and a length to width ratio of 10 or greater;

(ii) The specimen must be rigidly clamped in a horizontal position so that one half of its length protrudes from the face of the clamp;

(iii) The orientation of the specimen must be such that the specimen will suffer maximum damage when its free end is struck by the flat face of a steel billet;

(iv) The billet must strike the specimen so as to produce an impact equivalent to that resulting from a free vertical drop of 1.4 kg (3 lbs) through 1 m (40 in); and

(v) The flat face of the billet must be 25 mm (1 inch) in diameter with the edges rounded off to a radius of 3 mm $\pm$ 0.3 mm (.12 in $\pm$ 0.012 in).

(4) *Heat test.* The specimen must be heated in air to a temperature of not less than 800°C (1475°F), held at that temperature for a period of 10 minutes, and then allowed to cool.

(c) *Leaching assessment methods.* (1) For indispersible solid material—

(i) The specimen must be immersed for 7 days in water at ambient temperature. The water must have a pH of 6–8 and a maximum conductivity of 10 micromho per centimeter at 20° (68°F);

(ii) The water with specimen must then be heated to a temperature of 50°C ± 5°C (122°F ± 9°F) and maintained at this temperature for 4 hours.

(iii) The activity of the water must then be determined;

(iv) The specimen must then be stored for at least 7 days in still air of relative humidity not less than 90 percent at 30°C (86°F);

(v) The specimen must then be immersed in water under the same conditions as in paragraph (c)(1)(i) of this section, and the water with specimen must be heated to 50°C ± 5°C (122°F ± 9°F) and maintained at that temperature for 4 hours;

(vi) The activity of the water must then be determined. The sum of the activities determined here and in paragraph (c)(1)(iii) of this section must not exceed 2 kilobecquerels (kBq) (0.05 microcurie (μCi)).

(2) For encapsulated material—

(i) The specimen must be immersed in water at ambient temperature. The water must have a pH of 6–8 and a maximum conductivity of 10 micromho per centimeter;

(ii) The water and specimen must be heated to a temperature of 50°C ± 5°C (122°F ± 9°F) and maintained at this temperature for 4 hours;

(iii) The activity of the water must then be determined;

(iv) The specimen must then be stored for at least 7 days in still air at a temperature of 30°C (86°F) or greater;

(v) The process in paragraph (c)(2)(i), (ii), and (iii) of this section must be repeated; and

(vi) The activity of the water must then be determined. The sum of the activities determined here and in paragraph (c)(2)(iii) of this section must not exceed 2 kilobecquerels (kBq) (0.05 microcurie (μCi)).

(d) A specimen that comprises or simulates radioactive material contained in a sealed capsule need not be subjected to—

(1) The impact test and the percussion test of this section, provided that the specimen is alternatively subjected to the Class 4 impact test prescribed in ISO 2919–1980(e), ''Sealed Radioactive Sources Classification'' (see § 71.75(a)(5) for statement of availability); and

(2) The heat test of this section, provided the specimen is alternatively subjected to the Class 6 temperature test specified in the International Organization for Standardization document ISO 2919–1980(e), ''Sealed Radioactive Sources Classification.''

§ 71.77 Qualification of LSA–III Material

(a) LSA–III material must meet the test requirements of paragraph (b) of this section. Any differences between the specimen to be tested and the material to be transported must be taken into account in determining whether the test requirements have been met.

(b) *Leaching Test.* (1) The specimen, representing no less than the entire contents of the package, must be immersed for 7 days in water at ambient temperature;

(2) The volume of water to be used in the test must be sufficient to ensure that at the end of the test period the free volume of the unabsorbed and unreacted water remaining will be at least 10% of the volume of the specimen itself;

(3) The water must have an initial pH of 6–8 and a maximum conductivity 10 micromho/cm at 20°C (68°F); and

(4) The total activity of the free volume of water must be measured following the 7 day immersion test and must not exceed 0.1 A_2.

Subpart G—Operating Controls and Procedures

§ 71.81 Applicability of operating controls and procedures.

A licensee subject to this part, who, under a general or specific license, transports licensed material or delivers licensed material to a carrier for transport, shall comply with the requirements of this subpart G, with the quality assurance requirements of subpart H of this part, and with the general provisions of subpart A of this part.

§ 71.83 Assumptions as to unknown properties.

When the isotopic abundance, mass, concentration, degree of irradiation, degree of moderation, or other pertinent property of fissile material in any package is not known, the licensee shall package the fissile material as if the unknown properties have credible values that will cause the maximum neutron multiplication.

§ 71.85 Preliminary determinations.

Before the first use of any packaging for the shipment of licensed material—

(a) The licensee shall ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce the effectiveness of the packaging;

(b) Where the maximum normal operating pressure will exceed 35 kPa (5 lbf/in^2) gauge, the licensee shall test the containment system at an internal pressure at least 50 percent higher than the maximum normal operating pressure, to verify the capability of that system to maintain its structural integrity at that pressure; and

(c) The licensee shall conspicuously and durably mark the packaging with its model number, serial number, gross weight, and a package identification number assigned by NRC. Before applying the model number, the licensee shall determine that the packaging has been fabricated in accordance with the design approved by the Commission.

§ 71.87 Routine determinations.

Before each shipment of licensed material, the licensee shall ensure that the package with its contents satisfies the applicable requirements of this part and of the license. The licensee shall determine that—

(a) The package is proper for the contents to be shipped;

(b) The package is in unimpaired physical condition except for superficial defects such as marks or dents;

(c) Each closure device of the packaging, including any required gasket, is properly installed and secured and free of defects;

(d) Any system for containing liquid is adequately sealed and has adequate space or other specified provision for expansion of the liquid;

(e) Any pressure relief device is operable and set in accordance with written procedures;

(f) The package has been loaded and closed in accordance with written procedures;

(g) For fissile material, any moderator or neutron absorber, if required, is present and in proper condition;

(h) Any structural part of the package that could be used to lift or tie down the package during transport is rendered inoperable for that purpose, unless it satisfies the design requirements of § 71.45;

(i) The level of non-fixed (removable) radioactive contamination on the external surfaces of each package offered for shipment is as low as reasonably achievable, and within the limits specified in DOT regulations in 49 CFR 173.443;

(j) External radiation levels around the package and around the vehicle, if applicable, will not exceed the limits specified in § 71.47 at any time during transportation; and

(k) Accessible package surface temperatures will not exceed the limits specified in § 71.43(g) at any time during transportation.

§ 71.88 Air transport of plutonium.

(a) Notwithstanding the provisions of any general licenses and notwithstanding any exemptions stated directly in this part or included indirectly by citation of 49 CFR Chapter I, as may be applicable, the licensee shall assure that plutonium in any form, whether for import, export, or domestic shipment, is not transported by air or delivered to a carrier for air transport unless:

(1) The plutonium is contained in a medical device designed for individual human application; or

(2) The plutonium is contained in a material in which the specific activity is not greater than 0.002 μCi/g (70 Bq/g) of material and in which the radioactivity is essentially uniformly distributed; or

(3) The plutonium is shipped in a single package containing no more than an A_2 quantity of plutonium in any isotope or form, and is shipped in accordance with § 71.5; or

(4) The plutonium is shipped in a package specifically authorized for the shipment of plutonium by air in the Certificate of Compliance for that package issued by the Commission.

(b) Nothing in paragraph (a) of this section is to be interpreted as removing or diminishing the requirements of § 73.24 of this chapter.

(c) For a shipment of plutonium by air which is subject to paragraph (a)(4) of this section, the licensee shall, through special arrangement with the carrier, require compliance with 49 CFR 175.704, U.S. Department of Transportation regulations applicable to the air transport of plutonium.

§ 71.89 Opening instructions.

Before delivery of a package to a carrier for transport, the licensee shall ensure that any special instructions needed to safely open the package have been sent to, or otherwise made available to, the consignee for the consignee's use in accordance with 10 CFR 20.1906(e).

§ 71.91 Records.

(a) Each licensee shall maintain, for a period of 3 years after shipment, a record of each shipment of licensed material not exempt under § 71.10, showing where applicable—

(1) Identification of the packaging by model number and serial number;

(2) Verification that there are no significant defects in the packaging, as shipped;

(3) Volume and identification of coolant;

(4) Type and quantity of licensed material in each package, and the total quantity of each shipment;

(5) For each item of irradiated fissile material—

(i) Identification by model number and serial number;

(ii) Irradiation and decay history to the extent appropriate to demonstrate that its nuclear and thermal characteristics comply with license conditions; and

(iii) Any abnormal or unusual condition relevant to radiation safety;

(6) Date of the shipment;

(7) For fissile packages and for Type B packages, any special controls exercised;

(8) Name and address of the transferee;

(9) Address to which the shipment was made; and

(10) Results of the determinations required by § 71.87 and by the conditions of the package approval.

(b) The licensee shall make available to the Commission for inspection, upon reasonable notice, all records required by this part. Records are only valid if stamped, initialed, or signed and dated by authorized personnel or otherwise authenticated.

(c) The licensee shall maintain sufficient written records to furnish evidence of the quality of packaging. The records to be maintained include results of the determinations required by § 71.85; design, fabrication, and assembly records, results of reviews, inspections, tests, and audits; results of monitoring work performance and materials analyses; and results of maintenance, modification and repair activities. Inspection, test, and audit records must identify the inspector or data recorder, the type of observation, the results, the acceptability and the action taken in connection with any deficiencies noted. The records must be retained for three years after the life of the packaging to which they apply.

