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NUCLEAR REGULATORY COMMISSION

10 CFR Part 71

RIN: 3150 - AG-71

COMPATIBILITY WITH IAEA TRANSPORTATION SAFETY STANDARDS (TS-R-1) AND OTHER TRANSPORTATION SAFETY AMENDMENTS

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is proposing to amend its regulations on packaging and transporting radioactive material to make them compatible with the International Atomic Energy Agency (IAEA) standards and to codify other applicable requirements. These changes would be compatible with ST-1 (TS-R-1), the latest revision of the IAEA transportation standards. This rulemaking would also address the unintended economic impact of NRC's emergency final rule entitled "Fissile Material Shipments and Exemptions" (February 10, 1997; 62 FR 5907) and a petition for rulemaking submitted by International Energy Consultants, Inc. (PRM-71-12: February 19, 1998; 63 FR 8362).

DATES: The comment period closes (insert date 90 days after date of publication in the Federal Register). Comments received after this date will be considered if it is practicable to do so, but the Commission is able to assure consideration only for comments received on or before this date.

ADDRESSES: Submit comments to: Secretary, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001. Attention: Rulemaking and Adjudications Staff.

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

You may also provide electronic comments via the NRC's interactive rulemaking website at http://ruleforum.llnl.gov. This site provides the capability to upload comments as files (any format), if your web browser supports that function. For information about the interactive rulemaking website, contact Ms. Carol Gallagher at (301) 415-5905 (e-mail:CAG@nrc.gov).

Documents related to this action may be examined at the NRC Public Document Room (PDR) located at One White Flint North, 11555 Rockville Pike, Room 0-1F21, Rockville, MD. Documents created or received at the NRC after November 1, 1999, are also available electronically at the NRC's Public Electronic Reading Room on the Internet at http://www.nrc.gov/NRC/ADAMS/index.html. From this site, the public can gain entry into the NRC's Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. For more information, contact the NRC PDR Reference staff at 1-800-397-4209, 301-415-4737, or email to pdr@nrc.gov.

FOR FURTHER INFORMATION CONTACT: Naiem S. Tanious, telephone: (301) 415-6103; e-mail: nst@nrc.gov, Office of Nuclear Material Safety and Safeguards, USNRC, Washington, D.C. 20555-0001.

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I. Background

The Commission directed the NRC staff in Staff Requirements Memorandum (SRM) 00-0117 dated June 28, 2000, (1) to use an enhanced public-participation process (website and facilitated public meetings) to solicit public input on the Part 71 rulemaking, and (2) to publish the staff's Part 71 issues paper in the Federal Register (65 FR 44360; July 17, 2000) for public comment. The issues paper presented the NRC's plan to revise Part 71 and provided a summary of the changes being considered, both IAEA-related changes and NRC-initiated changes. The NRC published the issues paper to begin an enhanced public-participation process designed to solicit public input on the Part 71 rulemaking. This process included establishing an interactive website and holding three facilitated public meetings: a "roundtable" workshop at the NRC Headquarters, Rockville, MD, on August 10, 2000, and two “townhall” meetings - one in Atlanta, GA, on September 20, 2000, and a second in Oakland, CA, on September 26, 2000.

SRM-00-0117 also directed the staff to proceed, after completion of the public meetings, with the development of a proposed rule for submittal to the Commission by March 1, 2001. Oral and written comments received from the public meetings, by mail, and through the NRC website, in response to the issues paper, were considered in the drafting of the proposed changes contained herein.

Past NRC-IAEA Compatibility Revisions.

Recognizing that its international regulations for the safe transportation of radioactive material should be revised from time to time to reflect knowledge gained in scientific and technical advances and accumulated experience, IAEA invited Member States (the U.S. is a Member State) to submit comments and suggest changes to the regulations in 1969. 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. In 1979, a review of IAEA's transportation regulations was initiated that resulted in the publication of revised regulations in 1985 (Regulations for the Safe Transport of Radioactive Material, 1985 Edition, Safety Series No. 6).

The NRC also periodically revises its regulations for the safe transportation of radioactive material to make them compatible with those of the IAEA. On August 5, 1983 (48 FR 35600), the NRC published in the Federal Register a final revision to Part 71, "Packaging and Transportation of Radioactive Material." That revision, in combination with a parallel revision of the hazardous materials transportation regulations of the U.S. Department of

Transportation (DOT), brought U.S. domestic transport regulations into general accord with the 1973 edition of IAEA transport regulations. The last revision to Part 71 was published on September 28, 1995 (60 FR 50248), to make Part 71 compatible with the 1985 IAEA Safety Series No. 6. The DOT published its corresponding revision to Title 49 on the same date (60 FR 50291).

The last revision to the IAEA Safety Series No. 6 was named Safety Standards Series ST-1, published in December 1996, and was revised with minor editorial changes in June 2000, and was redesignated as TS-R-1. This rulemaking effort is to evaluate TS-R-1 for potential adoption in Part 71 regulations.

Historically, the NRC coordinated its Part 71 revisions with DOT, because DOT is the U.S. Competent Authority for transportation of hazardous materials. “Radioactive Materials” is a subset of “Hazardous Materials” in Title 49 regulations under DOT authority. Currently, DOT and NRC co-regulate transport of nuclear material in the United States. NRC is continuing with its coordinating effort with the DOT in this rulemaking process.

Scope of 10 CFR Part 71 Rulemaking.

As directed by the Commission, the NRC staff compared TS-R-1 to the previous version of Safety Series No. 6 to identify changes made in TS-R-1, and then identified affected sections of Part 71. Based on this comparison, the NRC staff identified eleven areas in Part 71 that needed to be addressed in this rulemaking process as a result of the IAEA regulations. The staff grouped the Part 71 IAEA compatibility changes into the following issues: (1) Changing Part 71 to the International System of units (SI) (also known as the metric system) exclusively; (2) Radionuclide specific exemption values; (3) Revision of A_1 and A_2 values; (4) Uranium hexafluoride (UF_6) package requirements; (5) Introduction of criticality safety index requirements; (6) Type C packages and low dispersible material; (7) Deep immersion test ;

(8) Grandfathering previously approved packages; (9) Adding and modifying Part 71 definitions; (10) Crush test for fissile material package design; and (11) Fissile material package design for transport by aircraft.

Eight additional NRC-initiated issues (numbers 12 through 19) were identified by Commission direction, and through staff consideration, for incorporation in the Part 71 rulemaking process. These NRC-initiated changes are: (12) Special package approvals; (13) Expansion of Part 71 quality assurance (QA) requirements to holders of, and applicants for, a Certificate of Compliance (CoC); (14) Adoption of the requirements of American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel (B&PV) Code for fabrication of spent fuel transportation packages; (15) Adoption of change authority; (16) Revisions to the fissile-exempt and general license provisions to address the unintended economic impact of the emergency rule (SRM-SECY-99-200); (17) Decision on Petition for Rulemaking PRM-71-12, which requested deletion of the double containment requirements for plutonium; (18) Surface contamination limits as applied to spent fuel and high-level waste packages (SRM-SECY-00-0117); and (19) Part 71 event reporting requirements. NRC published the first 18 issues in an issues paper in the Federal Register on July 17, 2000 (65 FR 44360).

The Part 71 rulemaking is being coordinated with DOT to ensure that consistent regulatory standards are maintained between NRC and DOT radioactive material transportation regulations, and to ensure coordinated publication of the final rules by both agencies. On December 28, 1999 (64 FR 72633), the DOT published an advance notice of proposed rulemaking regarding adoption of ST-1 in its regulations.

II. Summary of Public Comments

The NRC held three public meetings to discuss and hear public comments on the issues under consideration for this rule. These meetings were transcribed by a court reporter; the meeting transcripts and condensed summaries of the comments made in the meeting are available to the public on the NRC's interactive rulemaking website at http:/ruleforum.llnl.gov and the Public Document Room located at One White Flint North, 11555 Rockville Pike, Room 0-1F15, Rockville, MD. Also, the NRC received a total of 48 written comments on the issues paper during the meetings, by the mail, and through the website. All of these written comments have been placed on the NRC website.

This section provides a summary of general comments received at the public meetings that are not associated with any one issue, but rather with the NRC rulemaking process for this effort of the Part 71 revision. A summary of public comments associated with a specific issue is included later in the discussion section under that issue. Comments not specific to this rulemaking effort are not included, nor are they discussed for their relevancy to the scope of this proposed action.

August 10, 2000 Meeting.

Two commenters supported moving towards risk-informed regulation because they believe it will increase the safety of nuclear power plants by allowing the operators to focus on risk-significant issues.

Ten commenters wanted assurance that any changes to the NRC's regulations, whether in the context of conformity with international regulations, or solely affecting domestic shipments of radioactive materials, will not result in a reduction in transportation safety for the public.

Two commenters suggested that NRC provide more information about the specific changes that will be incorporated into a proposed rule. One of these commenters also suggested that NRC consider increasing the number of public meetings and having them early on in the process in locations that will potentially be affected by any changes in the transportation regulations. The commenter also requested that the public comment period for this proposed rule be extended. This commenter also suggested that possibly by coordinating public meetings for all rulemakings or actions related to transportation (e.g., the Package performance Study), the public will be better able to see the interrelation of the various NRC actions.

Two commenters voiced their concern about the public accessibility of documentation related to transportation regulations. Specifically, they were concerned about the legal implications (i.e., due process) of not providing access to documents, such as TS-R-1, TS-G-1.1 (supporting document for TS-R-1), and the ASME code, while requesting public input on potential changes to the regulations to enhance conformity with international and domestic standards and regulations. One commenter noted that without these materials, the underlying basis of a proposed rule cannot be fully explored before its incorporation into the regulations.

Two commenters were seeking clarification on the scope of the proposed changes. The commenters asked whether NRC intends to adopt all of the changes from IAEA's Safety Series 6 regulations that have been incorporated into the current TS-R-1 regulations, or just those identified in the proposed rule. One commenter also sought clarification as to whether the combined regulatory changes anticipated by NRC and DOT would cover all of the changes present in IAEA's TS-R-1 regulations.

Three commenters expressed concern over the possibility that the proposed changes in the transportation regulations could result in materials (including certain bulk materials) that were previously not regulated by NRC suddenly coming under NRC's jurisdiction, or actually becoming exempt in other jurisdictions. One commenter noted that this increased regulation could result in unnecessary concern on the part of the public as to the nature of the materials being transported. One commenter asked specifically if NRC was intending to start regulating naturally-occurring radioactive materials (NORM) and requested clarification on NRC's statutory authority to do so.

One commenter suggested that, in addition to NRC and DOT, State agencies play an important role in the regulation of radioactive materials. The commenter noted that currently 32 States have entered into agreements with the NRC to become Agreement States. As Agreement States they regulate use of radioactive material, and have regulations on transportation of radioactive material, including enforcement authority. The commenter is interested in being able to track possible changes in current regulations and how this could affect regulations at the State level.

Seven commenters were concerned about the harmonization of NRC's regulations with those of the IAEA. The commenters expressed concern over the value of harmonization compared to the costs of implementation, and they further questioned the magnitude of the safety benefits of such harmonization. One commenter questioned that if Member States were not adopting TS-R-1 uniformly, what impact could that have on licensee's ability to transport internationally. Two commenters noted that while the TS-R-1 standards are burdensome, NRC does not want to stop commerce, and that is a risk if NRC does not adopt or harmonize with the TS-R-1 standards.

Another commenter noted that the U.S. should have the right to adopt more stringent standards than those contained in TS-R-1. This commenter argued that uniform regulations should constitute a “minimum” set of requirements and should not be considered the highest standard that should be applicable.

One commenter suggested that NRC and DOT consider adopting a set of guiding principles to assure that harmonization is done in the best interest of public health and safety.

Another commenter suggested that NRC adopt the IAEA regulations using a similar philosophy as is currently used by NRC, that is by doing a safety check and ensuring that the level of safety is not diminished.

Two commenters were seeking clarification on the authority of the international organizations over the activities of the U.S. The commenters suggested that if these organizations are directly influencing what U.S. regulatory agencies do, then the public has the right to more knowledge about their activities. One commenter suggested that any activity to harmonize international regulations with those of the U.S. should be done in open, accountable, democratic forums.

September 20, 2000 Meeting,

Several commenters were frustrated with the rulemaking process. These commenters indicated that a lack of easy access to pertinent resources, including TS-R-1 and relevant sections of the regulations, made it difficult to understand the nature, need, and potential impacts of the proposed changes. These commenters suggested that NRC seek alternative publication methods for relevant documents, such as posting the documents on the NRC website.

Six commenters stated that NRC should only suggest changing existing standards if these changes improve or otherwise strengthen existing standards. Two commenters stated that attempting to affect any other change -- i.e., not increasing the protection of public health and safety and the environment -- is not worth its regulatory costs. However, if NRC is going to pursue these changes, then NRC should weigh heavily potential public and environmental costs. These commenters stated that while NRC is moving towards increased globalization, international standards should be considered a regulatory floor and not a ceiling. One commenter specifically cited that NRC should strengthen “double-casking requirements.”

Three commenters stated that the proposed changes should not be allowed because they would increase public exposure rates without adequately informing the public of any risks associated with the increase. These commenters acknowledged the existence of background exposure rates, but believed that NRC needs to fully inform the public before changing current standards.

Four commenters expressed an interest in better understanding the transportation process and the security arrangements associated with the proposed changes. One commenter specifically requested an explanation to what links existed between this rulemaking process and the NRC, the DOT, and DOE’s currently scheduled shipments of radioactive materials. Another commenter requested an explanation on what security arrangements exist and what preparations NRC and DOT have made to deal with accidents and other such security breaches.

One commenter suggested that the regulatory process be made as open and democratic as possible. This includes ensuring that supporting documents are not too expensive for the public to purchase, or otherwise access. Another commenter suggested that NRC hold additional public meetings to increase public involvement.

September 26, 2000 Meeting.

One commenter expressed his appreciation for the NRC using an enhanced rulemaking process and encouraged the NRC to continue using this process.

Three commenters requested an extension of the public comment period to allow for additional public meetings. One commenter suggested that NRC hold not only additional public meetings, but also representative group sessions where Agreement States' representatives from affected cities, citizens' groups, and industry representatives, discuss "the substantive issues that are implicated by ST-1."

One commenter wanted to ensure that DOT and NRC have a process where NRC would jointly study and, after a reconciliation process, be able to address public comments in a coordinated fashion.

Two commenters found it difficult to clearly identify what changes were being proposed. They requested additional details on the proposed changes and encouraged the NRC to define all of the terms and provide background information in the next iteration. Specifically, they requested information that would enable the public to understand and evaluate the context and rationale for the proposed actions.

Two commenters were concerned that NRC fully examine the impacts of the proposed changes on the U.S. Department of Energy (DOE) as well as other Federal agencies, such as the U.S. Environmental Protection Agency (EPA). One of the commenters stated that, to date, he has not seen any such detailed analysis, an analysis the commenter requested at an earlier time. The commenter stated that when NRC has previously relaxed its standards, DOE has followed suit and cited the example of transportation standards.

One commenter stated that NRC should view IAEA standards as minimum, not maximum, thresholds. The commenter requested that when NRC's regulations are more stringent than similar IAEA regulations, we retain that stringency. The commenter stated that he does not want NRC to lower its standards, and would prefer that international standards be raised.

Comments received on the website and by mail.

Several commenters indicated the importance of adopting uniform regulations by all countries to ensure safe and uninterrupted transportation of radioactive materials internationally. The commenters indicated that the IAEA serves a vital role in developing regulations governing the international shipment of radioactive materials, and without this guidance each country would develop its own regulations, thus making compatibility difficult, if not impossible, to achieve. These commenters strongly urged the NRC and DOT to make every effort to harmonize Part 71 with TS-R-1 regulations, as is reasonably achievable.

Several commenters indicated that the public was not involved in the process that developed the TS-R-1 requirements. As a result, there is no objective analysis available for the public to determine which requirements are appropriate to change, and which ones are not.

One commenter suggested that rather than NRC developing parallel regulations with DOT, NRC's regulations should only address those areas under NRC responsibility, such as fissile material and Type B shipments.

Several commenters indicated that NRC must involve interested members of the public, State and local governments, and Tribes, in a much broader framework in conjunction with the issuance of the proposed rule. One commenter argued that based on attendance at the public meetings, public participation has been inadequate and not representative. Another commenter noted that the public meetings were scheduled too close to the end of the public comment period, and that any meetings or hearings in conjunction with the proposed rule should be staged early in the comment process.

One commenter suggested that the issues paper did not contain sufficient detail indicating the NRC's positions with respect to each of the issues. The commenter stated that inclusion of this information, including any regulatory drivers, would be helpful in furthering the public's understanding of the basis of these proposed changes, most specifically with respect to adoption of TS-R-1 requirements.

One commenter raised the concern that the issues paper was not uniformly clear as to whether a proposed change would strengthen or weaken the protection of public health and safety in the U.S.

One commenter was concerned that the proposal to harmonize NRC's regulations with international standards does not take into account the special nature of transportation in the U.S. For example, the commenter noted that a significant portion of the transportation occurs over distances exceeding 2,400 miles and often in rural areas, where emergency responders are volunteers with limited training. The commenter stated that regulations should be developed to protect emergency responders and other personnel, who could be expected to be in contact with radioactive materials shipments.

Several commenters requested an extension of the public comment period for the issues paper. The commenters cited several examples of why an extension is necessary, including impeded access to relevant information, periods of time during which the PDR was not open to the public, and closure of the Bibliographic Retrieval System for a period of 5 days.

One commenter indicated that over the last several years, the majority of NRC rulemaking initiatives appear to be largely driven by concerns in providing regulatory relief for industry rather than in increasing safety for the public.

One commenter claimed that IAEA standards are colored by consideration of commercial purposes. The commenter requested that NRC set aside commercial considerations in reviewing possible adoption of IAEA standards as NRC is first responsible to the American public and not to the international or domestic nuclear industry.

Two commenters questioned whether NRC would take into account advances in science and engineering and accumulated experience since the development of the IAEA regulations 6 years ago. If not, one commenter argued that the proposed revisions to Part 71 could be outdated before they are issued.

One commenter requested that TS-R-1 be made available for review to fully judge the impact that the proposed changes may have on transportation programs. For example, the commenter noted that one proposed change would result in different shipping names, without specifying those changes.

One commenter suggested that NRC adopt a Transportation Safety Goal documenting the acceptable risk for the transportation of radioactive material.

The public comments were considered in drafting the proposed requirements for 18 of the 19 issues (issue 19 was added after publication of the issues paper). More details are provided under each issue.

NRC has made copies of publicly released documents available on the website at http://www.nrc.gov.NMSS/IMNS/transport/.html. Furthermore, The NRC plans to conduct additional public meetings during the proposed rule comment period. The dates and locations of these meetings will be noticed separately.

III. Discussion

This section is structured to present and discuss each issue separately (with cross references as appropriate). Each issue has four parts: Background, Discussion, NRC Proposed Position, and Affected Sections. The discussion section summarizes the public comments, NRC staff consideration of public comments and of technical and policy issues, and the regulatory analysis for that issue.

A. TS-R-1 Compatibility Issues

Issue 1. Changing Part 71 to the International System of Units (SI) Only

Background. TS-R-1 uses the SI units exclusively. This change is stated in TS-R-1, Annex II, page 199: “This edition of the Regulations for the Safe Transport of Radioactive Material uses the International System of Units (SI)”; the change to SI units exclusively is evident throughout TS-R-1. TS-R-1 also requires that activity values entered on shipping papers and displayed on package labels be expressed only in SI units (paragraphs 543 and 549). Safety Series No. 6 (TS-R-1's predecessor) used SI units as the primary controlling units, with subsidiary units in parentheses (Safety Series 6, Appendix II, page 97), and either units were permissible on labels and shipping papers (paragraphs 442 and 447).

The TS-R-1 change is in conflict with the NRC Metrication Policy issued on June 19, 1996 (61 FR 31169), which allows a dual-unit system to be used (SI units with customary units in parentheses). The NRC Metrication Policy was designed to allow market forces to determine the extent and timing for the use of the metric system of measurements. The NRC is committed, in that policy, to work with licensees and applicants and with national, international, professional, and industry standards-setting bodies (e.g., American National Standard Institute (ANSI), American Society for Testing and Materials (ASTM), American Society of Mechanical Engineers (ASME), et al.) to ensure metric-compatible regulations and regulatory guidance. The NRC encouraged its licensees and applicants, through its Metrication Policy, to employ the metric system wherever and whenever its use is not potentially detrimental to public health and safety, or its use is economic. The NRC did not make metrication mandatory by rulemaking because no corresponding improvement in public health and safety would result, but rather, costs would be incurred without benefit. As a result, licensees and applicants use both metric and customary units of measurement.

According to the NRC's Metrication Policy, the following documents should be published in dual units (beginning January 7, 1993): new regulations, major amendments to existing regulations, regulatory guides, NUREG-series documents, policy statements, information notices, generic letters, bulletins, and all written communications directed to the public. Documents specific to a licensee, such as inspection reports and docketed material dealing with a particular licensee, will be issued in the system of units employed by the licensee.

Currently, Part 71 uses the dual-unit system in accordance with the NRC Metrication Policy.

Discussion. Oral comments received at the public meetings, as well as written comments received on the issues paper, indicate opposition to the use of SI units only. Most commenters were opposed to switching to SI units only, and supported the continued use of the dual-unit system. In one comment, a radiopharmaceutical industry representative noted (August 10 meeting) that the Food and Drug Administration (FDA) requires the use of customary units (curie units), while shipping papers always list the activity in becquerels with curies in parentheses. The representative stated that while that presents some problems now, the industry is able to handle it. By moving to a system where the shipping papers are in SI units only, a situation would be created where the package contents are expressed in curies, while shipping papers and labels are expressed in becquerels. This could be confusing, especially when comparing the shipping papers to the contents. The implication is that this situation could create complications at the shipment destination as personnel would have to perform unit conversions to match package contents with the shipping papers. Furthermore, there was a concern that this could result in errors in patient administrations.   Other commenters indicated that this change would result in significant costs for industry, with no apparent safety benefit.

Another commenter indicated that, although the U.S. has adopted a policy of shifting to SI units, this policy has not been implemented. Several commenters argued that requiring the use of SI units only for domestic shipments of radioactive materials, when the balance of the nation’s activities are conducted in customary units, would cause confusion as well as possible safety issues if misunderstandings or miscalculations were to occur. The commenters noted that the majority of individuals (including emergency response workers) are more accustomed to using customary units, and by requiring the use of SI units, problems would occur in converting customary units to SI units. As a result, the commenters believed that this could result in an increased risk of inadvertent exposure of workers to radiation.

One commenter indicated that SI units are currently required to be used in certain cases for shipping and believed that such a change would pose little risk. However, the commenter added that any such change should be accompanied by a 3-year delay in the effective date to allow for proper transition.

NRC staff notes that the use of SI units only would conflict with the NRC’s Metrication Policy, which allows the use of a dual-unit system for measurements. The statement made in NRC’s final Metrication Policy, “...the NRC believed and continues to believe that if metrication were made mandatory by a rulemaking, no corresponding improvement in public health and safety would result but costs would be incurred without benefit,” still stands.

The NRC draft regulatory analysis (RA) indicates that maintaining the existing policy of allowing the use of dual units is appropriate from a safety, regulatory, and cost perspective. A change to require SI units only would necessitate an exemption by the Commission from its dual-units policy, and would result in an inconsistency between Part 71 and other Parts of the Commission's regulations. Further, anticipated costs to industry for implementing the new requirement (e.g., training, recalculations), estimated to be between $12.6 and $16.3 million, would be avoided if the dual-unit system is maintained. In addition, while NRC would incur $15,000 in costs by converting from one system of units to another, this cost is offset by a savings in resources for not proceeding with rulemaking activities to implement the change. As discussed by several commenters, the change to SI units only could result in the potential for adverse impact on the health and safety of workers and the general public as a result of unintended exposure in the event of shipping accidents, or medical dose errors, caused by confusion or erroneous conversion between the currently prevailing customary units and the new SI units by emergency responders or medical personnel.

The NRC considered the Commission policy on this issue, the above public comments, and the RA of the impact of this change, and concluded that adopting the IAEA use of SI units only in Part 71 would have both a cost impact and potentially negative impact on workers and public health and safety.

NRC Proposed Position. The NRC does not intend to change Part 71 to use SI units only, nor does it intend to impose on Part 71 licensees, certificate holders, or applicants for a Certificate of Compliance (CoC) the use of SI units only. While TS-R-1 uses SI units only, it does not specifically prohibit the use of a dual-unit system (SI units and customary units). Therefore, the NRC will continue to use the dual-unit system in Part 71.

Affected Sections. None (not adopted).

Issue 2. Radionuclide Exemption Values

Background. The DOT currently uses a specific activity threshold of 70 Bq/g (0.002 μCi/g) for defining a material as radioactive for transportation purposes. DOT regulations apply to all materials with specific activities that exceed this value. Materials are exempt from DOT's transportation regulations if the specific activity is equal to or below this value. The 70-Bq/g (0.002-μCi/g) specific activity value is applied collectively for all radionuclides present in a material.

Within § 71.10, the NRC uses the same specific activity threshold as a means of determining if a radioactive material is subject to the requirements of Part 71. Materials are exempt from the transportation requirements in Part 71 if the specific activity is equal to or below this value. Although the materials may be exempt from any additional transportation requirements under Part 71, the requirements for controlling the possession, use, and transfer of materials under Parts 30, 40, and 70 continue to apply, as appropriate, to the type, form, and quantity of material.

During the development of TS-R-1, it was recognized that there was no technical justification for the use of a single activity-based exemption (70-Bq/g) (0.002-μCi/g) value for all radionuclides. It was concluded that a more rigorous technical approach would be to base radionuclide exemptions on a uniform dose basis, rather than a uniform specific activity (also known as activity concentration) basis.

By 1994, the IAEA and other international health-related organizations had developed the *International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources*, IAEA Safety Series No. 115. (This document is sometimes referred to informally as the Basic Safety Standards, or BSS.) During the preparation of this document, a set of principles had been developed and accepted for determining when exemption from regulation was appropriate. One of the exemption criteria was that the effective dose expected to be incurred by a member of the public from a practice (e.g., medical use of radiopharmaceuticals in nuclear medicine applications) or a source within a practice should be unlikely to exceed a value of 10 μSv per year (1 mrem) per year. IAEA Member State researchers developed a set of exposure scenarios and pathways which could result in exposure to workers and members of the public. These scenarios and pathways were used to calculate radionuclide exemption activity concentrations and exemption activities which would not exceed the recommended dose (see Safety Series No. 115, Schedule I, “Exemptions”).

To investigate the exemption issue from a transportation perspective during the development of TS-R-1, IAEA Member State researchers calculated the activity concentration and activity for each radionuclide that would result in a dose of 10 μSv (1 mrem) per year to transport workers under various BSS and transportation-specific scenarios. Due to differences in radionuclide radiation emissions, exposure pathways, etc., the resulting radionuclide-specific activity concentrations varied widely. The appropriate activity concentrations for some radionuclides were determined to be less than 70 Bq/g (0.002 μCi/g), while the activity concentrations for others were much greater. However, the calculated dose to transport workers that would result from repetitive transport of each radionuclide at its exempt activity concentration was the same (10 μSv) per year (1 mrem) per year. For the single activity-based value, the opposite was true, i.e., the exempt activity concentration was the same for all radionuclides (70 Bq/g) (0.002 μCi/g), but the resulting doses under the same transportation scenarios varied widely, with annual doses ranging from much less than 10 μSv (1 mrem) per year for some radionuclides to greater than 10 μSv (1 mrem) per year for others. The radionuclide-specific activity concentration values minimized the variability in doses that were likely to result from exempt transport activities.

IAEA noted that the exempt activity concentrations calculated for transportation scenarios did not differ greatly from those found in Safety Series No. 115 (BSS), Table I-I, “EXEMPTION LEVELS: EXEMPT ACTIVITY CONCENTRATIONS AND EXEMPT ACTIVITIES OF RADIONUCLIDES (ROUNDED).” IAEA did not believe the differences warranted a second set of exemption values, and therefore adopted the Safety Series No. 115 (BSS) values in TS-R-1. These values are found in TS-R-1, paragraphs 401-406, and in Tables I and II. Note that some nuclides listed in Table I have a reference to footnote (b). These nuclides have the radiological contributions from their daughter products (progeny) already included in the listed value. For example, natural uranium [U (nat)] in Table I has a listed activity concentration for exempt material of 1 Bq/g (2.7 x 10^{-5} μCi/g). This means the activity concentration of the uranium is limited to 1 Bq/g (2.7 x 10^{-5} μCi/g), but the total activity concentration of an exempt material containing 1 Bq/g (2.7 x 10^{-5} μCi/g) of uranium will be higher (approximately 7 Bq/g (1.9 x 10^{-4} μCi/g)) due to the radioactivity of the daughter products.

The basis for the exemption values, as discussed in the draft Advisory Material for the Regulations for the Safe Transport of Radioactive Material, TS-G-1.1, paragraphs 107.5 and 401.3, indicates that materials with very low hazards can be safely exempted from the transportation regulations. If the exemptions did not exist, enormous amounts of material with only slight radiological risks, materials which are not ordinarily considered to be radioactive, would be unnecessarily regulated during transport.

Based on TS-R-1, paragraph 236, when both the activity concentration for exempt material and the activity limit for an exempt consignment are exceeded, the material or consignment must meet applicable transportation regulations. Paragraph 404 of TS-R-1 specifies how exemption values may be determined for mixtures of radionuclides.

Some of the lower activity concentration values might include naturally occurring radioactive material (NORM). As an example, ores may contain NORM. In regard to transporting NORM, one petroleum industry representative stated there are no findings that indicate the current standard fails to protect the public, and that there is no benefit in making the threshold more stringent. Further, it would have a significant impact on their operations. Other similar comments were received during the public meetings. The overall impact would be that some material formerly not subject to the radioactive material transport regulations may need to be transported as radioactive material and therefore meet the corresponding applicable DOT transport requirements.

IAEA recognized that application of the activity concentration exemption values to natural materials and ores might result in unnecessary regulation of these shipments, and established a further exemption for certain types of these materials. Paragraph 107(e) of TS-R-1 further exempts: "natural material and ores containing naturally occurring radionuclides which are not intended to be processed for use of these radionuclides provided the activity concentration of the material does not exceed 10 times the values specified in paragraphs 401-406."

Discussion. Comments were received on this issue during the public meetings, by mail, and on the NRC website. One commenter stated that the NRC should reference all DOT equivalent regulations (the radionuclide exemption values and all others) to prevent conflict between the NRC and DOT regulations. Two commenters cautioned that moving from one exemption value to different values for each radionuclide could result in more complicated compliance and enforcement scenarios. For example, one commenter indicated that the 70-Bq/g (0.002-μCi/g) exemption limit is also used as a standard by the U.S. Environmental Protection Agency (EPA) under the Resource Conservation and Recovery Act (RCRA) as the permit limit for the acceptance of material containing radioactive residuals. Any changes to this limit could result in the preclusion of certain materials for disposal at permitted disposal facilities. Some commenters indicated that the revised exemption values should apply not only to domestic shipments but to exported shipments as well.

One commenter indicated that this change will have a significant unintended impact on its operations because most of the oil and gas shipments would not be exempt under the new rule.

One commenter indicated that such a change would result in an increase in the number of shipments by requiring smaller quantities to be shipped due to the lower exemption values. Another commenter suggested that the use of radionuclide-specific exemption values would not result in an increase in the number of packages being shipped, but would result in more shipments being labeled as radioactive. The commenter argued that because many of these shipments are currently being made as “nonhazardous” shipments, many of the responses to accidents will be for minimal hazard materials representing insignificant risks that do not warrant increased response safety. The commenter stated that this would not result in increased safety, but would instead divert emergency response personnel from other, more significant, tasks.

Several commenters reflected a belief that, for some radionuclides, the new higher values would be a relaxation of the regulations, and thus will adversely impact public health and safety. A few commenters indicated that NRC should actually look at making the exemption values more stringent rather than reducing the level of protection currently afforded the public. One commenter suggested that, before adopting any of the exemption values contained in TS-R-1, NRC should scrutinize the values to determine whether they are justified as protective of human health and the environment.

A few commenters supporting the retention of the current Part 71 exemption values indicated that a move to radionuclide-specific exemption values would result in increased costs while yielding no additional safety benefit.

The overall impact would be that some previously exempted material may need to be transported as radioactive material and therefore would need to meet applicable DOT transport requirements. While these activity concentration values would impact certain sectors, the NRC staff believes that the impact of not adopting the international standard would be significantly greater. Therefore, the NRC is proposing to adopt the radionuclide exemption values to assure continued consistency between domestic and international regulations.

In § 71.10(b)(3), the 0.74-TBq (20-Ci) exemption for special form americium and special form plutonium would be removed, except for ^{244}Pu. This provision was originally provided in Part 71 to permit the transportation, in domestic commerce within the United States, of well-logging sealed sources containing up to 0.74 TBq (20 Ci) of radioactive material in Type A packages, even though that quantity of special form americium or plutonium was greater than the individual A_1 limits for these radionuclides. However, over time, the A_1 limits have been raised so that currently only ^{244}Pu has an A_1 limit less than 0.74 TBq (20 Ci) (i.e., 0.4 TBq) (10.81 Ci). Consequently, this exemption is unnecessary for special form americium and special form plutonium, but is still needed for ^{244}Pu.

To prevent an unnecessary economic impact on industry, the 0.74-TBq (20-Ci) exemption for special form ^{244}Pu, transported in domestic commerce, NRC staff believes should be retained as a new § 71.14(b)(2). Furthermore, an exception would be added to § 71.14(b)(1) indicating that paragraph (b)(1) does not apply to special form ^{244}Pu transported in domestic commerce. This exception to the exemption would provide regulatory consistency between paragraphs (b)(1) and (b)(2), while permitting the continued transportation, within the U.S. only, of well-logging sources in a Type A package — when the source contains more than an A_1 quantity of ^{244}Pu, but less than 0.74 TBq (20 Ci). For international shipments, the A_1 quantity limit for special form ^{244}Pu would continue to apply.

The NRC would include the TS-R-1 exemption values in a new table in Appendix A (Table A-2). Additionally, NRC recognized that changes were also required to Appendix A. Specifically, changes would be needed to paragraph II to correct the following problems: (1) The existing paragraph is not in plain language; (2) Guidance is needed on how to determine exempt material activity concentrations and exempt consignment activity limits for unlisted radionuclides; (3) The method of requesting Commission approval, if Table A-3 is not used, needs to be specified; and (4) The existing requirement on requesting NRC prior approval is not listed in the approved Information collection requirements of § 71.6.

The NRC draft RA indicates that adopting the radionuclide-specific exemption values contained in TS-R-1 is appropriate from a safety, regulatory, and cost perspective. Adoption of these values would provide a consistent level of protection for all radionuclides and result in enhanced regulatory efficiency for the NRC and consistency among NRC, IAEA, and DOT. In addition, adoption would result in a single system for determining if materials are subject to domestic or international regulations (e.g., an imported package from England or France, which is exempt, would also be exempt in the United States). NRC believes that this increase in regulatory efficiency and potential cost savings, in some cases, more than offsets the potential increased costs to industry. These costs are anticipated to include minor administrative and procedural changes to use radionuclide-specific exemptions. Also, industry would expend resources to identify the radionuclides in a material, measure the activity concentration of each radionuclide, and apply the “mixture rule” to ensure that a material is exempt. This is in contrast to the current approach of verifying that the material’s total concentration is less than 70 Bq/g (0.002 μCi/g). Further, because some low-level materials may be newly brought into the scope of the regulations, some additional costs may be incurred. However, NRC believes that these costs would be offset by the fact that some materials may be moved outside the scope of the regulations, resulting in a cost savings. Cost savings for shippers of low-level materials shipping both domestically and internationally would also be decreased because they would only have to ensure compliance with one set of requirements as opposed to two distinctly separate sets of requirements. Also, nonadoption of the TS-R-1 values could result in significant negative cost impacts on international commerce. Finally, NRC does not believe that adopting these values would have a significant effect on the total number of shipments domestically or internationally. The changes would also not significantly affect the way these materials are handled.

The NRC considered the above public comments and the draft RA of this change, and concluded that adopting the new IAEA, dose-based, exemption values would improve public health and safety by establishing a consistent dose-model application for minimizing potential dose to transport workers. Within the United States, DOT has the responsibility for regulating the classification of radioactive materials. DOT is also adopting the TS-R-1 exemption concentration activity and exempted consignment values, the NRC is proposing to make conforming changes to Part 71. While these activity concentration values will impact certain sectors, the impact of not adopting the international standard would be significantly greater.   By adopting the provision to allow natural material and ores containing NORM, which are not intended to be processed for the radionuclides, to have an activity 10 times the exemption value, the NRC believes that the impact on the mineral and petroleum industries will be minimized.

<u>NRC Proposed Position.</u>   The NRC is proposing to adopt the radionuclide exemption values in TS-R-1 to assure continued consistency between domestic and international regulations for the basic definition of radioactive material. This adoption into NRC regulations would not impact the Memorandum of Understanding (MOU) (July 2, 1979; 44 FR 38690) between DOT and NRC. The exemptions in existing § 71.10 would be revised to reflect the exempt concentration and exempt consignment values of Appendix A, Table A-2. In addition, provisions for 10 times applicable values would be included for NORM and other natural materials. These changes would conform this rule to DOT's proposed regulations.