§ 71.93 Inspection and tests.

(a) The licensee or certificate holder shall permit the Commission, at all reasonable times, to inspect the licensed material, packaging, premises, and facilities in which the licensed material or packaging is used, provided, constructed, fabricated, tested, stored, or shipped.

(b) The licensee shall perform, and permit the Commission to perform, any tests the Commission deems necessary or appropriate for the administration of the regulations in this chapter.

(c) The licensee shall notify the Administrator of the appropriate NRC Regional Office listed in appendix A of part 73 of this chapter, at least 45 days before fabrication of a package to be used for the shipment of licensed material having a decay heat load in excess of 5 kW or with a maximum normal operating pressure in excess of 103 kPa (15 lbf/in²) gauge.

§ 71.95 Reports.

The licensee shall report to the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, within 30 days—

(a) Any instance in which there is significant reduction in the effectiveness of any approved Type B, or fissile, packaging during use;

(b) Details of any defects with safety significance in Type B, or fissile, packaging after first use, with the means employed to repair the defects and prevent their recurrence; or

(c) Instances in which the conditions of approval in the certificate of compliance were not observed in making a shipment.

§ 71.97 Advance notification of shipment of irradiated reactor fuel and nuclear waste.

(a) As specified in paragraphs (b), (c) and (d) of this section, each licensee shall provide advance notification to the governor of a State, or the governor's designee, of the shipment of licensed material, through, or across the boundary of the State, before the transport, or delivery to a carrier, for transport, of licensed material outside the confines of the licensee's plant or other place of use or storage.

(b) Advance notification is required under this section for shipments of irradiated reactor fuel in quantities less than that subject to advance notification requirements of § 73.37(f) of this chapter. Advance notification is also required under this section for shipment of licensed material, other than irradiated fuel, meeting the following three conditions:

(1) The licensed material is required by this part to be in Type B packaging for transportation;

(2) The licensed material is being transported to or across a State boundary en route to a disposal facility or to a collection point for transport to a disposal facility; and

(3) The quantity of licensed material in a single package exceeds the least of the following:

(i) 3000 times the A_1 value of the radionuclides as specified in appendix A, Table A–1 for special form radioactive material;

(ii) 3000 times the A_2 value of the radionuclides as specified in appendix A, Table A–1 for normal form radioactive material; or

(iii) 1000 TBq (27,000 Ci).

(c) *Procedures for submitting advance notification.*

(1) The notification must be made in writing to the office of each appropriate governor or governor's designee and to the Administrator of the appropriate

NRC Regional Office listed in appendix A to part 73 of this chapter.

(2) A notification delivered by mail must be postmarked at least 7 days before the beginning of the 7-day period during which departure of the shipment is estimated to occur.

(3) A notification delivered by messenger must reach the office of the governor or of the governor's designee at least 4 days before the beginning of the 7-day period during which departure of the shipment is estimated to occur.

(i) A list of the names and mailing addresses of the governors' designees receiving advance notification of transportation of nuclear waste was published in the Federal Register on June 30, 1995 (60 FR 34306).

(ii) The list will be published annually in the Federal Register on or about June 30 to reflect any changes in information.

(iii) A list of the names and mailing addresses of the governors' designees is available on request from the Director, Office of State Programs, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

(4) The licensee shall retain a copy of the notification as a record for 3 years.

(d) *Information to be furnished in advance notification of shipment.* Each advance notification of shipment of irradiated reactor fuel or nuclear waste must contain the following information:

(1) The name, address, and telephone number of the shipper, carrier, and receiver of the irradiated reactor fuel or nuclear waste shipment;

(2) A description of the irradiated reactor fuel or nuclear waste contained in the shipment, as specified in the regulations of DOT in 49 CFR 172.202 and 172.203(d);

(3) The point of origin of the shipment and the 7-day period during which departure of the shipment is estimated to occur;

(4) The 7-day period during which arrival of the shipment at State boundaries is estimated to occur;

(5) The destination of the shipment, and the 7-day period during which arrival of the shipment is estimated to occur; and

(6) A point of contact, with a telephone number, for current shipment information.

(e) *Revision notice.* A licensee who finds that schedule information previously furnished to a governor or governor's designee, in accordance with this section, will not be met, shall telephone a responsible individual in the office of the governor of the State or of the governor's designee and inform that individual of the extent of the delay beyond the schedule originally reported. The licensee shall maintain a record of the name of the individual contacted for 3 years.

(f) *Cancellation notice.*

(1) Each licensee who cancels an irradiated reactor fuel or nuclear waste shipment for which advance notification has been sent shall send a cancellation notice to the governor of each State or to the governor's designee previously notified, and to the Administrator of the appropriate NRC Regional Office listed in appendix A of part 73 of this chapter.

(2) The licensee shall state in the notice that it is a cancellation and identify the advance notification that is being canceled. The licensee shall retain a copy of the notice as a record for 3 years.

§ 71.99 Violations.

(a) The Commission may obtain an injunction or other court order to prevent a violation of the provisions of—

(1) The Atomic Energy Act of 1954, as amended;

(2) Title II of the Energy Reorganization Act of 1974, as amended; or (3) A regulation or order issued pursuant to those Acts.

(b) The Commission may obtain a court order for the payment of a civil penalty imposed under section 234 of the Atomic Energy Act:

(1) For violations of—

(i) Sections 53, 57, 62, 63, 81, 82, 101, 103, 104, 107, or 109 of the Atomic Energy Act of 1954, as amended;

(ii) Section 206 of the Energy Reorganization Act;

(iii) Any rule, regulation, or order issued pursuant to the sections specified in paragraph (b)(1)(i) of this section; or

(iv) Any term , condition, or limitation of any license issued under the sections specified in paragraph (b)(1)(i) of this section.

(2) For any violation for which a license may be revoked under section 186 of the Atomic Energy Act of 1954, as amended.

§ 71.100 Criminal penalties.

(a) Section 223 of the Atomic Energy Act of 1954, as amended, provides for criminal sanctions for willful violation of, attempted violation of, or conspiracy to violate, any regulation issued under sections 161b, 161i, or 161o of the Act. For purposes of section 223, all the regulations in part 71 are issued under one or more of sections 161b, 161i, or 161o, except for the sections listed in paragraph (b) of this section.

(b) The regulations in part 71 that are not issued under sections 161b, 161i, or 161o for the purposes of section 223 are as follows: §§ 71.0, 71.2, 71.4, 71.6, 71.7, 71.9, 71.10, 71.31, 71.33, 71.35, 71.37, 71.38, 71.39, 71.41, 71.43, 71.45, 71.47, 71.51, 71.52, 71.53, 71.55, 71.59, 71.65, 71.71, 71.73, 71.74, 71.75, 71.77, 71.99, and 71.100.

Subpart H—Quality Assurance

§ 71.101 Quality assurance requirements.

(a) *Purpose.* This subpart describes quality assurance requirements applying to design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of components of packaging that are important to safety. As used in this subpart, ''quality assurance'' comprises all those planned and systematic actions necessary to provide adequate confidence that a system or component will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to control of the physical characteristics and quality of the material or component to predetermined requirements.

(b) *Establishment of program.* Each licensee shall establish, maintain, and execute a quality assurance program satisfying each of the applicable criteria of §§ 71.101 through 71.137 and satisfying any specific provisions that are applicable to the licensee's activities including procurement of packaging. The licensee shall apply each of the applicable criteria in a graded approach, i.e., to an extent that is consistent with its importance to safety.

(c) *Approval of program.* Before the use of any package for the shipment of licensed material subject to this subpart, each licensee shall obtain Commission approval of its quality assurance program. Each licensee shall file a description of its quality assurance program, including a discussion of which requirements of this subpart are applicable and how they will be satisfied, with the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

(d) *Existing package designs.* The provisions of this paragraph deal with packages that have been approved for use in accordance with this part before January 1, 1979, and which have been designed in accordance with the provisions of this part in effect at the time of application for package approval. Those packages will be accepted as having been designed in accordance with a quality assurance program that satisfies the provisions of paragraph (b) of this section.

(e) *Existing packages.* The provisions of this paragraph deal with packages that have been approved for use in accordance with this part before January 1, 1979; have been at least partially fabricated prior to that date; and for which the fabrication is in accordance with the provisions of this part in effect at the time of application for approval of package design. These packages will be accepted as having been fabricated and assembled in accordance with a quality assurance program that satisfies the provisions of paragraph (b) of this section.

(f) *Previously approved programs.* A Commission-approved quality assurance program that satisfies the applicable criteria of Appendix B of Part 50 of this chapter, and that is established, maintained, and executed with regard to transport packages, will be accepted as satisfying the requirements of paragraph (b) of this section. Before first use, the licensee shall notify the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, of its intent to apply its previously approved Appendix B program to transportation activities. The licensee shall identify the program by date of submittal to the Commission, Docket Number, and date of Commission approval.

§ 71.103 Quality assurance organization.

(a) The licensee [3] shall be responsible for the establishment and execution of the quality assurance program. The licensee may delegate to others, such as contractors, agents, or consultants, the work of establishing and executing the quality assurance program, or any part of the quality assurance program, but shall retain responsibility for the program. The licensee shall clearly establish and delineate, in writing, the authority and duties of persons and organizations performing activities affecting the safety-related functions of structures, systems, and components. These activities include performing the functions associated with attaining quality objectives and the quality assurance functions.

(b) The quality assurance functions are—

(1) Assuring that an appropriate quality assurance program is established and effectively executed; and

(2) Verifying, by procedures such as checking, auditing, and inspection, that activities affecting the safety-related functions have been performed correctly.

(c) The persons and organizations performing quality assurance functions must have sufficient authority and organizational freedom to—

(1) Identify quality problems;

(2) Initiate, recommend, or provide solutions; and

(3) Verify implementation of solutions.

(d) The persons and organizations performing quality assurance functions shall report to a management level that assures that the required authority and organizational freedom, including sufficient independence from cost and schedule, when opposed to safety considerations, are provided.

(e) Because of the many variables involved, such as the number of personnel, the type of activity being performed, and the location or locations where activities are performed, the organizational structure for executing the quality assurance program may take various forms, provided that the persons and organizations assigned the quality assurance functions have the required authority and organizational freedom.

(f) Irrespective of the organizational structure, the individual(s) assigned the responsibility for assuring effective execution of any portion of the quality assurance program, at any location where activities subject to this section are being performed, must have direct access to the levels of management necessary to perform this function.

§ 71.105 Quality assurance program.

(a) The licensee shall establish, at the earliest practicable time consistent with the schedule for accomplishing the activities, a quality assurance program that complies with the requirements of §§ 71.101 through 71.137. The licensee shall document the quality assurance program by written procedures or instructions and shall carry out the program in accordance with those procedures throughout the period during which the packaging is used. The licensee shall identify the material and components to be covered by the quality assurance program, the major organizations participating in the program, and the designated functions of these organizations.

(b) The licensee, through its quality assurance program, shall provide control over activities affecting the quality of the identified materials and components to an extent consistent with their importance to safety, and as necessary to assure conformance to the approved design of each individual package used for the shipment of radioactive material. The licensee shall assure that activities affecting quality are accomplished under suitably controlled conditions. Controlled conditions include the use of appropriate equipment; suitable environmental conditions for accomplishing the activity, such as adequate cleanliness; and assurance that all prerequisites for the given activity have been satisfied. The licensee shall take into account the need for special controls, processes, test equipment, tools, and skills to attain the required quality, and the need for verification of quality by inspection and test.

(c) The licensee shall base the requirements and procedures of its quality assurance program on the following considerations concerning the complexity and proposed use of the package and its components:

(1) The impact of malfunction or failure of the item to safety;

(2) The design and fabrication complexity or uniqueness of the item;

(3) The need for special controls and surveillance over processes and equipment;

(4) The degree to which functional compliance can be demonstrated by inspection or test; and

(5) The quality history and degree of standardization of the item.

(d) The licensee shall provide for indoctrination and training of personnel performing activities affecting quality, as necessary to assure that suitable proficiency is achieved and maintained. The licensee shall review the status and adequacy of the quality assurance program at established intervals. Management of other organizations participating in the quality assurance program shall review regularly the status and adequacy of that part of the quality assurance program which they are executing.

§ 71.107 Package design control.

(a) The licensee shall establish measures to assure that applicable regulatory requirements and the package design, as specified in the license for those materials and components to which this section applies, are correctly translated into specifications, drawings, procedures, and instructions. These measures must include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from standards are controlled. Measures must be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the materials, parts, and components of the packaging.

[3] While the term ''licensee'' is used in these criteria, the requirements are applicable to whatever design, fabrication, assembly, and testing of the package is accomplished with respect to a package prior to the time a package approval is issued.

(b) The licensee shall establish measures for the identification and control of design interfaces and for coordination among participating design organizations. These measures must include the establishment of written procedures, among participating design organizations, for the review, approval, release, distribution, and revision of documents involving design interfaces. The design control measures must provide for verifying or checking the adequacy of design, by methods such as design reviews, alternate or simplified calculational methods, or by a suitable testing program. For the verifying or checking process, the licensee shall designate individuals or groups other than those who were responsible for the original design, but who may be from the same organization. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, the licensee shall include suitable qualification testing of a prototype or sample unit under the most adverse design conditions. The licensee shall apply design control measures to items such as the following:

(1) Criticality physics, radiation shielding, stress, thermal, hydraulic, and accident analyses;

(2) Compatibility of materials;

(3) Accessibility for inservice inspection, maintenance, and repair;

(4) Features to facilitate decontamination; and

(5) Delineation of acceptance criteria for inspections and tests.

(c) The licensee shall subject design changes, including field changes, to design control measures commensurate with those applied to the original design. Changes in the conditions specified in the package approval require NRC approval.

§ 71.109 Procurement document control.

The licensee shall establish measures to assure that adequate quality is required in the documents for procurement of material, equipment, and services, whether purchased by the licensee or by its contractors or subcontractors. To the extent necessary, the licensee shall require contractors or subcontractors to provide a quality assurance program consistent with the applicable provisions of this part.

§ 71.111 Instructions, procedures, and drawings.

The licensee shall prescribe activities affecting quality by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall require that these instructions, procedures, and drawings be followed. The instructions, procedures, and drawings must include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

§ 71.113 Document control.

The licensee shall establish measures to control the issuance of documents such as instructions, procedures, and drawings, including changes, which prescribe all activities affecting quality. These measures must assure that documents, including changes, are reviewed for adequacy, approved for release by authorized personnel, and distributed and used at the location where the prescribed activity is performed. These measures must assure that changes to documents are reviewed and approved.

§ 71.115 Control of purchased material, equipment, and services.

(a) The licensee shall establish measures to assure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents. These measures must include provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by the contractor or subcontractor, inspection at the contractor or subcontractor source, and examination of products on delivery.

(b) The licensee shall have available documentary evidence that material and equipment conform to the procurement specifications before installation or use of the material and equipment. The licensee shall retain, or have available, this documentary evidence for the life of the package to which it applies. The licensee shall assure that the evidence is sufficient to identify the specific requirements met by the purchased material and equipment.

(c) The licensee shall assess the effectiveness of the control of quality by contractors and subcontractors at intervals consistent with the importance, complexity, and quantity of the product or services.

§ 71.117 Identification and control of materials, parts, and components.

The licensee shall establish measures for the identification and control of materials, parts, and components. These measures must assure that identification of the item is maintained by heat number, part number, or other appropriate means, either on the item or on records traceable to the item, as required throughout fabrication, installation, and use of the item. These identification and control measures must be designed to prevent the use of incorrect or defective materials, parts, and components.

§ 71.119 Control of special processes.

The licensee shall establish measures to assure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

§ 71.121 Internal inspection.

The licensee shall establish and execute a program for inspection of activities affecting quality by or for the organization performing the activity, to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. The inspection must be performed by individuals other than those who performed the activity being inspected. Examination, measurements, or tests of material or products processed must be performed for each work operation where necessary to assure quality. If direct inspection of processed material or products is not carried out, indirect control by monitoring processing methods, equipment, and personnel must be provided. Both inspection and process monitoring must be provided when quality control is inadequate without both. If mandatory inspection hold points, which require witnessing or inspecting by the licensee's designated representative and beyond which work should not proceed without the consent of its designated representative, are required, the specific hold points must be indicated in appropriate documents.

§ 71.123 Test control.

The licensee shall establish a test program to assure that all testing required to demonstrate that the packaging components will perform satisfactorily in service is identified and performed in accordance with written test procedures that incorporate the requirements of this part and the requirements and acceptance limits contained in the package approval. The test procedures must include provisions for assuring that all prerequisites for the given test are met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions. The licensee shall document and evaluate the test results to assure that test requirements have been satisfied.

§ 71.125 Control of measuring and test equipment.

The licensee shall establish measures to assure that tools, gauges, instruments, and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated, and adjusted at specified times to maintain accuracy within necessary limits.

§ 71.127 Handling, storage, and shipping control.

The licensee shall establish measures to control, in accordance with instructions, the handling, storage, shipping, cleaning, and preservation of materials and equipment to be used in packaging to prevent damage or deterioration. When necessary for particular products, special protective environments, such as inert gas atmosphere, and specific moisture content and temperature levels must be specified and provided.

§ 71.129 Inspection, test, and operating status.

(a) The licensee shall establish measures to indicate, by the use of markings such as stamps, tags, labels, routing cards, or other suitable means, the status of inspections and tests performed upon individual items of the packaging. These measures must provide for the identification of items that have satisfactorily passed required inspections and tests, where necessary to preclude inadvertent bypassing of the inspections and tests.

(b) The licensee shall establish measures to identify the operating status of components of the packaging, such as tagging valves and switches, to prevent inadvertent operation.

§ 71.131 Nonconforming materials, parts, or components.

The licensee shall establish measures to control materials, parts, or components that do not conform to the licensee's requirements to prevent their inadvertent use or installation. These measures must include, as appropriate, procedures for identification, documentation, segregation, disposition, and notification to affected organizations. Nonconforming items must be reviewed and accepted, rejected, repaired, or reworked in accordance with documented procedures.

§ 71.133 Corrective action.

The licensee shall establish measures to assure that conditions adverse to quality, such as deficiencies, deviations, defective material and equipment, and nonconformances, are promptly identified and corrected. In the case of a significant condition adverse to quality, the measures must assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken must be documented and reported to appropriate levels of management.

§ 71.135 Quality assurance records.