<u>Affected Sections.</u>   71.10, 71.88, Appendix A.

Issue 3. Revision of A_1 and A_2

<u>Background.</u> The international and domestic transportation regulations use established activity values to specify the amount of radioactive material that is permitted to be transported in a particular packaging and for other purposes. These values, known as the A_1 and A_2 values, indicate the maximum activity that is permitted to be transported in a Type A package. The A_1 values apply to special form radioactive material, and the A_2 values apply to normal form radioactive material. See § 71.4 for definitions.

In the case of a Type A package, the A_1 and A_2 values as stated in the regulations apply as package content limits. Additionally, fractions of these values can be used (e.g., $1x10^{-3}$ A_2 for a limited quantity of solid radioactive material in normal form), or multiples of these values (e.g., 3,000 A_2 to establish a highway route controlled quantity threshold value).

Based on the results from an updated Q-system (see TS-G-1.1, Appendix I), the IAEA has adopted new A_1 and A_2 values for radionuclides listed in TS-R-1 (see paragraph 201 and Table I). IAEA adopted these new values based on calculations which were performed using the latest dosimetric models recommended by the International Commission on Radiological Protection (ICRP) in Publication 60, "1990 Recommendations of the ICRP." A thorough review of the Q-system also included incorporation of data from updated metabolic uptake studies. In addition, several refinements were introduced in the calculation of contributions to the effective dose from each of the pathways considered. The pathways themselves are the same ones considered in the 1985 version of the Q-system (i.e., external photon dose, external beta dose, inhalation dose, skin and ingestion dose from contamination, and dose from submersion in gaseous radionuclides). The impact of these analyses is that, for each radionuclide, a thorough up-to-date radiological assessment has been performed of potential exposures to an individual should a Type A package of radioactive material be involved in an accident during transport. The new A_1 and A_2 values reflect that assessment.

While the dosimetric models and dose pathways within the Q-system were thoroughly reviewed and updated, the reference doses were unchanged. The reference doses are the dose values which are used to define a “not unacceptable” dose in the event of an accident. Consequently, while some revised A_1 and A_2 values are higher and some are lower, the potential dose following an accident is the same as with the previous A_1 and A_2 values. The revised dosimetric models are used internationally to calculate doses from individual radionuclides, and these refinements in the pathways calculations result in various changes to the A_1 and A_2 values. In other words, where an A_1 or A_2 value has increased, the potential dose is still the same - the use of the revised dosimetric models just shows that a higher activity of that radionuclide is actually required to produce the same reference dose. Conversely, where an A_1 or A_2 value has decreased, the revised models show that less activity of that nuclide is needed to produce the reference dose.

Discussion. Comments on the adoption of the new A_1 and A_2 values were received during the three public meetings and on the NRC website. One commenter stated that to conduct business internationally, there needs to be consistency between the international and domestic regulations. These commenters supported the adoption of the new values into Part 71. Other industry representatives, however, indicated the values should not change as they would need to modify the computer codes at their facility to maintain the ability to accurately meet the regulatory requirements for transportation. Other commenters were concerned about the safety aspects of transportation and the emergency responder’s exposure if the new values should be adopted.

Additional comments were received concerning the A_1 and A_2 values for molybdenum-99 and californium-252. Currently, in Part 71, the A_1 and A_2 values for these radionuclides are: molybdenum-99: A_1: 0.6 TBq (16.2 Ci); A_2: 0.5 TBq (13.5 Ci), and californium-252: A_1: 0.1TBq (2.7 Ci); A_2: 1.0×10^{-3} TBq (2.7 E-2 Ci). Further, Appendix A, Table A-1, the A_2 value for molybdenum-99 has a footnote that indicates for domestic use, the A_2 value is 0.74 TBq (20 Ci). The values from TS-R-1 for these radionuclides are: molybdenum-99: A_1: 1 TBq (27 Ci); A_2: 0.6 TBq (16.2 Ci), and californium-252: A_1: 5.0×10^{-2} TBq (1.35 Ci); A_2: 3.0×10^{-3} TBq (0.08 Ci). Pharmaceutical industry representatives indicated that a change to the new (lower) A_2 value for molybdenum-99 (16.2 Ci vs 20 Ci) would result in a significant increase in the number of packagings shipped, and in occupational doses. DOT is proposing to retain the current exception for molybdenum-99 for domestic commerce, and NRC also believes the current exception for this radionuclide should be retained.

Industry representatives also requested that the current A_1 and A_2 values for californium-252 be retained. Both NRC and DOT have learned that IAEA is considering changing the A_1 and A_2 values in TS-R-1 for californium-252 back to the values currently in Part 71 and 49 CFR. Therefore, NRC plans to retain the current Part 71 A_1 and A_2 values for californium-252 for domestic commerce, as a conforming action with DOT.

Several commenters opposed NRC's proposal to adopt the IAEA A_1 and A_2 values, arguing that any increase in allowable activity levels is unacceptable, could result in increased risk, and would violate the principle of maintaining safety. One commenter stated that the proposed adoption would change from an activity-based limit system to a dose-based limit system, which is unacceptable because dose-based limits are more difficult to verify and enforce than are activity-based limits.

Several commenters stated that NRC should provide a breakdown of which radionuclides would have increased activity levels, and which would remain the same, to allow for meaningful public comment on the proposed change.

Several commenters indicated that adoption of ICRP-60 into NRC regulations would result in another inconsistency within the regulations. Another commenter disagreed, arguing that NRC runs the risk of eroding public confidence in its regulatory role by accepting, then ignoring, the advice of international experts. The commenter argued that there should be a very strong justification if recommendations of the ICRP are to be discounted.

In general, the new A_1 and A_2 values are within a factor of about three of the earlier values; there are a few radionuclides where the new A_1 and A_2 values are outside this range. A few tens of radionuclides (out of more than 300) have new A_1 values higher than previous values by factors ranging between 10 and 100. This is due mainly to improved modeling for beta emitters. There are no new A_1 or A_2 values that are lower than the previous figures by more than a factor of 10. A few radionuclides previously listed are now excluded, but two additional ones have been added, both isomers of europium-150 and neptunium-236. Many A_1 and A_2 values remain unchanged.

The A_1 and A_2 values were revised by IAEA based on refined modeling of possible doses from radionuclides. The NRC staff believes adoption of the IAEA standard would be an overall benefit to public and worker health and international commerce by ensuring that the A_1 and A_2 values are consistent within and between international and domestic transportation regulations.

The NRC draft RA indicates that adopting the new A_1 and A_2 activity limits specified in TS-R-1 is appropriate from a safety, regulatory, and cost perspective. Adoption of these values would result in enhanced regulatory efficiency for the NRC and consistency between NRC, IAEA, and DOT, especially in the handling of imports and exports. Adoption would result in a single set of values for determining the activity limits for specifying the amount of radioactive material permitted to be transported in a particular package for both domestic and international shipments. In some cases, NRC believes that this increase in regulatory efficiency and potential cost savings more than offset the potential increased costs. These costs are anticipated to include revisions to shipping programs to implement the new values, modifications to shipping processes to assure compliance with the new values, and training. These costs, however, are expected to be minor because industry already has programs in place that use the A_1 and A_2 values. In addition, NRC would realize additional minor implementation costs in revising the values in Part 71. The NRC RA indicated no significant change in the number of shipments per year; therefore, accident frequency would not be affected.

NRC Proposed Position. The NRC is proposing to make a conforming change to Part 71 to adopt the new A_1 and A_2 values from TS-R-1 in Part 71, with the differences as discussed for molybdenum-99 and californium-252. This action would allow for continued consistency within and between international and domestic transportation regulations for radioactive materials. The DOT is also proposing to adopt the new TS-R-1 A_1 and A_2 values in their regulations.

Affected Sections. Appendix A**.**

Issue 4. Uranium Hexafluoride Package Requirements

Background. Requirements for uranium hexafluoride (UF_6) packaging and transportation are found in both NRC and DOT regulations. The DOT regulations contain requirements that govern many aspects of UF_6 packaging and shipment preparation, including a requirement that the UF_6 material be packaged in cylinders that meet the American National

Standard Institute ANSI N14.1 standard. NRC regulations address fissile materials and Type B packaging designs for all materials.

TS-R-1 contains detailed requirements for UF_6 packages designed for transport of more more than 0.1 kg UF_6. First, TS-R-1 requires the use of the International Organization for Standardization (ISO) 7195, "Packaging of Uranium Hexafluoride for Transport." Second, TS-R-1 requires that all packages containing more than 0.1 kg UF_6 must meet the "normal conditions of transport" drop test, a minimum internal pressure test and the hypothetical accident condition thermal test (para 630). However, TS-R-1 does allow a competent national authority to waive certain design requirements, including the thermal test for packages designed to contain greater than 9,000 kg UF_6, provided that multilateral approval is obtained. Third, TS-R-1 prohibits UF_6 packages from using pressure relief devices (para 631).   Fourth, TS-R-1 includes a new exception for UF_6 packages regarding the evaluation of criticality safety of a single package. This new exception (para 677(b)) allows UF_6 packages to be evaluated for criticality safety without considering the inleakage of water into the containment system. Consequently, a single fissile UF_6 package does not have to be subcritical assuming that water leaks into the containment system. This provision only applies when there is no contact between the valve body with the cylinder body under accident tests, and the valve remains leak-tight, and when there is quality controls in the manufacture, maintenance, and repair of packagings coupled with tests to demonstrate closure of each package before each shipment.

Discussion. One commenter indicated serious concerns about the safety margins for UF_6 packaging. The commenter cited the exception in TS-R-1, paragraph 677(b), which would allow UF_6 packages to be evaluated for criticality without considering the inleakage of water. The commenter cited a report describing one case where UF_6 packages with manufacturing defects were used. The commenter indicated that it would be imprudent and unwise public policy to assume that water could not leak into a package containing UF_6.

Another commenter stated that a justification for the reduced regulatory burden has not been established and cannot be done unless a risk study, which determines the level of conservatism currently contained in Part 71, is conducted. Without this analysis, the commenter argued, reduction of regulatory burden leading to inadvertent criticality could lead to loss of life, degradation of the environment, economic repercussions, and degradation of public confidence.

Also, comments at the public meetings supported the NRC view that ANSI N14.1 and ISO 7195 are equivalent. Further, other comments indicated that NRC-certified UF_6 packages already comply with TS-R-1 paragraphs 630 and 677(b).

The provisions of § 71.55(b) specify that a fissile material package must be designed, or the contents limited, so that a single package would be critically safe if water were to leak into the containment vessel. This is a design feature that assures criticality safety in transport, in the unanticipated event that water leaks into the containment vessel, and provides moderating materials for the fissile contents. The proposed new § 71.55(g) would except fissile UF_6 from the requirement that a single package must be critically safe with water inleakage. This is consistent with the worldwide practice in shipping fissile UF_6, and is consistent with ANSI N14.1 and ISO 7195 standards and DOT regulations.

The proposed rule language further restricts use of the exception to a maximum enrichment of 5 weight percent uranium-235. This is the maximum enrichment currently authorized in ANSI N14.1, ISO 7195, and DOT regulations in cylinders larger than 20.3 cm (8 inches) in diameter. For smaller cylinders, the exception is not needed because current enrichments are critically safe by geometry for a single package. The exception, with the enrichment limit, codifies current worldwide practice in shipping fissile uranium hexafluoride.

Large quantities of enriched (greater than 5 weight percent uranium-235) UF_6 would require packages that meet the water inleakage standards in § 71.55(b). The staff believes that it is not prudent to expand this exception to include UF_6 shipments with higher uranium enrichments.

The NRC draft RA indicates that revising the current requirements for uranium hexafluoride packages to include an exception from the requirement that single packages must be critically safe from water inleakage is appropriate from a safety, regulatory, and cost perspective. In developing the RA, the NRC first determined that there are no substantial differences between ANSI N14.1 standard and ISO 7195 standard for UF_6 packaging, and therefore, there would be no significant cost impacts from this change, because NRC currently requires conformance with ANSI N14.1, but regulatory efficiency would be enhanced by making Part 71 compatible with TS-R-1. The internal pressure test and drop test requirements are currently met by existing package designs that comply with ANSI N14.1. Therefore, there would be limited impact on licensees by this aspect of the NRC action. The NRC staff also considered the United States' earlier opposition (Taylor, 1996) to this change, i.e., the IAEA adopting the UF_6 package requirements. Most of the impact of adopting the TS-R-1 UF_6 provisions would fall on the 30-inch and 48-inch bare cylinders that are within the purview of the DOT and for which there is a "multilateral" approval option that could be used to mitigate most of this potential impact to licensees. Therefore, the adoption of the TS-R-1 requirements are not expected to have significant impact on fissile package designs for UF_6. Because the changes are not expected to have significant impacts on current package designs, changes in environmental impacts are expected to be negligible.

NRC Proposed Position. The NRC proposes the adoption of a new requirement, § 71.55(g), to address TS-R-1, paragraph 677(b), to exempt certain UF_6 packages from the requirements of § 71.55(b). The requirements in TS-R-1, paragraphs 629, 630, and 631, do not necessitate changes to Part 71 because NRC uses analogous national standards and addresses package design requirements in its design review process. All NRC-certified packages must be used in accordance with DOT requirements (including the UF_6 requirement in 49 CFR 173.420).

Affected Sections. 71.55.

Issue 5. Introduction of the Criticality Safety Index Requirements

Background. Historically, the IAEA and U.S. regulations (both NRC and DOT) have used a term known as the Transport Index (TI) to determine appropriate safety requirements during transport. TI has been used to control the accumulation of packages for both radiological safety and criticality safety purposes and to specify minimum separation distances from persons (radiological safety). The TI has been a single number which is the larger of two values: the "TI for criticality control purposes"; and the "TI for radiation control purposes." Taking the larger of the two values has ensured conservatism in limiting the accumulation of packages in conveyances and in-transit storage areas.

TS-R-1 (paragraph 218) has introduced the concept of a Criticality Safety Index (CSI) separate from the old TI. As a result, the TI was redefined in TS-R-1. The CSI is determined in the same way as the "TI for criticality control purposes," but now it must be displayed on shipments of fissile material (paragraphs 544 and 545) using a new "fissile material" label. The redefined TI is determined in the same way as the "TI for radiation control purposes" and continues to be displayed on the traditional "radioactive material" label.

Discussion. Comments received on this proposal indicated that the industry supports the use of the new label "CSI" in conjunction with the "TI" labels, and stated that separate labels are more meaningful and provide additional safety in transport, as long as the two labels are distinctive, so as to avoid confusion.

In general, public comments received at the meetings supported the use of the CSI. One commenter believed that using the TI as the means to control criticality safety does not provide emergency responders with information on the undamaged condition of the package. Other commenters suggested that NRC should provide the underlying technical justification for the term “equivalent safety,” because otherwise, this change would seemingly allow for more packages in a single shipment. This provides an equivalent safety because the CSI uses the same methodology (§ 71.59) that was used to calculate the criticality position of the current TI.

One industry commenter disagreed that the CSI requirement is appropriate. The commenter stated that the TI already incorporates the more restrictive value and provides adequate protection. The commenter believed there is no increase in safety by adding this new requirement and, in fact, it would result in more opportunities for human error. Further, the commenter indicated that any benefit for adding the CSI is far outweighed by the additional labor, material, training, and administration costs that would be borne by a company that ships thousands of packages each year.

The NRC RA indicates that introducing new criticality safety index requirements into Part 71 is appropriate from a safety, regulatory, and cost perspective. NRC would require that applicants for fissile material package design approvals clearly indicate the CSI value for the design. The CoCs the NRC issues for these designs would also need to clearly indicate the CSI value for authorized contents. The adoption of the CSI values would make Part 71 consistent with TS-R-1, therefore enhancing regulatory efficiency. The total annual estimated cost of the new label to the nuclear power licensees and material licensees is approximately $1.4 million on approximately 2.8 million shipments. Some of these costs would be offset by the fact that for some shipments of fissile material packages, the accumulation of packages for criticality control purposes and the accumulation of packages (including minimum separation distances from persons) for radiological control purposes are shipped independently (the most restrictive criteria would not control the other as is the case with the current dual-use TI). Further, increased efficiency in shipping some fissile material packages could occur by avoiding the situation where separation distance requirements (radiological safety) unduly restrict package accumulation (criticality safety). From a health and safety perspective, emergency responders in accident circumstances (thus public health and safety) benefit from more clearly displayed information upon arrival at the accident scene.

NRC Proposed Position. The NRC proposes to adopt the TS-R-1 (paragraph 218) which incorporates a CSI in Part 71 that would be determined in the same manner as the current Part 71 "TI for criticality control purposes." A TI will be determined in the same way as the "TI for radiation control purposes." The NRC believes the differentiation between criticality control and radiation protection would better define the hazards associated with a given package and, therefore, provide better package hazard information to emergency responders.

Affected Sections. 71.4, 71.18, 71.20, 71.59**.**

Issue 6. Type C Packages and Low Dispersible Material

Background. TS-R-1 has introduced two new concepts: the Type C package (paragraphs 230, 667-670, 730, 734-737) and the Low Dispersible Material (LDM). The Type C packages are designed to withstand severe accident conditions in air transport without loss of containment or significant increase in external radiation levels. The LDM has limited radiation hazard and low dispersibility; as such, it could continue to be transported by aircraft in Type B packages (i.e., LDM is excepted from the TS-R-1 Type C package requirements). U.S. regulations do not contain a Type C package or LDM category, but do have specific requirements for the air transport of plutonium (§§ 71.64 and 71.74). These specific NRC requirements for air transport of plutonium would continue to apply.

The Type C requirements apply to all radionuclides packaged for air transport that contain a total activity value above 3,000 A_1 or 100,000 A_2, whichever is lesser, for special form material, or above 3,000 A_2 for all other radioactive material. Below these thresholds, Type B packages would be permitted to be used in air transport. The Type C package performance requirements are significantly more stringent than those for Type B packages. For example, a 90-meter per second (m/s) impact test is required instead of the 9-meter drop test. A 60-minute fire test is required instead of the 30-minute requirement for Type B packages. There are other additional tests, such as a puncture/tearing test, imposed for Type C packages. These stringent tests are expected to result in package designs that would survive more severe aircraft accidents than Type B package designs.

The LDM specification was added in TS-R-1 to account for radioactive materials (package contents) that have inherently limited dispersibility, solubility, and external radiation levels. The test requirements for LDM to demonstrate limited dispersibility and leachability are a subset of the Type C package requirements (90-m/s impact and 60-minute thermal test) with an added solubility test, and must be performed on the material without packaging. The LDM must also have an external radiation level below 10 mSv/hr (1 rem/hr) at 3 meters. Specific acceptance criteria are established for evaluating the performance of the material during and after the tests (less than 100 A_2 in gaseous or particulate form of less than 100-mm aerodynamic equivalent diameter and less than 100 A_2 in solution). These stringent performance and acceptance requirements are intended to ensure that these materials can continue to be transported safely in Type B packages aboard aircraft.

In 1996, the NRC communicated to the IAEA that the NRC did not oppose the IAEA adoption of the newly created Type C packaging standards (letter dated May 31, 1996, from

James M. Taylor, EDO, NRC, to A. Bishop, President, Atomic Energy Control Board, Ottawa, Canada). However, Mr. Taylor stated in the letter that to be consistent with U.S. law, any plutonium air transport to, within, or over the U.S. will be subject to the more rigorous U.S. packaging standards.

Discussion: Comments from the public suggested that Type C standards might increase the number of shipments with smaller quantities of material using the same Type B containers to avoid the cost of developing Type C packages and to avoid the requirement of meeting the new Type C package standards. One commenter indicated that any proposal to change package design requirements should only be contemplated after a thorough technical review that has independently justified the change as protective.

However, one commenter stated that NRC should remove from its regulations the plutonium-specific requirements for air transport, and replace them with the Type C package requirements. Also, the commenter stated that because Type C package development would take a number of years, industry would work with the NRC to define tests, analyses, and criteria for demonstrating compliance with the Type C package standards.

One commenter questioned the rigorousness of the testing described in TS-R-1, indicating that the minimum acceptable impact speed should be increased to at least 129 m/s, as was mandated by Congress.

The staff evaluated the Type C package, and proposes that the NRC not adopt Type C or LDM requirements at this time. The bases for this staff proposal include: (1) IAEA is planning to develop aircraft accident severity information through a coordinated research project for further evaluation of the Type C and LDM requirements; (2) the fact that there are very few anticipated shipments affected by these requirements; (3) DOT rules permit the use of IAEA standards in nonplutonium import/export shipments of foreign certified Type C containers, so that international commerce is not impacted; (4) NRC's domestic regulations currently in place (§§ 71.64 and 71.74), based in specific statutory mandates, governing air transport of plutonium (plutonium air transport was a considerable factor in IAEA adoption of Type C provisions); and (5) comments made by the public on the issues generally disagreed with or questioned the rigor of the Type C tests, and supported NRC maintaining its current regulatory requirements for the safety of plutonium air shipments.

The DOT reviews the use of packages for import or export shipment. Consequently, foreign Type C packages could be approved by DOT for import and export only. The NRC does not believe that a Type C package is needed for domestic commerce, therefore, no provisions would be added to Part 71 relating to Type C packages. However, should DOT request that NRC perform a technical evaluation for a revalidation of a foreign Type C package design, NRC would evaluate the design against TS-R-1 Type C standards. Similarly, if requested by DOT, NRC would review a domestic Type C package design intended for use in international commerce against TS-R-1, and provide NRC's recommendation to DOT (Note that NRC revalidation of designs for DOT does not constitute NRC issuance of a certificate of compliance).

The NRC RA indicates that not adopting the TS-R-1 Type C or LDM provisions in Part 71 is appropriate from a safety, regulatory, and cost standpoint. There may be some reduction in regulatory efficiency as a result of the nonadoption of the TS-R-1 requirements, which could result in NRC case-by-case reviews to support international shipments. NRC would continue to use its proven, safe regulatory requirements for air transport of plutonium. Further, NRC staff resources are conserved by nonadoption, and no additional costs would be incurred by industry. These additional costs to industry would include implementation costs for the design of new packages to meet the Type C requirements rather than using existing Type B packages.

NRC Proposed Position. The NRC proposes not to adopt Type C or LDM requirements at this time.

Affected Sections. None (not adopted).

Issue 7. Deep Immersion Test

Background. TS-R-1 expanded the performance requirement for the deep water immersion test (paragraphs 657 and 730) from the requirements in the IAEA Safety Series No. 6, 1985 edition. Previously, the deep immersion test was only required for packages of irradiated fuel exceeding 37 PBq (1,000,000 Ci). The deep immersion test requirement is found in Safety Series No. 6, paragraphs 550 and 630, and basically stated that the test specimen be immersed under a head of water of at least 200 meters (660 ft) for a period of not less than one hour, and that an external gauge pressure of at least 2 MPa (290 psi) shall be considered to meet these conditions. The TS-R-1 expanded immersion test requirement (now called enhanced immersion test) now applies to all Type B(U) [Unilateral] and B(M) [Multilateral] packages containing more than 10^5 A_2, as well as Type C packages.

In its September 28, 1995 (60 FR 50264), rulemaking for Part 71 compatibility with the 1985 edition of Safety Series No. 6, the NRC addressed the new Safety Series No. 6 requirement for spent fuel packages by adding § 71.61, "Special requirements for irradiated nuclear fuel shipments." Currently, § 71.61 is more conservative than Safety Series No. 6 with respect to irradiated fuel package design requirements. It requires that a package for irradiated nuclear fuel with activity greater than 37 PBq (10^6 Ci) must be designed so 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*. The conservatism lies in the test criteria of no collapse, buckling, or inleakage as compared to the *"no rupture"* criteria found in Safety Series No. 6 and TS-R-1. The draft advisory document for TS-R-1 (TS-G-1.1, paragraphs 657.1 to 657.7) recognizes that leakage into the package and subsequent leakage from the package is possible while still meeting the IAEA requirement.

The Safety Series No. 6 test requirements were based on risk assessment studies that considered the possibility of a ship carrying packages of radioactive material sinking at various locations. The studies found that, in most cases, there would be negligible harm to the environment if a package were not recovered. However, should a large irradiated fuel package (or packages) be lost on the continental shelf, the studies indicated there could be some long term exposure to man through the food chain. The 200-meter (660-ft) depth specified in Safety Series No. 6 is equivalent to a pressure of 2 MPa (290 psi), and roughly corresponds to the continental shelf and to depths that the studies indicated radiological impacts could be important. Also, 200 meters (660 ft) was a depth at which recovery of a package would be possible, and salvage would be facilitated if the containment system did not rupture. (Reference Safety Series No. 7, paragraphs E-550.1-.3.)

The expansion in scope of the deep immersion test was due to the fact that radioactive materials, such as plutonium and high-level radioactive wastes, are increasingly being transported by sea in large quantities. The threshold defining a large quantity as a multiple of A_2 is considered to be a more appropriate criterion to cover all radioactive materials, and is based on a consideration of potential radiation exposure resulting from an accident.

Discussion. Several comments received at the public meetings, as well as written comments received on the issues paper, indicated support for retaining the current, more stringent, requirements contained in § 71.61 with respect to not allowing collapse, buckling, or inleakage of water in the containment vessel. One commenter was concerned that the term "rupture" seemed less stringent than "collapse, buckling, or inleakage of water." The commenter noted, however, that the issues paper does not include definitions for "rupture" or "buckling," so it is difficult to know which term is more or less stringent. Another commenter believed that the proposed test requirement of withstanding underwater pressure for at least an hour is insufficient. The commenter explained that it is unrealistic to expect to recover nuclear materials from the water within 1 hour after a major accident.

One commenter questioned whether there was sufficient technical justification for relaxing the current NRC test criteria for packages of irradiated nuclear fuel. The commenter stated that a lot of environmental damage can occur before a rupture develops, and that the proposal does nothing to ensure that packages are as safe as they can be.

Another commenter noted that TS-R-1 refers only to normal form material for the immersion test. Specifically, the commenter asked what the criteria are for a special form A_1 quantity, and whether the deep immersion test was necessary for B(U) packages for special form materials. NRC reviewed the IAEA regulations and believes that this requirement applies to both normal form and special form material. Similarly, one commenter noted that, in practicality, the quantities listed would be limited to irradiated fuel elements, and that shipment of radioisotopes rarely contain these amounts. This commenter suggested that the present criteria be maintained and extended to cover all packages with activity levels greater than or equal to 10^5 A_2 quantities with the note that this is more conservative than TS-R-1 requirements. The commenter stated this should eliminate the requirement for special review and certification of U.S. origin package designs. For nonirradiated fuel element shipments, the commenter believed there should be no impact on availability and shipping costs because there are few shipments of the required quantities of this material. Finally, the commenter questioned whether, with the application to B(U) packages containing A_1 special form sources, these packages are exempt from this test.

In response to the question about how to address the differences in acceptance standards, two commenters stated that due to the international nature of transportation activities, U.S. transportation regulations should be consistent with IAEA transportation regulations and, therefore, NRC should adopt the TS-R-1 requirements for the enhanced deep immersion test.

Two commenters also addressed whether U.S. origin package designs should be specifically reviewed and certified before shippers can export them. One commenter said that if the response is not specific to the deep immersion test, but applies to all package design criteria, then the shipment of U.S. certified package designs for import/export use beginning in mid-2001 is entirely dependent upon approval of these designs to TS-R-1 performance standards. The commenter believed that failure to grant U.S. Competent Authority certifications for these designs would seriously hinder the industrial radiography industry, and place U.S. package designers and manufacturers at a strong competitive disadvantage. The commenter added that several of its shipments were not acceptable in several countries when NRC and DOT failed to adopt Safety Series No. 6 in a timely manner.

Another commenter stated that NRC should clarify if previously approved packages would be grandfathered**,** or if they would have to be recertified by means of a deep immersion test.

The NRC proposes revising Part 71 requiring an enhanced water immersion test for packages used for radioactive contents with activity greater than $10^5 A_2$. Section 71.61 currently refers to packages for irradiated fuel with activity greater than 37 PBq (10^6 Ci); the water immersion test would need to be changed to apply to Type B packages containing greater than $10^5 A_2$ and Type C packages. Given that any package containing spent fuel with activity greater than 37 PBq (10^6 Ci) would also have an activity significantly greater than $10^5 A_2$, such a change would bound Type B spent fuel packages currently addressed in 10 CFR 71.61.

Therefore, a specific reference to special requirements for irradiated nuclear fuel shipments would no longer be required.

As mentioned earlier, there is a difference between the test acceptance criteria specified in TS-R-1 and § 71.61. Safety Series No. 6 refers to no rupture, while § 71.61 requires no collapse, buckling, or inleakage of water when subjected to the test conditions. In the September 28, 1995, rulemaking, NRC staff provided justification for the more specific NRC acceptance criteria. The rulemaking stated that: "NRC has since determined that the term 'rupture' cannot be determined by engineering analysis and that NRC has decided to change the acceptance criteria for the deep immersion test from 'rupture' to 'collapse, buckling, or inleakage of water'."

Given that the TS-R-1 background material does not provide any new information on defining the term "rupture" from that provided for Safety Series No. 6, the NRC intends to retain the current interpretation of "rupture" to mean "collapse, buckling, or inleakage of water," in any revision to § 71.61. During the comment period for the proposed rule, should information be provided about how the term "rupture" should be defined, or on how foreign countries have certified packages to this criterion, then the NRC will consider this in determining whether the "collapse, buckling, or inleakage of water" criteria should be revised before issuing the final rule.

The NRC RA indicates that revising Part 71 to require an enhanced water immersion test for packages used for radioactive contents with activity greater than 10^5 A_2 while retaining the current § 71.61 interpretation of "rupture" to mean "collapse, buckling, or inleakage of water," is appropriate from a safety, regulatory, and cost perspective. First, the proposed change would improve regulatory efficiency by bringing U.S. regulations in harmony with the standards contained in TS-R-1. This would improve the efficiency of handling imports and exports and would make U.S. standards compatible with other IAEA Members States.

Implementation of the proposed change could result in costs to licensees as they test and certify packages to the proposed standard. The NRC may incur costs for developing procedures, reviewing and approving test results, and recertifying packages. The proposed change may reduce impacts to public health in the case of an accident. A package tested to the new requirements would be able to withstand pressure at increased depths without collapsing, buckling, or allowing inleakage of water, thereby keeping the radioactive materials enclosed. The likelihood of a member of the public receiving a dose from a package resting in deep water is exceedingly small and would be even smaller if the proposed change were implemented in that the test would apply to a broad range of packages. Moreover, the duration of the test, 1 hour, is reasonable for a package resting in deep water, because the water pressure will be constant, and the 1 hour test will clearly establish if the package can withstand that pressure. A successfully-tested package would be able to withstand the pressure at this depth without rupturing, thereby keeping the radioactive materials enclosed and permitting a reasonable length of time for recovery. Retaining package integrity would prevent the possible expenses of restricting the area (to prevent users such as boaters or fishers from entering the vicinity) and remediating any contamination of the marine environment.

NRC Proposed Position. The NRC proposes to adopt the requirement for enhanced water immersion test for packages used for radioactive contents with activity greater than 10^5 A_2. The NRC intends to retain the current test requirements in § 71.61 of "one hour without collapse , buckling, or inleakage of water."

Affected Sections. 71.41, 71.51, 71.61.

Issue 8. Grandfathering Previously Approved Packages

Background. Historically, the IAEA, DOT, and NRC regulations have included transitional arrangements or “grandfathering” provisions whenever the regulations have undergone major revision. The purpose of grandfathering is to minimize the costs and impacts of implementing changes in the regulations on existing package designs and packagings. Grandfathering typically includes provisions that allow: (1) continued use of existing package designs and packagings already fabricated, although some additional requirements may be imposed; (2) completion of packagings that are in the process of being fabricated or that may be fabricated within a given time period after the regulatory change; and (3) limited modifications to package designs and packagings without the need to demonstrate full compliance with the revised regulations, provided that the modifications do not significantly affect the safety of the package.

Each transition from one edition of the IAEA regulations to another (and the corresponding revisions of the NRC and DOT regulations) has included grandfathering provisions. TS-R-1 includes provisions which apply to packages and special form sources previously approved in accordance with the 1973 and 1985 editions of the IAEA regulations. Previously, Safety Series No. 6 (1985) (as amended 1990) contained provisions applicable to packages approved under the 1967 and 1973 (as amended) editions of the IAEA regulations.

TS-R-1 grandfathering provisions (see TS-R-1, paragraphs 816 and 817) are more restrictive than those previously in place in Safety Series 6 (1985) (as amended 1990). The primary impact of these two paragraphs is that Safety Series 6 (1967) approved packagings are no longer grandfathered, i.e., cannot be used. The second impact is that fabrication of packagings designed and approved under Safety Series 6 (1985) (as amended 1990) must be completed by a specified date.

In TS-R-1, packages approved for use based on Safety Series 6 1973 (as amended) can continue to be used through their design life, provided the following conditions are satisfied: multilateral approval is obtained for international shipment, applicable TS-R-1 QA requirements and A_1 and A_2 activity limits are met, and, if applicable, the additional requirements for air transport of fissile material are met. While existing packagings are still authorized for use, no new packagings can be fabricated to this design standard. Changes in the packaging design or content that significantly affect safety require that the package meet current requirements of TS-R-1.

TS-R-1 further states that those packages approved for use based on Safety Series 6 (1985) (as amended 1990) may continue to be used until December 31, 2003, provided the following conditions are satisfied: TS-R-1 QA requirements and A_1 and A_2 activity limits are met and, if applicable, the additional requirements for air transport of fissile material are met. After December 31, 2003, use of these packages for foreign shipments may continue under the additional requirement of multilateral approval. Changes in the packaging design or content that significantly affect safety require that the package meet current requirements of TS-R-1. Additionally, new fabrication of this type packaging must not be started after December 31, 2006. After this date, subsequent package designs must meet TS-R-1 package approval requirements.

Discussion. Industry representatives were concerned that IAEA is adopting a 2-year revision cycle to TS-R-1. From a design approval point of view, the regulatory requirements to be met may not be understood, and, as a new design requirement is approved, new revisions to the regulations could conceivably be developed. In other words, industry may always be playing catch up with the regulations.

Previously, the IAEA standards permitted a package to be manufactured for two revision cycles of the IAEA standard. Because the IAEA standard was revised every 10 years, this equated to a 20-year period. However, IAEA is now changing to a 2-year revision cycle. Retaining the 2-year cycle provision would now equate to a 4-year allowable manufacturing period. This issue is under review by IAEA; therefore, the NRC is specifying in existing § 71.13 when packages can no longer be manufactured or used, rather than using a "two-revision cycle" process.

Additionally, a commenter expressed concern that beyond 2006, while packages could continue to be used under a valid CoC, no new packages could be manufactured based on any edition of Safety Series 6. Furthermore, after December 31, 2006, all ensuing packages would have to fully meet TS-R-1 requirements. The commenter stated that the licensing process for a package could be impacted. While NRC is aware and understands this concern, the proposed changes to § 71.13 are adequate to address the potential limitation on fabrication and use.

One commenter stated that the expense of designing and fabricating large Type B and spent fuel packages cannot be justified if the potential lifetime of the cask is limited to as short a period of time as 6 years. The commenter also believed that design and contents modifications should be allowed as specified in the current § 71.13(c). Conversely, one commenter stated that a 2-year updating cycle would force safety considerations in cask design up front, rather than continuing the attitude that casks be used as long as possible.

Another commenter urged NRC to include a grandfathering provision for continued transportation of packages, such as CoC packages at NRC, and DOT specification packages. The commenter explained that if NRC did not have a grandfathering provision, NRC would have to set aside hundreds of long-term disposal sites for the various Type B quantity containers currently in use at hospitals and research institutions.

Several commenters believed that grandfathering would allow the NRC to maintain an adequate level of safety for package designs. Some commenters stated that existing packages (even older ones) were safe and durable, because these packages must be maintained in accordance with the QA regulations of Part 71. Another commenter added that under current regulations, NRC may immediately recall a certification if a particular package created a safety concern.

One commenter voiced support for the proposal, assuming new regulations would continue to be more strict. Two commenters believed that while it is important for more stringent requirements to apply to all existing containers, relaxed provisions would effectively make newer containers less safe. In these instances, the commenters preferred that the older provisions remain in effect, instead of the newer, relaxed provisions. One commenter opposed grandfathering existing packages, and stated as a concern the unknown safety of older packages.

One commenter believed that NRC should incorporate specific requirements into the grandfathering provision to effectively maintain a good package program. The commenter explained that manufacturers of CoC containers or packages should be allowed to show, by calculations or testing, that upgraded standards and TS-R-1 have been achieved.

One commenter stated that the shorter cycle would put pressure on cask designers to make safety a more important design element.

In response to the question about the type and magnitude of package design changes that should be allowed for grandfathered packages before recertification is required, two commenters stated that TS-R-1 allows for a phase out of manufacturing of any packages that are not certified to the 1996 version of TS-R-1 by December 31, 2006. The commenters added that this provides a window for the design, testing, and certification of new packages, the reevaluation of existing packages to the 1996 specification, or a request for special certification.

The NRC recognizes that when the regulations change there is not necessarily an immediate need to discontinue use of packages that were approved under previous revisions of the regulations. Part 71, therefore, has always included provisions that would allow previously-approved designs to be upgraded and to be evaluated to the newer regulatory standards. NRC believes that packages approved under the 1967 edition of the regulations, and which have not been updated to later editions, may lack safety enhancements which have been included in the packages approved to the 1973 and 1985 editions. Therefore, the NRC believes that it is appropriate to begin a phased discontinuance of these earlier packages (1967-approved) to further improve transport safety. Since NRC adoption of the 1973 SS 6 provisions in 1983, these improved safety features have been included in all NRC-certified designs. Some of the improvements that affect transport safety include:

1. *The introduction of the A_1 and A_2 system.* Prior to the 1973 Edition of the IAEA regulations, the regulations were based on Transport Groups. The A_1 and A_2 system was intended to use a consistent safety basis for package contents based on radiological protection in transportation under normal and accident conditions.
2. *Standards for defining acceptable containment system performance.* The 1973 Edition of the IAEA regulations included for the first time activity limits for loss of radioactive contents from Type B packages under normal conditions of transport and under hypothetical accident conditions. The containment system performance requirements were tied to the A_1 and A_2 values, as described above.
3. *The immersion test for Type A fissile material packages.* The 1973 Edition of the IAEA regulations required that the 15-meter (50-ft) water immersion test, previously required as a hypothetical accident test only for Type B packages, also be applied to fissile material packages. This immersion test is important in considering the degree of

internal moderation (i.e., possible inleakage of water) in the criticality safety evaluation for fissile material packages in arrays.

4. *Maximum normal operating pressure (MNOP).* The 1973 Edition of the IAEA regulations added a revised definition of MNOP. The definition for MNOP was included in Part 71 and specifically excluded consideration of package venting and active cooling systems.

5. *Environmental test conditions.* The 1973 Edition of the IAEA Regulations specified for the first time the high and low temperatures, pressures, and weights that should be considered when evaluating the package under normal and accident condition tests.

6. *Quality Assurance (QA) requirements.* The requirements to apply QA to the design, fabrication, and use of transportation packages were proposed in Part 71 in 1973. Although the IAEA regulations did not adopt QA requirements until the 1985 Edition, NRC regulations required QA controls before IAEA adopted these provisions. QA program requirements are only imposed on packages approved for use after 1979. Packages approved under the IAEA 1973 Edition include QA in their design and fabrication, whereas, with a few exceptions (such as spent fuel casks), packages approved under earlier Editions do not include QA.

The NRC draft RA indicates that adopting the grandfathering provisions for packagings approved under Safety Series 6 (1985/1985 Amended) (known as "-85" packagings) and the associated expiration dates; as well as reflecting the "-96" designation, is appropriate from a safety, regulatory, and cost perspective. From a regulatory standpoint, the proposed revisions would result in enhanced regulatory efficiency by bringing NRC's requirements in harmony with those contained in TS-R-1. As described previously, NRC does not currently have sufficient information to quantify the economic impacts of adopting this provision. Should NRC receive comments providing detailed information on the potential economic impacts to industry, the RA would be revised accordingly. The proposed change would also result in implementation costs of approximately $2,000 to the NRC. The NRC would have to revise regulatory guides and NUREG-series documents to indicate which packages are covered by the "grandfathering of older packages" provision. Further, the proposed change could result in implementation and operation costs of approximately $1,000 to Agreement States if they adopt and implement parallel requirements. (The proposed change is not expected to affect implementation or operation costs of DOT.) Agreement States use regulatory guides and NUREG-series documents published by the NRC. Thus, Agreement States would only need to revise documents that they have specifically developed for their licensees (e.g., application materials). In terms of public health and safety, the existing and proposed requirements are believed to be equally protective. Thus, neither an increase nor a decrease in potential health and safety impacts is expected as a result of adopting the proposed administrative changes. Should the NRC become aware that a package or package design is unsafe, the NRC will take action to remove that package or design from service.

NRC Proposed Position. NRC supports this update to grandfathering from TS-R-1 and is proposing to adopt these changes into Part 71 to discontinue authorization to use packages approved under Safety Series 6 (1967). Based on this, NRC is proposing to make modifications to existing § 71.13 to phase out these types of packages. NRC realizes the impact this proposal may have on shipments using existing NRC-approved packages. Therefore, NRC proposes a 3-year transition period for the grandfathering provision on packages approved under Safety Series 6 (1967). This period would provide industry the opportunity to phase out old packages and phase in new ones, or demonstrate that current requirements are met.

For transitional arrangements for newer designs, NRC is proposing to incorporate into § 71.13(c) the provisions for packagings approved under Safety Series 6 (1985) (as amended 1990) (known as "-85" packagings) and the associated expiration dates. Additionally, § 71.13 does not currently contain the provisions for packagings approved under TS-R-1 (known as "-96" packagings). NRC is proposing to add existing § 71.13(e) to provide the "-96" designation.

In summary, the following conditions would apply: (1) Packages approved under Safety Series 6 (1967) may no longer be fabricated, but may be used for a 3-year period after adoption of a final rule; (2) Packages approved under Safety Series 6 1973 (as amended) may no longer be fabricated; however, the proposed rule would not impose any restrictions on the use of these packagings; (3) Packages approved under IAEA Safety Series 6 1985 (as amended 1990), and designated as "-85" in the identification number, may not be fabricated after December 31, 2006, but may continue to be used; (4) Package designs approved under any pre-1996 IAEA standards (i.e., packages with a "-85" or earlier identification number) may be resubmitted to the NRC for review against the current standards. If the package design described in the resubmitted application meets the current standards, the NRC may issue a new CoC for that package design with a "-96" designation.

Affected Sections. 71.13.

Issue 9. Changes to Various Definitions

Background. The changes contemplated by NRC in this proposed rulemaking would require changes to various definitions in § 71.4 to provide internal consistency and compatibility with TS-R-1. The terms must be clearly defined so that they can be used to accurately communicate requirements to licensees. By modifying existing definitions and adding new definitions, the licensee would benefit through more effective understanding of the requirements of Part 71.

Discussion. Eight commenters submitted information on changes to various definitions in the proposed rule. One commenter stated that the definitions should be adopted to the extent the terms are used in the updated regulations. Another commenter urged NRC to be clear, consistent, and precise, particularly regarding the definitions of "rupture," "collapse," "buckling," and "inleakage." Two other commenters stated that the TS-R-1 definition identifies the specific types of packaging allowed for Class 7, and unless DOT revises its regulations, there will be a domestic conflict. Therefore, these commenters do not recommend this change. The commenters added that NRC should consider definitions that explain the differences among "uniformly distributed," "distributed throughout," and "homogeneous."

Another commenter stated that the existing regulation defines special form radioactive material that has been demonstrated to comply with specific tests. The commenter added that TS-R-1, paragraph 225, introduces the term "low dispersible radioactive material," but fails to provide any guidance as to what characteristics qualify the material. Another commenter stated that the definition for "low dispersible radioactive material" should indicate that this does not refer to surface contamination, but rather activation of a solid material. This commenter also suggested adding the term "sealed source'' to mean (for use of A_1 values) encapsulated radioactive material that was designed and manufactured under a specific license and has been assigned a sealed source identification registry number.

One commenter stated that the proposed definitions of "confinement system" and "package" are indistinguishable for packages intended to transport fissile material. The commenter urged NRC to use only one term or to clearly distinguish between the two definitions. The commenter added that if the definition of "confinement system" is added, the term "competent authority" must also be defined, and if the definition of "package" is incorporated, definitions of "excepted" and "industrial" must be added. Another commenter stated that the confinement system definitions should be revised to include fuel assemblies, the PWR basket, and the shipping cask, because all three provide different levels and degrees of confinement.

The NRC draft RA indicates that revising Part 71 to modify existing and add new definitions is appropriate from a safety, regulatory, and cost perspective. The proposed changes would provide greater internal consistency and compatibility with TS-R-1. By modifying existing definitions and adding new definitions, licensees would benefit through a more effective understanding of the requirements of Part 71. The proposed changes would result in implementation costs to the NRC. The NRC would have to revise regulatory guides and NUREG-series documents to include the new or revised definitions of § 71.4. The proposed changes could affect implementation and operation costs of Agreement States because they would have to adopt the revision to the various definitions in § 71.4. (The proposed change is not expected to affect implementation or operation costs of DOT.) Because Agreement States use regulatory guides and NUREG-series documents published by the NRC, they would only need to revise documents that they have developed specifically for their licensees (e.g., application materials).

Additionally, as a means of improving use and understanding of Part 71, the following existing definitions from § 71.4 would be modified: A_1, A_2, and *Low Specific Activity, specifically LSA-III*. The definitions that are structured in § 71.4 are presented in italicized print as a means of distinguishing them from the corresponding text. The definition of *LSA-III* material would be modified to reference the testing provisions for *LSA-III* material found in § 71.77. Other definitions (e. g. *Special form radioactive material*) reference appropriate requirements within Part 71 that must be followed.

Lastly, within the issues paper, NRC posed the idea of adopting the following definitions from TS-R-1: *Confinement System* (TS-R-1, paragraph 209) and *Quality Assurance* (TS-R-1, paragraph 232). NRC is excluding the definition of *Confinement system* because it is included within the broader definition of *Containment system*. Further, NRC's use of *Quality assurance* is somewhat different from that of the IAEA, and NRC will retain the description of *Quality assurance* found in Subpart H.

NRC Proposed Position. The NRC is proposing to adopt the TS-R-1 definition of *Criticality Safety Index (CSI)*. Additionally, the following definitions would be revised to improve their clarity: A_1, A_2, and *LSA-III*. Note: Additional changes to § 71.4 would be made by other Issues.

Affected Sections. 71.4.

Issue 10. Crush Test for Fissile Material Package Design

Background. In TS-R-1, the crush test requirements have been broadened to apply to fissile material package designs (regardless of package activity). Previously, IAEA Safety Series No. 6 and Part 71 have required the crush test for certain Type B packages. This broadened application was created in recognition that the crush environment was a potential accident force that should be protected against for both radiological safety purposes (packages containing more than 1,000 A_2 in normal form) and criticality safety purposes (fissile material package design).

Under requirements for packages containing fissile material, TS-R-1, paragraph 682(b), requires tests specified in paragraphs 719-724 followed by whichever of the following is the more limiting: (1) the drop test onto a bar as specified in paragraph 727(b) and either the crush test as indicated in paragraph 727(c) for packages having a mass not greater than 500 kg

(1,100 lbs) and an overall density not greater than 1,000 kg/m^3 (62.4 lbs/ft^3) based on external dimensions, or the 9-meter (30-ft) drop test as defined in paragraph 727(a) for all other packages; or (2) the water immersion test as specified in paragraph 729.

Both the Safety Series No. 6, paragraph 548, and the current § 71.73 require the crush test for packages having a mass not greater than 500 kg (1,100 lbs), an overall density not greater than 1,000 kg/m^3 (62.4 lbs/ft^3) based on external dimensions, and radioactive contents greater than 1,000 A_2 not as special form radioactive material. Under TS-R-1, the criterion for radioactive contents greater than 1,000 A_2 has been eliminated for packages containing fissile material. The 1,000 A_2 criterion still applies to Type B packages and is also applied to the IAEA newly created Type C package category.

Discussion. Several commenters provided feedback regarding crush test requirements for packages containing fissile material. A number of commenters urged NRC to keep the current regulations requiring the crush test and the free drop test. One commenter stated that the crush test was especially useful for large packages. Another commenter supported the test and stated that U.S. transportation activities should be consistent with IAEA transportation regulations. Similarly, one commenter stated that the testing sequence as required in TS-R-1 should be adopted to assure international uniformity. One commenter recommended removing the optional requirement of either a crush or a drop test, and replacing it with a requirement to conduct both tests.

One commenter requested NRC to improve the realism associated with crush tests. The commenter stated that the crush test should be a physical test rather than using a computer model simulating a test. Additionally, the test should use full-scale packages that are loaded with nonradioactive materials to provide improved test reliability. This commenter stated that crush tests should be included for all package sizes, and the test parameters should be increased to reflect real-world conditions.

A few commenters stated that the proposed requirement to use the free drop test or the crush test is problematic because the results of these tests are different and could require reanalysis of current packages.

One commenter stated that elimination of the 1,000 A_2 activity limit, without providing for flexibility in test sequencing, would be an unfair and costly burden. The commenter stated that Part 71 should be changed to conform to TS-R-1 in all aspects, or not be changed at all. Another commenter stated that the impact of the elimination of the 1,000 A_2 activity limit for fissile material packages having a mass not greater than 500 kg (1,100 lbs), and overall density not greater than 1,000 kg/m^3 (62.4 lbs/ft^3), based on external dimensions, is currently unknown. The commenter noted that shipping companies must use international standards established in TS-R-1 to allow international trade. Another commenter supported the removal of the 1,000 A_2 threshold for fissile packages on the grounds that A_2 levels are intended as an index of radiological hazard rather than criticality potential, and it is inconsistent with TS-R-1.

The NRC believes that full compliance with TS-R-1 requirements for fissile material packages would require changes to the hypothetical accident conditions test sequencing of § 71.73 and would require performance of the 9-meter free drop test or the crush test, but not both, as presently required by § 71.73. The TS-R-1 test requirements are essentially the same as those contained in Safety Series No. 6. In the previous NRC rulemaking for compatibility with Safety Series No. 6 (1985 edition), NRC staff addressed this difference in test requirements. In the June 8, 1988; 53 FR 21550, proposed rule, the NRC stated that: “IAEA applies the crush test in place of the 9-meter drop test for the lightweight packages specified. In the absence of experience using the crush test, and because the crush test and drop test evaluate different features of a package, NRC is requiring both the crush test and the 9-meter drop test for the lightweight packages." Further, in the September 28, 1995; 60 FR 50248, final rule, the NRC stated: "NRC is requiring both the crush test and drop test, for lightweight packages, to ensure that the package response to both crush test and drop forces is within applicable limits."

The NRC draft RA indicates that revising Part 71 to adopt the TS-R-1 requirements for a crush test for fissile material package design, while maintaining the current testing sequence, is appropriate from a safety, regulatory, and cost perspective. Not adopting the requirement would result in an inconsistency between Part 71 requirements and TS-R-1, which could affect international shipments, and fissile material package designs would continue to not be evaluated for criticality safety against this potential accident condition. However, the NRC believes that further information on the impact of the TS-R-1 requirement for fissile material package testing is required. Imposing the crush test requirement on fissile material package designs may impact the industry through costs imposed to demonstrate compliance and may lead to the redesign of packages. Under present Part 71 standards and Safety Series No. 6, the 1,000 A_2 criterion, used to identify packages that must meet the crush test, essentially exempts all packages designed to contain uranium enriched to five percent or less (due to an unlimited A_2 value). For fissile material package designs, this would only apply to designs for plutonium contents. However, if TS-R-1 is adopted, only the weight and density criteria would apply to fissile uranium material packages, and packages that were previously exempted because of the 1,000 A_2 criterion would now require crush testing. The potential impact on the industry is unknown as data on the number of packages shipped under § 71.55, where the 1,000 A_2 value allowed exemption from crush testing, are unknown.

NRC Proposed Position. The NRC proposes to adopt the requirement for a crush test for fissile material packages, and eliminate the 1000 A_2 criterion. However, because there is no new information that addresses concerns from the previous rulemaking regarding the difference in test requirements between Part 71 and Safety Series No. 6, the NRC proposes not to change the testing sequence nor to change the drop and crush test requirements in this revision.

Affected Sections. 71.73.

Issue 11. Fissile Material Package Design for Transport by Aircraft

Background. TS-R-1 introduced new requirements for fissile material package designs that are intended to be transported aboard aircraft. TS-R-1 requires that shipped-by-air fissile material packages with quantities greater than excepted amounts (which would include all NRC-certified fissile packages) be subjected to an additional criticality evaluation. Specifically, TS-R-1, paragraph 680, requires that packages must remain subcritical, assuming reflection by 20 centimeters of water but no water inleakage (i.e., moderation) when subjected to the tests for Type C packages.[1] The specification of no water ingress is given because the objective of this requirement is protection from criticality events resulting from mechanical rearrangement of the geometry of the package (i.e., fast criticality). The provision also states that if a package takes credit for "special features," this package can only be presented for air transport if it is shown that these features remain effective even under the Type C package test conditions followed by a water immersion test. "Special features" generally mean features that could

[1] The TS-R-1 imposition of Type C and LDM requirements (see Issue 6) was in recognition that severe aircraft accidents could result in forces exceeding those of the "accident conditions of transport" that are imposed on Type B and fissile package designs. Because the hypothetical accident conditions for Type B packages are the same as those applied to package designs for fissile material, there was also a need to consider how these more severe test conditions should be applied to fissile package designs transported by air.

prevent water inleakage (and therefore credit could be taken in criticality analyses) under the hypothetical accident conditions. Special features are permitted under current § 71.55(c).

TS-R-1, paragraph 680, requirements for packages to be transported by air, are in addition to the normal condition and accident tests that the package already must meet. Thus:

Type A fissile package by air must:

(A) withstand normal conditions of transport with respect to release, shielding, and maintaining subcriticality (single package and 5xN array[2]);

(B) withstand accident condition tests with respect to maintaining subcriticality (single package and 2xN array); and

(C) comply with TS-R-1, paragraph 680, with respect to maintaining subcriticality (single package);

Type B fissile package by air must:

(A) withstand normal conditions of transport *and* Type B tests with respect to release, shielding, and maintaining subcriticality (single package and 5xN array/normal and 2xN array/accident); and

(B) comply with TS-R-1, paragraph 680, with respect to maintaining subcriticality.

There are no provisions in TS-R-1 for “grandfathering” (Issue 8) fissile material package designs, which will be transported by air. TS-R-1, paragraphs 816 and 817, state that these packages are not allowed to be grandfathered. Consequently, all fissile package designs intended to be transported by aircraft would have to be evaluated before their use.

Discussion.   Five commenters provided information regarding our proposal of the TS-R-1 provisions for fissile material package design for transport by aircraft. One commenter expressed concern about the comprehensibility of the regulations for Type B or below quantities

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[2] N represents the maximum number of fissile material packages that can be shipped on a single conveyance.

of fissile materials. The commenter was aware that the IAEA went through efforts to try to clarify the requirements, but asserted that the regulations need to be understood consistently by the people who approve package designs for transport of fissile materials by air. The commenter stated that this is a critical issue for industry because the International Civil Aviation Organization (ICAO) will adopt TS-R-1 in early 2001 and, therefore, shipments must meet the requirements in TS-R-1 for fissile materials. The commenter encouraged Federal agencies, including NRC and DOT, to push the concept of clarification of the rules and consider a streamlined approval process for designs of air transport of fissile material. Another commenter stated that TS-R-1 writers are working to develop a table that takes into consideration mass, enrichment, and moderation to define an acceptable limit for shipment by air.

One commenter asked when and in what situations the transportation of fissile level material by air would be required.

Two commenters supported the inclusion of these requirements as they are generally in parallel with those in place for surface mode accidents.

The NRC draft RA indicates that adopting TS-R-1 paragraph 680 for criticality evaluation (only applicable to air transport) is reasonable from a safety, regulatory, and cost perspective. Adopting this change would provide the NRC with the regulatory framework for approving package designs that will be used internationally. NRC costs would be reduced while maintaining consistency with international requirements, because the NRC is not adopting the Type C packaging tests for domestic use. Shippers will be required to meet these requirements even if the NRC does not adopt them, because the ICAO is adopting regulations consistent with TS-R-1 effective July 1, 2001. U.S. domestic air carriers require compliance with the ICAO regulations even for domestic shipments.

NRC Proposed Position. The NRC proposes to adopt TS-R-1, paragraph 680, Criticality evaluation, in a new proposed § 71.55(f) that only applies to air transport. Section 71.55 specifies the general package requirements for fissile materials, and the existing paragraphs of § 71.55 are unchanged. Because (1) the NRC is deferring adoption of the Type C packaging tests (see Issue 6), (2) TS-R-1, paragraph 680, references the Type C tests, and (3) paragraph 680 applies to more than Type C packages, only the salient text would be inserted into § 71.55(f), and would apply to domestic shipments.

Affected Sections. 71.55.

B. NRC-Initiated Issues

Issue 12. Special Package Authorizations

Background. The basic concept for radioactive material transportation is that radioactive contents are placed in an authorized container, or packaging, and then shipped. The packaging, together with its contents, is called the package. In general, the transportation regulations in TS-R-1, Part 71, and Title 49 are based on the shipment of radioactive contents in a separate, authorized packaging. There are a few exceptions, however. For example, TS-R-1 provides that the least radioactive of the *Low Specific Activity materials (LSA-I)* and *Surface Contaminated Objects (SCO-I)* may be shipped unpackaged, provided certain conditions are met. Title 49 permits shipment of LSA-I materials in bulk, where the conveyance (e.g., truck or freight container) serves as the packaging.

In other cases involving larger quantities of radioactive material, the content to be shipped may itself be a container. A storage tank containing a radioactive residue is an example. It is not necessary for the shipper to place the tank within an authorized packaging, if the shipper demonstrates that the tank satisfies the requirements for the packaging. DOT and NRC have jointly provided guidance on such shipments (see *"Categorizing and Transporting Low Specific Activity Materials and Surface Contaminated Objects,"* NUREG-1608, RAMREG-003, July 1998).

As older nuclear facilities are decommissioned, DOT and NRC are being asked to approve the shipment of large components, including reactor vessels and steam generators. These components may contain significant quantities of radioactive material, but they are so large that it is not practical to fabricate authorized packagings for them. Because these components were not contemplated when the regulations were developed, the regulations do not specifically address them.

Basically, large components can be shipped under DOT regulations if the components meet the definition of *Surface Contaminated Object (SCO)* or *Low Specific Activity (LSA) material* (see 49 CFR 173.403 for SCO and LSA definitions). For example, steam generators that meet the SCO definition are exempt from Part 71 and are shipped under Title 49, following guidance provided in NRC Generic Letter 96-07 dated December 5, 1996. This method has been applied to several shipments of steam generators and small reactor vessels to the low level waste disposal facility at Barnwell, SC. NRC and DOT intend to continue employing this approach and method for steam generators and similar components that can be shipped under DOT regulations.

Large components that exceed the SCO and LSA definitions are subject to Part 71. An example is the Trojan reactor vessel. By letter dated March 31, 1997, Portland General Electric Company (PGE) requested approval of the Trojan Reactor Vessel Package (TRVP) (including internals) for transport to the disposal facility operated by U.S. Ecology on the Hanford Nuclear Reservation near Richland, Washington. The TRVP contained approximately 74 PBq (2 million Ci) in the form of activated metal and 5.7 TBq (155 Ci) in the form of internal surface contamination, was filled with low-density concrete and weighed approximately 900 metric tons (1,000 tons). Normally, large curie contents are required to be shipped in a Type B packaging, but the TRVP was too large and massive to be shipped within another packaging.

PGE acknowledged that the TRVP could not meet Type B regulations and applied for a Type B package CoC for the TRVP itself, either under § 71.41(c), "Demonstration of compliance," or § 71.8, "Specific exemptions." Section 71.41(c) provides that "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." Section 71.41(c) has been used to accommodate minor deviations in test environments (e.g., initial temperatures), and was not intended to be used to establish new test conditions for Type B packages. The use of this provision in the Trojan case would essentially have resulted in establishing new (and less rigorous) Type B test conditions that the Trojan vessel could meet. A CoC for a Type B package could then have been issued for Trojan, but the level of performance reflected in that Certificate would have been significantly different from that in other Type B Certificates. NRC decided against using § 71.41(c), and to use the § 71.8 exemption provision - the only other option available.

Section 71.8 provides that NRC may grant any exemption from the requirements of the regulations in Part 71 that it determines is authorized by law and will not endanger life or property nor the common defense and security. The exemption approach had three impacts on the TRVP review. First, the NRC's categorical exclusion from preparing an Environmental Assessment (EA) pursuant to the National Environmental Protection Act (NEPA) for package approvals (§ 51.22(c)(13)) does not apply to packages authorized under an exemption. Consequently, an EA of the proposed exemptions was required. Second, DOT's regulations that govern radioactive material shipments do not recognize packages approved via NRC exemption. PGE was therefore required to obtain an exemption from DOT regulations in 49 CFR Part 173 for the TRVP shipment. Third, use of the exemption option provided a mechanism for NRC to consider the operational and administrative controls, which were proposed by PGE to influence shipment risk factors. Considering the statements and representations contained in the application, as supplemented, and the conditions specified in the package approval, NRC concluded that the TRVP, as exempted, met the requirements of Part 71, and recommended that the Commission approve the exemptions and the TRVP shipment.

Currently, no regulatory provisions exist in Part 71 for dealing with nonstandard packages, other than the exemption provisions and § 71.41(c). The NRC's policy is to avoid the use of exemptions for recurring licensing actions. Therefore, as a lesson learned from the Trojan approval, the NRC staff identified large component package authorizations as an issue for consideration in this proposed rule.

Discussion. Numerous comments were received on the special package approvals issue in response to the issues paper from the public meetings and from NRC's website. One of the commenters supported the idea of creating a system for providing special package approvals without using the existing exemption requirements. This commenter noted that his agency found it very useful to realize that there are packages or materials outside the current scope of NRC regulations that still need to be transported as they cannot stay where they are. The commenter agreed that it is appropriate to have a method to address these issues.

A number of commenters did not support the development of a special package approvals regulation. These commenters believed the issue of special package approvals should be conducted on a case-by-case basis, using the current exemption process. One commenter noted that "hot decommissioning" and "hot" shipping introduce a new regimen, and therefore, the commenter believed that the only way for the NRC to proceed is with a case-by-case, very individual and specialized exemption or allowance, if at all. The commenter went on to say that the people who are on the first lines, the first responders and the emergency management coordinators at the local level, and the people who are in transport corridor communities have a right to information that a specialized process (i.e., an exemption process) would provide. The commenter stated that the concerns of the public who are in these transport corridor communities are not being given adequate weight in decisionmaking, and the opportunities for discussion are too limited. Finally, this commenter stated that removing the exemption process for big, unusual shipments could set the stage for applying this concept to other types of materials to be exempted from testing and packaging requirements which the commenter believed would be a bad precedent.

Two commenters expressed concern over the definition of a "special large object." One commenter stated that if special provisions are added, then the term "large" must be defined with respect to both size and weight. Another commenter requested that NRC consider revisions to Part 71 to address large objects in general, that would include reactor vessels.

Three commenters spoke to the issue of Type B quantities. The first commenter stated that there could be overlap between orphan sources and Type B quantities. This commenter recommended that Type B orphan sources be included in a separate rule from the special large packages. The second commenter would like to see collaboration between the NRC and DOT to address the possibility of initiating a program that would minimize package review costs of decommissioning Type B quantities of cobalt-60 and cesium-137. Two commenters stated that there have been cases where a Type B package has been damaged in a way that it will continue to secure and shield the sources, but does not meet compliance standards. The commenters noted that in these types of cases, a special arrangement certificate would be beneficial to allow transport of the damaged equipment for disposal.

Several commenters did not believe that NRC's use of the shipment of the Trojan reactor vessel was an adequate basis for determining whether or not to remove the requirement for exemptions for special packages and replace it with other provisions. One commenter noted that because the Trojan vessel was shipped by barge, a lot of the risk of exposure that would normally be present in other transport modes was removed (e.g., a truck being caught in traffic). This commenter also stated that moving to a risk-informed decisionmaking process for special package approvals may result in a situation where the public is "informed to more risk while the industry is exposed to less regulation." Another commenter noted that if NRC is using the shipment of the Trojan reactor vessel as its baseline for determining whether to revise its regulations, care should be taken to limit the scope of this special approval to NRC's responsibilities and expertise. The commenter noted that as the Trojan approval process moved along, there was a difference of opinion as to the extent of NRC's evaluation of river and barging conditions, when in reality, these issues are the jurisdiction of the Coast Guard, and if the Coast Guard had approved the waterway and the conveyance, it should not be necessary for this information to be a part of an application to NRC subject to NRC review and approval. Other commenters disagreed. One commenter added that significant experience has already been gained in exempting the Trojan reactor vessel, a precedent has been established, and the possibility exists that the requirements placed on the shipment of the Trojan reactor vessel might have been more restrictive than might have been determined as necessary. Two commenters stated that the Trojan shipment review is a point of reference for the basis of other similar shipments, but that each case should still be assessed on its own merits.

A number of commenters raised specific issues that NRC should consider when deciding whether to propose a special package approval process and how that process should be defined. Two commenters noted that the system has been defined as to how these materials should be moved and what kind of information needs to be provided to the regulators to move the materials. These commenters further noted that any change to Part 71, with respect to these special shipments, needs to be specific to those items that are going to be regulated under the MOU between the NRC and DOT. The two commenters added that the majority of those items that get moved are large components and would fall under the DOT's jurisdiction under the MOU. Thus, DOT would regulate items like steam generators and demineralizers and pressurizers, all of which are pieces and parts of reactors that are being decommissioned. NRC would regulate items like reactor pressure vessels (e.g., the Trojan reactor pressure vessel).

One commenter did not support the adoption of an analog of the IAEA special arrangements provisions in Part 71. The commenter did not support the adoption of this type of provision in Part 71 because the IAEA special arrangements were specifically designed for movement internationally, whereas most of these items would be moved domestically.

One commenter provided input on the specific issue of what additional determinations should be included in an application for a special package approval. The commenter noted that a precedent has already been established with the requirement that a transportation plan be provided with the exemption requests. The transportation plan contains safety features that would be substituted for the current codified requirements that would provide an equivalent order of safety, considerations of the entire safety system versus independent components of safety, emergency response plans, and risk-informed considerations.

The NRC processing of one-time exemptions for nonstandard packages, such as the Trojan vessel, represents expenditure of considerable staff resources. Once the application for exemption is received, the staff spends a significant amount of time reviewing the application and preparing an EA. The Commission itself has been involved in the approval of these actions. Rather than exempting nonstandard packages from regulations, as was necessary for

Trojan, the staff is proposing that regulatory requirements be added to Part 71 which would address nonstandard packages. These special packages are likely to increase in number as a result of future decommissioning activities.

The NRC is proposing a regulatory mechanism to address large component shipments. In this regard, NRC has considered TS-R-1, paragraph 312, entitled: SPECIAL ARRANGEMENT:

> Consignments for which conformity with the other provisions of these regulations is impracticable shall not be transported except under special arrangement. Provided the competent authority is satisfied that conformity with the other provisions of the regulations is impracticable and that the requisite standards of safety established by these regulations have been demonstrated through means alternative to the other provisions, the competent authority may approve special arrangement transport operations for single or a planned series of multiple consignments. The overall level of safety in transport shall at least be equivalent to that which would be provided if all the applicable requirements had been met. For international consignments of this type, multilateral approval shall be required.

The Special Arrangement paragraph is intended to provide competent authorities (DOT in the U.S.) the authority to approve shipments that don't completely conform to the transportation safety standards, provided the overall level of safety established by the regulations is maintained. DOT consults with NRC regarding the approvals for shipment of packages containing larger quantities of radioactive material and/or fissile materials. NRC is proposing to add this provision to § 71.41.

The NRC draft RA indicates that adopting the special package authorization requirements proposed for incorporation into Part 71 is appropriate from a safety, regulatory, and cost perspective. The proposed action would result in enhanced regulatory efficiency by standardizing the requirements to provide greater regulatory certainty and clarity, and would ensure consistent treatment among licensees requesting authorization for shipment of special packages. This increase in regulatory efficiency, however, would depend in part on modifications to DOT's regulations to recognize NRC special package exemptions. Further, NRC experience in handling the one-time exemption(s) during the transition period would be used in crafting the new requirements. As a result, applications for one-time exemptions would be eliminated, resulting in savings in licensee staff resources and NRC staff resources. Because the new section is expected to be better streamlined for handling these nonstandard packages, considerable savings would be realized, both in NRC and licensee staff time. These expected NRC savings are estimated to be approximately $500,000. Special package shipments are likely to increase regardless of the outcome of this rulemaking, as a result of future decommissioning activities. The justification for authorizing special packages for shipment is a decreased risk of radiation exposure to the public and workers as opposed to the shipment alternatives. NRC believes that standardizing the method for reviewing these packages would provide adequate review without imposing unnecessary administrative burdens on NRC staff associated with the processing of exemption-based reviews.

NRC Proposed Position. Based on the above considerations and the public comments, NRC proposes a special package authorization that would apply only in limited circumstances, and only to one-time shipments of large components. Further, any such special package authorization would be issued on a case-by-case basis, and would require the applicant to demonstrate that the proposed shipment would not endanger life or property nor the common defense and security, following the basic process used by applicants to obtain nonspecial package authorizations from NRC.

NRC proposes to adopt a provision that is an analogous to TS-R-1, paragraph 312, for Part 71 with respect to the approval of large component packages. The applicant would need to provide reasonable assurance that the special package, considering operational procedures and administrative controls employed during the shipment, would not encounter conditions beyond those for which it had been analyzed and demonstrated to provide protection. NRC would review applications for large component special package authorizations. Approval would be based on a staff determination that the applicant met the requirements of Subpart D. If approved, the NRC would issue a CoC or other approval (i.e., special package authorization letter).

NRC would consult with DOT on making the determinations required to issue an NRC special package authorization. The efficiency of the NRC special package process in part depends on a modification by DOT of its regulations to recognize NRC special package authorizations, so that a DOT exemption would not be required for use of the NRC authorization. DOT is proposing this change in its companion TS-R-1 compatibility rulemaking.

Affected Sections. 71.41.

Issue 13. Expansion of Part 71 Quality Assurance Requirements to Certificate of Compliance (CoC) Holders

Background. The Commission recently issued a final rule to expand the QA provisions of Part 72, Subpart G, to specifically include certificate holders and applicants for a CoC (see 64 FR 56114; October 15, 1999). In development of the proposed rule for Part 72, the NRC staff submitted a rulemaking plan to the Commission in SECY-97-214.[3] In a Staff

[3] SECY-97-214, "Changes to 10 CFR Part 72, Expand Applicability to Include Certificate Holders and Applicants and Their Contractors and Subcontractors," dated September 24, 1997.

Requirements Memorandum (SRM) to SECY-97-214, the Commission approved the staff's rulemaking plan and directed the staff to also consider whether conforming changes to the QA regulations in Part 71 would be necessary because of the existence of dual-purpose cask designs. In a memorandum from the Executive Director for Operations to the Commission, dated December 3, 1997, the NRC staff indicated that expansion of the Part 71 QA provisions to include certificate holders and applicants for a CoC would be made as part of the rulemaking to conform Part 71 to IAEA Standard TS-R-1. Furthermore, in the final rule expanding QA regulations in Part 72, Subpart G, the Commission did not include contractors or subcontractors (e.g., fabricators) within the scope of the revised Part 72, Subpart G. The Commission took this action in response to comments on the associated proposed rule. In the response to Comments 3 and 9 in the final Part 72 rule, the Commission indicated that Part 72 licensees, certificate holders, and applicants for a CoC are responsible for assuring that their contractors and subcontractors (e.g. fabricators) are implementing adequate QA programs. Similarly, Part 71 licensees, certificate holders, and applicants for a CoC are responsible under § 71.115 for assuring that their contractors and subcontractors (e.g. fabricators) are implementing adequate QA programs.

Under Part 71, the NRC reviews and approves applications for Type B and fissile material packages for the transport of radioactive material. The NRC's approval of a package is documented in a CoC. Applicants for a CoC are currently required by § 71.37 to describe their QA program for the design, fabrication, assembly, testing, maintenance, repair, modification, and use of the proposed package. Further, existing § 71.101(a) describes QA requirements that apply to design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of components of packagings that are important to safety. Type B packages are intended to transport radioactive material that contains quantities of radionuclides greater than the A_1 or A_2 limits for each radionuclide (see Appendix A to Part 71 for examples of A_1 or A_2 limits). Fissile material packages are intended to transport fissile material in quantities greater than the Part 71, Subpart C, general license limits for fissile material (e.g., existing §§ 71.18, 71.20, 71.22, and 71.24).

Although CoCs are legally binding documents, certificate holders or applicants for a CoC and their contractors and subcontractors have not clearly been brought into the scope of Part 71 requirements. This is because the terms "certificate holder" and "applicant for a certificate of compliance" do not appear in Part 71, Subpart H; rather, Subpart H only mentions "licensee" in these regulations. Consequently, the NRC has not had a clear basis to cite certificate holders and applicants for a CoC for violations of Part 71 requirements in the same way it has licensees.

The NRC Enforcement Policy [4] and its implementing program was established to support the NRC's overall safety mission in protecting public health and safety and the environment. Consistent with this purpose, enforcement actions are used as a deterrent to emphasize the importance of compliance with requirements and to encourage prompt identification and comprehensive correction of the violations. Enforcement sanctions consist of Notices of Violation (NOVs), civil penalties, and orders of various types. In addition to formal enforcement actions, the NRC also uses related administrative actions such as Notices of Nonconformance (NONs), Confirmatory Action Letters, and Demands for Information to supplement its enforcement program. The NRC expects licensees, certificate holders, and applicants for a CoC to adhere to any obligations and commitments that result from these actions and would not hesitate to issue appropriate orders to ensure that these obligations and commitments are met. The nature and extent of the enforcement action are intended to reflect

[4] NUREG-1600, “General Statement of Policy and Procedures for NRC Enforcement Actions,” dated May 2000.

the seriousness of the violation involved. An NOV is a written notice setting forth one or more violations of a legally binding requirement.