The licensee shall maintain sufficient written records to describe the activities affecting quality. The records must include the instructions, procedures, and drawings required by § 71.111 to prescribe quality assurance activities and must include closely related specifications such as required qualifications of personnel, procedures, and equipment. The records must include the instructions or procedures which establish a records retention program that is consistent with applicable regulations and designates factors such as duration, location, and assigned responsibility. The licensee shall retain these records for 3 years beyond the date when the licensee last engages in the activity for which the quality assurance program was developed. If any portion of the written procedures or instructions is superseded, the licensee shall retain the superseded material for 3 years after it is superseded.

§ 71.137 Audits.

The licensee shall carry out a comprehensive system of planned and periodic audits, to verify compliance with all aspects of the quality assurance program, and to determine the effectiveness of the program. The audits must be performed in accordance with written procedures or checklists by appropriately trained personnel not having direct responsibilities in the areas being audited. Audited results must be documented and reviewed by management having responsibility in the area audited. Follow-up action, including reaudit of deficient areas, must be taken where indicated.

Appendix A to Part 71—Determination of A_1 and A_2

I. Values of A_1 and A_2 for individual radionuclides, which are the bases for many activity limits elsewhere in these regulations are given in Table A–1. The curie (Ci) values specified are obtained by converting from the Terabecquerel (TBq) figure. The curie values are expressed to three significant figures to assure that the difference in the TBq and Ci quantities is one tenth of one percent or less. Where values of A_1 or A_2 are unlimited, it is for radiation control purposes only. For nuclear criticality safety, some materials are subject to controls placed on fissile material.

II. For individual radionuclides whose identities are known, but which are not listed in Table A–1, the determination of the values of A_1 and A_2 requires Commission approval, except that the values of A_1 and A_2 in Table A–2 may be used without obtaining Commission approval.

III. In the calculations of A_1 and A_2 for a radionuclide not in Table A–1, a single radioactive decay chain, in which 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 those daughter nuclides shall be considered as mixtures of different nuclides.

IV. For mixtures of radionuclides whose identities and respective activities are known, the following conditions apply:

(a) For special form radioactive material, the maximum quantity transported in a Type A package:

$$\sum_{I} \frac{B(i)}{A_1(i)} \text{ less than or equal to } 1$$

(b) For normal form radioactive material, the maximum quantity transported in a Type A package:

$$\sum_{I} \frac{B(i)}{A_2(i)} \text{ less than or equal to } 1$$

Where B(i) is the activity of radionuclide I and A_1(i) and A_2(i) are the A_1 and A_2 values for radionuclide I, respectively.

Alternatively, an A_1 value for mixtures of special form material may be determined as follows:

$$A_1 \text{ for mixture} = \frac{1}{\sum_{I} \frac{f(i)}{A_1(i)}}$$

Where f(i) is the fraction of activity of nuclide I in the mixture and A_1(i) is the appropriate A_1 value for nuclide I.

An A_2 value for mixtures of normal form material may be determined as follows:

$$A_2 \text{ for mixture} = \frac{1}{\sum_{I} \frac{f(i)}{A_2(i)}}$$

Where f(i) is the fraction of activity of nuclide I in the mixture and A_2(i) is the appropriate A_2 value for nuclide I.

V. 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 A_1 or A_2 value, as appropriate, for the radionuclides in each group may be used in applying the formulas in paragraph IV. Groups may be based on the total alpha activity and the total beta/gamma activity when these are known, using the lowest A_1 or A_2 values for the alpha emitters and beta/gamma emitters.

TABLE A–1.—A_1 AND A_2 VALUES FOR RADIONUCLIDES

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity	
						(TBq/g)	(Ci/g)
Ac-225	Actinium(89)	0.6	16.2	1×10^{-2}	0.270	2.1×10^{3}	5.8×10^{4}
Ac-227		40	1080	2×10^{-5}	5.41×10^{-4}	2.7	7.2×10^{1}
Ac-228		0.6	16.2	0.4	10.8	8.4×10^{4}	2.2×10^{6}
Ag-105	Silver(47)	2	54.1	2	54.1	1.1×10^{3}	3.0×10^{4}
Ag-108m		0.6	16.2	0.6	16.2	9.7×10^{-1}	2.6×10^{1}
Ag-110m		0.4	10.8	0.4	10.8	1.8×10^{3}	4.7×10^{3}
Ag-111		0.6	16.2	0.5	13.5	5.8×10^{3}	1.6×10^{5}
Al-26	Aluminum(13)	0.4	10.8	0.4	10.8	7.0×10^{-4}	1.9×10^{-2}
Am-241	Americium(95)	2	54.1	2×10^{-4}	5.41×10^{-3}	1.3×10^{-1}	3.4
Am-242m		2	54.1	2×10^{-4}	5.41×10^{-3}	3.6×10^{-1}	9.7×10^{5}
Am-243		2	54.1	2×10^{-4}	5.41×10^{-3}	7.4×10^{-3}	2.0×10^{-1}
Ar-37	Argon(18)	40	1080	40	1080	3.7×10^{3}	9.9×10^{4}
Ar-39		20	541	20	541	1.3×10^{3}	3.4×10^{1}
Ar-41		0.6	16.2	0.6	16.2	1.5×10^{6}	4.2×10^{7}
Ar-42		0.2	5.41	0.2	5.41	9.6	2.6×10^{2}
As-72	Arsenic(33)	0.2	5.41	0.2	5.41	6.2×10^{4}	1.7×10^{6}
As-73		40	1080	40	1080	8.2×10^{2}	2.2×10^{4}
As-74		1	27.0	0.5	13.5	3.7×10^{3}	9.9×10^{4}
As-76		0.2	5.41	0.2	5.41	5.8×10^{4}	1.6×10^{6}
As-77		20	541	0.5	13.5	3.9×10^{4}	1.0×10^{6}
At-211	Astatine(85)	30	811	2	54.1	7.6×10^{4}	2.1×10^{6}
Au-193	Gold(79)	6	162	6	162	3.4×10^{4}	9.2×10^{5}
Au-194		1	27.0	1	27.0	1.5×10^{4}	4.1×10^{5}
Au-195		10	270	10	270	1.4×10^{2}	3.7×10^{3}
Au-196		2	54.1	2	54.1	4.0×10^{3}	1.1×10^{5}
Au-198		3	81.1	0.5	13.5	9.0×10^{3}	2.4×10^{5}
Au-199		10	270	0.9	24.3	7.7×10^{3}	2.1×10^{5}
Ba-131	Barium(56)	2	54.1	2	54.1	3.1×10^{3}	8.4×10^{4}
Ba-133m		10	270	0.9	24.3	2.2×10^{4}	6.1×10^{5}
Ba-133		3	81.1	3	81.1	9.4	2.6×10^{2}
Ba-140		0.4	10.8	0.4	10.8	2.7×10^{3}	7.3×10^{4}
Be-7	Beryllium(4)	20	541	20	541	1.3×10^{4}	3.5×10^{5}
Be-10		20	541	0.5	13.5	8.3×10^{-4}	2.2×10^{-2}
Bi-205	Bismuth(83)	0.6	16.2	0.6	16.2	1.5×10^{-3}	4.2×10^{4}
Bi-206		0.3	8.11	0.3	8.11	3.8×10^{3}	1.0×10^{5}