However, when the NRC has identified a failure to comply with Part 71 QA requirements by certificate holders or applicants for a CoC, it has issued an NON rather than an NOV. Although an NON and an NOV appear to be similar, the Commission prefers the issuance of an NOV because: (1) the issuance of an NOV effectively conveys to both the person violating the requirement and the public that a violation of a legally binding requirement has occurred; (2) the use of graduated severity levels associated with an NOV allows the NRC to effectively convey to both the person violating the requirement and the public a clearer perspective on the safety and regulatory significance of the violation; and (3) violation of a regulation reflects the NRC's conclusion that potential risk to public health and safety could exist. Therefore, the NRC believes that limiting the available enforcement sanctions to administrative actions is insufficient to address the performance problems observed in industry.

Discussion. Sixteen commenters provided comments regarding the possible expansion of QA requirements to holders of, and applicants for, a CoC. Of these, three supported expanding the QA requirements. Two commenters stated that the cask design and fabricating industry should be allowed flexibility to make design changes to the casks that would not impact safety. One of the commenters stated that cask designers and fabricators should be held responsible as are parties on the nuclear power reactor side.

Four commenters did not support the overall proposed change to expand the QA requirements of Part 71. One commenter stated that it is the responsibility of the purchaser, user, or licensee of the cask or shipping container to ensure the container's QA, and therefore, NRC already has enforcement authority over that particular container. Two commenters stated that extending the responsibility to the fabricator or certificate holder would encourage fabricators to get out of business because of the regulatory and paper burden of the proposed provision. Another commenter stated that there is confusion between what is in the current regulations and what is in the proposed regulations. Another commenter stated that NRC could be regulating packages for which NRC is not responsible under the MOU between the NRC and the DOT. A commenter stated that NRC currently has adequate QA control on the Part 71 packages under Subpart H. The commenters did not believe that issuing an NOV instead of an NON would result in additional compliance.

Several commenters noted the need for consistency in the QA provisions between Parts 71 and 72, which should be maintained for dual purpose casks used for storage and transportation of spent nuclear fuel and high-level radioactive waste. Additionally, one commenter noted that a distinction has never been established between Part 71 and Part 72 packages used to transport/store spent fuel and the Part 71 packages used to transport sealed radioactive sources. The commenter suggested that “Part 50 reactor licensees be specifically exempted from participation in nuclear power specific QA activities.”

Representatives of DOT and the U.S. Department of Energy (DOE) questioned whether this provision would apply to Type A packages. The NRC intends that this proposed change would apply only to NRC certificate holders and applicants for a CoC and only for package designs that are regulated by NRC (e.g., Type B or fissile packages).

The principal changes to Subpart H would involve adding the terms "certificate holder" and "applicant for a CoC" to indicate that these persons are also covered by the section, although in some cases, only "certificate holder" would be added, because an applicant for a CoC would not be expected to accomplish these specific activities. Additional conforming changes would be made to various sections in Part 71 to ensure greater consistency between Part 71 and Part 72.

The NRC draft RA indicates that expanding the QA provisions of Part 71, Subpart H, to certificate holders and applicants for a CoC is appropriate from a safety, regulatory, and cost perspective. First, adopting these requirements would ensure that the regulatory scheme of Part 71 would remain more consistent with other NRC regulations in that certificate holders and applicants for a CoC would be responsible for the behavior of their contractors and subcontractors. Also, because this action would be limited to certificate holders and applicants for a CoC, it may not be as likely to be challenged as an expansion of NRC authority. Inclusion of certificate holders and applicants for a CoC would make it possible for NRC to issue NOVs and orders, if appropriate, for violation to the regulatory requirements; this would allow the NRC to conduct its business of protecting public health and safety more efficiently and effectively. This proposed rule would not authorize the NRC to issue civil penalties to Part 71 certificate holders or applicants for a CoC, who are found to be in violation of regulatory requirements. Alternatively, contractors and subcontractors of licensees, certificate holders, and applicants do have responsibility for safety, and omitting them from Part 71 would continue the present difficulty that NRC has encountered in reaching these persons with its enforcement tools. Certificate holders and applicants for a CoC would incur costs associated with understanding and implementing the new regulations, as well as in preparing and submitting reports similar to those described in SECY 99-174. SECY 99-174 states that “Additional requirements for recordkeeping and reporting for certificate holders are needed to include records required to be kept as a condition of the CoC. This will provide an enforcement basis equivalent to the recordkeeping and reporting regulations for licensees.” These costs are estimated to be approximately $400,000 per year for the certificate holders and applicants for a CoC. NRC would incur costs associated with monitoring certificate holders and applicants for a CoC and maintaining and reviewing the records for certificate holder submittals. These costs are estimated to be approximately $80,000 per year. By specifically listing certificate holders and applicants for a CoC in Part 71, inspection deficiencies noted by NRC might result in an NOV. This authority would allow NRC to issue orders or take other enforcement actions (except civil penalties) necessary to ensure that certificate holders and applicants for a CoC comply with Part 71 requirements, similar to NRC enforcement actions in other program areas. However, this benefit is difficult to quantify and is estimated to be small.

The NRC is proposing to expand the QA provisions of Part 71, Subpart H, to specifically include certificate holders and applicants for a CoC. This expansion is necessary to enhance NRC's ability to enforce nonconformance by the certificate holders and applicants for a CoC. The NRC is also proposing to add a new section (§ 71.9) on employee protection to Part 71. Currently, regulations on employee protection are contained in the individual parts under which the NRC issued a specific license. Consequently, this regulation was not deemed necessary for a Part 71 general licensee. However, the equivalent requirement for certificate holders or applicants for a CoC does not exist. The NRC believes that employee protection regulations should be added for the employees of certificate holders and applicants for a CoC to provide greater regulatory equivalency between Part 71 licensees and certificate holders. Therefore, the NRC would add a requirement on employee protection to Part 71.

NRC Proposed Position. The NRC is proposing to expand the QA provisions of Part 71, Subpart H, to specifically include certificate holders and applicants for a CoC.

In addition to the changes to Subpart H, conforming changes would also be made to: § 71.0, "Purpose and scope"; § 71.1, "Communications and records" ; § 71.6, "Information collection requirements: OMB approval"; § 71.7, "Completeness and accuracy of information"; § 71.91, "Records"; § 71.93, "Inspection and tests", and § 71.100, " Criminal penalties." Additionally, § 71.11 would be redesignated as § 71.8; and a new § 71.9, "Employee protection," would be added.

Affected Sections. §§ 71.0, 71.1, 71.6, 71.7, 71.8 , 71.9, 71.91, 71.93, 71.100, and 71.101 through 71.137.

Issue 14. Adoption of ASME Code

Background. NRC considered the adoption of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code (B&PV) Code, Section III, Division 3, for two reasons. First, previous NRC inspections at vendor and fabricator shops (for fabrication of spent fuel storage canisters and transportation casks) identified quality control (QC) and QA problems. Some of these problems would have been prevented with improved QA programs, and may have been prevented had fabrication occurred under more prescriptive requirements such as the ASME Code requirements. Second, Public Law 104-113, "National Technology Transfer and Advancement Act," enacted in 1996, requires that Federal agencies use, as appropriate, consensus standards (e.g., the ASME B&PV (Boiler and Pressure Vessel) Code), except when there are justified reasons for not doing so.

Currently, no ASME Code requirements exist in Part 71 for fabrication/construction of spent fuel transportation packages.

Discussion. NRC received numerous comments regarding the adoption of the ASME Code. Four commenters stated they favored adoption of the ASME Code. One commenter favored using ASME codes for all components used in the containment boundary of all products that are used in transportation and storage of radioactive materials. This commenter also supported an explanatory guideline in the ASME Code that speaks to the subject of categorization of materials, whereby all manufacturers are using the same criteria. Another commenter stated that using ASME standards would improve current problems with casks and the current lack of QA. One commenter stated that some benefits of a third party authorized nuclear inspector (ANI) would accrue to industry. These benefits are that common standards would decrease complexity and interpretation, lower cost, and increase safety.

Eight commenters stated concerns or disapproval of the adoption of the ASME Code. One commenter was concerned with the adoption of the guidelines before a full review of the affects on transportation. Another commenter stated concern over adopting voluntary standards into regulations. Specifically, this concern was directed at the inconsistency between industry standards and regulations. Similarly, another commenter noted that changes within ASME might occur quickly, and it would be difficult to follow these changes. One commenter recommended that incorporation of the ASME Code by reference is the appropriate regulatory mechanism, following the precedent set by § 50.55(a) for the ASME Code, Section III, Division 1. Several commenters recommended that NRC place industry standards in regulatory guides, which would allow for simpler updating, recognize that other methods of demonstrating compliance are available, and satisfy the Congressional mandate to consider the use of consensus standards. One commenter stated a concern about the enforceability of the standard if it is not placed in the regulations. Conversely, another commenter noted that the regulatory burden is significantly increased when voluntary standards are changed to regulations, and compliance may not always practical or be accomplished.

Other commenters were concerned about the widespread impact of the adoption. One commenter stated that there is no technical justification for adoption of the ASME Code, and it would have significant adverse impact on the ability of the U.S. Navy to refuel and defuel the U.S. nuclear powered warships. Another commenter stated that overseas market impacts need to be considered in the rulemaking. Another commenter stated that when an applicant commits to certain standards in his or her safety requirements during the license approval process, it becomes a license condition, and NRC can enforce it.

One commenter stated that if the ASME Code is adopted, the development of it and the information involved must be publicly available. Two commenters specifically asked if the proposed change applies to all packages, dual-purpose spent fuel packages, or to all CoC holders. Another commenter questioned how, or whether, the requirement will change if the industry standard changes in the future.

During the early period of spent fuel storage and transportation cask fabrication, NRC inspection staff consistently identified QC and QA problems at the vendor/fabricator facilities. At that time, NRC believed that these problems might have been prevented had fabrication occurred under ASME Code requirements. Therefore, there was an impetus to place consideration of the ASME Code requirements in the Part 71 rulemaking. However, since then, due to increased attention by the NRC and industry, the overall frequency and significance of QA and QC problems at fabricators and vendors have decreased.

With respect to conformance to Public Law 104-113, the ASME issued a consensus standard in May 1997, entitled: "Containment Systems and Transport Packages for Spent Fuel and High Level Radioactive Waste," ASME B&PV Code, Section III, Division 3. The ASME Code requires the presence of an ANI during construction to ensure that the Code requirements are met, and the stamping of components (i.e., the transportation cask's containment) constructed to the Code. NRC staff participated, and continues to participate, in the ASME subcommittee that developed the Code requirements. It is the NRC staff's understanding, through participation in the subcommittee, that the ASME Code document is undergoing extensive review and modification and that a major revision will be issued. Therefore, NRC staff believes that inclusion of the ASME Code in Part 71 is not appropriate at this time.

Public Law 104-113 requires that Federal agencies use consensus standards in lieu of government-unique standards, if this use is not impractical or inconsistent with other existing laws. Because a major revision to the ASME Code is forthcoming and because the changes in that revision are not yet available for staff and stakeholder review, the NRC staff considers it an imprudent use of NRC and stakeholder resources to initiate rulemaking on the current Code revision only to have the Code requirements change during the Part 71 rulemaking. After the ASME Code revision is issued, the NRC staff can then consider its incorporation through the rulemaking process, or consider adopting and accepting the ASME Code as an acceptable method for complying with NRC requirements through endorsement in regulatory guidance.

The NRC draft RA indicates that not adopting the ASME Code requirements in Part 71 is appropriate from a safety, regulatory, and cost perspective. While NRC resources would be conserved by not adopting the ASME Code, the proposed action would retain the current status.   However, the proposed action would result in no benefits or negative impacts on industry.

After consideration of the public comments and the NRC recently learning of the extensive review and revision of the ASME Code, the staff recommends not to incorporate the ASME Code, Section III, Division 3, requirements into Part 71. However, adoption of the ASME Code into Part 71 will be considered by the NRC staff in a future rulemaking or guidance document.

NRC Proposed Position.   The NRC staff recommends not incorporating the ASME Code, Section III, Division 3 requirements into Part 71.

Affected Sections.   None (not adopted).

Issue 15. Change Authority for Dual-purpose Package Certificate Holders

Background: The Commission recently approved a final rule to expand the provisions of § 72.48, "Changes, Tests, and Experiments," to include Part 72 certificate holders

(64 FR 53582; October 4, 1999). Part 72 certificate holders are allowed under the amended § 72.48 to make certain changes to a spent fuel storage cask's design or procedures used with the storage cask and to conduct tests and experiments, without prior NRC review and approval. Part 71 does not contain any similar provisions to permit a certificate holder to change the design of a Part 71 transportation package, without prior NRC review and approval. The NRC has issued separate Certificate of Compliance (CoCs) under Parts 71 and 72 for dual-purpose spent fuel casks and transportation packages (i.e., a container intended for both the storage and transportation of spent fuel). This has created the situation where an entity holding both a Part 71 and Part 72 CoC would be allowed under Part 72 to make certain changes to the design of a dual-purpose cask, e.g., changes that affected a component or design feature that has a storage function, without obtaining prior NRC approval. However, the same entity would not be allowed under Part 71 to make changes to the design of this same dual-purpose cask (package), e.g., changes that affect the same component or design feature, if that component or feature also has a transportation function, without obtaining prior NRC approval, even when the same physical component and change is involved (i.e., the change involves a component that has both storage and transportation functions).

In SECY-99-130[5] and SECY-99-054,[6] NRC indicated that comments had been received on the § 72.48 proposed rule (63 FR 56098; October 21, 1998) that requested similar authority be created in Part 71, particularly with respect to dual-purpose casks. In SECY-99-054, NRC staff recommended that an authority similar to § 72.48 be created for spent fuel transportation packages intended for domestic use only. NRC staff also recommended that this authority be

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

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

limited to Parts 50 and 72 licensees shipping spent fuel and the Part 71 certificate holder. NRC indicated that providing change authority under Part 71 would be addressed in the current rulemaking. The Commission directed the staff to implement recommendations contained in SECY-99-130 and SECY-99-054, in an SRM dated June 22, 1999.

NRC staff also identified that other supporting changes to Part 71 would be required to ensure consistency with the process contained in § 72.48. These changes would include using common terminology such as “changes to the cask design, as described in the final safety analysis report” (FSAR), and a process for requesting amendments to a CoC. Additionally, requirements for (1) certificate holders to periodically update the FSAR for a transportation package would be required to ensure an accurate “licensing” basis is available when future proposed changes are evaluated, and (2) licensees to possess a copy of the FSAR as well as the CoC before making a shipment.

NRC believes that the current IAEA standard TS-R-1 does not contain any equivalent provisions for changing a transportation package's design, without prior review by the agency that certified the design. NRC is the reviewing agency for Type B and fissile material package approvals. Therefore, any application of "change authority" to Part 71 CoCs would only apply to packages intended for the domestic transport of spent fuel.

Discussion. The NRC has received 48 public comments on this issue in response to the issue paper, public meetings, and the website. Industry representatives and certain members of the public support the issue. Public interest organizations, State representatives, and other members of the public generally oppose the issue. The DOE also opposes this issue. Groups in favor of this issue pointed to similar provisions in Parts 50 and 72 where such changes have been safely made. Groups opposed to this issue believe that all changes to a transport package's design should be submitted to the NRC for prior review and approval.

These commenters believed this is necessary because transportation packages are on the public roadways and railways, hence the public believes there is more immediate and greater exposure to the radioactive contents of the package in an accident. The following is a more detailed description of these comments.

Seven commenters supported the effort to expand the provisions contained in § 72.48 to include Part 71 certificate holders. Two commenters also requested that NRC expand the authority for all packages, not just dual-purpose spent nuclear fuel packages.

Three commenters requested that NRC be consistent and revoke the change, test, and experiment authority for Part 72 certificate holders. One commenter opposed allowing the ability to make any changes to casks without prior NRC approval. Similarly, one commenter sought assurance that NRC would continue to be able to monitor industry performance (i.e., maintain regulatory oversight capability), and be able to undo or revise changes or force amendments when necessary.

One commenter, opposed to the expansion of authority, referenced a Government Accounting Office (GAO) report that highlighted problems with transportation casks fabricated by Westinghouse, claiming that 20 out of 40 casks had been found to be defective. Another commenter was opposed to any action, such as moving to performance- or risk-based management, that would increase the level and type of public risk.

Another commenter stated that he does not support allowing change authority because the definition of "minimal" has historically been ill-defined. This commenter also expressed his belief that Issue 15 (change authorization issue), as currently proposed, would not result in Part 71 conforming with TS-R-1. The commenter cited as evidence the text in the issues paper that states, "the current IAEA standard ST-1 does not contain any equivalent provisions for changing a transportation package's design, without prior review by the competent authority."

Most commenters expressed interest in receiving additional information from NRC about what changes might be allowable, and clarification that these allowable changes would only be for activities not important to safety (e.g., switching to nonreactive paints). One commenter also suggested that NRC and DOT be careful in determining allowable, nonsafety changes because with the effort to lengthen the certificate revalidation cycle, it is conceivable that these changes would just be rolled into the new certification without review. This commenter also questioned how NRC plans to address the issue of conformity with other nations' package requirements and certificates.

NRC staff believes that the capability to make minor changes to a transportation package is similar to the capability to make minor changes to a reactor facility, to a spent fuel storage facility, or a spent fuel storage cask design. The Commission has recently issued a final rule which authorized Part 72 certificate holders to make minor changes to a spent fuel storage cask's design. Therefore, NRC believes that extending this authority to Part 71 packages is consistent with previous Commission actions.

The current regulatory structure of Part 71 requires all design changes to a transportation package, which would change the CoC or included drawings, be submitted to the NRC for prior review and approval. However, a package user (i.e., a Part 71 general licensee) is not currently required to obtain a copy of the safety analysis report (SAR) and understand it before shipping radioactive material. Rather, the licensee is only required to obtain a copy of the CoC and any referenced documents, determine that the package is properly configured for shipment (i.e., meets the requirements of §§ 71.85 and 71.87), determine that the intended radioactive contents are within the conditions of the CoC, implement any procedure required by the CoC, and accomplish these activities under an NRC-approved QA program (in accordance with Part 71, Subpart H). Consequently, a licensee is not required to understand the technical bases of the Part 71 regulations on normal conditions of transport, hypothetical accident conditions, and criticality control (i.e., §§ 71.71, 71.73, and 71.55, respectively), before the licensee can use the package to transport radioactive material. Therefore, NRC staff believes that a significant increase in burden would be imposed on licensees to understand these technical bases, if they were permitted to make changes under a "change authority" regulation.

NRC staff also notes that Part 71 does not contain some of the regulatory foundations which support the recent revision to § 72.48. For example, under § 72.48, a licensee is required to evaluate proposed changes to the cask design against the FSAR (as updated), and to periodically incorporate these changes into the FSAR to ensure that an accurate licensing basis is maintained for use in evaluating future proposed changes. Additionally, a Part 71 licensee need not own the package it is using to transport radioactive material. Instead the licensee is considered a "registered user" of the package. This second circumstance, when coupled with a Part 71 change authority, might create a situation in which one licensee could make an authorized change to a package, without prior NRC approval, transfer that package to another registered user, without forwarding all change summaries to the next user, who would then be unable to verify or recognize that the package is in conformance with the CoC (i.e., acceptable for use under the requirements of Subpart G (e.g., § 71.87)).

The design drawings for a transportation package are directly incorporated by reference into the Part 71 CoC, whereas the design drawings for a spent fuel storage cask are contained in the FSAR. While changes to a design (as described in the FSAR) are permitted, changes to the CoC (or any drawings incorporated into the CoC by reference) would not be permitted. As a consequence, these referenced drawings limit the population of potential changes that a licensee or certificate holder could make under a Part 71 authority equivalent to § 72.48.

Based upon review of the potential impacts, NRC believes that adding the necessary regulatory requirements (i.e., foundations) to Part 71 to support a change authority equivalent to § 72.48, would unnecessarily increase the burden on all licensees without providing a corresponding benefit. Providing this change authority would also increase the complexity of the Part 71 regulations.

The NRC believes the issue of inconsistent change authority between Parts 71 and 72 for a dual-purpose spent fuel package should be resolved. Performance of Parts 50 and 72 licensees and the Part 76 certificate holder in implementing the change processes of Parts 50, 72, and 76, has demonstrated that these types of changes can be made safely, without prior NRC approval. However, NRC staff also believes that the scope of this authority should be limited to dual-purpose packages, rather than all NRC-certified spent fuel packages, and limited to only the certificate holders.

Accordingly, the NRC staff considers the best approach in resolving these conflicts is through the use of a parallel regulatory structure in Part 71. While the NRC staff would retain the current process for existing transportation packages, a new process for approving dual-purpose transportation packages would be added to Part 71. Authority to make changes to a dual-purpose package design would be provided, and new requirements on the issuance and review of an SAR would also be provided. These new regulations would only apply to Type B(DP) dual-purpose packages intended for the domestic transportation and storage of spent fuel. Because IAEA standard TS-R-1 does not contain any provisions to permit a certificate holder to make changes to the design of a package without prior review and approval by the "competent authority" that issued the certificate, a Type B(DP) could not be approved for international use.

To provide a clear distinction between these new and existing packages, the new packages would be classified as Type B(DP), would have a unique "B(DP)" identifier, and for reasons discussed below, these packages would not be required to meet TS-R-1 standards and could not be used in international transport. For a Type B(DP) package, requirements on submitting an FSAR, periodically updating the FSAR, applying for an amendment to the CoC, and changing the design of the dual-purpose package, without prior NRC approval, would be consolidated in a new Subpart I to Part 71. To provide greater consistency between the Parts 71 and 72 CoCs, the NRC staff would use the same term for both CoCs (i.e., 20 years), and would synchronize the CoCs' expiration dates. Further the NRC staff would use the same 20-year term for a QA program approval to design or fabricate a Type B(DP) package.

Additionally, a general license, new § 71.18, would be added to Subpart C that would require a licensee shipping spent fuel in a Type B(DP) package to have both a copy of the CoC and the current updated FSAR before making the shipment. Licensees would not be authorized under this proposed rule to make changes to a Type B(DP) package's design by themselves, but would be required to obtain certificate holder (i.e., the package designer) review and approval of the proposed change. Further, should the evaluation of the proposed change indicate that prior NRC approval is required, then only the certificate holder would be authorized to submit an application to the NRC to amend the CoC.

NRC believes that approval of proposed changes to the design of a Type B(DP) package, or submitting a request to modify a package's design, should be restricted to the certificate holder. As described above, licensees have not previously been required to understand the design bases for a transportation package or the technical bases of the Part 71 regulations.

The NRC believes that the new parallel structure provides a choice to applicants desiring to obtain transportation certification for a spent fuel storage and transportation package. This proposed structure (in Subpart I) would not restrict an applicant's right to obtain a CoC for a spent fuel transportation package under the existing requirements in Subpart D. Applicants can weigh the costs and benefits associated with each approach against the needs of its customers and determine which approach is better. Consequently, the NRC believes the new parallel structure is voluntary and does not impose a backfit.

Additional conforming changes would be made to § 71.0 to include Type B(DP) packages within the scope of Part 71; to § 71.4 to add a definition for *Certificate of compliance*, *Type B(DP) packages* and *structures, systems, and components important to safety*; to § 71.6 to reflect the new recordkeeping and reporting requirements created by the addition of new Subpart I (required under the Paperwork Reduction Act); to add a new § 71.10 to provide for public availability of applications; to § 71.51 to exclude Type B(DP) packages; and to § 71.100 to indicate which of these new sections (i.e., § 71.18 and Subpart I) would be subject to criminal penalties.

The NRC draft RA indicates that the proposed expansion of Part 71 to include a new § 71.175, "Changes, tests, and experiments," to include Part 71 certificate holders is reasonable from a regulatory, cost, and safety perspective. As noted, however, the NRC has very limited data from which to draw this conclusion. The NRC believes that not adopting these provisions may be awkward and appears to result in a regulatory inconsistency. Specifically, this inconsistency appears in situations where a certificate holder for a dual-purpose cask design could not modify the design of a component that had both storage and transport functions without prior NRC approval, irrespective of the certificate holder's authority under § 72.48 to modify the design of a storage cask. While the adoption of this change would not be consistent with the requirements in TS-R-1, the NRC believes the benefits to be gained by allowing Part 50 and Part 72 licensees and the Part 71 certificate holder to revise the cask design for a dual-purpose cask outweigh the potential impacts of this inconsistency. Further, these impacts would be offset by restricting this authority to packages intended for domestic shipments only. Preliminary estimates indicate that NRC costs would decline slightly by adopting this change, because the NRC would not have to review as many license amendments each year. This cost savings was determined to be negligible in the § 72.48 regulatory analysis, and would be offset by the agency having to adopt new document controls to handle the “minimal change” submission required every two years for licensees making “minimal changes.” For the 350 recordkeeping licensees listed in the Part 71 Supporting Statement, professional judgment was used to assume that, in any given year, 50 percent of licensees will perform a “minimal change” as described in § 72.48 over a 2-year period. Submittals under § 72.48 are required every 2 years, therefore approximately 88 submittals are expected per year. The cost savings of reporting “minimal changes” versus preparing license amendments is estimated at approximately $2.4 million per year. The 350 licensees would incur a one-time recordkeeping cost of approximately $2.3 million the first year this change is implemented.

NRC Proposed Position. The NRC proposes to add a new type of package (dual-purpose) to Part 71 [i.e., Type B(DP)]. Type B(DP) transportation packages would be certified for the storage of spent fuel under Part 72 and for transportation of spent fuel under Part 71. Type B(DP) packages would be restricted to use in domestic commerce. Requirements on the submission, review, amendment, and issuance of a CoC for a Type B(DP) package would be contained in a new Subpart I to Part 71. A new general license providing for the use of a Type B(DP) package would be added to Subpart C (§ 71.18). Certificate holders for Type B(DP) packages would also be required to submit, and periodically update, an FSAR describing the package's design. Additionally, only the certificate holder for a Type B(DP) package would be allowed under Subpart I to make changes to the package's design.

Additionally, conforming changes would be made to §§ 71.0, 71.4, 71.6, 71.10, 71.17, and 71.100

Affected Sections. §§ 71.0, 71.4, 71.6, 71.10, 71.17, 71.18, 71.100, and 71.151 through 71.177.

Issue 16. Fissile Material Exemptions and General License Provisions

Background. The NRC published an emergency final rule amending Part 71 on shipments of small quantities of fissile material (62 FR 5907; February 10, 1997). This rule revised the regulations on fissile exemptions in § 71.53 and the fissile general licenses in §§ 71.18 and 71.22. The NRC determined that good cause existed, pursuant to § 553(b)(B) of the Administrative Procedure Act (APA) (5 U.S.C. 553(b)(B)), to publish this final rule without notice and prepromulgation opportunity for public comment. Further, the NRC also determined that good cause existed, under Section 553(d)(3) of the APA (5 U.S.C. 553(d)(3)), to make this final rule immediately effective. Notwithstanding the final status of the rule, the NRC provided for a 30-day postpromulgation public comment period. The NRC subsequently published in the Federal Register (64 FR 57769; October 27, 1999) a response to the postpromulgation comments received on the emergency final rule and a request for information on any unintended economic impacts caused by the emergency final rule.

The NRC issued this emergency final rule in response to a regulatory defect in the fissile exemption regulation in § 71.53 which was identified by an NRC licensee. The licensee was evaluating a proposed shipment of a special fissile material and moderator mixture (beryllium oxide mixed with a low concentration of high-enriched uranium). The licensee concluded that while § 71.53 was applicable to the proposed shipment, applying the requirements of § 71.53 could, in certain circumstances, result in an inadequate level of criticality safety (i.e., an accidental nuclear criticality was possible in certain unique circumstances).[7]

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[7] For transportation purposes, “nuclear criticality” means a condition in which an uncontrolled, self-sustaining, and neutron-multiplying fission chain reaction occurs. “Nuclear criticality” is generally a concern when sufficient concentrations and masses of fissile material and neutron moderating material exist together in a favorable configuration. Neutron moderating material cannot achieve criticality by itself in any concentration or configuration. However, it can enhance the ability of fissile material to achieve criticality by slowing down neutrons or reflecting neutrons.

The NRC staff confirmed the licensee's analysis that this beryllium oxide and high-enriched uranium mixture created the potential for inadequate criticality safety during transportation. An added factor in the urgency of the situation was that under the NRC regulations in §§ 71.18, 71.20, 71.22, 71.24, and 71.53, these types of fissile material shipments could be made without prior approval of the NRC. For many years, the NRC allowed these shipments of small quantities of fissile material based on the NRC's understanding of the level of risk involved with these shipments, as well as industry's historic transportation practices. This experience base had led the NRC (and its predecessor, the Atomic Energy Commission (AEC)) to conclude that shipments made under the fissile exemption provisions of Part 71 typically required minimal regulatory oversight (i.e., the NRC considered these types of shipments to be inherently safe).[8]

All public comments on the emergency final rule supported the need for limits on special moderators (i.e., moderators with low neutron-absorption properties such as beryllium, graphite, and deuterium). However, the commenters stated that the restrictions were far too limiting (to the point that some inherently safe packages were excluded from the fissile exemption) and could lead to undue cost burdens with no benefit to safety. In addition, the commenters believe that the consignment mass limits set to deter undue accumulation of fissile mass would be extremely costly. Therefore, the commenters recommended that further rulemaking was necessary to resolve these excessive restrictions. Based on the public comments on the emergency final rule, NRC staff contracted with Oak Ridge National Laboratory (ORNL) to

[8] The NRC's regulations in Part 71 ensure protection of public health and safety by requiring that Type AF, B, or BF packages used for transportation of large quantities of radioactive materials be approved by the NRC. This approval is based upon the NRC's review of applications which contain an evaluation of the package's response to a specific set of rigorous tests to simulate both normal conditions of transport (NCT) and hypothetical accident conditions (HAC). However, certain types of packages are exempted from the testing and NRC prior approval; these are fissile material packages that either contain exempt quantities (§ 71.53), or are shipped under the general license provisions of §§ 71.18, 71.20, 71.22, or 71.24.

review the fissile material exemptions and general license provisions, study the regulatory and technical bases associated with these regulations, and perform criticality model calculations for different mixtures of fissile materials and moderators. The results of the ORNL study were documented in NUREG/CR-5342,[9] and the NRC published a notice of the availability of this document in the Federal Register (63 FR 44477; August 19, 1998). The ORNL study confirmed that the emergency final rule was needed to provide safe transportation of packages with special moderators that are shipped under the general license and fissile material exemptions, but the regulations may be excessive for shipments where water moderation is the only concern. The ORNL study recommended that the NRC revise Part 71.

Subsequently, the NRC published a Federal Register notice that responded to public comments on the emergency final rule and requested additional information on the cost impact of the emergency final rule from the public, industry, and the DOE (64 FR 57769; October 27, 1999). The Commission requested this cost impact information because the NRC staff was not successful in obtaining this information. Specifically, the NRC requested information on the cost of shipments made under the fissile material exemptions and general license provisions of Part 71 before the publication of the emergency final rule, and those costs and/or changes in costs resulting from implementation of the emergency rule. One commenter agreed with the NRC approach, but stated that, ”the limits for those materials containing no special moderators can and should be increased, hopefully back to their pre-emergency rule levels.”

As part of NUREG/CR-5342, ORNL performed computer model calculations of k_{eff} (k-effective) for various combinations of fissile material and moderating material, including beryllium, carbon, deuterium, silicon-dioxide, and water, to verify the accuracy of current minimum critical mass values. These minimum critical mass values were then applied to the

[9] NUREG/CR-5342, "Assessment and Recommendations for Fissile-Material Packaging Exemptions and General Licenses Within 10 CFR Part 71," July 1998.

regulatory structure contained in Part 71, and revised mass limits for both the general license and exemption provisions to Part 71 were determined. Also, ORNL researched the historical bases for the fissile material exemption and general license regulations in Part 71 and discussed the impact of the emergency final rule's restrictions on NRC licensees. ORNL concluded that the restrictions imposed by the emergency final rule were necessary to address concerns relative to uncontrolled accumulation of exempt packages (and thus fissile mass) in a shipment and the potential for inadequate safety margin for exempt packages with large quantities of special moderators.

Based on its new k_{eff} calculations, ORNL suggested that: (1) the mass limits in the general license and exemption provisions could be safely increased and thereby provide greater flexibility to licensees shipping fissile radioactive material; and (2) additional revisions to Part 71 were appropriate to provide increased clarification and simplification of the regulations. NUREG/CR-5342 is available electronically in the Reference Library area of the NRC's Home Page under Technical Reports (http://www.nrc.gov) or the NRC's Public Electronic Reading Room. Also, copies of NUREG/CR-5342 may be obtained by writing to the Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328. Copies are also available from the National Technical Information Service, 5285 Port Royal Road, Springfield, VA 22161-0002. A copy is also available for inspection and copying, for a fee, at the NRC Public Document Room in the NRC Headquarters at One White Flint North, Room O-1F21, 11555 Rockville Pike, Rockville, MD 20852-2738.

Discussion. The NRC has received public comments on this issue in response to the issues paper, public meetings, and the workshop. Industry representatives, public interest organizations, Agreement States, and members of the public supported the issue. None of the comments presented new issues from those previously presented in the response to the emergency final rule or the Commission's request for additional cost information.

Addressing the emergency final rule, one commenter agreed with the necessity for the rule, but stated that there are issues yet to be resolved for water moderated shipments. In comparison, another commenter took issue with our stated goal and NRC's methods. This commenter believed that if NRC adopts these provisions, then NRC will be unable to conform with TS-R-1. The commenter cited as evidence a statement in the issues paper, "IAEA standard ST-1 (nee TS-R-1) contains language on fissile exemptions and restrictions on the use of special moderators. However, ST-1 does not currently contain provisions on general licenses for shipment of fissile material."

Similarly, one commenter raised the importance of coordinating regulatory actions on fissile material exemptions with the international community. The commenter noted the international community's interest in fissile material exemptions and encouraged NRC to listen to its international counterparts at the next IAEA meeting; the commenter's goal being to ensure that NRC is not out of step with the rest of the world (i.e., fissile material exempt in the U.S. is not elsewhere and vice-versa).

One commenter raised questions concerning specific recommendations in NUREG/CR-5342. The commenter was concerned in how recommendations 3 and 4 would introduce unnecessary complexity and noted that this concern vanishes if the TS-R-1 definitions for regulated material are adopted. The commenter also stated that recommendation 17 could seemingly eliminate the fissile excepted category, which is something the commenter did not want to see occur. If such a change is necessary, the commenter requested that the NRC instead revise the excepted package's definition to reduce the amount of fissile material present and ensure that 10 CFR 71.53 and 49 CFR 173.453 are consistent with TS-R-1 (i.e., with respect to upper limits on a package's fissile material, as well as the total amount of fissile material in a fissile exempt consignment).

The current restrictions on fissile exempt and general license shipments under § 71.53, and §§ 71.18 through 71.24, respectively, are burdensome for a large number of shipments that actually contain no special moderating materials (i.e., packages that are shipped with water considered as the potential moderating material). This problem was clearly expressed in public comments on the emergency final rule. Another regulatory problem is that the current fissile exempt and general license provisions are cumbersome and outdated; this was one of the main conclusions of the ORNL study. Therefore, the NRC would update, simplify, and streamline these sections of Part 71 to eliminate regulatory confusion.

The proposed revisions in Table 16-1 are based on public comments received on the February 10, 1997, emergency final rule, on the subsequent Commission's direction in SRM-SECY-99-200 regarding the unintended economic impact of that emergency final rule, and on the latest public comments received on the July 2000 issues paper. Altogether, ORNL suggested 17 changes to the Part 71 regulations in NUREG/CR-5342. A summary of these changes and the NRC's assessment and recommendation are contained in Table 16-1. NUREG/CR-5342 contains a more detailed discussion of the proposed changes listed in Table 16-1 and ORNL's supporting calculations.

Table 16-1

Summary of Recommended Changes in NUREG/CR-5342

Description of Issue	**NRC Staff Recommendation**
Issue 16-1: Definitions for "consignment," "consignor," and "shipper" should be provided to reduce confusion between regulations in 49 CFR Part 173 and 10 CFR Part 71.	Disagree. These changes are not necessary with the use of mass ratio limits and a criticality safety index when combined with the current requirement in § 71.59.
Issue 16-2: Plutonium-238 should be removed from the definition of "fissile material," because ^{238}Pu is only fissionable, not fissile.	Agree.
Issue 16-3: The exemption for radioactive material in § 71.10(a) should be revised to exclude fissile material. ORNL's concern was that a large quantity of a low-concentration fissile material could pose a criticality safety concern. The revised k_{eff} calculations indicate that a 43 Bq/g (1.16 x 10^{-3} μCi/g) limit for fissile material (^{235}U) would be necessary. However, other fissile nuclides have higher limits (e.g, 6,230 Bq/g (0.168 μCi/g) for ^{233}U or 66,000 Bq/g (1.784 μCi/g) for ^{241}Pu) or the Appendix A, new Table A-2, values are only 10 Bq/g (2.7 x 10^{-4} μCi/g) (e.g., ^{239}Pu).	Disagree. The existing exception to the exemption in paragraph (b) would be maintained (i.e., the reference to the fissile exemption in new § 71.15). However, no change would be made to paragraph (a) because the values in Table A-2 are less than 43 Bq/g (1.16 x 10^{-3} μCi/g) or the fissile nuclides have criticality limits which would be higher than the exempt concentration limits of Table A-2.
Issue 16-4: The exemption for radioactive material in existing § 71.10 should be revised to require shipment in an acceptable package as required by existing § 71.11 to improve safety.	Agree.
Issue 16-5: Section 71.53 should be relocated from Subpart E — Package Approval Standards, to Subpart B — Exemptions, to provide greater consistency in Part 71. (Note: § 71.53 would also be redesignated as § 71.15.)	Agree.
Issue 16-6: The NRC or DOT should keep a database of shipments made under the fissile exemption or general licenses. Section 71.97 should be revised to require licensees to keep these records and report this information.	Disagree. The licensee's burden in keeping and reporting these records is not commensurate with the safety risk for fissile exemption shipments.

Issue 16-7: The provisions for plutonium-beryllium (Pu-Be) shipments should be removed from the four general licenses of existing §§ 71.18, 71.20, 71.22, and 71.24 and consolidated in a new general license. The mass limits for Pu-Be shipments should be reduced, because the revised k_{eff} calculations indicate potential safety problems exist with the current limits.	Agree.
Issue 16-8: The general licenses of existing §§ 71.18, 71.20, 71.22, and 71.24 should be consolidated into one general license to simplify the regulations and consistently apply the criticality safety index (CSI).	Agree.
Issue 16-9: The distinction between quantities of ^{235}U that can be shipped in a uniform distribution and nonuniform distribution should be eliminated from the general licenses. The bounding nonuniform quantities should be used to simplify compliance with the rule.	Agree.
Issue 16-10: Restrictions on the quantities of Be, C, and D_2O to less than 0.1% should be removed for the general licenses. A maximum of 500g of Be, C, and D_2O per package should be imposed to preclude the potential for these materials to be effective as reflector materials.	Agree.
Issue 16-11: A separate mass control or restriction for moderators having a hydrogen density greater than water should be retained for general licenses. For mixtures of moderators, lower mass limits should be imposed if more than 15% of the moderating material has a moderating effectiveness greater than the hydrogen density of water. Use of a 15% mixture limit would reduce confusion when mixtures of moderators are present in a shipment.	Agree.
Issue 16-12: Package mass limits for general licenses may be increased to reflect results of new analyses and still maintain equivalence of safety as provided for certified packages.	Agree. Also, minimum package requirements should be established. However, imposing § 71.43 requirements would be excessive for the commensurate risk from these shipments. Instead, the DOT Type A package requirements should be used.
Issue 16-13: Package mass limits for general licenses should be revised to reflect the new k_{eff} calculations. These mass limits can be safely increased.	Agree.

Issue 16-14: The mass-limit based exemption in existing § 71.53(a) should be changed to a mass-ratio based approach. In contrast to concentration-based approaches with consignment limits that are now in use in the fissile exemptions, the mass-ratio approach should provide a simpler, more cost-effective approach to preventing the formation of system configurations having inadequate subcritical margins as a result of transport scenarios (§§ 71.71 and 71.73).	Agree.
Issue 16-15: If a mass-ratio approach is used, the restrictions on Be, C, and D_2O in existing § 71.53(a), (c), and (d) should be removed.	Agree.
Issue 16-16: The exemption for uranyl nitrate solutions in § 71.53(c) should include a packaging requirement from existing § 71.43.	Agree, in part. Minimum package requirements should be established. However, § 71.43 is excessive for the commensurate risk from these shipments. The DOT Type A package requirements should be used.
Issue 16-17: The exemption for uranium enriched to less than 1 wt % ^{235}U in existing § 71.53(b) should be modified to remove the homogeneity requirements and lattice prevention requirement. Instead, retain the 0.1% Be, C, and D_2O limit because of the difficulty in defining and applying "homogenous" and "lattice arrangement" restrictions.	Agree.

In addition to the recommendations contained in NUREG/CR-5342, the Commission directed the NRC staff, in SRM-M970122B on SECY-96-268, to issue additional guidance in instances where fissile materials may be mixed in the same shipping container with different moderators (i.e., materials of differing moderator effectiveness). Therefore, the NRC would add a note to Table 71-1 in existing § 71.22 to use reduced mass limits if more than 15 percent of the moderating materials in a package have a moderating effectiveness greater than the average hydrogen density of H_2O (see Issue 16-11 in Table 16-1 above).

The NRC believes these changes would provide greater flexibility in the shipment of fissile material under the fissile exemption and general license regulations. The NRC would revise these requirements using a risk-informed approach, and address the burden and excessiveness issues raised in the public comments on the emergency final rule. The NRC would use a graduated regulatory approach in establishing requirements for the shipment of fissile material. The graduated approach would involve three tiers of regulations consisting of: (1) the fissile material exemptions with low fissile mass limits and minimal requirements (i.e., the new § 71.15); (2) the fissile general licenses with higher mass limits and packaging and QA requirements (i.e., the new §§ 71.22 and 71.23); and (3) the Type AF, BF, B(U)F, or B(M)F fissile material packages with large mass limits that require prior NRC approval of the package design (i.e., the existing § 71.55). The NRC believes this approach would establish a risk-informed framework by imposing progressively stricter requirements as the quantity of fissile material being shipped increases (i.e., the criticality hazard increases). In accomplishing this risk-informed approach, some mass limits in the general licenses would increase, and others would decrease. These changes would reflect the new k_{eff} calculations in NUREG/CR-5342. To counterbalance the increases in mass limits in the general licenses, requirements would be added on the use of a Type A package, a CSI, and an NRC-approved QA program.

Overall, the NRC would amend Part 71 as follows: (1) revise § 71.10, "Exemption for low level material," to exclude fissile material, also redesignate § 71.10 as 71.14 ; (2) redesignate § 71.53 as §71.15, "Exemption from classification as fissile material," and revise the fissile exemptions; (3) consolidate the existing four general licenses in §§ 71.18, 71.20, 71.22, and 71.24 into one general license in new § 71.22, revise the mass limits, and add Type A package, CSI, and QA requirements; and (4) consolidate the existing general licenses requirements for plutonium-beryllium sealed sources, which are contained in existing §§ 71.18 and 71.22 into one general license in new § 71.20 and revise the mass limits. Additionally, conforming changes would be made to § 71.4, "Definitions"; to § 71.59, "Standards for arrays of fissile material packages"; and § 71.100, “Criminal penalties.”

The NRC draft RA indicates that incorporating revisions to the fissile material exemption and general license provisions in Part 71 is appropriate from a safety, regulatory, and cost perspective. As stated earlier, there is a shortage of data on the fissile material general license and exempt shipments; consequently, the NRC was not successful in obtaining data to quantify the economic impact which would result from adopting some or all of the 17 recommendations in NUREG/CR-5342. The impact of these amendments on the licensees and the NRC would be both positive and negative, depending on the specific recommendation. Recommendations 1, 2, and 5 would enhance regulatory efficiency due to the increase in clarity of the NRC regulations. Recommendations 3, 4, 6, 9, and 12 would increase costs to licensees. Recommendations 7, 8, 10, 13, 14, and 15 would eliminate the potential for criticality accidents, which would, in turn, yield environmental and public health and safety benefits. Finally, recommendations 11, 16, and 17 would result in savings to licensees.

NRC Proposed Position. The NRC proposes revisions to the fissile material exemptions and the general license provisions in Part 71.

Affected Sections. 71.4, 71.10, 71.11, 71.18, 71.20, 71.22, 71.24, 71.53, 71.59, and 71.100.

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

Background: In 1974, the AEC issued a final rule which imposed special requirements on the shipment of plutonium (39 FR 20960; June 17, 1974). These requirements are located in 10 CFR 71.63 and apply to shipments of radioactive material containing quantities of plutonium in excess of 0.74 TBq (20 curies). Section 71.63 contains two principal requirements. First, the plutonium contents of the package must be in solid form (§ 71.63(a)). Second, the packaging containing the plutonium must provide a separate inner containment (i.e., the “double containment” requirement) (§ 71.63(b)). In addition, the AEC specifically excluded from the double containment requirement of § 71.63(b) plutonium in the form of reactor fuel elements, metal or metal alloys, and other plutonium-bearing solids that the Commission (AEC or NRC) may determine, on a case-by-case basis, do not require double containment. This regulation remained essentially unchanged from 1974 until 1998, when vitrified high-level waste in sealed canisters was added to the list of exempt forms of plutonium in § 71.63(b) (63 FR 32600; June 15, 1998). The double containment requirement is in addition to the existing Subparts E and F requirements imposed on Type B packagings (e.g., the normal conditions of transport and hypothetical accident conditions of §§ 71.71 and 71.73, respectively, and the fissile package requirements of §§ 71.55 and 71.59). Part 71 does not impose a double containment requirement for any radionuclide other than plutonium. Additionally, IAEA standard TS-R-1 does not provide for a double containment requirement (in lieu of the single containment Type B package standards) for any radionuclide.

The AEC issued this regulation at a time when AEC staff anticipated widespread reprocessing of commercial spent fuel, and existing shipments of plutonium were made in the form of liquid plutonium nitrate. Because of physical changes to the plutonium that was expected to be reprocessed (i.e., higher levels of burnup in commercial reactors for spent fuel, which would then be reprocessed), and regulatory concerns with the possibility of package leakage, the AEC issued a regulation that imposed the double containment requirement when the package contained more than 0.74 TBq (20 Ci) of plutonium. This double containment was in addition to the existing Type B package standards on packages intended for the shipment of greater than an A_1 or A_2 quantity of plutonium.

NRC staff has reviewed the available regulatory history for § 71.63, and has provided a recapitulation of the supporting information which led to the issuance of this regulation. NRC staff has extracted the following information from several SECY papers the AEC staff submitted to the Commission on this regulation. NRC staff believes this information is relevant and will provide stakeholders with perspective in understanding the bases for this regulation, and thereby assist stakeholders in evaluating the staff's proposed changes to this regulation.

In SECY-R-702,[10] the AEC staff identified two considerations that were the genesis of the rulemaking that led to § 71.63. AEC staff stated:

> First, increasingly larger quantities of plutonium will be recovered from power reactor spent fuel. Second, the specific activity of the plutonium will increase with higher reactor fuel burnup resulting in greater pressure generation potential from plutonium nitrate solutions in shipping containers, greater heat generation, and higher gamma and neutron radiation levels. These changes will make the present nitrate packages obsolete. Thus, from both safety and economic considerations, the transportation of plutonium as [liquid] nitrate will soon require substantial redesign of packages to handle larger quantities as well as to deal with the higher levels of gas evolution (pressurization), heat generation, and gamma and neutron radiation.
>
> There is little doubt that larger plutonium nitrate packages could be designed to meet regulatory standards. The increased potential for human error and the consequences of such error in the shipment of plutonium nitrate are not so easily controlled by regulation. Even though such packages may be adequately designed, their loading and closure requires high operation performance by personnel on a continuing basis. As the number of packages to be shipped increases, the probability of leakage through improperly assembled and closed packages also increases.... More refined or stringent regulatory requirements,

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[10] SECY-R-702, "Consideration of Form for Shipping Plutonium," June 1, 1973.

such as double containment, would not sufficiently lessen this concern because of the necessary dependence on people to affect engineered safeguards.

In SECY-R-74-5,[11] AEC staff summarized the factors relevant to consideration of a proposed rule following a June 14, 1973, meeting to discuss SECY-R-702, between the Regulatory and General Manager's staffs (i.e., the rulemaking and operational sides of the AEC). The AEC stated:

> As a result of this meeting [on June 14, 1973], the [Regulatory and General Manager's] staffs have agreed that the basic factors pertinent to the consideration of form for shipment of plutonium are:
>
> 1. The experience with shipping plutonium as an aqueous nitrate solution in packages meeting current regulatory criteria has been satisfactory to date.
> 2. The changing characteristic of plutonium recovered from power reactors will make the existing packaging obsolete for plutonium nitrate solutions and possibly for solid form. Economic factors will probably dictate considerably larger shipments (and larger packages) than currently used.
> 3. It is expected that packages can be designed to meet regulatory standards for either aqueous solutions or solid plutonium compounds. Just as in any situation involving the packaging of radioactive materials, a high level of human performance is necessary to assure against leakage caused by human error in packaging. As the number of plutonium shipments increases, as it will, and packages become larger and more complex in design, the probability of such human error increases.

[11] SECY-R-74-5, "Consideration of Form for Shipping Plutonium," dated July 6, 1973.

4. The probability of human error with the packaging for liquid, anticipated to be more complex in design, is probably greater than with the packaging for solid. Furthermore, should a human error occur in package preparation or closure, the probability of liquid escaping from the improperly prepared package is greater than for most solids and particularly for solid plutonium materials expected to be shipped.
5. Staff studies reported in SECY-R-62 and SECY-R-509[12] conclude that the consequences of release of solid or aqueous solutions do not differ appreciably. Therefore, this paper (SECY-R-702) does not deal with the consequences of releases.
6. It is therefore concluded that safety would be enhanced if plutonium were shipped as a solid rather than in solution.

The arguments for requiring a solid form of plutonium for shipment are largely subjective, in that there is no hard evidence on which to base statistical probabilities or to assess quantitatively the incremental increase in safety which is expected. The discussion in the regulatory paper, SECY-R-702, is not intended to be a technical argument which incontrovertibly leads to a conclusion. It is, rather, a presentation of the rationale which has led the Regulatory staff to its conclusion that a possible problem may develop and that the proposed action is a step towards increased assurance against the problem developing. In SECY-R-74-172, [13] AEC staff submitted a final rule to the Commission for approval.

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[12] SECY-R-62, "Shipment of Plutonium," and SECY-R-509, "Plutonium Handling and Storage," dated October 16, 1970. These papers concluded that there is no scientific or technical reason to prohibit shipment of plutonium nitrate and recommended that Commission (AEC) efforts be directed toward providing improved safety criteria for shipping containers.

[13] SECY-R-74-172, “Consideration of Form for Shipping Plutonium,” April 18, 1974.

The proposed rule had contained a requirement that the plutonium be contained in a special form capsule. However, in response to comments from the AEC General Manager, the final rule changed this requirement to a separate inner container (i.e., the double containment requirement). The AEC staff indicated in a response to a public comment in Enclosure B (to SECY-R-74-172) that "[t]he need for the inner containment is based on the desire to provide a substitute for not requiring the plutonium to be in a 'nonrespirable' form."

The NRC staff believes the regulatory history of § 71.63 indicates that the AEC's decision to require a separate inner container for shipments of plutonium in excess of 0.74 TBq (20 Ci) was based on policy and regulatory concerns (i.e., "that a possible problem may develop and that the proposed action [in SECY-R-702] is a step towards increased assurance against the problem developing"). Because of the expectation of a significant increase in the number of liquid plutonium nitrate shipments, the AEC used a defense-in-depth philosophy (i.e., the double containment and solid form requirements), to ensure that respirable plutonium would not be released to the environment during a transportation accident. However, the regulatory history does indicate that the AEC's concerns did not involve the adequacy of existing liquid plutonium nitrate packages. Rather, the AEC's regulatory concern was on the increased possibility of human error combined with an expected increase in the number of shipments would yield an increased probability of leakage during shipment. The AEC's policy concern was based on an economic decision on whether the AEC should require the reprocessing industry to build new, larger liquid plutonium-nitrate shipping containers, capable of handling higher burnup reactor spent fuel, or to build new, dry, powdered plutonium-dioxide shipping containers. The regulatory history indicates that the AEC staff judged that new, larger, higher burnup-capacity liquid plutonium-nitrate packages could be designed, approved, built, and safely used. However, one of the AEC's principal underlying assumptions for this rule was obviated in 1979 when the Carter administration decided that reprocessing of civilian spent fuel and reuse of plutonium was not desirable. Consequently, the expected plutonium reprocessing economy and widespread shipments of liquid plutonium nitrate within the U.S. never materialized.

On June 15, 1998, in response to a petition for rulemaking from the DOE (PRM-71-11), the Commission issued a final rule revising § 71.63(b) to add vitrified high-level waste contained in a sealed canister to the list of forms of plutonium exempt from the double containment requirement (June 15, 1998; 63 FR 32600). In its original response to PRM-71-11, NRC proposed in SECY-96-215[14] to make a "determination" under § 71.63(b)(3) that vitrified HLW contained in a sealed canister did not require double containment. However, the Commission in an SRM on SECY-96-215, dated October 31, 1996, disapproved the staff's approach and directed that resolution of this petition be addressed through rulemaking (the June 15, 1998, final rule was the culmination of this effort). In addition to disapproving the use of a "determination" process, the Commission also directed the staff to "... also address whether the technical basis for 10 CFR 71.63 remains valid, or whether a revision or elimination of portions of 10 CFR 71.63 is needed to provide flexibility for current and future technologies." In SECY-97-218[15], NRC staff responded to the SRM's direction and stated "[t]he technical basis remains valid and the provisions provide adequate flexibility for current and future technologies."

Petition: The NRC received a petition for rulemaking from International Energy Consultants, Inc. (IEC), dated September 25, 1997. The petition was docketed as PRM-71-12 and was published for public comment (63 FR 8362; February 19, 1998). Based on a request

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[14] SECY-96-215, "Requirements for Shipping Packages Used to Transport Vitrified Waste Containing Plutonium," dated October 8, 1996.

[15] SECY-97-218, "Special Provisions for Transport of Large Quantities of Plutonium (Response to Staff Requirements Memorandum - SECY-96-215)," dated September 29, 1997.

from General Atomic, the comment period was extended to July 31, 1998 (see 63 FR 34335; June 24, 1998). Nine public comments were received on the petition. Four commenters supported the petition, and five commenters opposed the petition.

The petitioner requested that § 71.63(b) be removed. The petitioner argued that the double containment provisions of § 71.63(b) cannot be supported technically or logically. The petitioner stated that based on the "Q-system for the Calculation of A_1 and A_2 Values," an A_2 quantity of any radionuclide has the same potential for damaging the environment and the human species as an A_2 quantity of any other radionuclide.

NRC staff believes that the Q-values are based upon radiological exposure hazard models which calculate the allowable quantity limit (the A_1 or A_2 value) necessary to produce a known exposure (i.e., one A_2 of plutonium-239 or one A_2 of cobalt-60 will both yield the same radiation dose under the Q-system models, even though the A_2 values for these nuclides are different [e.g., one A_2 of plutonium-239 = 2×10^{-4} TBq of plutonium and one A_2 of cobalt-60 = 1 TBq of cobalt]). The Q-system models take into account the exposure pathways of the various radionuclides, typical chemical forms of the radionuclide, methods for uptake into the body, methods for removal from the body, the type of radiation the radionuclide emits, and the bodily organs the radionuclide preferentially affects. The specific A_1 and A_2 values for each nuclide are developed using radiation dosimetry approaches recommended by the World Health Organization and the International Commission on Radiological Protection (ICRP). The models are periodically reviewed by international health physics experts (including representatives from the United States), and the A_1 and A_2 values are updated during the IAEA revision process, based upon the best available data. (Note that changes to the A_1 and A_2 values as a result of changes to the models in TS-R-1 are also discussed in Issue 3.) These values are then promulgated by the IAEA in safety standards such as TS-R-1. When the IAEA has revised the A_1 and A_2 values in previous revisions of its transport regulations, these revised values have been adopted by the NRC and DOT into the transportation regulations in 10 CFR Part 71 and 49 CFR Part 173, respectively.

NRC's review of the current A_1 and A_2 values in Appendix A to Part 71, Table A-1, reveals that 5 radionuclides have an A_2 value lower than plutonium (i.e., plutonium-239), and 11 radionuclides have an A_2 value that is equal to plutonium-239. Because the models used to determine the A_1 and A_2 values all result in the same radiation exposure (i.e., hazard), a smaller A_1 and A_2 value for one radionuclide would indicate a greater potential hazard to humans than a radionuclide with larger A_1 and A_2 value. Thus, the overall Table A-1 can also be viewed as a relative hazard ranking (for transportation purposes) of the listed radionuclides. In that light, requiring double containment for plutonium alone is not consistent with the relative hazard rankings in Table A-1.

The petitioner also argued that the Type B package requirements should be applied consistently for any radionuclide, whenever a package's contents exceed an A_2 limit. However, Part 71 is not consistent by imposing the double containment requirement for plutonium. The petitioner believes that if Type B package standards are sufficient for a quantity of a particular radionuclide which exceeds the A_2 limit, then Type B package standards should also be sufficient for any other radionuclide which also exceeds the A_2 limit. The petitioner stated that:

> While, for the most part, Part 71 regulations embrace this simple logical congruence, the congruence fails under 10 CFR 71.63(b) wherein packages containing plutonium must include a separate inner container for quantities of plutonium having a radioactivity exceeding 20 curies [0.74 TBq] (with certain exceptions).

The petitioner further stated that:

> If the NRC allows this failure of congruence to persist, the regulations will be vulnerable to the following challenges: (1) the logical foundation of the adequacy

of A_2 values as a proper measure of the potential for damaging the environment and the human species, as set forth under the Q-System, is compromised; (2) the absence of a limit for every other radionuclide which, if exceeded, would require a separate inner container, is an inherently inconsistent safety practice; and (3) the performance requirements for Type B packages, as called for by 10 CFR Part 71, establish containment conditions under different levels of package trauma. The satisfaction of these Type B package standards should be a matter of proper design work by the package designer and proper evaluation of the design through regulatory review. The imposition of any specific package design feature such as that contained in 10 CFR 71.63(b) is gratuitous. The regulations are not formulated as package design specifications, nor should they be.

NRC agrees that the Part 71 regulations are not formulated as package design specifications; rather, the Part 71 regulations establish performance standards for a package's design. The NRC reviews the application to evaluate whether the package's design meets the performance requirements of Part 71. Consequently, the NRC can then conclude that the design of the package provides reasonable assurance that public health and safety and the environment are adequately protected.

The petitioner also believes that the continuing presence of § 71.63(b) engenders excessively high costs in the transport of some radioactive materials without a clearly measurable net safety benefit. The petitioner stated that this is so, in part, because the ultimate release limits allowed under Part 71 package performance requirements are identical with or without a "separate inner container," and because the presence of a "separate inner container" promotes additional exposures to radiation through the additional handling required for the "separate inner container." Consequently, the petitioner asserted that the presence or absence of a separate inner container barrier does not affect the standard to which the outer container barrier must perform in protecting public health and safety and the environment. Therefore, the petitioner concluded that given that the outer containment barrier provides an acceptable level of safety, the separate inner container is superfluous and results in unnecessary cost and radiation exposure. According to the petitioner, these unnecessary costs involve both the design, review, and fabrication of a package, as well as the costs of transporting the package. And the unnecessary radiation exposure involves workers having to handle (i.e., seal, inspect, or move) the "separate inner container."

As an alternative to the primary petition, the petitioner believes that an option to eliminate both § 71.63(a) and (b) should also be considered. Section 71.63(a) requires that plutonium in quantities greater than 0.74 TBq (20 Ci) be shipped in solid form. This option would have the effect of removing § 71.63 entirely. The petitioner believes that the arguments set forth to support the elimination of § 71.63(b) also support the elimination of § 71.63(a). The petitioner did not provide a separate regulatory or cost analysis supporting the request to remove § 71.63(a).

Comments on the Petition: The four commenters supporting the petition essentially stated that the IAEA's Q-system accurately reflects the dangers of radionuclides, including plutonium, and that elimination of § 71.63(a) and (b) would make the regulations more performance based, reduce costs and personnel exposures, and be consistent with the IAEA standards.

The five commenters opposing the petition essentially stated that: (1) Plutonium is very dangerous, especially in liquid form, and therefore additional regulatory requirements are warranted; (2) Existing regulations are not overly burdensome, especially in light of the total expected transportation cost; (3) TRUPACT-II packages meet current § 71.63(b) requirements

(TRUPACT-II is a package developed by DOE to transport transuranic wastes {including plutonium} to the Waste Isolation Pilot Plant (WIPP) and has been issued a Part 71 CoC, No. 9218); (4) A commenter (the Western Governors' Association) has worked for over 10 years to ensure a safe transportation system for WIPP, including educating the public about the TRUPACT-II package; (5) Any change now would erode public confidence and be detrimental to the entire transportation system for WIPP shipments; and (6) Additional personnel exposure due to double containment is insignificant.

Discussion: The NRC has received 48 public comments on this issue in response to the issue paper, public meetings, and the workshop. Industry representatives and some members of the public support the petition. Public interest organizations, Agreement States, State representatives, the Western Governor's Association, and other members of the public oppose the petition. Several commenters believe that Congress, in approving the Waste Isolation Pilot Plant Land Withdrawal Act (the Act), Public Law 102-579 (106 Stat. 4777), Section 16(a), which mandates the NRC certify the design of packages used to transport transuranic waste to WIPP, expected those packages to have a double containment. The NRC researched this issue, and Section 16(a) of the Act does not contain any explicit provisions mandating the use of a double containment in packages transporting transuranic waste to or from WIPP. Section 16(a) of the Act states, in part, "[n]o transuranic waste may be transported by or for the Secretary [of the DOE] to or from WIPP, except in packages the design of which has been certified by the Nuclear Regulatory Commission..." Furthermore, the NRC has reviewed the legislative history[16] associated with the Act and has not identified any discussions

[16] See Congressional Record Vol. 137, November 5, 1991, pages S15984 - 15997 (Senate approval of S. 1671); Cong. Rec. Vol. 138, July 21, 1992, pages H6301 - 6333 (House approval of H.R. 2637); Cong. Rec. Vol. 138, October 5, 1992, pages H11868 - 11870 (House approval of Conference Report on S. 1671); Cong. Rec. Vol. 138, October 8, 1992 (Senate approval of Conference Report on S. 1671); and Cong. Rec. Vol. 138, October 5, 1992, pages H12221 - 12226 (Conference Report on S. 1671 - (H.) Rpt. 102-1037).

on the use of double containment for the shipment of transuranic waste. The legislative history does mention that the design of these packages will be certified by the NRC; however, this language is identical to that contained in the Act itself. Therefore, the NRC believes the absence of specific language in Section16(a) of the Act requiring double containment should be interpreted as requiring the NRC to apply its independent technical judgment in establishing standards for package designs and in evaluating applications for certification of package designs, to ensure that such packages would provide reasonable assurance that public health and safety and the environment would be adequately protected. In carrying out its mission, the courts have found that the NRC has broad latitude in establishing, maintaining, and revising technical performance criteria necessary to provide reasonable assurance that public health and safety and the environment are adequately protected. An example of these technical performance criteria is the Type B package design standards. Accordingly, the NRC believes that the proposed revision of a technical package standard (i.e., removal of the double containment requirement for plutonium from the Type B package standards) is not restricted by the mandate of § 16(a) of the Act for the NRC to certify the design of packages intended to transport transuranic material to and from WIPP.

Other commenters stated that stakeholders' expectations were that packages intended to transport transuranic material to and from WIPP would include a double containment provision. Consequently, the commenters believed that removal of the double containment requirement would decrease public confidence in the NRC's accomplishment of its mission in the approval of the design of packages for the transportation of transuranic waste to and from WIPP. The commenters believed the public would view elimination of the double containment requirement as a relaxation in safety. The presence of a separate inner container provides defense-in-depth through an additional barrier to the release of plutonium during a transportation accident. In addition, the commenters believed that plutonium is so inherently deadly, that defense-in-depth is appropriate. The NRC agrees that a double containment does provide an additional barrier. However, the NRC also believes this rationale is not risk-informed nor performance-based. The NRC believes that the use of Type B package standards provides a risk-informed approach to the transportation of radioactive material. The NRC and AEC have not required an additional containment barrier for Type B packages transporting any radionuclides other than plutonium and, before 1974, the AEC did not require double containment for plutonium.

In response to some of the comments opposed to the petition, the NRC believes that removal of § 71.63(b) would not invalidate the design of existing packages intended for the shipment of plutonium. These packages could continue to be used with a separate inner container. The NRC agrees with the commenters that a quantitative cost analysis was not provided by the petitioner.

The NRC has issued Part 71 CoC No. 9218 to the DOE for the TRUPACT-II package (Docket No. 71-9218), for the transportation of transuranic waste (including plutonium) to and from the WIPP. The TRUPACT-II package complies with the current § 71.63(b) requirements and has a separate inner container. The TRUPACT-II SAR indicates that the weight of the inner container and its lid is approximately 2,620 lbs. Hypothetically, elimination of the separate inner container would increase the available payload for the TRUPACT-II package from the current 7,265 to 9,885 lbs. Thus, removal of the double containment requirement would potentially increase the TRUPACT-II's available payload by 36 percent. Further, the removal of the inner container from the TRUPACT-II would also potentially increase the available volume. The NRC believes that the proposed rule would not invalidate the existing TRUPACT-II design, and thus, DOE could continue to use the TRUPACT-II to ship transuranic waste to and from WIPP, or DOE could consider an alternate Type B package.

Additionally, based on comments received in the public meetings, the NRC believes that a misperception exists with respect to TRUPACT-II shipments; removal of the § 71.63(b) double containment requirement would not result in loose plutonium waste being placed inside a TRUPACT-II package. Based upon information contained in the safety analysis report, plutonium wastes (i.e., used gloves, anti-Cs, rags, etc.) are placed in plastic bags, and these bags are sealed inside lined 55-gallon steel drums. Plutonium residues are placed inside cans which are then sealed inside a pipe overpack (a 6-inch or 12-inch stainless steel cylinder with a bolted lid), and the pipe overpack is then sealed inside a lined 55-gallon steel drum. The 55-gallon drums are then sealed inside the TRUPACT-II inner containment vessel, and finally the inner containment vessel is sealed inside the TRUPACT-II package. Consequently, the TRUPACT-II shipping practices employ multiple barriers, and removal of the inner containment vessel would not be expected to produce a significant incremental increase in the possibility of leakage during normal transportation. The NRC notes that some NRC regulations have established additional requirements for plutonium (e.g., the special nuclear material license application provisions of § 70.22(f)).

The NRC believes that the Type B packaging standards, in and of themselves, provide reasonable assurance that public health and safety and the environment would be adequately protected during the transportation of radioactive material. This belief is supported by an excellent safety record in which no fatalities or injuries have been attributed to material transported in a Type B package. Type B packaging standards have been in existence for approximately 40 years and have been incorporated into the Part 71 regulations by both the NRC and its predecessor, the AEC. The NRC's Type B package standards are based on IAEA's Type B package standards. Moreover, IAEA's Type B package standards have never required a separate inner container for packages intended to transport plutonium, nor for any other radionuclide. The NRC believes that while U.S. shipments of plutonium subject to

§ 71.63(b) have consisted primarily of solid plutonium contaminated wastes, other European countries have reprocessed plutonium in their reactor fuel cycles and have transported liquid plutonium nitrate. The NRC is not aware of any accidents involving a Type B liquid plutonium nitrate package which has led to the significant failure of the package and release of the contents.

Therefore, the NRC believes that imposition of an additional packaging requirement (in the form of a separate inner container) is fundamentally inconsistent with the position that Type B packaging standards, in and of themselves, provide reasonable assurance that public health and safety and the environment would be adequately protected during the transportation of (any type of) radioactive material. Thus, the NRC believes that § 71.63(b) is not consistent with the Type B packaging standards contained in Part 71.

The NRC also believes that the regulatory history of § 71.63 demonstrates that the AEC's decision was based on policy and regulatory concerns. However, the NRC also agrees that the use of a double containment does provide defense-in-depth and does decrease the absolute risk of the release of respirable plutonium to the environment during a transportation accident. Consequently, while the defense-in-depth afforded by a double containment does reduce risk, the NRC believes the question which should be focused on is whether the double containment requirement is risk-informed. The NRC is unaware of any risk studies that would provide either a qualitative or quantitative indication of the risk reduction associated with the use of double containment in transportation of plutonium. Rather, the NRC would look to the demonstrated performance record of existing Type B package standards to conclude that double containment is not necessary.

In summary, the AEC staff indicated (in SECY-R-702 and SECY-R-74-5), that liquid plutonium nitrate packages were safe, and new, larger, packages to handle higher burnup reactor spent fuel could also be designed. NRC believes that the AEC's assumption for initiating this requirement was that large scale reprocessing of civilian reactor spent fuel and reuse of plutonium would occur. Former President Carter's administration's decision to forgo the reprocessing of civilian reactor spent fuel and reuse of plutonium obviated the AEC's assumption. Consequently, the AEC's supposition that a human error occurring while sealing a package of liquid plutonium nitrate was more likely to occur with the expected increase in shipments of plutonium nitrate was also obviated by the Government's decision to forgo the reprocessing of civilian reactor spent fuel. In SECY-97-218, NRC staff indicated that the separate inner container provided an additional barrier to the release of plutonium in an accident. NRC continues to believe that a separate inner container provides an additional barrier to the release of plutonium in an accident, just as a package with triple containment would provide an even greater barrier to the release of plutonium in an accident. However, this type of approach is not risk-informed nor performance-based. Consequently, based upon review of the petition, comments on the petition, and research into the regulatory history of the double containment requirement, the NRC agrees that a separate inner container is not necessary for Type B packages containing solid plutonium. NRC believes that the worldwide performance record over 40 years of Type B packages demonstrates that a single containment barrier is adequate. Therefore, the NRC agrees with the petitioner and believes that § 71.63(b) is not technically necessary to provide a reasonable assurance that public health and safety and the environment will be adequately protected during the transportation of plutonium.

While the NRC believes a case can be made for elimination of the separate inner container requirement in § 71.63(b), elimination of the solid form requirement in § 71.63(a) is not as clear. While the same arguments can be made on the obviation of the AEC's basis for originally promulgating § 71.63(a) (i.e., the elimination of reprocessing of plutonium), the same regulatory inconsistency between Type B package standards and the inner containment requirement does not exist for the liquid versus solid form argument. The NRC considers the contents of a package when it is evaluating the adequacy of a packaging's design. The approved content limits and the approved packaging design together define the CoC for a package. However, other than criticality controls and the liquid form requirement of § 71.63(a), Subparts E and F do not contain any restrictions on the contents of a package. Thus, while the inner containment requirement in § 71.63(b) can be seen as conflicting with the Type B package standard because the inner containment affects the packaging's design, the solid form requirement of § 71.63(a) does not conflict with the packaging requirements of the Type B package standard because the solid form requirement affects only the contents of the package, not the packaging itself.

The NRC expects that cost and dose savings would accrue from the removal of § 71.63(b). However, because no shipments of liquid plutonium nitrate are contemplated in the U.S., NRC does not expect cost or dose savings to accrue from the removal of § 71.63(a). Further, the AEC's original bases have been obviated by former President Carter’s administration's decision to not pursue a commercial fuel cycle involving the reprocessing of plutonium.

After weighing this information, the NRC continues to believe that the Type B package standards, when evaluated against 40 years of use worldwide, and millions of safe shipments of Type B packages, together provide reasonable assurance that public health and safety and the environment would be adequately protected during the transportation of radioactive material. The NRC believes that, in this case, the reasonable assurance standard, provided by the Type B package requirements, provides an adequate basis for the public's confidence in the NRC's actions.

NRC Proposed Position: The NRC would adopt, in part, the recommended action of PRM-71-12. Specifically, the NRC would remove the double containment requirement of

§ 71.63(b). However, the NRC would retain the package contents requirement in § 71.63(a). Shipments whose contents contain greater than 0.74 TBq (20 Ci) of plutonium must be made with the contents in solid form.

Affected Sections. 71.63.

Issue 18. Contamination Limits As Applied to Spent Fuel and High Level Waste (HLW) Packages

Background. In the period of December 1997 through April 1998, the French Nuclear Installations Safety Directorate inspected a French nuclear power plant and railway terminal used by the La Hague reprocessing plant. The inspectors noticed that, since the beginning of the 1990's, a high percentage of spent fuel packages and/or railcars had a level of removable surface contamination that exceeded IAEA regulatory limits by as much as a factor of 1000. Subsequent investigations found that the contamination incidents involved shipments from other European countries, and the French transport authorities notified their counterparts of their findings. Subsequently, French, German, Swiss, Belgian, and Dutch spent fuel shipments were temporarily suspended.

After estimating the occupational and public doses from the contamination incidents, the European transport authorities concluded that these incidents did not have any radiological consequence. The contamination was believed to be caused by contact of the spent fuel package surface with contaminated water from the spent fuel storage pool during package handling operations. The authorities concluded that there were deficiencies in the contamination measurement procedures and the distribution of that information.

Media reports on these incidents focused attention on IAEA's regulations for removable contamination on package surfaces. TS-R-1 contains contamination limits for all packages of

4.0 Bq/cm^2 for beta and gamma and low toxicity alpha emitting radionuclides, and 0.4 Bq/cm^2 for all other alpha emitting radionuclides. Although TS-R-1 uses the term limit, IAEA considers these “limits” to be guidance values, or derived values, above which appropriate action should be considered. In cases of contamination above the limit, that action is to decontaminate to below the limits.