TABLE A–1.—A_1 AND A_2 VALUES FOR RADIONUCLIDES—Continued

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity	
						(TBq/g)	(Ci/g)
Bi-207		0.7	18.9	0.7	18.9	1.9	5.2×10^{1}
Bi-210m		0.3	8.11	3×10^{-2}	0.811	2.1×10^{-5}	5.7×10^{-4}
Bi-210		0.6	16.2	0.5	13.5	4.6×10^{3}	1.2×10^{5}
Bi-212		0.3	8.11	0.3	8.11	5.4×10^{5}	1.5×10^{7}
Bk-247	Berkelium(97)	2	54.1	2×10^{-4}	5.41×10^{-3}	3.8×10^{-2}	1.0
Bk-249		40	1080	8×10^{-2}	2.16	6.1×10^{1}	1.6×10^{3}
Br-76	Bromine(35)	0.3	8.11	0.3	8.11	9.4×10^{4}	2.5×10^{6}
Br-77		3	81.1	3	81.1	2.6×10^{4}	7.1×10^{5}
Br-82		0.4	108	0.4	10.8	4.0×10^{4}	1.1×10^{6}
C-11	Carbon(6)	1	270	0.5	13.5	3.1×10^{7}	8.4×10^{8}
C-14		40	1080	2	54.1	1.6×10^{-1}	4.5
Ca-41	Calcium(20)	40	1080	40	1080	3.1×10^{-3}	8.5×10^{-2}
Ca-45		40	1080	0.9	24.3	6.6×10^{2}	1.8×10^{4}
Ca-47		0.9	24.3	0.5	13.5	2.3×10^{4}	6.1×10^{5}
Cd-109	Cadmium(48)	40	1080	1	27.0	9.6×10^{1}	2.6×10^{3}
Cd-113m		20	541	9×10^{-2}	2.43	8.3×10^{4}	2.2×10^{2}
Cd-115m		0.3	8.11	0.3	8.11	9.4×10^{2}	2.5×10^{4}
Cd-115		4	108	0.5	13.5	1.9×10^{4}	5.1×10^{5}
Ce-139	Cerium(58)	6	162	6	162	2.5×10^{2}	6.8×10^{3}
Ce-141		10	270	0.5	13.5	1.1×10^{3}	2.8×10^{4}
Ce-143		0.6	16.2	0.5	13.5	2.5×10^{4}	6.6×10^{5}
Ce-144		0.2	5.41	0.2	5.41	1.2×10^{2}	3.2×10^{3}
Cf-248	Californium(98)	30	811	3×10^{-3}	8.11×10^{-2}	5.8×10^{1}	1.6×10^{3}
Cf-249		2	54.1	2×10^{-4}	5.41×10^{-3}	1.5×10^{-1}	4.1
Cf-250		5	135	5×10^{-4}	1.35×10^{-2}	4.0	1.1×10^{2}
Cf-251		2	54.1	2×10^{-4}	5.41×10^{-3}	5.9×10^{-2}	1.6
Cf-252		0.1	2.70	1×10^{-3}	2.70×10^{-2}	2.0×10^{1}	5.4×10^{2}
Cf-253		40	1080	6×10^{-2}	1.62	1.1×10^{3}	2.9×10^{4}
Cf-254		3×10^{-3}	8.11×10^{-2}	6×10^{-4}	1.62×10^{-2}	3.1×10^{2}	8.5×10^{3}
Cl-36	Chlorine(17)	20	541	0.5	13.5	1.2×10^{-3}	3.3×10^{-2}
Cl-38		0.2	5.41	0.2	5.41	4.9×10^{6}	1.3×10^{8}
Cm-240	Curium(96)	40	1080	2×10^{-2}	0.541	7.5×10^{2}	2.0×10^{4}
Cm-241		2	54.1	0.9	24.3	6.1×10^{2}	1.7×10^{4}
Cm-242		40	1080	1×10^{-2}	0.270	1.2×10^{2}	3.3×10^{3}
Cm-243		3	81.1	3×10^{-4}	8.11×10^{-3}	1.9	5.2×10^{1}
Cm-244		4	1080	4×10^{-4}	1.08×10^{-2}	3.0	8.1×10^{5}
Cm-245		2	54.1	2×10^{-4}	5.41×10^{-3}	6.4×10^{-3}	1.7×10^{-1}
Cm-246		2	54.1	2×10^{-4}	5.41×10^{-3}	1.1×10^{-2}	3.1×10^{-1}
Cm-247		2	54.1	2×10^{-4}	5.41×10^{-3}	3.4×10^{-6}	9.3×10^{-5}
Cm-248		4×10^{-2}	1.08	5×10^{-5}	1.35×10^{-3}	1.6×10^{-4}	4.2×10^{-3}
Co-55	Cobalt(27)	0.5	13.5	0.5	13.5	1.1×10^{5}	3.1×10^{6}
Co-56		0.3	8.11	0.3	8.11	1.1×10^{3}	3.0×10^{4}
Co-57		8	216	8	216	3.1×10^{2}	8.4×10^{3}
Co-58m		40	1080	40	1080	2.2×10^{5}	5.9×10^{6}
Co-58		1	27.0	1	27.0	1.2×10^{3}	3.2×10^{4}
Co-60		0.4	10.8	0.4	10.8	4.2×10^{1}	1.1×10^{3}
Cr-51	Chromium(24)	30	811	30	811	3.4×10^{3}	9.2×10^{4}
Cs-129	Cesium(55)	4	108	4	108	2.8×10^{4}	7.6×10^{5}
Cs-131		40	1080	40	1080	3.8×10^{3}	1.0×10^{5}
Cs-132		1	27.0	1	27.0	5.7×10^{3}	1.5×10^{5}
Cs-134m		40	1080	9	243	3.0×10^{5}	8.0×10^{6}
Cs-134		0.6	16.2	0.5	13.5	4.8×10^{1}	1.3×10^{3}
Cs-135		40	1080	0.9	24.3	4.3×10^{-5}	1.2×10^{-3}
Cs-136		0.5	13.5	0.5	13.5	2.7×10^{3}	7.3×10^{4}
Cs-137		2	54.1	0.5	13.5	3.2	8.7×10^{1}
Cu-64	Copper(29)	5	135	0.9	24.3	1.4×10^{5}	3.9×10^{6}
Cu-67		9	243	0.9	24.3	2.8×10^{4}	7.6×10^{5}
Dy-159	Dysprosium(66)	20	541	20	541	2.1×10^{2}	5.7×10^{3}
Dy-165		0.6	16.2	0.5	13.5	3.0×10^{5}	8.2×10^{6}
Dy-166		0.3	8.11	0.3	8.11	8.6×10^{3}	2.3×10^{5}
Er-169	Erbium(68)	40	1080	0.9	24.3	3.1×10^{3}	8.3×10^{4}
Er-171		0.6	16.2	0.5	13.5	9.0×10^{4}	2.4×10^{6}
Es-253	Einsteinium(99) [a]	40	1080	5×10^{-1}	1.35		
Es-254		30	811	3×10^{-3}	8.11×10^{-2}		
Es-254m		0.6	16.2	0.4	10.8		
Es-255							
Eu-147	Europium(63)	2	54.1	2	54.1	1.4×10^{3}	3.7×10^{4}
Eu-148		0.5	13.5	0.5	13.5	6.0×10^{2}	1.6×10^{4}
Eu-149		20	541	20	541	3.5×10^{2}	9.4×10^{3}
Eu-150		0.7	18.9	0.7	18.9	6.1×10^{4}	6.7×10^{6}

TABLE A–1.—A_1 AND A_2 VALUES FOR RADIONUCLIDES—Continued

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity	
						(TBq/g)	(Ci/g)
Eu-152m		0.6	16.2	0.5	13.5	8.2×10^{4}	2.2×10^{6}
Eu-152		0.9	24.3	0.9	24.3	6.5	1.8×10^{2}
Eu-154		0.8	21.6	0.5	13.5	9.8	2.6×10^{2}
Eu-155		20	541	2	54.1	1.8×10^{1}	4.9×10^{3}
Eu-156		0.6	16.2	0.5	13.5	2.0×10^{3}	5.5×10^{4}
F-18	Fluorine(9)	1	27.0	0.5	13.5	3.5×10^{5}	9.5×10^{7}
Fe-52	Iron(26)	0.2	5.41	0.2	5.41	2.7×10^{5}	7.3×10^{6}
Fe-55		40	1080	40	1080	8.8×10^{1}	2.4×10^{3}
Fe-59		0.8	21.6	0.8	21.6	1.8×10^{3}	3.0×10^{-4}
Fe-60		40	1080	0.2	5.41	7.4×10^{-4}	2.0×10^{-2}
Fm-255	Fermium(100) [b]	40	1080	0.8	21.6		
Fm-257		40	1080	7×10^{-3}	1.89×10^{-3}		
Ga-67	Gallium(31)	6	162	6	162	2.2×10^{4}	6.0×10^{5}
Ga-68		0.3	8.11	0.3	8.11	1.5×10^{6}	4.1×10^{7}
Ga-72		0.4	10.8	0.4	10.8	1.1×10^{5}	3.1×10^{6}
Gd-146	Gadolinium(64)	0.4	10.8	0.4	10.8	6.9×10^{2}	1.9×10^{4}
Gd-148		3	81.1	3×10^{-4}	8.11×10^{-3}	6.7	2.9×10^{1}
Gd-153		10	270	5	135	1.3×10^{2}	3.5×10^{3}
Gd-159		4	108	0.5	13.5	3.9×10^{4}	1.1×10^{6}
Ge-68	Germanium(32)	0.3	8.11	0.3	8.11	2.6×10^{2}	7.1×10^{3}
Ge-71		40	1080	40	1080	5.8×10^{3}	1.6×10^{5}
Ge-77		0.3	8.11	0.3	8.11	1.3×10^{5}	3.6×10^{6}
H-3	Hydrogen(1)	See T-Tritium					
Hf-172	Hafnium(72)	0.5	13.5	0.3	8.11	4.1×10^{1}	1.1×10^{3}
Hf-175		3	81.1	3	81.1	3.9×10^{2}	1.1×10^{4}
Hf-181		2	54.1	0.9	24.3	6.3×10^{2}	1.7×10^{4}
Hf-182		4	108	3×10^{-2}	0.811	8.1×10^{-6}	2.2×10^{-4}
Hg-194	Mercury(80)	1	27.0	1	27.0	1.3×10^{-1}	3.5
Hg-195m		5	135	5	135	1.5×10^{4}	4.0×10^{5}
Hg-197m		10	270	0.9	24.3	2.5×10^{4}	6.7×10^{5}
Hg-197		10	270	10	270	9.2×10^{3}	2.5×10^{5}
Hg-203		4	108	0.9	24.3	5.1×10^{2}	1.4×10^{4}
Ho-163	Holmium(67)	40	1080	40	1080	2.7	7.6×10^{1}
Ho-166m		0.6	16.2	0.3	8.11	6.6×10^{-2}	1.8
Ho-166		0.3	8.11	0.3	8.11	2.6×10^{4}	7.0×10^{5}
I-123	Iodine(53)	6	162	6	162	7.1×10^{4}	1.9×10^{6}
I-124		0.9	24.3	0.9	24.3	9.3×10^{3}	2.5×10^{5}
I-125		20	541	2	54.1	6.4×10^{2}	1.7×10^{4}
I-126		2	54.1	0.9	24.3	2.9×10^{3}	8.0×10^{4}
I-129		Unlimited	Unlimited	Unlimited	Unlimited	6.5×10^{-6}	1.8×10^{-4}
I-131		3	81.1	0.5	13.5	4.6×10^{3}	1.2×10^{5}
I-132		0.4	10.8	0.4	10.8	3.8×10^{5}	1.0×10^{7}
I-133		0.6	16.2	0.5	13.5	4.2×10^{4}	1.1×10^{6}
I-134		0.3	8.11	0.3	8.11	9.9×10^{5}	2.7×10^{7}
I-135		0.6	16.2	0.5	13.5	1.3×10^{5}	3.5×10^{6}
In-111	Indium(49)	2	54.1	2	54.1	1.5×10^{4}	4.2×10^{5}
In-113m		4	108	4	108	6.2×10^{5}	1.7×10^{7}
In-114m		0.3	8.11	0.3	8.11	8.6×10^{2}	2.3×10^{4}
In-115m		6	162	0.9	24.3	2.2×10^{5}	6.1×10^{6}
Ir-189	Iridium(77)	10	270	10	270	1.9×10^{3}	5.2×10^{4}
Ir-190		0.7	18.9	0.7	18.9	2.3×10^{3}	6.2×10^{4}
Ir-192		1	27.0	0.5	13.5	3.4×10^{2}	9.2×10^{3}
Ir-193m		10	270	10	270	2.4×10^{3}	6.4×10^{4}
Ir-194		0.2	5.41	0.2	5.41	3.1×10^{4}	8.4×10^{5}
K-40	Potassium(19)	0.6	16.2	0.6	16.2	2.4×10^{-7}	6.4×10^{-6}
K-42		0.2	5.41	0.2	5.41	2.2×10^{5}	6.0×10^{6}
K-43		1.0	27.0	0.5	13.5	1.2×10^{5}	3.3×10^{6}
Kr-81	Krypton(36)	40	1080	40	1080	7.8×10^{-4}	2.1×10^{-2}
Kr-85m		6	162	6	162	3.0×10^{5}	8.2×10^{6}
Kr-85		20	541	10	270	1.5×10^{1}	3.9×10^{2}
Kr-87		0.2	5.41	0.2	5.41	1.0×10^{6}	2.8×10^{7}
La-137	Lanthanum(57)	40	1080	2	54.1	1.6×10^{-3}	4.4×10^{-2}
La-140		0.4	10.8	0.4	10.8	2.1×10^{4}	5.6×10^{5}
Lu-172	Lutetium(71)	0.5	13.5	0.5	13.5	4.2×10^{3}	1.1×10^{5}
Lu-173		8	216	8	216	5.6×10^{1}	1.5×10^{3}
Lu-174m		20	541	8	216	2.0×10^{2}	5.3×10^{3}
Lu-174		8	216	4	108	2.3×10^{1}	6.2×10^{2}
Lu-177		30	811	0.9	24.3	4.1×10^{3}	1.1×10^{5}
MFP	For mied fission products, use formula for mitures or Table A-2						
Mg-28	Magnesium(12)	0.2	5.41	0.2	5.41	2.0×10^{5}	5.4×10^{6}