The current TS-R-1 limits for removable package surface contamination were derived from a radiological model developed for the 1961 Edition of the IAEA regulations. The exposure pathways considered in the model included external irradiation of the skin, and ingestion and inhalation from resuspension of the contamination in air. The model uses values for the degree to which surface contamination is resuspended in air, making it available for inhalation, and for the number of hours of exposure to the resuspended contamination. The values were chosen to represent occupational conditions at shipper and carrier facilities, in which workers manually handled many packages throughout the year. These exposure conditions are much greater than the public would experience from brief exposure to packages in transport. The values also exceed real occupational resuspension rates and exposure times and were believed to result in worker doses that would be well within the annual occupational dose limit. Exposure at the contamination limit does not pose a significant health hazard to workers. Therefore, members of the public, few of whom would ever be expected to encounter contaminated packages in transit, and then only briefly, are also protected against contamination hazards by the limit.

TS-R-1 further provides that in transport, “...the magnitude of individual doses, the number of persons exposed, and the likelihood of incurring exposure shall be kept as low as reasonable, economic and social factors being taken into account...” The IAEA contamination regulations have been applied to radioactive material packages in international commerce for almost 40 years, and practical experience demonstrates that the regulations can be applied successfully. With respect to Contamination limits, TS-R-1 contains no changes from previous versions of IAEA's regulations.

Part 71 does not contain contamination limits, but § 71.87(i) requires that licensees determine that the level of removable contamination on the external surface of each package offered for transport is as low as is reasonably achievable, and within the limits specified in DOT regulations in 49 CFR 173.443. The DOT contamination limits differ from TS-R-1 in that the contamination limits apply to the wipe material used to survey the surface of the package, not the surface itself. Also, the contamination limits are only 10 percent of the TS-R-1 values (e.g., wipe limit of 0.4 Bq/cm^2 (2200 dpm/100 cm^2) for beta and gamma and low toxicity alpha emitting radionuclides), because the DOT limits are based on the assumption that the wipe removes 10 percent of the surface contamination. In this regard, the DOT and TS-R-1 limits are equivalent.

The DOT contamination regulations contain an additional provision for which there is no counterpart in TS-R-1. Section 173.443(b) provides that, for packages transported as exclusive use (see 49 CFR 173.403 for exclusive use definition) shipments by rail or public highway only, the removable contamination on any package at any time during transport may not exceed 10 times the contamination limits (e.g., wipe contamination of 4 Bq/cm^2 (22,000 dpm/100 cm^2) for beta and gamma and low toxicity alpha emitting radionuclides). In practice, this means that packages transported as exclusive use shipments (this includes spent fuel packages) that meet the contamination limits at shipment departure may have 10 times that contamination upon arrival at the destination. This provision is intended to address a phenomenon known as "cask-weeping," in which surface contamination that is nonremovable at the beginning of a shipment becomes removable during the course of the shipment. Nonremovable contamination is not measurable using wipe surveys and is not subject to the removable contamination limits. At the destination facility, a package exhibiting cask-weeping can exceed the contamination limits by a considerable margin, even though the package met the limits at the originating facility, and was not subjected to any further contamination sources during shipment. Environmental conditions are believed to affect the cask-weeping phenomenon.

Spent fuel packages and shipments differ from those considered in the 1961 model used to develop package surface contamination limits. Workers are exposed to only a few spent fuel packages per year at most, so their exposure time to package contamination is less than that modeled. Unlike the packages in the model, however, spent fuel package surface areas and radiation levels are significant. Exposure to the package radiation level while performing either contamination survey or decontamination activities contributes to worker dose, and this impact was not considered in the model.

The IAEA has plans to establish a Coordinated Research Project (CRP) to review contamination models, approaches to reduce package contamination, strategies to address cask-weeping, and possible recommendations for revisions to the contamination standard that consider risks, costs, and practical experience. IAEA establishes CRPs to facilitate investigation of radioactive material transportation issues by key IAEA Member States. IAEA will then consider a CRP report and any further actions or remedies that may be warranted at periodic meetings (at TRANSSC). NRC informed IAEA that NRC supports the IAEA initiative to establish the CRP and that the NRC would participate in the IAEA reviw of surface contamination standards.

Discussion. During the three public meetings, NRC has received verbal public comments on the contamination issue. One commenter agreed that external contamination on packages of radioactive material in transport is a significant problem and is the source of actual or perceived hazard that can cause damage to the nuclear industry. The commenter would prefer not to change contamination limits (i.e., continuing to use TS-R-1 limits) unless there is a sound technical basis for doing so.

NRC was requested to clarify its discussion of the 4 Bq/cm^2. The commenter stated that the current limit for removable contamination levels in 49 CFR 173.443 is 0.4 Bq/cm^2 before shipment, unless an assessment method with higher efficiency is used, in which case the limit may be as high as 10 times 0.4 Bq/cm^2 (i.e., 4 Bq/cm^2) (22,000 dpm/100 cm^2).

Four commenters stated they understood that existing surface contamination limits (i.e., 4 Bq/cm^2) (2200 dpm/100 cm^2) were intended for small and not large packages and that using the limit for large packages, while it may reduce public exposure rates, would conceivably increase worker exposure rates. Another commenter added that worker exposure could actually increase when double containment is required, and expressed concern about how this issue with contamination limits impacts international shipments. Some commenters stated that it was doubtful that worker exposure rates could be reduced, even if allowable surface contamination rates were significantly increased.

Several commenters addressed the issue that workers would be exposed to radiation while measuring the surface contamination level. Three of the commenters acknowledged that this is true regardless of the level of the package contamination limit. Two commenters suggested that NRC consider other ways to protect workers, including cask design. Another commenter stated that if the radiation is too great for workers to get close enough to measure it, it is too great to transport it.

Absent public objection to the current standard and an overall significantly improved approach, NRC is planning no revisions to Part 71 regarding surface contamination in this proposed rule. The NRC intends to use the information it collects from public comments on this issue to continue to support DOT in U.S. participation in the IAEA CRP and to work with DOT and other IAEA Member States on this issue. Because IAEA has adopted a 2-year revision cycle for TS-R-1, a revision based on the CRP's results could be incorporated into TS-R-1 more quickly than under the previous 10-year revision cycle.

NRC Proposed Position. The NRC proposes no changes to Part 71 for this issue.

Affected Sections. None (not adopted).

Issue 19. Modifications of Event Reporting Requirements

Background. The Commission recently issued a final rule to revise the event reporting requirements in 10 CFR Part 50 (see 65 FR 63769; October 20, 2000). This final rule revised the verbal and written event notification requirements for power reactor licensees in §§ 50.72 and 50.73. In SECY-99-181,[17] NRC staff informed the Commission that public comments on the proposed Part 50 rule had suggested that conforming changes also be made to the event notification requirements in 10 CFR Part 72 (Licensing Requirements for the Independent Storage of Spent Fuel) and 10 CFR Part 73 (Physical Protection of Plants and Material). In response, the Commission directed the NRC staff to study whether conforming changes should be made to Parts 72 and 73. During this study, the NRC also reviewed the Part 71 event reporting requirements in § 71.95, and concluded that similar changes could be made to the Part 71 event reporting requirements.

Discussion. This issue was not included in the Part 71 issues paper (65 FR 44360; July 17, 2000). Therefore, there were no public comments on this issue.

The current regulations in § 71.95 require that a licensee submit a written report to the NRC within 30 days of three events: (1) a significant decrease in the effectiveness of a packaging while it is in use to transport radioactive material; (2) details of any defects with

[17] SECY-99-181, "Proposed Plans and Schedules to Modify Reporting Requirements Other than 10 CFR 50.72 and 50.73 for Power Reactors and Material Licensees," dated July 9, 1999.

safety significance found after first use of the cask; and (3) failure to comply with conditions of the CoC during use.

The NRC has identified three principal concerns with the existing requirements in § 71.95. First, the existing requirements only apply to licensees and not to certificate holders. Second, the existing requirements do not contain any direction on the content of these written reports. Third, inconsistencies existed in reporting time frames as a result of the Commission decision in the October 20, 2000, final rule which reduced the reporting burden on reactor licensees in the Part 50 final rule by changing the time for submittal of written reports from 30 days to 60 days.

With respect to the first concern, NRC believes that events involving a significant reduction in effectiveness of a packaging during its use to transport radioactive material may call into question the design bases for the packaging. Examples of a significant reduction in effectiveness might involve an event that causes a package to exceed the 2-mSv per hour (200-mrem) per hour dose limit or exceed the Type B package requirements of § 71.51. In these cases, the cause of the reduction in effectiveness may be due to a design flaw. Because the certificate holder has the most in-depth understanding of the design basis for a packaging, the NRC believes that it is appropriate for the certificate holder to work with the licensee to jointly determine the root cause(s) for an event that resulted in a significant decrease in packaging effectiveness. Similarly, identification of safety-significant defects after first use of a packaging may reveal flaws with the packaging's basic design. Therefore, the NRC would revise § 71.95 to require that the licensee request certificate holder input before submitting a written report for the criteria in new paragraphs (a)(1) and (a)(2). The licensee would also be required to provide the certificate holder with a copy of the written event report, after the report is submitted to the NRC. This would permit the certificate holder to monitor and trend package performance information arising from package use by multiple licensees. In new paragraph (a)(3), the NRC would retain the existing requirement for licensees to report instances of failure to follow the conditions of the CoC while a packaging was in use.

With respect to the second concern, NRC believes that direction should be provided on the expected contents of these written reports. Currently, no direction is provided to licensees on the form or content of these written reports. The NRC believes that standards for the contents of written reports should be unambiguous. The NRC uses this information to determine if inspection and enforcement follow-up is required for the event or if a generic safety issue exists. Consequently, sufficient information must be provided to the NRC to fulfill its responsibilities to protect public health and safety and the environment. Therefore, NRC would add new paragraphs (c) and (d) to § 71.95 which would provide guidance on the content of these written reports. This new requirement is consistent with the written report requirements for Parts 50 and 72 licensees (i.e., §§ 50.73 and 72.75) and the direction from the Commission in SECY-99-181 to consider conforming event notification requirements to the recent changes made to Part 50. The NRC would also update the submission location for the written reports from the Director, Office of Nuclear Material Safety and Safeguards, to the NRC Document Control Desk. This action is consistent with previous Commission direction to standardize the location for incoming documents and correspondence and would bring Part 71 into greater conformity with Parts 50 and 72. Additionally, the NRC would remove the specific location for submission of written reports from § 71.95(c) and requires that reports be submitted in accordance with § 71.1. This action is also consistent with the approach taken in Parts 50 and 72 and would reduce future NRC burden should the submission address change. This proposed change to § 71.1 is identical to a change made to § 72.4 in a recent Part 72 final rule (see 64 FR 33178; June 22, 1999).

With respect to the third concern, the NRC staff believes that lengthening the period for submitting reports from 30 days to 60 days will reduce the burden on licensees, while still providing the staff with the necessary information to fulfill the NRC's mission. The NRC uses written event reports for trending, analysis, and long-term followup of a licensee's corrective actions. In contrast, immediate reporting of events to the NRC provides indication of significant events when immediate action to protect public health and safety may be required or where the NRC needs timely and accurate information to respond (see 48 FR 39039; August 29, 1983, on the basis for Part 50 event reporting). For transportation events, the NRC receives early notifications in the NRC's Operations Center either from a licensee, when a licensee declares an emergency under its emergency plan — for a transportation event, or from the DOT's National Response Center, when a shipper notifies DOT of an accident involving radioactive material. Consequently, extending the submission time for written event reports to 60 days would not adversely affect the NRC's ability to promptly respond to an event, because these written reports are not used as the basis for immediate or short term actions.

The Commission concluded in the October 20, 2000, final rule revising Part 50 event reporting requirements (65 FR 63769) that the length of time to submit a written report should be extended to permit a thorough evaluation of the event, identification of the root causes, and development of corrective actions. The Commission also indicated that a licensee's submission of written reports should not be unnecessarily delayed to take advantage of the full 60-day period. The NRC took this action because some events required a significant amount of time to evaluate the event, identify the root causes, and identify the corrective actions; and consequently, a supplemental written event report was necessary. In addition, a 60-day period is more consistent with the NRC's desire that the licensee and the certificate holder both be involved in the analysis of an event. The Commission indicated that the licensee's burden, in submitting a supplemental written event report, would be reduced by providing sufficient time to complete the original written event report.

Accordingly, the NRC staff believes the Commission's rationale for lengthening the reporting period from 30 days to 60 days for Part 50 written event reports is also valid for Part 71 written event reports.

The NRC draft RA indicates that adoption of the conforming change to Part 71 for event reporting requirements is appropriate from a safety, regulatory, and cost perspective. Regulatory efficiency within NRC would increase with adoption of this proposed change and would result in greater conformity among Parts 50, 71, and 72. Further, NRC burden (and thus costs) would be reduced should the submission address change in the future. There would be a one-time implementation cost for licensees for revising procedures and for training. A key benefit of the proposed amendments would be a reduction in the recurring annual reporting burden on licensees, as a result of reducing the efforts associated with reporting events of little or no risk or safety significance. It is anticipated that the NRC's recurring annual review efforts for telephone notifications and written reports would not be significantly reduced.

NRC Proposed Position. The NRC proposes a reduction in regulatory burden for licensees by lengthening the § 71.95 event reporting submission period from 30 to 60 days.

Affected Sections. 71.95.

IV. Section-By-Section Analysis

Several sections In Part 71 would be redesignated in this rulemaking to improve consistency and ease of use. For some sections, only the section number would be changed. However, for other sections, revisions would also be made to the regulatory language. The following table is provided to aid the public in understanding the proposed numerical changes to sections of Part 71.

Redesignation Table	
New § number	Old § number
§ 71.8	§ 71.11
§ 71.9	New section
§ 71.10	New section
§ 71.11 (Reserved)	NA
§ 71.12	§ 71.8
§ 71.13	§ 71.9
§ 71.14	§ 71.10
§ 71.15	§ 71. 53
§ 71.16 (Reserved)	NA
§ 71.17	§ 71.12
§ 71.18	New section
§ 71.19	§ 71.13
§ 71.20	§ 71.14
§ 71.21	§ 71.16
§ 71.22	§ 71.18
§ 71.23	§ 71.20
§ 71.24 (Reserved)	§ 71.22 (Section removed)
§ 71.25 (Reserved)	§ 71.24 (Section removed)
§ 71.53 (Reserved)	§ 71.53 (Section redesignated)

Subpart A—General Provisions

10 CFR 71.0 Purpose and scope.

Paragraph (d) would be reformatted into four subparagraphs to simplify this regulation, to better use plain language, and to reflect the existence of the new Type B(DP) package approval process in new Subpart I. Paragraph (d)(1) would indicate that general licenses for which no NRC package approval is required are issued in new §§ 71.20 through 71.23. This is changed from the current sentence, because of the removal of existing §§ 71.22 and 71.24 (redesignated §§ 71.24 and 71.25). A new sentence would be added referring to the requirement for a CoC to be issued for a Type B(DP) package to be used under the new general license in new § 71.18. Paragraph (d)(2) would indicate that an application for package approval — for package types other than Type B(DP) — must be completed in accordance with Subpart D. Paragraph (d)(3) would indicate that an application for a Type B(DP) package must be completed in accordance with Subpart I. Paragraph (d)(4) would continue to require a licensee transporting, or delivering material to a carrier for transport, to meet the requirements of the applicable portions of Subparts A, G, and H.

New paragraph (e) would be added to indicate that persons who hold, or apply for, a Part 71 CoC for Type AF, Type B, Type BF, Type B(U)F, Type B(M)F, and Type B(DP) packages are within the scope of Part 71 regulations.

Existing paragraphs (e) and (f) would be redesignated as new paragraphs (f) and (g), respectively. The rule text in new paragraph (f) would be the same as new paragraph (e) text. New paragraph (g) would be revised to reflect the redesignation of existing § 71.11 as new § 71.8.

10 CFR 71.1 Communications and reports.

In § 71.1, paragraph (a) would be revised to indicate that documents submitted to the NRC should be addressed to the attention of the "NRC Document Control Desk," not the "Director of the Office of Nuclear Material Safety and Safeguards." Provisions would also be added to provide requirements when a due date for a document falls on a Saturday, Sunday, or Federal holiday. In that case, the document would be due the next Federal work day. This change would be identical to a change made to § 72.4 in a recent Part 72 final rule (see 64 FR 33178; June 22, 1999).

10 CFR 71.2 Interpretations.

No changes were made to the text of this section; however, it is included in the revision of this subpart for completeness.

10 CFR 71.3 Requirement for license.

No changes were made to the text of this section; however, it is included in the revision of this subpart for completeness.

10 CFR 71.4 Definitions.

The existing definitions for A_1, *fissile material*, *low specific activity material (LSA)*, *package*, and *transport index (TI)* would be revised as conforming changes. New definitions for A_2; *certificate of compliance*; *criticality safety index (CSI)*; *deuterium*; *DOT*; *graphite*; *spent fuel*; and *structures, systems, and components important to safety* would be added as conforming changes.

The definition of A_1 would be revised to split the current combined definition for A_1 and A_2 into two individual definitions. This approach is consistent with standard in TS-R-1.

Furthermore, no change would be made to the current technical content of the definition for A_1; however, the text would be revised to improve readability.

A definition for A_2 would be added, because the current joint definition for A_1 and A_2 would be split into two definitions. [See also definition for A_1.]

A definition for *certificate of compliance* would be added. This definition would be similar to the definition for the same term found in § 72.3

A definition of *criticality safety index* (CSI) would be added.

A definition of *deuterium* would be added to indicate that, for the purposes of new §§ 71.15 and 71.22, the definition of "deuterium" is found in 10 CFR 110.2 applies.

A definition of *U.S. Department of Transportation (DOT)* would be added.

The definition of *fissile material* would be revised by removing ^{238}Pu from the list of fissile nuclides; clarifying that “fissile material” means the fissile nuclides themselves, not materials containing fissile nuclides; and redesignating the reference to exclusions from fissile material controls from § 71.53 to new § 71.15.

A definition of *graphite* would be added to indicate that, for the purposes of new §§ 71.15 and 71.22, the definition of *nuclear grade graphite* found in § 110.2 applies.

The definition of *low specific activity material* (LSA), for LSA-III material, would be revised to reflect the existence of § 71.77 (§ 71.77 provides requirements on the qualification of LSA-III material).

The definition of *package* would be revised by clarifying in paragraph (1) that *fissile material package* also means a Type AF, Type BF, Type B(U)F, or Type B(M)F package. New paragraph (2) would be added defining *Type A packages* in accordance with DOT regulations contained in 49 CFR Part 173. Existing paragraph (2) defining *Type B packages* would be redesignated as paragraph (3). No changes would be made to the redesignated text. New paragraph (4) would be added defining a *Type B(DP) package*.

A definition of *spent nuclear fuel* or *spent fuel* would be added. This definition is the same as that currently found in § 72.3.

A definition for *structures, systems, and components important to safety* would be added for Type B(DP) packages. This definition would be similar to the definition currently found in § 72.3

The definition for *transport index* (TI) would be revised to reflect the new definition of *criticality safety index*; however, the method for determining the TI of a package, based on the package's radiation dose rate, would remain unchanged.

10 CFR 71.5 Transportation of licensed material.

No changes were made to the text of this section; however, it is included in the revision of this subpart for completeness.

10 CFR 71.6 Information collection requirements: OMB approval.

This section would be redesignated from Subpart B—Exemptions, to Subpart A—General Provisions. Paragraph (b) of this section would be revised as a conforming change to reflect the addition of new information collection requirements in §§ 71.18, 71.151, 71.153, 71.155, 71.157, 71.159, 71.161, 71.165, 71.167, 71.171, 71.173, 71.175, and 71.177. Additionally, the existing information collection requirement in Appendix A to Part 71, Paragraph II, was inadvertently omitted from the list of approved information collection requirements in a previous rulemaking; consequently, NRC staff would add Appendix A, Paragraph II to paragraph (b) to correct this error. Furthermore, § 71.6a would be removed, because no such section currently exists in Part 71.

10 CFR 71.7 Completeness and accuracy of information.

This section would be redesignated from Subpart B—Exemptions, to Subpart A—General Provisions. Further, paragraphs (a) and (b) would be revised by adding the terms "certificate holder" and "applicant for a CoC."

10 CFR 71.8 Deliberate misconduct.

This section would be redesignated from Subpart B—Exemptions, to Subpart A—General Provisions. Further, in Subpart A, § 71.11 would be redesignated as § 71.8. However, the current text of § 71.11 would not be changed in the redesignated § 71.8.

§ 10 CFR 71.9 Employee protection.

This section would be redesignated from Subpart B—Exemptions, to Subpart A—General Provisions. New § 71.9 would be added to provide requirements on employee protection. Currently, requirements relating to the protection of employees against firing or other discrimination when the employee engages in certain "protected activities" are provided under the Parts of Title 10 for which a specific license was issued to possess radioactive material. However, no provisions were provided in Part 71 relating to the protection of employees against firing or other discrimination when employees engage in certain "protected activities" when they are the employees of a certificate holder or applicant for a CoC. The NRC believes these employees should also be afforded the same rights and protection as are currently afforded employees of licensees. The new section would be identical to the existing § 72.10, "Employee protection." In including licensees in the new § 71.9 , the NRC recognizes that the potential for duplication occurs for licensees regulated under multiple 10 CFR Parts. However, the NRC believes that by including licensees along with certificate holders and applicants for a CoC, improved regulatory clarity would be achieved, and any potential confusion would be minimized.

10 CFR 71.10 Public Inspection of Application.

A new section would be added indicating that applications and documents submitted to the Commission in connection with an application for a package approval shall be available for public review in accordance with the provisions of 10 CFR Parts 2 and 9. This new section would be similar to existing § 72.20. Existing § 71.10 would be redesignated § 71.14 with changes to the text.

10 CFR 71.11 (Reserved)

This section would be redesignated from Subpart B-Exemptions, to Subpart A-General Provisions, and would be reserved. Existing § 71.11 would be redesignated as § 71.8.

Subpart B - Exemptions

10 CFR 71.12 Specific exemptions.

Existing § 71.8 would be redesignated as § 71.12. No changes would be made to the contents of this section. Existing § 71.12 would be redesignated as § 71.17, with changes to the text as discussed under § 71.17, below.

10 CFR 71.13 Exemption of physicians.

Existing § 71.9 would be redesignated as § 71.13. No changes would be made to the contents of this section. Existing § 71.13 would be redesignated as § 71.19, with changes to the text as discussed under § 71.19, below.

10 CFR 71.14 Exemption for low-level materials.

Existing § 71.10 would be redesignated as § 71.14. Existing § 71.14 would be redesignated as § 71.20, with no changes to the text.

In new § 71.14. paragraph (a) would be revised by removing the existing single 70 Bq/g (0.002 μCi/g) specific activity value and replacing it with "Activity Concentration for Exempt Material" found in Table A-2 in Appendix A to Part 71. Additionally, paragraph (a) would be reformatted by adding two new paragraphs. Paragraph (a)(1) would provide an exemption for natural radioactive materials and ores.   Paragraph (a)(2) would provide an exemption for radioactive material based on its specific activity, not based on the material being in a package.

Paragraph (b) would be revised to consolidate the exemption provisions for LSA and SCO material. The LSA and SCO exemptions contained in existing paragraphs (b)(2) and (c) of this section would be consolidated into a revised paragraph (b)(3); and existing paragraph (c) would be removed. The reference to material exempt from classification as fissile material would be revised from § 71.53 to § 71.15 , because of the redesignation of the section.

Existing paragraph (b)(3) would be removed. The 0.74 TBq (20 Ci) exemption for special form americium and special form plutonium would be removed. However, the 0.74 TBq (20 Ci) exemption for special for special form plutonium-244, transported in domestic commerce, would be retained as new paragraph (b)(2). Furthermore, an exception would be added to paragraph (b)(1) indicating that paragraph (b)(1) does not apply to a package containing greater than an A_1 quantity of special form plutonium-244 transported in domestic commerce. For international shipments, the A_1 quantity limit for special form plutonium-244 would continue to apply.

10 CFR 71.15 *Exemption from classification as fissile material*.

Existing § 71.11 would be redesignated to § 71.8 . Existing § 71.53 would be redesignated as § 71.15 **,** and relocated to Subpart B with the other Part 71 exemptions. This section would be revised by providing mass-ratio based limits in classifying fissile-exempt material. This approach would remove the concentration- and consignment-based limits of the current § 71.53 and return to package-based mass limits, with required minimum ratios of nonfissile-to-fissile mass.

The title would be changed to "Exemption from classification as fissile material."

New paragraphs (a) and (b) would be added and would allow for increasing quantities of fissile material to be shipped, would provide a concurrent increase in the required mass ratio to ensure criticality safety, and would allow shipment of fissile material in bulk packaging (i.e., large freight containers). The nonfissile material would be limited to noncombustible material which is insoluble in water. In paragraph (a), the fissile mass per package would be limited to 15 grams with a nonfissile-to-fissile mass ratio of 200:1; and the nonfissile material would be restricted to iron. In paragraph (b), the allowed fissile mass is raised to 350 grams per package, but the ratio of nonfissile-to-fissile material is also raised to 2000:1. The mass of any lead, graphite, beryllium, and deuterium in the package cannot be included in determining the nonfissile material mass, and the nonfissile material that is counted in the ratio must be noncombustible and insoluble in water.

Current § 71.53, paragraph (b), would be redesignated as paragraph (c)**,** and would be revised to limit beryllium, graphite, and hydrogenous material enriched in deuterium to less than 0.1 percent of the fissile material mass. The current homogenous distribution and lattice requirements would be removed.

Current § 71.53, paragraph (c), would be redesignated as paragraph (d), and would be reformatted and revised to clarify that the nitrogen to uranium atomic ratio, for shipments of liquid uranyl nitrate, must be greater than or equal to 2.0. A new requirement would be added specifying the use of DOT Type A packaging.

Current § 71.53, paragraph (d), would be redesignated as paragraph (e), and would be reformatted and revised to clarify the mass limits for plutonium. No substantive changes would be made to this paragraph.

10 CFR 71.16 (Reserved)

This section would be redesignated from Subpart C—General Licenses, to Subpart B—Exemptions, and would be reserved. Further, existing § 71.16 would be redesignated as § 71.21. However, the current text of § 71.16 would not be changed in the redesignated § 71.21.

Subpart C—General Licenses

§ 71.17 General license: NRC-approved package.

Existing § 71.12 would be redesignated as § 71.17. Paragraph (a) would be revised as a conforming change to indicate that this general license does not apply to Type B(DP) packages.

Paragraph (c)(3) would be revised using plain language, and to reflect the NRC's requirement to address information submitted to the NRC to the attention of the NRC's Document Control Desk, in accordance with § 71.1.

10 CFR 71.18 General license: NRC-approved Type B(DP) package.

This new section would be added to provide a general license for the transportation of spent fuel in Type B(DP) packages. The structure of this new section would be similar to the existing § 71.12(a) through (d).

10 CFR 71.19 Previously approved package.

Existing § 71.13 would be redesignated as § 71.19. Paragraph (a) would be revised to reflect the current package designators (e.g., B(U)F, B(M)F, AF). Additionally, an expiration date for grandfathering these packages would be established. Paragraph (b) would be updated to remove the LSA packages, as these packages no longer exist. A new paragraph (c) would be added to reflect the type B(U) and B(M) packages that have met the requirements of IAEA Safety Series 6 1985 or 1985 (as amended 1990). Additionally, a date by which fabrication of these packages must be complete would be added. Existing paragraph (c) would be redesignated as paragraph (d). Existing paragraph (d) would be redesignated as paragraph (e), and updated to reflect the identification number suffix of "-96" for previously approved package designs that have been resubmitted for review by the NRC and have been approved, and to remove the package designated as Type A from this paragraph.

10 CFR 71.20 General license: DOT specification container.

Existing § 71.14 would be redesignated as § 71.20. No changes would be made to the contents of this section.

10 CFR 71.21 General license: Use of foreign approved package.

Existing § 71.16 would be redesignated as § 71.21. No changes would be made to the contents of this section.

10 CFR 71.22 General license: Fissile material.

Existing § 71.18 would be redesignated as § 71.22. This section would be amended by consolidating and simplifying the current fissile general license provisions contained in existing §§ 71.18, 71.20, 71.22, and 71.24 into a new § 71.22. The new § 71.22, while retaining some of the provisions of the existing general licenses, would principally use mass-based limits and a CSI. Concentration-based limits would be removed. Exceptions relating to plutonium-beryllium sealed sources in existing §§ 71.18 and 71.22 would be relocated to new § 71.23. The values contained in new Tables 71-1 and 71-2 would be revised from the values contained in the table in existing § 71.22 and in Table 1 in existing § 71.20, respectively; and are based on new minimum critical mass calculations described in NUREG/CR-5342. In some instances**,** the allowable mass limit has been increased from the current limits in existing §§ 71.18, 71.20, 71.22, and 71.24; in other instances**,** the allowable mass limit has been reduced. The values contained in new Tables 71-1 and 71-2 would be used as the variables X, Y, and Z in the equation in paragraph (e).

The title would be revised to indicate that this general license is not restricted to a specific type of fissile material shipment.

Paragraph (a) would be revised to require that fissile material shipped under this general license would be contained in a DOT Type A package. Additionally, while the existing exception from Subparts E and F requirements is maintained, the DOT Type A package regulations of 49 CFR Part 173 would also be specified.

Paragraph (b) would remain unchanged.

Paragraph (c) would be revised to remove the specific gram limits for uranium and plutonium, but would retain the existing Type A quantity limit. Revised gram limits would be relocated to new Table 71-1, which would be associated with new paragraphs (d) and (e). A requirement would also be added to limit the amount of special moderating materials beryllium, graphite, and hydrogenous material enriched in deuterium present in a package to less than 500 g.

Existing paragraph (d) would be removed. Revised gram limits for fissile material mixed with material having a hydrogen density greater than water (i.e., a moderating effectiveness greater than H_2O) would be placed in new Table 71-1. A note would be added to new Table 71-1 to indicate that reduced mass limits apply when more than 15 percent of a mixture of moderating materials contains moderating material with a hydrogen density greater than H_2O.

New paragraph (d) would be added to require that shipments of fissile material packages be labeled with a CSI, that the CSI per package be less than or equal to 10.0, and that the sum of the CSIs in a shipment of multiple fissile material packages comply with the array requirements of § 71.59(c) (i.e., the maximum number of packages on a conveyance would be limited by the sum of the CSIs to less than 50 for a nonexclusive use vehicle and to less than 100 for an exclusive use vehicle).

Existing paragraphs (e) and (f) would be removed.

New paragraph (e) would be added to require that the CSI be calculated via a new equation for any of the fissile nuclides. Guidance on applying the equation and the mass limit input values of Tables 71-1 and 71-2 would also be contained in this paragraph.

10 CFR 71.23 General license: Plutonium-beryllium special form material.

The existing § 71.20, "General license: Fissile material, limited moderator per package," would be removed. A new section on the shipment of plutonium-beryllium (Pu-Be) special-form fissile material (i.e., sealed sources) would be added as a new § 71.23. New § 71.23 would consolidate regulations on shipment of Pu-Be sealed sources contained in existing §§ 71.18 and 71.22 into one location in Part 71 and would use an approach consistent with the revised

§ 71.18. The § 71.23 would reduce the maximum quantity of fissile plutonium Pu-Be sealed sources that could be shipped on a single conveyance through changes in the mass limits and calculation of the CSI. Currently, a Pu-Be sealed source package can contain up to 400 g of fissile plutonium with a CSI = to 10.0. Consequently, the current conveyance limits are 4,000 g per shipment for an exclusive-use vehicle and 2000 g per shipment for a nonexclusive use vehicle. The new § 71.23 would increase the maximum CSI per package from 10 to 100; however, the maximum quantity of plutonium per conveyance (i.e., shipment) would be reduced to 1000 g. The 1000 g per shipment limit and a 240 g of fissile plutonium limit are equivalent to those in new § 71.22(f) (1,000 g per shipment and 200 g of fissile plutonium). The 240 g versus 200 g of fissile plutonium per package is due to the increased confidence that the fissile plutonium within a sealed source capsule would not escape from the capsule during an accident and reconfigure itself into an unfavorable geometry.

New § 71.23 would be titled: “General license: Plutonium-beryllium special form material.”

Paragraph (a) would describe the applicability of this section, exceptions to the requirements of Subparts E and F, and the requirement to ship Pu-Be sealed sources in DOT Type A packages.

Paragraph (b) would require that shipments of Pu-Be sealed sources be made under an NRC-approved QA program.

Paragraph (c) would require a 1,000 g per package and per shipment limit. In addition, plutonium-239 and plutonium-241 may constitute only 240 g of the 1,000 g limit.

Paragraph (d) would require that a CSI be calculated per paragraph (e), and the CSI must be less than 100.0. For shipments of multiple packages, the CSI limits of § 71.59(c) would also apply.

Paragraph (e) would provide an equation to calculate the CSI for Pu-Be sources. This equation would be based upon the 240 g mass limit for fissile nuclide plutonium-239 and plutonium-241 in paragraph (c).

10 CFR 71.24 (Reserved)

10 CFR 71.25 (Reserved)

Existing §§ 71.22 and 71.24 would be redesignated as §§ 71.24 and 71.25. Furthermore, new §§ 71.24 and 71.25 would be removed and reserved.

Subpart D—Application for Package Approval

10 CFR 71.41 Demonstration of Compliance.

Paragraph (a) would be revised to require that a Type B package which contains radioactive contents with activity greater than 10^5 A_2 of any radionuclide must meet the enhanced deep immersion test found in § 71.61. New paragraph (d) would be added to provide special package authorizations.

10 CFR 71.51 Additional Requirements for Type B Packages.

Paragraph (a) would be revised to remove the reference to § 71.52, because the requirements of § 71.52 have expired. Paragraph (d) would be added to require that, for other than Type B(DP) packages, a package which contains radioactive contents with activity greater than 10^5 A_2 of any radionuclide must also meet the enhanced deep immersion test found in § 71.61.

10 CFR *71.53 Fissile material exemptions (Reserved).*

This section would be deleted and reserved, its contents would be moved to § 71.15. Section 71.53 will be reserved.

10 CFR 71.55 General requirements for fissile material packages.

New paragraphs (f) and (g) would be added. Paragraph (f) would specify design and testing for fissile material package design for transport by aircraft, and paragraph (g) would address UF_6 criticality exception from § 71.55(b). Additionally, as a conforming change, paragraph (b) would be updated to support new paragraph (g).

10 CFR 71.59 Standards for arrays of fissile material packages.

Paragraphs (b) and (c) would be revised to use the term CSI (criticality safety index).

Paragraph (b) would be revised by adding two new paragraphs. Paragraph (b)(1) would provide direction on calculating the CSI for a fissile material package. The approach in new paragraph (b)(1) would be the same as existing paragraph (b). Paragraph (b)(2) would provide new direction on calculating the CSI for packages shipped under the general license provisions of new §§ 71.22 and 71.23, through the use of the CSI equations defined in §§ 71.22(d) and 71.23(e). Also, paragraph (b)(2) would indicate that packages shipped under the general license provisions of new § 71.22 or 71.23 should use the CSI determined by those sections, rather than the calculation of § 71.59(b).

Paragraph (c) of this section would be revised to provide direction to licensees when the CSI, as calculated by new §§ 71.22(d), 71.23(e), and 71.59(b), is exactly equal to 10.0, and to use plain language. Subparagraph (1) would be revised by replacing the term "[n]ot in excess of 10," with the term "[l]ess than or equal to 10.0." Paragraph (c)(2) would be revised by replacing the term "[i]n excess of 10," with the term "[g]reater than 10.0." These two changes would provide greater clarity and mathematical consistency between paragraphs (c)(1) and (c)(2).

10 CFR 71.61 Special requirements for Type B packages containing more than 10^5 A_2.

This section would be revised to require an enhanced water immersion test for packages used for radioactive contents with activity greater than 10^5 A_2. The title of this section would also be revised to reflect the scope has been broadened beyond irradiated nuclear fuel.

10 CFR 71.63 Special requirement for plutonium shipments.

The title would be revised to reflect only a single "requirement" rather than multiple requirements.

Paragraph (b) would be removed.

The designation of the remaining text as paragraph (a) would be removed, because only one paragraph would remain. The text of former paragraph (a) would be revised to use plain language. The 0.74-TBq (20-Ci) limit and solid form requirement would be retained.

10 CFR 71.73 Hypothetical accident conditions.

A new paragraph (c)(2) is added to require a crush test for fissile material pakages.

10 CFR 71.88 Air transport of plutonium.

Paragraph (a)(2) would be revised to remove the 70 Bq/g (0.002 μCi/g) specific activity value and substitute activity concentration values for plutonium found in Appendix A, Table A-2, of this part. This revision would be a conforming change to the revision to new § 71.14 to ensure consistent treatment of plutonium between these two sections.

Subpart G—Operating Controls and Procedures

10 CFR 71.91 Records.

As a conforming change to Subpart H, paragraphs (b) and (c) would be redesignated as paragraphs (c) and (d), respectively, and would be revised by adding the terms certificate holder and applicant for a CoC. New paragraph (b) would be added to require a certificate holder to keep records on the model, serial number, and date of manufacture of a packaging. These requirements are similar to the requirements in paragraph (a), though less information is required. No change would be made to paragraph (a).

10 CFR 71.93 Inspection and tests.

As a conforming change to Subpart H, paragraphs (a) and (b) would be revised by adding the terms certificate holder and applicant for a CoC. Paragraph (c) would be revised to require the certificate holder to notify the NRC before it begins fabrication of a packaging that can contain material having a decay heat load in excess of 5 kW or a maximum normal operating pressure of 103 kPa [kilo Pascals] (15 lbf/in^2) gauge. This notification could be for either fabricating a single packaging or the beginning of a campaign for fabricating multiple packagings. This notification would be in accordance with the requirements of § 71.1, rather than to an NRC Regional Administrator. This change in notification location is consistent with current Commission policy and would reduce confusion in identifying the appropriate Regional Administrator when the certificate holder and fabrication location are overseas. Licensees would be removed from this paragraph because the NRC believes that requiring a licensee, who does not own the packaging, to notify the NRC in advance of a packaging fabrication, when the licensee may not use the packaging for years, is inappropriate and an unreasonable burden. The NRC believes that requiring certificate holders and applicants for a CoC to notify the NRC in advance of fabricating a packaging(s) would allow the NRC adequate opportunity to inspect these activities. This change would be similar to the current requirement in § 72.232(d) for Part 72 certificate holders or applicants for a CoC to notify the NRC 45 days before starting the fabrication of the first storage cask under a Part 72 CoC. This action would improve the harmonization between these two regulations in Parts 71 and 72, particularly regarding dual-purpose casks (i.e., casks intended to both store and transport spent fuel).

10 CFR 71.95 Reports.

The existing introductory text and paragraphs (a) and (b) would be combined into a new paragraph (a) which would require a licensee, after requesting the certificate holder's input, to submit a written report to the NRC in certain circumstances. The requirement for the licensee to request input from the certificate holder during development of the written event report would ensure that design deficiency issues have been throughly considered. The licensee would also be required to provide the certificate holder with a copy of the written event report, after the report is submitted to the NRC. This would permit the certificate holder to monitor and trend the package performance information, arising from package use by multiple licensees. Additionally, requirements on timing and submission location for the written repots would be relocated to new paragraph (c). Furthermore, the 30-day reporting requirement would be reduced to a 60-day reporting requirement.

The existing paragraph (c) has been redesignated as paragraph (b) and revised for clarity.

New paragraphs (c) and (d) would be added to provide requirements on the timing, submission location, form, and content of the written reports.

10 CFR 71.100 Criminal penalties.

Section 223 of the Atomic Energy Act of 1954, as amended, [the Act] 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. The Commission stated in a final rule on "Clarification of Statutory Authority for Purposes of Criminal Enforcement" (57 FR 55082; November, 24, 1992), that substantive rules under sections 161b, 161i, or 161o of the Act include those rules that create "duties, obligations, conditions, restrictions, limitations, and prohibitions." For the NRC to consider the possibility of criminal sanctions for willful violation of, attempted violation of, or conspiracy to violate, any substantive regulations, the NRC must have clearly identified to affected parties which regulations in Part 71 are substantive rules. Accordingly, paragraph (b) of this section identifies those Part 71 regulations that the NRC does not consider as substantive regulations. Thus, willful violation of, attempted violation of, or conspiracy to violate any of the regulations listed in paragraph (b) is not subject to possible criminal sanctions.

Paragraph (b) of this section would be revised as a conforming change. The NRC has reviewed new §§ 71.10, 71.151, 71.153, 71.155, 71.157, 71.159, 71.161, 71.163, 71.165, 71.167, and 71.169 and considers that these regulations are not substantive rules. Therefore, new §§ 71.10 and 71.151 through 71.169 would be added to the list of sections in paragraph (b). The NRC reviewed new §§ 71.9, 71.18, 71.23, 71.171, 71.173, 71.175, and 71.177**,** and considers that these regulations are substantive rules. Therefore, these sections would not be added to paragraph (b). Additionally, the NRC has reviewed the existing §§ 71.9, 71.10, and 71.53 and concluded these sections should be recharacterized as substantive rules. Therefore, new §§ 71.13, 71.14, and 71.18 would not be included in paragraph (b).

Additionally, existing §§ 71.52 and 71.53 would be removed from paragraph (b), because these section numbers have been removed from Part 71.

Subpart H—Quality Assurance

10 CFR 71.101 Quality assurance requirements.

Paragraph (a) would be revised by adding two new sentences to the end of the paragraph specifying responsibilities for certificate holders and applicants for a CoC.

Paragraph (b) would be revised to add the terms "certificate holder" and "applicant for a CoC." The second sentence would be revised to provide greater clarity and consistency within Subpart H by referring to "the QA requirement's importance to safety."

Paragraph (c) would be revised by redesignating the existing text as paragraph (c)(1), and new text would be added on submitting QA programs in accordance with the requirements of § 71.1. New paragraph (c)(2) would be added to provide equivalent requirements on the submission of QA programs for certificate holders and applicants for a CoC.

Paragraph (f) would be revised to allow the use of existing NRC-approved Part 71 and Part 72 QA programs, in lieu of submitting a new QA program. Additionally, the terms "certificate holder" and "applicant for a CoC" would be added.

Paragraph (g) would be revised by making a minor change to clarify that § 34.31(b) is located in Chapter I of Title 10 of the Code of Federal Regulations. Additionally, as a conforming change, § 71.12(b) would be redesignated as § 71.17(b).

10 CFR 71.103 Quality assurance organization.

Paragraph (a) would be revised by adding the terms "certificate holder" and "applicant for a CoC." Further, the fourth sentence would be revised to improve clarity and consistency within Subpart H and with Part 72, Subpart G, by referring to "the functions of structures, systems, and components that are important to safety."

10 CFR 71.105 Quality assurance program.

Paragraphs (a) through (d) would be revised by adding the terms "certificate holder" and "applicant for a CoC."

10 CFR 71.107 Package design control.

Paragraph (a) would be revised by adding the terms "certificate holder" and "applicant for a CoC." Further, the last sentence would be revised to improve clarity and consistency within Subpart H by referring to "processes that are essential to the functions of the materials, parts, and components that are important to safety."

Paragraph (b) would be revised by adding the terms "certificate holder" and "applicant for a CoC." Additionally, the last sentence would be revised by replacing the text "[c]hanges in the conditions specified in the package approval require NRC approval...." with "[c]hanges in the conditions specified in the CoC require NRC prior approval...."

10 CFR 71.109 Procurement document control.

This section would be revised by adding the terms "certificate holder" and "applicant for a CoC."

10 CFR 71.111 Instructions, procedures, and drawings.

This section would be revised by adding the terms "certificate holder" and "applicant for a CoC."

10 CFR 71.113 Document control.

This section would be revised by adding the terms "certificate holder" and "applicant for a CoC."

10 CFR 71.115 Control of purchased material, equipment, and services.

Paragraphs (a) through (c) would be revised by adding the terms "certificate holder" and "applicant for a CoC."

10 CFR 71.117 Identification and control of materials, parts, and components.

This section would be revised by adding the terms "certificate holder" and "applicant for a CoC."

10 CFR 71.119 Control of special processes.

This section would be revised by adding the terms "certificate holder" and "applicant for a CoC."

10 CFR 71.121 Internal inspection.

This section would be revised by adding the terms "certificate holder" and "applicant for a CoC."

10 CFR 71.123 Test control.

This section would be revised by adding the terms "certificate holder" and "applicant for a CoC."

10 CFR 71.125 Control of measuring and test equipment.

This section would be revised by adding the terms "certificate holder" and "applicant for a CoC."

10 CFR 71.127 Handling, storage, and shipping control.

This section would be revised by adding the terms "certificate holder" and "applicant for a CoC."

10 CFR 71.129 Inspection, test, and operating status.

Paragraph (a) would be revised by adding the terms "certificate holder" and "applicant for a CoC."

Paragraph (b) would remain unchanged.

10 CFR 71.131 Nonconforming materials, parts, or components.

This section would be revised by adding the terms "certificate holder" and "applicant for a CoC."

10 CFR 71.133 Corrective Action.

This section would be revised by adding the terms "certificate holder" and "applicant for a CoC."

10 CFR 71.135 Quality assurance records.

This section would be revised by adding the terms "certificate holder" and "applicant for a CoC."

10 CFR 71.137 Audits.

This section would be revised by adding the terms "certificate holder" and "applicant for a CoC."

Subpart I—Application for Type B(DP) Package Approval

New Subpart I would be added to provide requirements on the application, review, approval, and amendment of a CoC for a Type B(DP) package. Requirements would also be provided on the submission and periodic updating of a final safety analysis report. Additionally, requirements would be added authorizing a certificate holder to make minor changes to the design of a Type B(DP) package, without prior NRC approval, if certain tests were met. Further, identification would be made of which sections in Part 71 also apply to packages approved under this new subpart.

10 CFR 71.151 Procedures for applying for a Type B(DP) package approval.

This new section would describe the process for submitting an application to the NRC to request approval of a Type B(DP) package design. This section would be similar to § 72.230.

10 CFR 71.153 Contents of application.

This new section would provide requirements on what information must be contained in an application for a Type B(DP) package approval. This section would be similar to § 71.31.

10 CFR 71.155 Package description.

This new section would provide requirements on the description of a Type B(DP) package (both the packaging and its contents) which must be contained in an application for package approval. This section would be similar to § 71.33.

10 CFR 71.157 Package evaluation.

This new section would provide requirements which an application for a Type B(DP) package must demonstrate compliance with (i.e., sections in Subparts E and F). Additionally, because the Type B(DP) package is a fissile material package, the applicant would be required to: (1) determine and provide the number "N" which is used in determining the maximum number of fissile packages on a conveyance; and (2) provide any special controls, precautions, or handling instructions. This section would be similar to § 71.35.

10 CFR 71.159 Quality assurance.

This new section would require~~s~~ a certificate holder to describe the quality assurance program, which meets the requirements of Subpart H of Part 71, that would be used to design, fabricate, test, repair and modify a Type B(DP) package. This section would be similar to § 71.37.

10 CFR 71.161 Requirement for additional information.

This new section would require a certificate holder to provide the Commission any information the NRC requires to determine if a CoC should be modified, suspended, or revoked. This section would be similar to § 71.39.

10 CFR 71.163 Issuance of an NRC certificate of compliance.

This new section would provide direction to the NRC staff on criteria for approving a Type B(DP) CoC. This section would be similar to § 72.238.

10 CFR 71.165 Conditions for package reapproval.

This new section would provide direction to a certificate holder who desires to renew a Type B(DP) CoC or a Part 71 quality assurance program approval. This section would be similar to § 71.38.

10 CFR 71.167 Application to amend a certificate of compliance.

This new section would provide direction to a certificate holder who wishes to amend the CoC for a Type B(DP) package. This section would be similar to § 72.244.

10 CFR 71.169 Issuance of an amendment to a certificate of compliance.

This new section would provide direction to the NRC staff on issuance of an amendment to a Type B(DP) package CoC. This section would be similar to § 72.246.

10 CFR 71.171 Inspections and tests.

This new section would require a certificate holder to permit and to make provisions for NRC inspections at facilities used to design, fabricate, or test a Type B(DP) package. This section would also require a certificate holder to make records available and to perform tests the Commission deems necessary. This section would be similar to § 72.232.

10 CFR 71.173 Recordkeeping and reports.

This new section would provide requirements on submitting reports to the NRC and on maintaining records of fabricated Type B(DP) packages. This section would be similar to § 72.242.

10 CFR 71.175 Changes.

This new section would provide requirements permitting a Part 71 certificate holder to make changes to the design of a Type B(DP) package~~,~~ without prior NRC approval. The certificate holder would be required to periodically submit to the NRC a summary of any changes made under § 71.175. This section would be similar to § 72.48.

10 CFR 71.177 Safety analysis report updating.

This new section would provide requirements for a Type B(DP) certificate holder on: (1) an initial submittal of a final safety analysis report (FSAR) to the NRC~~,~~ **;** (2) submitting periodic updates of the FSAR to the NRC~~,~~ **;** and (3) providing a copy of the updated FSAR to each licensee using the Type B(DP) package. This section would be similar to § 72.248.

Appendix A to Part 71 — Determination of A_1 and A_2

No changes were made in Paragraphs I, III, and V; however, these paragraphs would be included due to revising Appendix A in its entirety.

Paragraph II would be revised to use plain language and would be redesignated as subparagraph II(a). The intent of existing paragraph II would not be changed; however, the reference to existing Table A-2 would be revised as a conforming change to the new Table A-3. New paragraph II(b) would be added to provide direction on determining exempt material activity concentration and exempt consignment activity values when a radionuclide has been identified as a constituent of a proposed shipment, but the individual radionuclide is not listed in Table A-2. Consequently, the structure of paragraphs II(a) and II(b) would be the same. New paragraph II(c) would be added to provide direction to licensees on how to submit requests for Commission prior approval of either A_1 and A_2 values or exempt material activity concentration and exempt consignment activity values, for radionuclides that are not listed in Tables A-1 and A-2, respectively.

Paragraph IV would be revised by adding new paragraphs (e) and (f) to provide equations to use in determining a consolidated exempt material activity concentration and exempt consignment activity values when a shipment contains multiple radionuclides. The existing text describing an alternative method for calculating the A_1 or A_2 value of a mixture would be redesignated as paragraphs (c) and (d). No changes would be made from the existing equations.

APPENDIX A, TABLE A-1 — A_1 and A_2 VALUES FOR RADIONUCLIDES

This Table would be revised to reflect the values from TS-R-1**.**

APPENDIX A, TABLE A-2 — EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES

A new Table A-2 would be added to Appendix A of Part 71. This table would contain the values of Exempt Material Activity Concentrations and Exempt Consignment Activity Limits for selected radionuclides. Table A-2 is referenced in new § 71.14(a)(2)**,** and is used by § 71.14 to determine when concentrations of material are not considered radioactive material, for the purposes of transportation.

APPENDIX A, TABLE A-3 — GENERAL VALUES FOR A_1 AND A_2

The existing Table A-2 would be redesignated as new Table A-3**,** and the values would be revised to reflect the changes from IAEA TS-R-1.

APPENDIX A, TABLE A-4 — ACTIVITY MASS RELATIONSHIPS FOR URANIUM

The existing Table A-3 would be redesignated as new Table A-4. No changes would be made to the values contained in new Table A-4.

V. Criminal Penalties

For the purposes of Section 223 of the Atomic Energy Act (AEA), the Commission is proposing to issue amendments to amend 10 CFR Part 71: §§ 71.XX etc, under one or more of sections 161b, 161i, or 161o of the AEA. Willful violations of the rule would be subject to criminal enforcement.

The following is a list of substantive rule sections being revised or added in this rulemaking: §§ 71.1, 71.3, 71.5, 71.8, 71.9, 71.12, 71.13, 71.14, 71.15, 71.17, 71.18, 71.19, 71.20, 71.21, 71.22, 71.23, 71.61, 71.63, 71.88, 71.91, 71.93, 71.95, 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, 71.137, 71.171, 71.173, 71.175, and 71.177

VI. Issues of Compatibility for Agreement States

Under the "Policy Statement on Adequacy and Compatibility of Agreement State Programs" which became effective on September 3, 1997 (62 FR 46517), NRC program elements (including regulations) are placed into four compatibility categories. In addition, NRC program elements also are identified as having particular health and safety significance or as being reserved solely to the NRC. Compatibility Category A are those program elements that are basic radiation protection standards and scientific terms and definitions that are necessary to understand radiation protection concepts. An Agreement State should adopt Category A program elements in an essentially identical manner in order to provide uniformity in the regulation of agreement material on a nationwide basis. Compatibility Category B are those program elements that apply to activities that have direct and significant effects in multiple jurisdictions. An Agreement State should adopt Category B program elements in an essentially identical manner. Compatibility Category C are those program elements that do not meet the criteria of Category A or B, but the essential objectives of which an Agreement State should adopt to avoid conflict, duplication, gaps, or other conditions that would jeopardize an orderly pattern in the regulation of agreement material on a nationwide basis. An Agreement State should adopt the essential objectives of the Category C program elements. Compatibility Category D are those program elements that do not meet any of the criteria of Category A, B, or C, above, and, thus, do not need to be adopted by Agreement States for purposes of compatibility. A bracket around a category means that the section may have been adopted elsewhere and it is not necessary to adopt it again. Health and Safety (H&S) are program elements that are not required for compatibility (i.e., Category D), but are identified as having a particular health and safety role (i.e., adequacy) in the regulation of agreement material within the State. Although not required for compatibility, the State should adopt program elements in this category based on those of NRC that embody the essential objectives of the NRC program elements because of particular health and safety considerations. Compatibility Category NRC are those program elements that address areas of regulation that cannot be relinquished to Agreement States pursuant to the Atomic Energy Act, as amended, or provisions of Title 10 of the Code of Federal Regulations. These program elements should not be adopted by

Agreement States. The following table lists the Part 71 revisions and their corresponding categorization under the "Policy Statement on Adequacy and Compatibility of Agreement State Programs."

Part 71 - PACKAGING AND TRANSPORTATION OF RADIOACTIVE MATERIAL

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
§ 71.0	Purpose and Scope	D	This provision does not meet any of the criteria for designations Category A, B, C, or health and safety. Thus, it does not need to be adopted by Agreement States.
§ 71.1	Communications and Records	D	This provision does not meet any of the criteria for designations Category A, B, C, or health and safety. Thus, it does not need to be adopted by Agreement States.
§ 71.2	Interpretations	D	This provision does not meet any of the criteria for designations Category A, B, C, or health and safety. Thus, it does not need to be adopted by Agreement States.
§ 71.3.	Requirements for license	D	This provision does not meet any of the criteria for designations Category A, B, C, or health and safety. Thus, it does not need to be adopted by Agreement States.

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
§ 71.4	Definitions		
	A_1	[B]	This definition is designated Compatibility Category B because it applies to activities that have direct and significant effects in multiple jurisdictions. An Agreement State should adopt Category B program elements in an essentially identical manner. The bracket, “B,” indicates that if a State has adopted this definition in another portion of its regulations, such as the State’s DOT regulations, then the adoption of this definition is not necessary.
	A_2	[B]	This definition is designated Compatibility Category B because it applies to activities that have direct and significant effects in multiple jurisdictions. An Agreement State should adopt Category B program elements in an essentially identical manner. The bracket, “B,” indicates that if a State has adopted this definition in another portion of its regulations, such as the State’s DOT regulations, then the adoption of this definition is not necessary.

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
	Certificate of compliance (CoC)	D	This definition is not required for compatibility since it defines a term which pertains to an area reserved to NRC. A State may adopt this definition for purposes of clarity or communication. This definition can be adopted by Agreement States since it in and of itself does not convey any authority whereby a State can regulate in an exclusive NRC jurisdiction. However, if a State chooses to define the term then the definition should be essentially identical. In addition, this term does not meet any of the criteria of the Category A, B, C or health and safety and this term is widely accepted as an area of sole responsibility of the NRC.
	Criticality safety Index	B	This definition is designated Compatibility Category B because it applies to activities that have direct and significant effects in multiple jurisdictions. An Agreement State should adopt Category B program elements in an essentially identical manner. In addition, this definition is needed for a common understanding beyond a plain dictionary meaning of the term in order to implement 10 CFR § 71.22 and § 71.23.

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
	Deuterium	B	This definition is designated Compatibility Category B because it applies to activities that have direct and significant effects in multiple jurisdictions. An Agreement State should adopt Category B program elements in an essentially identical manner. In addition, this definition is needed for a common understanding beyond a plain dictionary meaning of the term in order to implement § 71.15.
	DOT	D	This term does not meet any of the criteria of the Category A, B, C or health and safety because it is a widely accepted abbreviation for the U. S. Department of Transportation.
	Fissile material	[B]	This definition is designated Compatibility Category B because it applies to activities that have direct and significant effects in multiple jurisdictions. An Agreement State should adopt Category B program elements in an essentially identical manner. The bracket, “B,” indicates that if a State has adopted this definition in another portion of its regulations, such as the State’s DOT regulations, then the adoption of this definition is not necessary.
	Graphite	B	This definition is needed for a common understanding beyond a plain dictionary meaning of the term in order to implement § 71.15, which has direct and significant transboundary effects.

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
	High-level radioactive waste or HLW	D	This definition is not required for compatibility since it defines a term which pertains to an area reserved to NRC. A State may adopt this definition for purposes of clarity or communication. This definition can be adopted by Agreement States since it in and of itself does not convey any authority whereby a State can regulate in an exclusive NRC jurisdiction. However, if a State chooses to define the term then the definition should be essentially identical.
	Low Specific Activity (LSA) material	[B]	This definition is designated Compatibility Category B because it applies to activities that have direct and significant effects in multiple jurisdictions. An Agreement State should adopt Category B program elements in an essentially identical manner. The bracket, "B," indicates that if a State has adopted this definition in another portion of its regulations, such as the State's DOT regulations, then the adoption of this definition is not necessary.
	Package	[B]	This definition is designated Compatibility Category B because it applies to activities that have direct and significant effects in multiple jurisdictions. An Agreement State should adopt Category B program elements in an essentially identical manner. The bracket, "B," indicates that if a State has adopted this definition in another portion of its regulations, such as the State's DOT regulations, then the adoption of this definition is not necessary.

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
	Spent Nuclear Fuel or Spent Fuel	D	This definition is not required for compatibility since it defines a term which pertains to an area reserved to NRC. A State may adopt this definition for purposes of clarity or communication. This definition can be adopted by Agreement States since it in and of itself does not convey any authority whereby a State can regulate in an exclusive NRC jurisdiction. However, if a State chooses to define the term then the definition should be essentially identical.
	Structures, systems, and components important to safety (SSCs)	D	This definition is not required for compatibility since it defines a term which pertains to an area reserved to NRC. A State may adopt this definition for purposes of clarity or communication. This definition can be adopted by Agreement States since it in and of itself does not convey any authority whereby a State can regulate in an exclusive NRC jurisdiction. However, if a State chooses to define the term then the definition should be essentially identical.
	Transport Index	[B]	This definition is designated Compatibility Category B because it applies to activities that have direct and significant effects in multiple jurisdictions. An Agreement State should adopt Category B program elements in an essentially identical manner. The bracket, "B," indicates that if a State has adopted this definition in another portion of its regulations, such as the State's DOT regulations, then the adoption of this definition is not necessary.

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
§ 71.5	Transportation of Licensed Material	B	This provision is designated Compatibility Category B because it applies to activities that have direct and significant effects in multiple jurisdictions. An Agreement State should adopt Category B program elements in an essentially identical manner.
§ 71.6	Information collection requirements: OMB approval	D	This provision does not meet any of the criteria for designations Category A, B, C, or health and safety. Thus, it does not need to be adopted by Agreement States.
§ 71.7	Completeness and accuracy of Information	D	This provision does not meet any of the criteria for designations Category A, B, C, or health and safety. Thus, it does not need to be adopted by Agreement States.
§ 71.8	Deliberate misconduct	C	The Commission determined in response to SECY-97-156 that Agreement States should adopt the essential objectives of this provision. If deliberate misconduct and wrongdoing issues involving Agreement State licensees were not pursued and closed by Agreement States, then a potential gap may be created between NRC and Agreement State programs.
§ 71.9	Employee Protection	D	This provision does not meet any of the criteria for designations Category A, B, C, or health and safety. Thus, it does not need to be adopted by Agreement States.
§ 71.10	Public Inspection of Application	D	This provision does not meet any of the criteria for designations Category A, B, C, or health and safety. Thus, it does not need to be adopted by Agreement States.

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
§ 71.14	Exemptions for low level material	B- paragraph (a) NRC- paragraphs (b) and (c)	Paragraph (a) is designated as a compatibility Category B because of it significant transboundary impacts with respect to the implementation of the "Exempt Activity Concentration Values," for individual radionuclides in Appendix A, which is designated as a compatibility category B. Paragraphs (b) and (c) are designated compatibility category "NRC." This provision is reserved to the NRC because it delineates NRC's authority from that of DOT's in the area of transportation of radioactive materials. These provisions relinquish to DOT the control of types of shipment that are of low risk both from radiation and criticality standpoints. Further, to ensure that only low criticality risk shipments are included in the area of DOT authority, these provisions restrict the exemption to Type A and low-specific-activity (LSA) or surface contaminated objects (SCOs) that either contain no fissile material or satisfy the fissile material exemption requirements in § 71.15. Finally, this provision is reserved to the NRC because this exemption does not relieve licensees from DOT requirements by reason of NRC's authority. Thus, Agreement States should not adopt this provision in order to retain their ability to implement all of 49 CFR as directed by DOT.

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
§ 71.15	Exemptions from classification as fissile material	[B]	This provision is designated Compatibility Category B because it applies to activities that have direct and significant effects in multiple jurisdictions. An Agreement State should adopt Category B program elements in an essentially identical manner. The bracket, "B," indicates that if a State has adopted this definition in another portion of its regulations, such as the State's DOT regulations, then the adoption of this definition is not necessary.
§ 71.17	General license: NRC-approved package	B	This provision is designated Compatibility Category B because it applies to activities that have direct and significant effects in multiple jurisdictions. An Agreement State should adopt Category B program elements in an essentially identical manner.
§ 71.18	General license: NRC-approved Type B(DP) package	NRC	This provision is reserved to the NRC because it addresses packages intended for both the storage and transportation of spent fuel.

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
§ 71.19	Previously approved package	NRC	This provision is reserved to the NRC because it addresses packages intended for both the storage and transportation of spent fuel.
§ 71.22	General license: Fissile material	[B]	§ 71.22 was previously entitled, "General license: Fissile material, limited quantity, controlled shipment." It was designated a Compatibility Category D. As a part of this amendment, this section was removed. This provision is designated Compatibility Category B because it applies to activities that have direct and significant effects in multiple jurisdictions. An Agreement State should adopt Category B program elements in an essentially identical manner. The bracket, "B," indicates that if a State has adopted this definition in another portion of its regulations, such as the State's DOT regulations, then the adoption of this definition is not necessary.
§ 71.23	General license: Plutonium-beryllium special form material	[B]	This provision is designated Compatibility Category B because it applies to activities that have direct and significant effects in multiple jurisdictions. An Agreement State should adopt Category B program elements in an essentially identical manner. The bracket, "B," indicates that if a State has adopted this definition in another portion of its regulations, such as the State's DOT regulations, then the adoption of this definition is not necessary.

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
§ 71.24	[RESERVED]		§ 71.24 was previously entitled, "General license: Fissile material, limited moderator, controlled shipment." It was designated a Compatibility Category NRC. As a part of this amendment, this section was removed.
§ 71.25	[RESERVED]		§ 71.25 is a new section that is reserved.
§71.41	Demonstration of Compliance	NRC	This provision is designated NRC because it addresses an area reserved to NRC's regulatory authority.
§ 71.51	Additional requirements for Type B packages	NRC	This provision is designated NRC because it addresses an area reserved to NRC's regulatory authority, which is the approval of Type B packages.
§ 71.53	[RESERVED]		§ 71.53 was previously entitled, "Fissile material exemptions." It was designated a Compatibility Category NRC. As a part of this amendment, the provision was removed.
§ 71.55	General requirements for fissile material packages	NRC	This provision is designated NRC because it addresses an area reserved to NRC's regulatory authority.
§ 71.59	Standards for arrays of fissile material packages	NRC	This provision is designated NRC because it addresses an area reserved to NRC's regulatory authority.
§ 71.61	Special requirements for Type B packages containing more than $10^5 A_2$	NRC	This provision is designated NRC because it addresses an area reserved to NRC's regulatory authority.

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
§ 71.63	Special requirements for plutonium shipments	NRC	This provision is designated NRC because it addresses an area reserved to NRC's regulatory authority.
§ 71.73	Hypothetical accident conditions	NRC	This provision is designated NRC because it addresses an area reserved to NRC's regulatory authority.
§ 71.88	Air transport of plutonium	B	This provision is designated Compatibility Category B because it applies to activities that have direct and significant effects in multiple jurisdictions. An Agreement State should adopt Category B program elements in an essentially identical manner.
§ 71.91	Records	D	This provision does not meet any of the criteria for designations Category A, B, C, or health and safety. Thus, it does not need to be adopted by Agreement States.
§ 71.93	Inspection and tests	D	This provision does not meet any of the criteria for designations Category A, B, C, or health and safety. Thus, it does not need to be adopted by Agreement States.
§ 71.95	Reports	D	This provision does not meet any of the criteria for designations Category A, B, C, or health and safety. Thus, it does not need to be adopted by Agreement States.
71.100	Criminal penalties	D	This provision does not meet any of the criteria for designations Category A, B, C, or health and safety. Thus, it does not need to be adopted by Agreement States.

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
§ 71.101	Quality assurance requirements	D- Paragraphs (a), (b), (c)(1) and (f) are designated D's for those States which have no licensees that use Type B packages. C- Paragraphs (a), (b) and (c)(1) are designated C's for those States which have licensees that use Type B packages. D- paragraph (f) C- paragraph (g) NRC- paragraph (c)(2), (d) and (e)	Paragraphs (a), (b), and (c)(1) are designated Category C and the essential objectives of these provisions should be adopted by those Agreement States which have licensees who use Type B packages. These provisions are designated Category C's because the quality assurance of Type B packages is an activity that is needed in order to avoid a nationwide regulatory gap in the regulation of the transportation of radioactive materials. If these provisions are not adopted, this could result in undesirable consequences in multiple jurisdictions. The essential objective of paragraph (a) is that each licensee who uses a Type B package is responsible for the quality assurance requirements which apply to the use of a package. The essential objective of paragraph (b) is that each licensee who uses a Type B package shall establish, maintain and execute a quality assurance program. The essential objective of paragraph (c)(1) is that each licensee who uses a Type B package shall prior to the use of any package for the shipment of any material subject to this part, shall obtain approval of its quality assurance program by the regulatory agency. Paragraph (f) is not required for compatibility because the States have the flexibility to determine whether they wish to accept a previously approved quality assurance program.

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
§ 71.103	Quality assurance organization	D- for those States which have no licensees that use Type B packages. [C]- Paragraph (a) is designated [C] for those States which have licensees that use Type B packages. C-Paragraph (b) is designated C for those States which have licensees that use Type B packages. D- paragraphs (d), (e), and (f)	For paragraph (a), those States which have licenses that use Type B packages, and have adopted the essential objectives of §71.101(a), it is not necessary for them to adopt this provision again. Paragraphs (b) is designated as a Category C and the essential objectives of these provisions should be adopted by those Agreement States which have licensees who use Type B packages. This provision is designated Category C because the quality assurance of Type B packages is an activity that is needed in order to avoid a nationwide regulatory gap in the regulation of the transportation of radioactive materials. If these provisions are not adopted, this could result in undesirable consequences in multiple jurisdictions. The essential objective of paragraph (b) is that each licensee who uses a Type B package should verify by procedures such as checking, auditing, and inspection, that activities affecting the safety-related functions have been performed correctly.

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
§ 71.105	Quality assurance program	D- for those States which have no licensees that use Type B packages C- Paragraphs (a) and (b) are designated as C for those States which have licensees that use Type B packages. D- paragraph (c)	Paragraphs (a) and (b) are designated Category C's for those States which have licensees that use Type B packages. These provisions are designated Category C's because the quality assurance of Type B packages is an activity that is needed in order to avoid a nationwide regulatory gap in the regulation of the transportation of radioactive materials. If these provisions are not adopted, this could result in undesirable consequences in multiple jurisdictions. The essential objectives of paragraph (a) are that each licensee who uses a Type B package 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, and shall identify the material and components covered by the quality assurance program. The essential objective of paragraph (b) is that each licensee who uses a Type B package shall through its quality assurance program provide control over activities affecting the quality of the identified materials and components to an extent to assure that Type B packages are shipped and maintained in accordance with the certificate of compliance or other approval.

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
§ 71.107	Package design control	NRC	This provision is reserved to the NRC because it addresses the design, fabrication, modification, and approval of Type B packages.
§ 71.109	Procurement document control	NRC	This provision is reserved to the NRC because it addresses the design, fabrication, modification, and approval of Type B packages.
§ 71.111	Instructions, procedures, and drawings	NRC	This provision is reserved to the NRC because it addresses the design, fabrication, modification, and approval of Type B packages.
§ 71.113	Document control	NRC	This provision is reserved to the NRC because it addresses the design, fabrication, modification, and approval of Type B packages.
§ 71.115	Control of purchased material, equipment, and services	NRC	This provision is reserved to the NRC because it addresses the design, fabrication, modification, and approval of Type B packages.
§ 71.117	Identification and control of materials, parts, and components	NRC	This provision is reserved to the NRC because it addresses the design, fabrication, modification, and approval of Type B packages.
§ 71.119	Control of special processes	NRC	This provision is reserved to the NRC because it addresses the design, fabrication, modification, and approval of Type B packages.
§ 71.121	Internal Inspection	NRC	This provision is reserved to the NRC because it addresses the design, fabrication, modification, and approval of Type B packages.
§ 71.123	Test control	NRC	This provision is reserved to the NRC because it addresses the design, fabrication, modification, and approval of Type B packages.

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
§ 71.125	Control of measuring and test equipment	NRC	This provision is reserved to the NRC because it addresses the design, fabrication, modification, and approval of Type B packages.
§ 71.127	Handling, storage, and shipping control	D- for those States which have no licensees that use Type B packages [C]- for those States which have licensees that use Type B packages	For those States which have licensees that use Type B packages~~,~~ and have adopted the essential objectives of §71.105 (b), it is not necessary for them to adopt this provision again.
§ 71.129	Inspection, test, and operating status	D	This provision does not meet any of the criteria for designations Category A, B, C, or health and safety. Thus, it does not need to be adopted by Agreement States.
§ 71.131	Nonconforming materials, parts, or components	D	This provision does not meet any of the criteria for designations Category A, B, C, or health and safety. Thus, it does not need to be adopted by Agreement States.

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
§71.133	Corrective action	D- for those States which have no licensees that use Type B packages C- for those States which have licensees that use Type B packages	This provision is designated Category C for those States which have licensees that use Type B packages. This provision is designated Category C because the quality assurance of Type B packages is an activity that is needed in order to avoid a nationwide regulatory gap in the regulation of the transportation of radioactive materials. If this provision is not adopted, this could result in undesirable consequences in multiple jurisdictions. The essential objective of this provision is that each licensee who uses a Type B package 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.
§71.135	Quality assurance records	D- for those States which have no licensees that use Type B packages C- for those States which have licensees that use Type B packages.	This provision is designated a Category C for those States which have licensees that use Type B packages. This provision is designated Category C because the quality assurance of Type B packages is an activity that is needed in order to avoid a nationwide regulatory gap in the regulation of the transportation of radioactive materials. If this provision is not adopted, this could result in undesirable consequences in multiple jurisdictions. The essential objective of this provision is that each licensee who uses a Type B package shall maintain sufficient written records to demonstrate compliance with the quality assurance program.

REGULATION SECTION	SECTION TITLE	COMPATIBILITY CATEGORY	COMMENTS
§71.137	Audits	D- for those States which have no licensees that use Type B packages C- for those States which have licensees that use Type B packages.	This provision is designated a Category C for those States which have licensees that use Type B packages. This provision is designated Category C because the quality assurance of Type B packages is an activity that is needed in order to avoid a nationwide regulatory gap in the regulation of the transportation of radioactive materials. If this provision is not adopted, this could result in undesirable consequences in multiple jurisdictions. The essential objective of this provision is that each licensee who uses a Type B package shall carry out a 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, and the audits must be performed by appropriately trained personnel.
§71.151 through §71.177	Subpart I - Type B(DP) Package Approval	NRC	Subpart I is designated Category NRC because it addresses Type B (DP) package approval, an area reserved to NRC's regulatory authority.
Appendix A	Determination of A1 and A2	B	This provision is designated a Category B because it applies to activities that have direct and significant effects in multiple jurisdictions.

VII. Plain Language

The Presidential Memorandum dated June 1, 1998, entitled, “Plain Language in Government Writing,” directed that the Federal government’s writing be in plain language. This memorandum was published June 10, 1998 (63 FR 31883). In complying with this directive, editorial changes have been made in the proposed revision to improve the organization and readability of the existing language of paragraphs being revised. These types of changes are not discussed further in this document. The NRC requests comments on the proposed rule specifically with respect to the clarity and effectiveness of the language used. Comments should be sent to the address listed under the “ADDRESSES” heading.

VIII. Voluntary Consensus Standards

The National Technology Transfer and Advancement Act of 1995, Pub. L. 104-113, requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standard bodies unless the use of such a standard is inconsistent with applicable law or otherwise impractical. In this proposed rule, the NRC is presenting amendments to its transportation regulations that would make them compatible with the IAEA transportation standards. This action does not constitute the establishment of a standard that establishes generally-applicable requirements.