TABLE A–1.—A_1 AND A_2 VALUES FOR RADIONUCLIDES—Continued

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity	
						(TBq/g)	(Ci/g)
Mn-52	Manganese(25)	0.3	8.11	0.3	8.11	1.6×10^{4}	4.4×10^{5}
Mn-53		Unlimited	Unlimited	Unlimited	Unlimited	6.8×10^{-5}	1.8×10^{-3}
Mn-54		1	27.0	1	27.0	2.9×10^{2}	7.7×10^{3}
Mn-56		0.2	5.41	0.2	5.41	8.0×10^{5}	2.2×10^{7}
Mo-93	Molybdenum(42)	40	1080	7	189	4.1×10^{-2}	1.1
Mo-99		0.6	16.2	0.5	13.5[c]	1.8×10^{4}	4.8×10^{5}
N-13	Nitrogen(7)	0.6	16.2	0.5	13.5	5.4×10^{7}	1.5×10^{9}
Na-22	Sodium(11)	0.5	13.5	0.5	13.5	2.3×10^{2}	6.3×10^{3}
Na-24		0.2	5.41	0.2	5.41	3.2×10^{5}	8.7×10^{6}
Nb-92m	Niobium(41)	0.7	18.9	0.7	18.9	5.2×10^{3}	1.4×10^{5}
Nb-93m		40	1080	6	162	8.8	2.4×10^{2}
Nb-94		0.6	16.2	0.6	16.2	6.9×10^{-3}	1.9×10^{-1}
Nb-95		1	27.0	1	27.0	1.5×10^{3}	3.9×10^{4}
Nb-97		0.6	16.2	0.5	13.5	9.9×10^{5}	2.7×10^{7}
Nd-147	Neodymium(60)	4	108	0.5	13.5	3.0×10^{3}	8.1×10^{4}
Nd-149		0.6	16.2	0.5	13.5	4.5×10^{5}	1.2×10^{7}
Ni-59	Nickel(28)	40	1080	40	1080	3.0×10^{-3}	8.0×10^{-2}
Ni-63		40	1080	30	811	2.1	5.7×10^{1}
Ni-65		0.3	8.11	0.3	8.11	7.1×10^{5}	1.9×10^{7}
Np-235	Neptunium(93)	40	1080	40	1080	5.2×10^{1}	1.4×10^{3}
Np-236		7	189	1×10^{-3}	2.70×10^{-2}	$4.710\text{-}4^{-4}$	1.3×10^{-2}
Np-237		2	54.1	$2\times10_{-4}$	$5.41\times10_{-3}$	2.6×10^{-5}	7.1×10^{-4}
Np-239		6	162	0.5	13.5	8.6×10^{3}	2.3×10^{5}
Os-185	Osmium(76)	1	27.0	1	27.0	2.8×10^{2}	7.5×10^{3}
Os-191m		40	1080	40	1080	4.6×10^{4}	1.3×10^{6}
Os-191		10	270	0.9	24.3	1.6×10^{3}	4.4×10^{4}
Os-193		0.6	16.2	0.5	13.5	2.0×10^{4}	5.3×10^{5}
Os-194		0.2	5.41	0.2	5.41	1.1×10^{1}	3.1×10^{2}
P-32	Phosphorus(15)	0.3	8.11	0.3	8.11	1.1×10^{4}	2.9×10^{5}
P-33		40	1080	0.9	24.3	5.8×10^{3}	1.6×10^{5}
Pa-230	Protactinium(91)	2	54.1	0.1	2.70	1.2×10^{3}	3.3×10^{4}
Pa-231		0.6	16.2	6×10^{-5}	1.62×10^{-3}	1.7×10^{-3}	4.7×10^{-2}
Pa-233		5	135	0.9	24.3	7.7×10^{2}	2.1×10^{4}
Pb-201	Lead(82)	1	27.0	1	27.0	6.2×10^{4}	1.7×10^{6}
Pb-202		40	1080	2	54.1	1.2×10^{-4}	3.4×10^{-3}
Pb-203		3	81.1	3	81.1	1.1×10^{4}	3.0×10^{5}
Pb-205		Unlimited	Unlimited	Unlimited	Unlimited	4.5×10^{-6}	1.2×10^{-4}
Pb-210		0.6	16.2	9×10^{-3}	0.243	2.8	7.6×10^{1}
Pb-212		0.3	8.11	0.3	8.11	5.1×10^{4}	1.4×10^{6}
Pd-103	Palladium(46)	40	1080	40	1080	2.8×10^{3}	7.5×10^{4}
Pd-107		Unlimited	Unlimited	Unlimited	Unlimited	1.9×10^{-5}	5.1×10^{-4}
Pd-109		0.6	16.2	0.5	13.5	7.9×10^{4}	2.1×10^{6}
Pm-143	Promethium(61)	3	81.1	3	81.1	1.3×10^{2}	3.4×10^{3}
Pm-144		0.6	16.2	0.6	16.2	9.2×10^{1}	2.5×10^{3}
Pm-145		30	811	7	189	5.2	1.4×10^{2}
Pm-147		40	1080	0.9	24.3	3.4×10^{1}	9.3×10^{2}
Pm-148m		0.5	13.5	0.5	13.5	7.9×10^{2}	2.1×10^{4}
Pm-149		0.6	16.2	0.5	13.5	1.5×10^{4}	4.0×10^{5}
Pm-151		3	81.1	0.5	13.5	2.7×10^{4}	7.3×10^{5}
Po-208	Polonium(84)	40	1080	2×10^{-2}	0.541	2.2×10^{1}	5.9×10^{2}
Po-209		40	1080	2×10^{-2}	0.541	6.2×10^{-1}	1.7×10^{1}
Po-210		40	1080	2×10^{-2}	0.541	1.7×10^{2}	4.5×10^{3}
Pr-142	Praseodymium(59)	0.2	5.41	0.2	5.41	4.3×10^{4}	1.2×10^{6}
Pr-143		4	108	0.5	13.5	2.5×10^{3}	6.7×10^{4}
Pt-188	Platinum(78)	0.6	16.2	0.6	16.2	2.5×10^{3}	6.8×10^{4}
Pt-191		3	81.1	3	81.1	8.7×10^{3}	2.4×10^{5}
Pt-193m		40	1080	9	243	5.8×10^{3}	1.6×10^{5}
Pt-193		40	1080	40	1080	1.4	3.7×10^{1}
Pt-195m		10	270	2	54.1	6.2×10^{3}	1.7×10^{5}
Pt-197m		10	270	0.9	24.3	3.4×10^{5}	1.0×10^{7}
Pt-197		20	541	0.5	13.5	3.2×10^{4}	8.7×10^{5}
Pu-236	Plutonium(94)	7	189	7×10^{-4}	1.89×10^{-2}	2.0×10^{1}	5.3×10^{2}
Pu-237		20	541	20	541	4.5×10^{2}	1.2×10^{4}
Pu-238		2	54.1	2×10^{-4}	5.41×10^{-3}	6.3×10^{-1}	1.7×10^{1}
Pu-239		2	54.1	2×10^{-4}	5.41×10^{-3}	2.3×10^{-3}	6.2×10^{-2}
Pu-240		2	54.1	2×10^{-4}	5.41×10^{-3}	8.4×10^{-3}	2.3×10^{-1}
Pu-241		40	1080	1×10^{-2}	0.270	3.8	1.0×10^{2}
Pu-242		2	54.1	2×10^{-4}	5.41×10^{-3}	1.5×10^{-4}	3.9×10^{-3}
Pu-244		0.3	8.11	2×10^{-4}	5.41×10^{-3}	6.7×10^{-7}	1.8×10^{-5}
Ra-223	Radium(88)	0.6	16.2	3×10^{-2}	0.811	1.9×10^{3}	5.1×10^{4}

TABLE A–1.—A_1 AND A_2 VALUES FOR RADIONUCLIDES—Continued

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity	
						(TBq/g)	(Ci/g)
Ra-224		0.3	8.11	6×10^{-2}	1.62	5.9×10^{3}	1.6×10^{5}
Ra-225		0.6	16.2	2×10^{-2}	0.541	1.5×10^{3}	3.9×10^{4}
Ra-226		0.3	8.11	2×10^{-2}	0.541	3.7×10^{-2}	1.0
Ra-228		0.6	16.2	4×10^{-2}	1.08	1.0×10^{1}	2.7×10^{2}
Rb-81	Rubidium(37)	2	54.1	0.9	24.3	3.1×10^{5}	8.4×10^{6}
Rb-83		2	54.1	2	54.1	6.8×10^{2}	1.8×10^{4}
Rb-84		1	27.0	0.9	24.3	1.8×10^{3}	4.7×10^{4}
Rb-86		0.3	8.11	0.3	8.11	3.0×10^{3}	8.1×10^{4}
Rb-87		Unlimited	Unlimited	Unlimited	Unlimited	3.2×10^{-9}	8.6×10^{-8}
Rb (natural)		Unlimited	Unlimited	Unlimited	Unlimited	6.7×10^{6}	1.8×10^{8}
Re-183	Rhenium(75)	5	135	5	135	3.8×10^{2}	1.0×10^{4}
Re-184m		3	81.1	3	81.1	1.6×10^{2}	4.3×10^{3}
Re-184		1	27.0	1	27.0	6.9×10^{2}	1.9×10^{4}
Re-186		4	108	0.5	13.5	6.9×10^{3}	1.9×10^{5}
Re-187		Unlimited	Unlimited	Unlimited	Unlimited	1.4×10^{-9}	3.8×10^{-8}
Re-188		0.2	5.41	0.2	5.41	3.6×10^{4}	9.8×10^{5}
Re-189		4	108	0.5	13.5	2.5×10^{4}	6.8×10^{5}
Re (natural)		Unlimited	Unlimited	Unlimited	Unlimited		2.4×10^{-8}
Rh-99	Rhodium(45)	2	54.1	2	54.1	3.0×10^{3}	8.2×10^{4}
Rh-101		4	108	4	108	4.1×10^{1}	1.1×10^{3}
Rh-102m		2	54.1	0.9	24.3	2.3×10^{2}	6.2×10^{3}
Rh-102		0.5	13.5	0.5	13.5	4.5×10^{1}	1.2×10^{3}
Rh-103m		40	1080	40	1080	1.2×10^{6}	3.3×10^{7}
Rh-105		10	270	0.9	24.3	3.1×10^{4}	8.4×10^{5}
Rn-222	Radon(86)	0.2	5.41	4×10^{-3}	0.108	5.7×10^{3}	1.5×10^{5}
Ru-97	Ruthenium(44)	4	108	4	108	1.7×10^{4}	4.6×10^{5}
Ru-103		2	54.1	0.9	24.3	1.2×10^{3}	3.2×10^{4}
Ru-105		0.6	16.2	0.5	13.5	2.5×10^{5}	6.7×10^{6}
Ru-106		0.2	5.41	0.2	5.41	1.2×10^{2}	3.3×10^{3}
S-35	Sulfur(16)	40	1080	2	54.1	1.6×10^{3}	4.3×10^{4}
Sb-122	Antimony(51)	0.3	8.11	0.3	8.11	1.5×10^{4}	4.0×10^{5}
Sb-124		0.6	16.2	0.5	13.5	6.5×10^{2}	1.7×10^{4}
Sb-125		2	54.1	0.9	24.3	3.9×10^{1}	1.0×10^{3}
Sb-126		0.4	10.8	0.4	10.8	3.1×10^{3}	8.4×10^{4}
Sc-44	Scandium(21)	0.5	13.5	0.5	13.5	6.7×10^{5}	1.8×10^{7}
Sc-46		0.5	13.5	0.5	13.5	1.3×10^{3}	3.4×10^{4}
Sc-47		9	243	0.9	24.3	3.1×10^{4}	8.3×10^{5}
Sc-48		0.3	8.11	0.3	8.11	5.5×10^{4}	1.5×10^{6}
Se-75	Selenium(34)	3	81.1	3	81.1	5.4×10^{2}	1.5×10^{4}
Se-79		40	1080	2	54.1	2.6×10^{-3}	7.0×10^{-2}
Si-31	Silicon(14)	0.6	16.2	0.5	13.5	1.4×10^{6}	3.9×10^{7}
Si-32		40	1080	0.2	5.41	3.9	1.1×10^{2}
Sm-145	Samarium(62)	20	541	20	541	9.8×10^{1}	2.6×10^{3}
Sm-147		Unlimited	Unlimited	Unlimited	Unlimited	8.5×10^{-1}	2.3×10^{-8}
Sm-151		40	1080	4	108	9.7×10^{-1}	2.6×10^{1}
Sm-153		4	108	0.5	13.5	1.6×10^{4}	4.4×10^{5}
Sn-113	Tin(50)	4	108	4	108	3.7×10^{2}	1.0×10^{4}
Sn-117m		6	162	2	54.1	3.0×10^{3}	8.2×10^{4}
Sn-119m		40	1080	40	1080	1.4×10^{2}	3.7×10^{3}
Sn-121m		40	1080	0.9	24.3	2.0	5.4×10^{1}
Sn-123		0.6	16.2	0.5	13.5	3.0×10^{2}	8.2×10^{3}
Sn-125		0.2	5.41	0.2	5.41	4.0×10^{3}	1.1×10^{5}
Sn-126		0.3	8.11	0.3	8.11	1.0×10^{-3}	2.8×10^{-2}
Sr-82	Strontium(38)	0.2	5.41	0.2	5.41	2.3×10^{3}	6.2×10^{4}
Sr-85m		5	135	5	135	1.2×10^{6}	3.3×10^{7}
Sr-85		2	54.1	2	54.1	8.8×10^{2}	2.4×10^{4}
Sr-87m		3	81.1	3	81.1	4.8×10^{5}	1.3×10^{7}
Sr-89		0.6	16.2	0.5	13.5	1.1×10^{3}	2.9×10^{4}
Sr-90		0.2	5.41	0.1	2.70	5.1	1.4×10^{2}
Sr-91		0.3	8.11	0.3	8.11	1.3×10^{5}	3.6×10^{6}
Sr-92		0.8	21.6	0.5	13.5	4.7×10^{5}	1.3×10^{7}
T	Tritium(1)	40	1080	40	1080	3.6×10^{2}	9.7×10^{3}
Ta-178	Tantalum(73)	1	27.0	1	27.0	4.2×10^{6}	1.1×10^{8}
Ta-179		30	811	30	811	4.1×10^{1}	1.1×10^{3}
Ta-182		0.8	21.6	0.5	13.5	2.3×10^{2}	6.2×10^{3}
Tb-157	Terbium(65)	40	1080	10	270	5.6×10^{-1}	1.5×10^{1}
Tb-158		1	27.0	0.7	18.9	5.6×10^{-1}	1.5×10^{1}
Tb-160		0.9	24.3	0.5	13.5	4.2×10^{2}	1.1×10^{4}
Tc-95m	Technetium(43)	2	54.1	2	54.1	8.3×10^{2}	2.2×10^{4}
Tc-96m		0.4	10.8	0.4	10.8	1.4×10^{6}	3.8×10^{7}

TABLE A–1.—A_1 AND A_2 VALUES FOR RADIONUCLIDES—Continued

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity	
						(TBq/g)	(Ci/g)
Tc-96		0.4	10.8	0.4	10.8	1.2×10^{4}	3.2×10^{5}
Tc-97m		40	1080	40	1080	5.6×10^{2}	1.5×10^{4}
Tc-97		Unlimited	Unlimited	Unlimited	Unlimited	5.2×10^{-5}	1.4×10^{-3}
Tc-98		0.7	18.9	0.7	18.9	3.2×10^{-5}	8.7×10^{-4}
Tc-99m		8	216	8	216	1.9×10^{5}	5.3×10^{6}
Tc-99		40	1080	0.9	24.3	6.3×10^{-4}	1.7×10^{-2}
Te-118	Tellurium(52)	0.2	5.41	0.2	5.41	6.8×10^{3}	1.8×10^{5}
Te-121m		5	135	5	135	2.6×10^{2}	7.0×10^{3}
Te-121		2	54.1	2	54.1	2.4×10^{3}	6.4×10^{4}
Te-123m		7	189	7	189	3.3×10^{2}	8.9×10^{3}
Te-125m		30	811	9	243	6.7×10^{2}	1.8×10^{4}
Te-127m		20	541	0.5	13.5	3.5×10^{2}	9.4×10^{3}
Te-127		20	541	0.5	13.5	9.8×10^{4}	2.6×10^{6}
Te-129m		0.6	16.2	0.5	13.5	1.1×10^{3}	3.0×10^{4}
Te-129		0.6	16.2	0.5	13.5	7.7×10^{5}	2.1×10^{7}
Te-131m		0.7	18.9	0.5	13.5	3.0×10^{4}	8.0×10^{5}
Te-132		0.4	10.8	0.4	10.8	1.1×10^{4}	3.0×10^{5}
Th-227	Thorium(90)	9	243	1×10^{-2}	0.270	1.1×10^{3}	3.1×10^{4}
Th-228		0.3	8.11	4×10^{-4}	1.08×10^{-2}	3.0×10^{1}	8.2×10^{2}
Th-229		0.3	8.11	3×10^{-5}	8.11×10^{-4}	7.9×10^{-3}	2.1×10^{-1}
Th-230		2	54.1	2×10^{-4}	5.41×10^{-3}	7.6×10^{-4}	2.1×10^{-2}
Th-231		40	1080	0.9	24.3	2.0×10^{4}	5.3×10^{5}
Th-232		Unlimited	Unlimited	Unlimited	Unlimited	4.0×10^{-9}	$1.1\times10_{-7}$
Th-234		0.2	5.41	0.2	5.41	8.6×10^{2}	2.3×10^{4}
Th (natural)		Unlimited	Unlimited	Unlimited	Unlimited	8.1×10^{-9}	2.2×10^{-7}
Ti-44	Titanium(22)	0.5	13.5	0.2	5.41	6.4	1.7×10^{2}
Tl-200	Thallium(81.1)	0.8	21.6	0.8	21.6	2.2×10^{4}	6.0×10^{5}
Tl-201		10	270	10	270	7.9×10^{3}	2.1×10^{5}
Tl-202		2	54.1	2	54.1	2.0×10^{3}	5.3×10^{4}
Tl-204		4	108	0.5	13.5	1.7×10^{1}	4.6×10^{2}
Tm-167	Thulium(69)	7	189	7	189	3.1×10^{3}	8.5×10^{4}
Tm-168		0.8	21.6	0.8	21.6	3.1×10^{2}	8.3×10^{3}
Tm-170		4	108	0.5	13.5	2.2×10^{2}	6.0×10^{3}
Tm-171		40	1080	10	270	4.0×10^{1}	1.1×10^{3}
U-230	Uranium(92)	40	1080	1×10^{-2}	0.270	1.0×10^{3}	2.7×10^{4}
U-232		3	81.1	3×10^{-4}	8.11×10^{-3}	8.3×10^{-1}	2.2×10^{1}
U-233		10	270	1×10^{-3}	2.70×10^{-2}	3.6×10^{-4}	9.7×10^{-3}
U-234		10	270	1×10^{-3}	2.70×10^{-2}	2.3×10^{-4}	6.2×10^{-3}
U-235		Unlimited	Unlimited	Unlimited	Unlimited	8.0×10^{-8}	2.2×10^{-6}
U-236		10	270	1×10^{-3}	2.70×10^{-2}	2.4×10^{-6}	6.5×10^{-5}
U-238		Unlimited	Unlimited	Unlimited	Unlimited	1.2×10^{-8}	3.4×10^{-7}
U (natural)		Unlimited	Unlimited	Unlimited	Unlimited	2.6×10^{-8}	7.1×10^{-7}
U (enriched 5% or less).		Unlimited	Unlimited	Unlimited	Unlimited		(See Table A-3)
U (enriched more than 5%).		10	270	1×10^{-3}	2.70×10^{-2}		(See Table A-3)
U (depleted) ...		Unlimited	Unlimited	Unlimited	Unlimited		(See Table A-3)
V-48	Vanadium(23)	0.3	8.11	0.3	8.11	6.3×10^{3}	1.7×10^{5}
V-49		40	1080	40	1080	3.0×10^{2}	8.1×10^{3}
W-178	Tungsten(74)	1	27.0	1	27.0	1.3×10^{3}	3.4×10^{4}
W-181		30	811	30	811	2.2×10^{2}	6.0×10^{3}
W-185		40	1080	0.9	24.3	3.5×10^{2}	9.4×10^{3}
W-187		2	54.1	0.5	13.5	2.6×10^{4}	7.0×10^{5}
W-188		0.2	5.41	0.2	5.41	3.7×10^{2}	1.0×10^{4}
Xe-122	Xenon(54)	0.2	5.41	0.2	5.41	4.8×10^{4}	1.3×10^{6}
Xe-123		0.2	5.41	0.2	5.41	4.4×10^{5}	1.2×10^{7}
Xe-127		4	108	4	108	1.0×10^{3}	2.8×10^{4}
Xe-131m		40	1080	40	1080	3.1×10^{3}	8.4×10^{4}
Xe-133		20	541	20	541	6.9×10^{3}	1.9×10^{5}
Xe-135		4	108	4	108	9.5×10^{4}	2.6×10^{6}
Y-87	Yttrium(39)	2	54.1	2	54.1	1.7×10^{4}	4.5×10^{5}
Y-88		0.4	10.8	0.4	10.8	5.2×10^{2}	1.4×10^{4}
Y-90		0.2	5.41	0.2	5.41	2.0×10^{4}	5.4×10^{5}
Y-91m		2	54.1	2	54.1	1.5×10^{6}	4.2×10^{7}
Y-91		0.3	8.11	0.3	8.11	9.1×10^{2}	2.5×10^{4}
Y-92		0.2	5.41	0.2	5.41	3.6×10^{5}	9.6×10^{6}
Y-93		0.2	5.41	0.2	5.41	1.2×10^{5}	3.3×10^{6}
Yb-169	Ytterbium(70)	3	81.1	3	81.1	8.9×10^{2}	2.4×10^{4}

TABLE A–1.—A_1 AND A_2 VALUES FOR RADIONUCLIDES—Continued

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity	
						(TBq/g)	(Ci/g)
Yb-175		30	811	0.9	24.3	6.6×10^{3}	1.8×10^{5}
Zn-65	Zinc(30)	2	54.1	2	54.1	3.0×10^{2}	8.2×10^{3}
Zn-69m		2	54.1	0.5	13.5	1.2×10^{5}	3.3×10^{6}
Zn-69		4	108	0.5	13.5	1.8×10^{6}	4.9×10^{7}
Zr-88	Zirconium(40)	3	81.1	3	81.1	6.6×10^{2}	1.8×10^{4}
Zr-93		40	1080	0.2	5.41	9.3×10^{-5}	2.5×10^{-3}
Zr-95		1	27.0	0.9	24.3	7.9×10^{2}	2.1×10^{4}
Zr-97		0.3	8.11	0.3	8.11	7.1×10^{4}	1.9×10^{6}

[a] International shipments of Einsteinium require multilateral approval of A_1 and A_2 values.
[b] International shipments of Fermium require multilateral approval of A_1 and A_2 values.
[c] 20 Ci for Mo99 for domestic use.

TABLE A–2.—GENERAL VALUES FOR A_1 AND A_2

Contents	A_1		AA_2	
	(TBq)	(Ci)	(TBq)	(Ci)
Only beta- or gamma-emitting nuclides are known to be present	0.2	5	0.02	0.5
Alpha-emitting nuclides are known to be present, or no relevant data are available	0.10	2.70	$2\times\times10^{-5}$	$5.41\times\times10^{-4}$

TABLE A–3.—ACTIVITY-MASS RELATIONSHIPS FOR URANIUM

Uranium enrichment [1] wt % U–235 present	Specific activity	
	TBq/g	Ci/g
0.45	$1.8\times\times10^{-8}$	$5.0\times\times10^{-7}$
0.72	$2.6\times\times10^{-8}$	$7.1\times\times10^{-7}$
1.0	$2.8\times\times10^{-8}$	$7.6\times\times10^{-7}$
1.5	$3.7\times\times10^{-8}$	$1.0\times\times10^{-6}$
5.0	$1.0\times\times10^{-7}$	$2.7\times\times10^{-6}$
10.0	$1.8\times\times10^{-7}$	$4.8\times\times10^{-6}$
20.0	$3.7\times\times10^{-7}$	$1.0\times\times10^{-5}$
35.0	$7.4\times\times10^{-7}$	$2.0\times\times10^{-5}$
50.0	$9.3\times\times10^{-7}$	$2.5\times\times10^{-5}$
90.0	$2.2\times\times10^{-6}$	$5.8\times\times10^{-5}$
93.0	$2.6\times\times10^{-6}$	$7.0\times\times10^{-5}$
95.0	$3.4\times\times10^{-6}$	$9.1\times\times10^{-5}$

[1] The figures for uranium include representative values for the activity of the uranium-235 which is concentrated during the enrichment process.

Dated at Rockville, MD this 13th day of September 1995.

For the Nuclear Regulatory Commission.

James M. Taylor,

Executive Director for Operations.

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