IX. Environmental Assessment: Finding of No Significant Environmental Impact

The Commission has prepared a draft environmental assessment entitled: Draft Environmental Assessment (EA) of Major Revision of 10 CFR Part 71, February 2001, on this proposed regulation. The draft EA is available on the NRC rulemaking website, also available for inspection in the NRC Public Document Room, 11555 Rockville Pike, Room 0-1F21, Rockville, MD. The Commission requests public comments on the draft EA. Comments on the draft EA may be submitted to the NRC as indicated under the ADDRESSES heading. The following is a brief summary of the draft EA.

The EA grouped the proposed action into 19 different changes to Part 71, which could be adopted either all together as one list or independently in a partial list. Of these 19 changes, the following four meet the NRC's categorical exclusion criteria:

- Changes to Various Definitions (issue 9);
- Expansion of Part 71 Quality Assurance Requirements to Certificate of Compliance (CoC) Holders (issue 13);
- Change Authority for Dual-Purpose Package Certificate Holders (issue 15); and
- Modifications of Event Reporting Requirements (issue 19).

None of the remaining 15 changes are expected to cause a significant impact to human health, safety, or the environment, whether promulgated altogether or individually. In fact, most of the changes would have negligible effects or result in slight improvements in health, safety, and environmental protection. In particular, the following changes are primarily administrative in nature, would not cause any new negative impacts, and would result in the beneficial effect of simplifying and/or harmonizing the NRC's regulations with TS-R-1:

- Changing Part 71 to the International System of Units (SI) Only (Issue 1);

- Revision of A_1 and A_2 (issue 3);
- A new requirement to display the Criticality Safety Index on shipping packages of fissile material (Issue 5);
- A provision to “grandfather” older shipping packages under the Part 71 requirements in existence when their Certificates of Compliance were issued (issue 8); and
- Procedures for approval of special arrangements for shipment of special packages (issue 12).

The following changes would result in slight net improvements in health, safety, and environmental protection:

- Addition of uranium hexafluoride package requirements (issue 4);
- Strengthening the requirements in § 71.61 to ensure package containment in deep submersion scenarios (issue 7);
- Adoption of the crush test for fissile material package design (issue 10);
- Adoption of fissile material package design requirements for transport by aircraft (issue 11); and
- Adoption of the ASME Code for spent fuel transportation casks (issue 14).

The proposal to change the existing 70 Bq/g (0.002 μCi/g) level to radionuclide-specific activity limits (issue 2) is expected to have mixed, although overall minor, effects. For radionuclides with new exemption values that are lower than the current limit, there could be a decrease in the number of exempted shipments and a commensurate slight increase in the level of protection. For radionuclides with new exemption values that are higher than the current limit, there could be an increase in the number of exempted shipments and a commensurate slight increase in associated radiation exposures. However, IAEA and the NRC have determined that this change would not significantly increase the risk to individuals.

The addition of the Type C package and low level dispersible material concepts (issue 6) would result in mixed, although overall minor, effects. If the same number of packages are handled, the radiation doses to workers loading and unloading Type C packages shipped by air will be slightly higher than the doses to workers loading and unloading other kinds of packages shipped by other means. At the same time, “incident-free” doses during the shipping of Type C packages are expected to be slightly reduced compared to baseline conditions, while the risks associated with accidents during shipping could be slightly increased or decreased depending on the shipping scenario.

Changes to transportation regulations for fissile materials actually consist of 17 individual recommendations for revisions to Part 71 (issue 16). Ten of these recommendations are expected to result in no impact, as they simply clarify definitions, consolidate related requirements into single sections, or streamline the regulations. Four of the recommendations will result in small improvements to health, safety, and environmental protection by eliminating confusion among licensees and/or providing added assurance for critical safety. The last two recommendations, which would revise exemptions for low-level material and remove or modify provisions related to the shipment of Pu-Be neutron sources, are expected to significantly improve criticality safety.

Changes to the requirements for plutonium shipments in § 71.63 (PRM-71-12) could result in a slight increase in the probability and consequences of accidental releases, primarily when and if plutonium is shipped in liquid form. However, most plutonium shipments are either related to the disposition of plutonium wastes or to the production of mixed oxides, neither of which involve the shipment of a liquid solution of plutonium.

No changes have been identified for the issue related to surface contamination limits as applied to spent fuel and high level waste (issue 18). The issue was included in the proposed rule in response to Commission direction in SRM-SECY-00-0117. NRC is seeking input on whether the Agency should address this issue in future rulemaking activities. As a result, no regulatory options were developed, and therefore no environmental assessment conducted.

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[18], 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, the NRC 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 assessment of the impacts associated with this rulemaking are evaluated in “Environmental Assessment of Major Revision to Packaging and Transportation of Radioactive Material Regulations (10 CFR Part 71),'' dated February 2000.

NUREG-0170 established the nonaccident 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 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

[18] 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, 11555 Rockville Pike, Room 0-1F21, Rockville, MD.

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, nonaccident related radiation exposures and accident related impacts associated with this rulemaking would not change from the impact of the current Part 71 requirements (i.e., no increase or decrease). Nonradiological traffic injuries and nonradiological traffic deaths would not change. 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, 11555 Rockville Pike, Room 0-1F21, Rockville, MD. The environmental assessment is also available on the NRC rulemaking website.

X. Paperwork Reduction Act Statement

This proposed rule (or proposed policy statement) amends information collection requirements that are subject to the requirements of the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). This rule (or policy statement) has been submitted to the Office of Management and Budget for review and approval of the paperwork requirements.

The public reporting burden for this information collection is estimated to average ____ hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the information collection. The U.S. Nuclear regulatory Commission is seeking public comment on the potential impact of the information collections contained in the proposed rule (or proposed policy statement) and on the following issues:

1. Is the proposed information collection necessary for the proper performance of the functions of the NRC, including whether the information will have practical utility?
2. Is the estimate of burden accurate?
3. Is there a way to enhance the quality, utility, and clarity of the information to be collected?
4. How can the burden of the information collection be minimized, including the use of automated collection techniques?

Send comments on any aspect of this proposed information collection, including suggestions for reducing the burden, to the Records Management Branch (T-6E6), U.S. Nuclear Regulatory Commission, Washington DC 20555-0001, or by Internet electronic mail at BJS1@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0008), Office of Management and Budget, Washington, DC 20503.

Comments to OMB on the information collections or on the above issues should be submitted by (insert date 30 days after publication in the Federal Register). Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given to comments received after this date.

XI. Public Protection Notification

If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and person is not required to respond to, the information collection.

XII. Regulatory Analysis

The Commission has prepared a draft regulatory analysis entitled: “Draft Regulatory Analysis of Major Revision of 10 CFR Part 71 - Proposed Rule”. To support the discussions of the proposed changes, selected material from this regulatory analysis has been included earlier under each issue. The analysis examines the costs and benefits of the alternatives considered by the Commission. The draft analysis is available on the NRC rulemaking website, also available for inspection at the NRC Public Document Room, 11555 Rockville Pike, Room 0-1F21, Rockville, MD. The Commission requests public comments on the draft regulatory analysis. Comments on the draft analysis may be submitted to the NRC as indicated under the ADDRESSES heading.

XIII. 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 will not, if promulgated, have a significant economic impact on a substantial number of small entities. This proposed 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 (§ 2.810).

XIV. Backfit Analysis

The NRC has determined that a backfit analysis is not required for this proposed rule because these amendments do not involve any provisions that would require backfits as defined in § 50.109(a)(1).

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. 553, the Commission is proposing to revise 10 CFR Part 71 as follows:

PART 71 -- PACKAGING AND TRANSPORTATION OF RADIOACTIVE MATERIAL

1. The authority citation for Part 71 continues to read as follows:

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.

2. Subparts A, B, and C to Part 71 are revised to read as follows:

Sec.

71.0 Purpose.

71.1 Communications and records.

71.2 Interpretations.

71.3 Requirement for license.

71.4 Definitions.

71.5 Transportation of licensed material.

71.6 Information collection requirements: OMB approval.

71.7 Completeness and accuracy of information.

71.8 Deliberate misconduct.

71.9 Employee protection.

71.10 Public inspection of application.

71.11 [Reserved]

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[19]) 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)(1) Exemptions from the requirement for license in § 71.3 are specified in § 71.14. General licenses for which no NRC package approval is required are issued in §§ 71.20 through 71.23. The general license in § 71.17 requires that an NRC certificate of compliance or other package approval be issued for the package to be used under this general license. The general license in § 71.18 requires that an NRC certificate of compliance or other package approval be issued for the Type B(DP) package to be used under this general license.

(2) Application for package approval, other than Type B(DP) packages, must be completed in accordance with subpart D of this part, demonstrating that the design of the

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[19] Postal Service manual (Domestic Mail Manual), Section 124.3, which is incorporated by reference at 39 CFR 111.1

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.

(3) Application for Type B(DP) package approval must be completed in accordance with subpart I of this part, demonstrating that the design of the package to be used satisfies the applicable package approval standards contained in subpart E of this part, as related to the tests of subpart F of this part.

(4) A licensee transporting licensed material, or delivering licensed material to a carrier for transport, shall comply with the operating control~~s~~ requirements of subpart G of this part; the quality assurance requirements of subpart H of this part; and the general provisions of subpart A of this part, including DOT regulations referenced in § 71.5.

(e) The regulations of this part apply to any person holding or applying for a certificate of compliance, issued pursuant to this part, for a package intended for the transportation of radioactive material, outside the confines of a licensee's facility or authorized place of use.

(f) 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.

(g) This part also gives notice to all persons who knowingly provide to any licensee, certificate holder, quality assurance program approval holder, applicant for a license, certificate, or quality assurance program approval, or to a contractor, or subcontractor of any of them, components, equipment, materials, or other goods or services, that relate to a licensee’s, certificate holder’s, quality assurance program approval holder’s, or applicant’s activities subject to this part, that they may be individually subject to NRC enforcement action for violation of § 71.8.

§ 71.1 Communications and records.

(a) Except where otherwise specified, all communications and reports concerning the regulations in this part and applications filed under them should be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001. Written communications, reports, and applications may be delivered in person to the U.S. NRC, ATTN: Document Control Desk, at One White Flint North, 11555 Rockville Pike, Rockville, MD 20852-2738 between 7:30 a.m. and 4:15 p.m., Federal workdays. If the submittal deadline date falls on a Saturday, Sunday, or a Federal holiday, the next Federal work day becomes the official due date.

(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. This value is either listed in Appendix A of this part, Table A-1, or may be derived in accordance with the procedures prescribed in Appendix A of this part.

A_2 means the maximum activity of radioactive material, (other than special form material), LSA, and SCO material, permitted in a Type A package. This value is either listed in Appendix A of this part, Table A-1, or may be derived in accordance with the procedures 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.

Certificate of compliance (CoC) means the certificate issued by the Commission under either subpart D or I of this part which approves the design of a package for the transportation of radioactive material.

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.

Criticality safety index (CSI) means the dimensionless number (rounded up to the next tenth) assigned to and placed on the label of a fissile material package, to designate the degree of control of accumulation of packages containing fissile material during transportation. Determination of the criticality safety index is described in §§ 71.22, 71.23, and 71.59.

Deuterium means, for the purposes of §§ 71.15 and 71.22, the definition of *Deuterium* as found in § 110.2 of this chapter.

DOT means the U.S. Department of Transportation.

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 the radionuclides uranium-233, uranium-235, plutonium-239, and plutonium-241, or any combination of these radionuclides. Fissile material means the fissile nuclides themselves, not material containing fissile nuclides. 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 § 71.15.

Graphite means, for the purposes of §§ 71.15 and 71.22, the definition of *Nuclear grade graphite* as found in § 110.2 of this chapter.

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 material (LSA) 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;

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

(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 distributed throughout, 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) that satisfy the requirements of § 71.77, in which:

(i) The radioactive material is 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 2×10^{-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 or Type *AF package, Type BF package, Type B(U)F package,* or *Type B(M)F package* means a fissile material packaging together with its fissile material contents.

(2) *Type A package* means a Type A packaging together with its radioactive contents. A Type A package is defined and must comply with the DOT regulations in 49 CFR Part 173.

(3) *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 lbs/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 § 71.19.

(4) *Type B(DP) package* means a Type B(DP) packaging together with its radioactive contents. A Type B(DP) package is a dual-purpose package intended for both the transportation and storage of spent fuel. A type B(DP) package is also a fissile material package. A Type B(DP) package is issued both a certificate of compliance approving the design of a spent-fuel transportation package, in accordance with subpart I of this part, and a certificate of compliance approving the design of a spent fuel storage cask, in accordance with subpart L of part 72 of this chapter.

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.

Spent nuclear fuel or *Spent fuel* means fuel that has been withdrawn from a nuclear reactor following irradiation, has undergone at least one year's decay since being used as a source of energy in a power reactor, and has not been chemically separated into its constituent elements by reprocessing. Spent fuel includes the special nuclear material, byproduct material, source material, and other radioactive materials associated with fuel assemblies.

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.

Structures, systems, and components important to safety (SSCs) means those features of a Type B(DP) package whose functions are—

(1) To maintain the conditions required to safely transport the package's contents;

(2) To prevent damage to the package during transport; or

(3) To provide reasonable assurance that the radioactive material contents can be received, handled, transported, and retrieved without undue risk to public health and safety and the environment.

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 nonfixed 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 4×10^4 Bq/cm^2 (1.0 microcurie/cm^2) for beta and gamma and low toxicity alpha emitters, or 4×10^3 Bq/cm^2 (0.1 microcurie/cm^2) for all other alpha emitters; and

(iii) The nonfixed 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 4×10^4 Bq/cm^2 (1 microcurie/cm^2) for beta and gamma and low toxicity alpha emitters, or 4×10^3 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 nonfixed 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 nonfixed 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 (TI) 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 the number determined by multiplying the maximum radiation level in millisievert (mSv) per hour at 1 meter (3.3 ft) from the external surface of the package by 100 (equivalent to the maximum radiation level in millirem per hour at 1 meter (3.3 ft)).

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.

§ 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 (44 U.S.C. 3501 et seq.). The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number. 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, 71.7, 71.8, 71.12, 71.13, 71.17, 71.18, 71.19, 71.20, 71.31, 71.33, 71.35, 71.37, 71.38, 71.39, 71.41, 71.47, 71.85, 71.87, 71.89, 71.90, 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, 71.137, 71.151, 71.153, 71.155, 71.157, 71.159, 71.161, 71.165, 71.167, 71.171, 71.173, 71.175, 71.177, and Appendix A.

§ 71.7 Completeness and accuracy of information.

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

(b) Each licensee, certificate holder, or applicant for a license or CoC must notify the Commission of information identified by the licensee, certificate holder, or applicant for a license or CoC as having, for the regulated activity, a significant implication for public health and safety or common defense and security. A licensee, certificate holder, or an applicant for a license or CoC violates this paragraph only if the licensee, certificate holder, or applicant for a license or CoC fails to notify the Commission of information that the licensee, certificate holder, or applicant for a license or CoC 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 which is already required to be provided to the Commission by other reporting or updating requirements.

§ 71.8 Deliberate misconduct.

(a) This section applies to any--

(1) Licensee;

(2) Certificate holder;

(3) Quality assurance program approval holder;

(4) Applicant for a license, certificate, or quality assurance program approval;

(5) Contractor (including a supplier or consultant) or subcontractor, to any person identified in paragraphs (a)(4) of this section; or

(6) Employees of any person identified in paragraphs (a)(1) through (a)(5) of this section.

(b) A person identified in paragraph (a) of this section who knowingly provides to any entity, listed in paragraphs (a)(1) through (a)(5) of this section any components, materials, or other goods or services that relate to a licensee's, certificate holder's, quality assurance program approval holder's or applicant's activities subject to this part may not:

(1) Engage in deliberate misconduct that causes or would have caused, if not detected, a licensee, certificate holder, quality assurance program approval holder, or any applicant to be

in violation of any rule, regulation, or order; or any term, condition or limitation of any license, certificate or approval issued by the Commission; or

(2) Deliberately submit to the NRC, a licensee, a certificate holder, quality assurance program approval holder, an applicant for a license, certificate or quality assurance program approval, or a licensee's, applicant's, certificate holder's, or quality assurance program approval holder's contractor or subcontractor, information that the person submitting the information knows to be incomplete or inaccurate in some respect material to the NRC.

(c) A person who violates paragraph (b)(1) or (b)(2) of this section may be subject to enforcement action in accordance with the procedures in 10 CFR part 2, subpart B.

(d) For the purposes of paragraph (b)(1) of this section, deliberate misconduct by a person means an intentional act or omission that the person knows:

(1) Would cause a licensee, certificate holder, quality assurance program approval holder, or applicant for a license, certificate, or quality assurance program approval to be in violation of any rule, regulation, or order; or any term, condition, or limitation of any license or certificate issued by the Commission; or

(2) Constitutes a violation of a requirement, procedure, instruction, contract, purchase order, or policy of a licensee, certificate holder, quality assurance program approval holder, applicant, or the contractor or subcontractor of any of them.

§ 71.9 Employee protection.

(a) Discrimination by a Commission licensee, certificate holder, an applicant for a Commission license or a CoC, or a contractor or subcontractor of any of these, against an employee for engaging in certain protected activities, is prohibited. Discrimination includes discharge and other actions that relate to compensation, terms, conditions, or privileges of employment. The protected activities are established in section 211 of the Energy Reorganization Act of 1974, as amended, and in general are related to the administration or enforcement of a requirement imposed under the Atomic Energy Act of 1954, as amended, or the Energy Reorganization Act of 1974, as amended.

(1) The protected activities include**,** but are not limited to:

(i) Providing the Commission or his or her employer information about alleged violations of either of the statutes named in paragraph (a) of this section or possible violations of requirements imposed under either of those statutes;

(ii) Refusing to engage in any practice made unlawful under either of the statutes named in paragraph (a) of this section or under these requirements if the employee has identified the alleged illegality to the employer;

(iii) Requesting the Commission to institute action against his or her employer for the administration or enforcement of these requirements;

(iv) Testifying in any Commission proceeding, or before Congress, or at any Federal or State proceeding regarding any provision (or proposed provision) of either of the statutes named in paragraph (a) of this section; and

(v) Assisting or participating in, or is about to assist or participate in, these activities.

(2) These activities are protected even if no formal proceeding is actually initiated as a result of the employee's assistance or participation.

(3) This section has no application to any employee alleging discrimination prohibited by this section who, acting without direction from his or her employer (or the employer's agent), deliberately causes a violation of any requirement of the Energy Reorganization Act of 1974, as amended, or the Atomic Energy Act of 1954, as amended.

(b) Any employee who believes that he or she has been discharged or otherwise discriminated against by any person for engaging in protected activities specified in paragraph (a)(1) of this section may seek a remedy for the discharge or discrimination through an administrative proceeding in the Department of Labor. The administrative proceeding must be initiated within 180 days after an alleged violation occurs. The employee may do this by filing a complaint alleging the violation with the Department of Labor, Employment Standards Administration, Wage and Hour Division. The Department of Labor may order reinstatement, back pay, and compensatory damages.

(c) A violation of paragraph (a), (e), or (f) of this section by a Commission licensee, certificate holder, applicant for a Commission license or a CoC, or a contractor or subcontractor of any of these may be grounds for:

(1) Denial, revocation, or suspension of the license or the CoC;

(2) Imposition of a civil penalty on the licensee or applicant; or

(3) Other enforcement action.

(d) Actions taken by an employer, or others, which adversely affect an employee may be predicated upon nondiscriminatory grounds. The prohibition applies when the adverse action occurs because the employee has engaged in protected activities. An employee's engagement in protected activities does not automatically render him or her immune from discharge or discipline for legitimate reasons or from adverse action dictated by nonprohibited considerations.

(e)(1) Each licensee, certificate holder, and applicant for a license or CoC must prominently post the revision of NRC Form 3, "Notice to Employees," referenced in § 19.11(c). This form must be posted at locations sufficient to permit employees protected by this section to observe a copy on the way to or from their place of work. The premises must be posted not later than 30 days after an application is docketed and remain posted while the application is pending before the Commission, during the term of the license or CoC, and for 30 days following license or CoC termination.

(2) Copies of NRC Form 3 may be obtained by writing to the Regional Administrator of the appropriate U.S. Nuclear Regulatory Commission Regional Office listed in Appendix D to part 20 of this chapter or by calling the NRC Information and Records Management Branch at 301-415-7230.

(f) No agreement affecting the compensation, terms, conditions, or privileges of employment, including an agreement to settle a complaint filed by an employee with the Department of Labor pursuant to section 211 of the Energy Reorganization Act of 1974, as amended, may contain any provision which would prohibit, restrict, or otherwise discourage an employee from participating in a protected activity as defined in paragraph (a)(1) of this section including, but not limited to, providing information to the NRC or to his or her employer on potential violations or other matters within NRC's regulatory responsibilities.

§ 71.10 Public inspection of application.

Applications for approval of a package design under this part, which are submitted to the Commission, may be made available for public inspection, in accordance with provisions of parts 2 and 9 of this chapter. This includes an application to amend or revise an existing package design, any associated documents and drawings submitted with the application, and any responses to NRC requests for additional information.

§ 71.11 [Reserved.]

Subpart B - Exemptions

§ 71.12 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.

§ 71.13 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.

§ 71.14 Exemption for low-level materials.

(a) A licensee is exempt from all the requirements of this part with respect to shipment or carriage of the following low-level materials:

(1) Natural material and ores containing naturally occurring radionuclides that are not intended to be processed for use of these radionuclides, provided the activity concentration of the material does not exceed 10 times the values specified in Appendix A, of this part.

(2) Materials for which the activity concentration is not greater than the activity concentration values specified in Appendix A, of this part, or for which the consignment activity is not greater than the limit for an exempt consignment found in Appendix A, of this part.

(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 do not contain any fissile material, or the material is exempt from classification as fissile material under § 71.15:

(1) The package contains no more than a Type A quantity of radioactive material. *Exception*. This paragraph does not apply to a package — transported within the United States — containing special greater than an A_1 quantity form plutonium-244;

(2) The package — transported within the United States — contains no more than 0.74 TBq (20 Ci) of special form plutonium-244; or

(3) The package contains only LSA or SCO radioactive material, provided —

(i) ~~t~~**T**hat the LSA or SCO material has an external radiation dose of less than or equal to 10 mSv/h (1 rem/h), at a distance of 3 m from the unshielded material; or

(ii) ~~t~~**T**hat the package is classified as LSA-I or SCO-I.

(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.

§ 71.15 Exemption from classification as fissile material.

Fissile material~~s~~ meeting the requirements of at least one of the paragraphs (a) through (e) of this section**,** are exempt from classification as fissile material and from the fissile material package standards of §§ 71.55 and 71.59, but are subject to all other requirements of this part, except as noted.

(a) The mass ratio of iron to fissile material is greater than 200:1 and the package contents contain less than 15 g of fissile material. The fissile material may be contained in individual or bulk packaging.

(b) The mass ratio of noncombustible, insoluble-in-water, material (including both the contents and packaging) to fissile material is greater than 2000:1 and the package contents contain less than 350 g of fissile material. Lead, beryllium, graphite, and hydrogenous material enriched in deuterium may be present in the package, but must not be included in determining the mass ratio for the package. The fissile material may be contained in individual or bulk packaging.

(c) Uranium enriched in uranium-235 to a maximum of 1 percent by weight, and with total plutonium and uranium-233 content of up to 1 percent of the mass of uranium-235, provided that the mass of any beryllium, graphite, and hydrogenous material enriched in deuterium present in the package is less than 0.1 percent of the fissile mass.

(d) Liquid solutions of uranyl nitrate enriched in uranium-235 to a maximum of 2 percent by weight, provided that:

(1) the total plutonium and uranium-233 content does not exceed 0.1 percent of the mass of uranium-235;

(2) the nitrogen to uranium atomic ratio (N/U) is greater than or equal to 2.0; and

(3) the material must be contained in at least a DOT Type A package**.**

(e) Plutonium with a total mass of less than 1000 grams, provided that: plutonium-239, plutonium-241, or any combination of these radionuclides, constitutes less than 20 percent by mass of the total quantity of plutonium in the package.

§ 71.16 [Reserved]

Subpart C - General Licenses

§ 71.17 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 (other than a Type B(DP) 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 NRC, 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. A licensee shall submit this information in accordance with § 71.1.

(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 § 71.19.

§ 71.18 General license: NRC-approved Type B(DP) 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 Type B(DP) package for which a license, certificate of compliance (CoC), 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 CoC, or other approval, of the Type B(DP) package; a copy of the updated final safety analysis report for the package; and the drawings and other documents referenced in the CoC, or other 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, CoC, 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 NRC, 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. A licensee shall submit this information in accordance with § 71.1.

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

(e) This general license does not authorize a Type B(DP) packages to be transported by air.

§71.19 Previously approved package.

(a) A Type B package previously approved by NRC, but not designated as B(U), B(M), B(U)F, B(M)F, in the identification number of the NRC Certificate of Compliance, or Type AF packages approved under Safety Series No. 6 (1967 Edition), may be used under the general license of § 71.17 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.

(4) § 71.19(a) will expire 3 years after the effective date of the final rule.

(b) A Type B(U) package, a Type B(M) 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 § 71.17 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) A Type B(U) package, a Type B(M) package, or a fissile material package previously approved by the NRC, but without the designation "-96" in the identification number of the NRC Certificate of Compliance, may be used under the general license of § 71.17 with the following additional conditions:

(1) Fabrication of the package must be satisfactorily completed by December 31, 2006, as demonstrated by application of its model number in accordance with § 71.85(c); and

(2) After December 31, 2003, 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.

(d) 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.

(e) NRC will revise the package identification number to designate previously approved package designs as B(U), B(M), AF, or BF, as appropriate, and with the identification number suffix "-96" after receipt of an application demonstrating that the design meets the requirements of this part.

§ 71.20 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.

§ 71.21 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.

§ 71.22 General license: Fissile material.

(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, if the material is shipped in accordance with this section. The fissile material need not be contained in a package which meets the standards of subparts E and F of this part; however, the material must be contained in a Type A package. The Type A package must also meet the DOT requirements of 49 CFR 173.417(a).

(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) The general license applies only when a package's contents:

(1) Contain less than a Type A quantity of fissile material; and

(2) Contain less than 500 total grams of beryllium, graphite, or hydrogenous material enriched in deuterium.

(d) The general license applies only to fissile material packages labeled with a CSI which:

(1) has been determined in accordance with paragraph (e) of this section;

(2) has a value less than or equal to 10.0; and

(3) for a shipment of multiple fissile material packages, contained in a single conveyance, the sum of the CSIs must meet the requirements of § 71.59(c).

(e)(1) The value for the CSI must be greater than or equal to the number calculated by the following equation:

$$\mathrm{CSI} = 10\left[\frac{\text{grams of } {}^{235}\mathrm{U}}{X} + \frac{\text{grams of } {}^{233}\mathrm{U}}{Y} + \frac{\text{grams of Pu}}{Z}\right];$$

(2) The calculated CSI must be rounded up to the first decimal place;

(3) The values of *X*, *Y*, and *Z* used in the CSI equation must be taken from Tables 71-1 or 71-2, as appropriate;

(4) If Table 71-2 is used to obtain the value of *X*, then the values for the terms in the equation for uranium-233 and plutonium must be assumed to be zero; and

(5) Table 71-1 values for *X*, *Y*, and *Z* must be used to determine the CSI if:

(i) uranium-233 is present in the package;

(ii) the mass of plutonium exceeds 1 percent of the mass of uranium-235;

(iii) the uranium-235 is of unknown enrichment; or

(iv) substances having a moderating effectiveness (i.e., an average hydrogen density greater than H_2O) [e.g., certain hydrocarbon oils or plastics] are present in any form, except as polyethylene used for packing or wrapping.

TABLE 71-1. MASS LIMITS FOR GENERAL LICENSE PACKAGES CONTAINING MIXED QUANTITIES OF FISSILE MATERIAL OR URANIUM-235 OF UNKNOWN ENRICHMENT PER § 71.22(e)

Fissile material	Fissile material mass mixed with moderating substances having an average hydrogen density less than or equal to H_2O. (grams)	Fissile material mass mixed with moderating substances having an average hydrogen density greater than H_2O.[a] (grams)
^{235}U (*X*)	60	38
^{233}U (*Y*)	43	27
^{239}Pu or ^{241}Pu (Z)	37	24

[a] When mixtures of moderating substances are present, the lower mass limits shall be used if more than 15 percent of the moderating substance has an average hydrogen density greater than H_2O.

TABLE 71-2 — MASS LIMITS FOR GENERAL LICENSE PACKAGES CONTAINING URANIUM-235 OF KNOWN ENRICHMENT PER § 71.22(e)

Uranium enrichment in weight percent of ^{235}U not exceeding	Fissile material mass of ^{235}U (X). (grams)
24	60
20	63
15	67
11	72
10	76
9.5	78
9	81
8.5	82
8	85
7.5	88
7	90
6.5	93
6	97
5.5	102
5	108
4.5	114
4	120
3.5	132
3	150
2.5	180
2	246
1.5	408
1.35	480
1	1,020
0.92	1.800

§ 71.23 General license: Plutonium-beryllium special form material.

(a) A general license is issued to any licensee of the Commission to transport fissile material in the form of plutonium-beryllium (Pu-Be) special form sealed sources, or to deliver Pu-Be sealed sources to a carrier for transport, if the material is shipped in accordance with this section. This material need not be contained in a package which meets the standards of subparts E and F of this part; however, the material must be contained in a Type A package. The Type A package must also meet the DOT requirements of 49 CFR 173.417(a).

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

(c)The general license applies only when a package's contents:

(1) Contain less than a Type A quantity of material; and

(2) Contain less than 1000 g of plutonium, provided that: plutonium-239, plutonium-241, or any combination of these radionuclides, constitutes less than 240 g of the total quantity of plutonium in the package.

(d) The general license applies only to packages labeled with a CSI which:

(1) has been determined in accordance with paragraph (e) of this section;

(2) has a value less than or equal to 100.0; and

(3) for a shipment of multiple fissile material packages, contained in a single conveyance, the sum of the CSI must meet the requirements of § 71.59(c).

(e)(1) The value for the CSI must be greater than or equal to the number calculated by the following equation:

$$\text{CSI} = 10\left[\frac{\text{grams of } ^{239}\text{Pu} + \text{grams of } ^{241}\text{Pu}}{24}\right]; \text{ and}$$

(2) The calculated CSI must be rounded up to the first decimal place.

§ 71.24 [Reserved]

§ 71.25 [Reserved]

3. In § 71.41, paragraph (a) is revised and a new paragraph (d) is added to read as follows:

§ 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 requirements for Type B packages containing more than 10^5 A_2 "), 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.

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(d) Packages for which compliance with the other provisions of these regulations is impracticable shall not be transported except under special package authorization. Provided the applicant demonstrates that compliance with the other provisions of the regulations is impracticable and that the requisite standards of safety established by these regulations have been demonstrated through means alternative to the other provisions, a special package authorization may be approved for one-time shipments. The applicant shall demonstrate that the overall level of safety in transport for these shipments is at least equivalent to that which would be provided if all the applicable requirements had been met.

4. In § 71.51, the introductory text of paragraph (a) is revised, and a new paragraph (d) is added to read as follows:

§ 71.51 Additional requirements for Type B packages.

(a) 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:

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(d) For packages which contain radioactive contents with activity greater than 10^5 A_2, the requirements of § 71.61 must be met. Except for Type B(DP) packages, a package must meet the requirements of § 71.61, if the package contents contain radioactive material with an activity greater than 10^5 A_2.

5. Section 71.53 is removed and reserved.

§ 71.53 [Reserved]

6. In § 71.55, paragraph (b) is revised, and new paragraphs (f) and (g) are added to read as follows:

§ 71.55 General requirements for fissile material packages.

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(b) Except as provided in paragraph (c) or (g) 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:

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(f) For fissile material package designs to be transported by air:

(1) The package must be designed and constructed, and its contents limited so that it would be subcritical, assuming reflection by 20 cm (7.9 in) of water but no water inleakage, when subjected to sequential application of:

(i) The free drop test in §71.73(c)(1);

(ii) The crush test in §71.73(c)(2);

(iii) A puncture test, for packages of 250 kg or more, consisting of a free drop of the specimen through a distance of 3 m (120 in) in a position for which maximum damage is expected at the conclusion of the test sequence, onto the upper end of a solid, vertical, cylindrical, mild steel probe mounted on an essentially unyielding, horizontal surface. The probe must be 20 cm (7.9 in) in diameter, with the striking end forming the frustum of a right circular cone with the dimensions of 30 cm height, 2.5 cm top diameter, and a top edge rounded to a radius of not more than 6 mm (0.25 in). For packages less than 250 kg, the puncture test must be the same, except that a 250 kg probe must be dropped onto the specimen which must be placed on the surface; and

(iv) The thermal test in §71.73(c)(4), except that the duration of the test must be 60 minutes.

(2) The package must be designed and constructed, and its contents limited so that it would be subcritical, assuming reflection by 20 cm (7.9 in) of water but no water inleakage, when subjected to an impact on an unyielding surface at a velocity of 90 m/s normal to the surface, at such orientation so as to result in maximum damage. A separate, undamaged specimen can be used for this evaluation.

(3) Allowance may not be made for the special design features in paragraph (c) of this section, unless water leakage into or out of void spaces is prevented following application of the

tests in paragraphs (f)(1) and (f)(2) of this section, and subsequent application of the immersion test in § 71.73(c)(5).

(g) Packages containing uranium hexafluoride only are excepted from the requirements of paragraph (b) of this section provided that:

(1) Following the tests specified in § 71.73 (“Hypothetical accident conditions”), there is no physical contact between the valve body and any other component of the packaging, other than at its original point of attachment, and the valve remains leak tight;

(2) There is an adequate quality control in the manufacture, maintenance and repair of packagings;

(3) Each package is tested to demonstrate closure before each shipment; and

(4) The uranium is enriched to not more than 5 weight percent uranium-235.

7. In § 71.59, paragraphs (b) and (c) are revised to read as follows:

§ 71.59 Standards for arrays of fissile material packages.

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(b) The CSI must be determined by —

(1) Dividing the number 50 by the value of "N" derived using the procedures specified in paragraph (a) of this section. The value of the CSI 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 CSI greater than zero must be rounded up to the first decimal place; or

(2) The number calculated by the equations in either §§ 71.22 or 71.23, for packages shipped under the general license provisions of §§ 71.22 or 71.23.

(c) Where a fissile material package is assigned a CSI value —

(1) Less than or equal to 10.0, that package may be shipped by any carrier, and that carrier must provide adequate criticality control by limiting the sum of the CSIs to less than 50.0 in a nonexclusive use vehicle, and to less than 100.0 in an exclusive use vehicle.

(2) Greater than 10.0, that package must be shipped by exclusive use vehicle or other shipper controlled system specified by DOT for fissile material packages. The shipper must provide adequate criticality control by limiting the sum of the CSIs to less than 100.0 in an exclusive use vehicle.

8. Section 71.61 is revised to read as follows:

§ 71.61 Special requirements for Type B packages containing more than 10^5 A_2.

A Type B package containing more than 10^5 A_2 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.

9. Section 71.63 is revised to read as follows:

§ 71.63 Special requirement for plutonium shipments.

Shipments containing plutonium must be made with the contents in solid form, if the contents contain greater than 0.74 TBq (20 Ci) of plutonium.

10. In § 71.73, paragraph (c)(2) is revised to read as follows:

§ 71.73 Hypothetical accident conditions.

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(c) ★ ★ ★

(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-lb) 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 dimension, and radioactive contents greater than 1000 A_2 not as special form radioactive material. For packages containing fissile material, the radioactive contents greater than 1000 A_2 criterion does not apply.

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11. In § 71.88, paragraph (a)(2) is revised to read as follows:

§ 71.88 Air transport of plutonium

(a) ★ ★ ★

(2) The plutonium is contained in a material in which the specific activity is less than or equal to the activity concentration values for plutonium specified in Appendix A, Table A-2 of this part, and in which the radioactivity is essentially uniformly distributed; or

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12. In § 71.91, paragraphs (b) and (c) are revised, and a new paragraph (d) is added to read as follows:

§ 71.91 Records.

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(b) Each certificate holder shall maintain, for a period of 3 years after the life of the packaging to which they apply, records identifying the packaging by model number, serial number, and date of manufacture.

(c) The licensee, certificate holder, and an applicant for a CoC, 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.

(d) The licensee, certificate holder , and an applicant for a CoC, 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. These records must be retained for 3 years after the life of the packaging to which they apply.

13. Section 71.93 is revised to read as follows:

§ 71.93 Inspection and tests.

(a) The licensee, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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 certificate holder and applicant for a CoC shall notify the NRC, in accordance with § 71.1, 45 days in advance of starting fabrication of the first packaging under a CoC. This paragraph applies to any packaging used for the shipment of licensed material which has either—

(1) A decay heat load in excess of 5 kW; or

(2) A maximum normal operating pressure in excess of 103 kPa (15 lbf/in^2) gauge.

14. Section 71.95 is revised to read as follows:

§ 71.95 Reports.

(a) The licensee, after requesting the certificate holder's input, shall submit a written report to the Commission of—

(1) Instances in which there is a significant reduction in the effectiveness of any NRC-approved Type B or Type A(F) packaging during use; or

(2) Details of any defects with safety significance in any NRC-approved Type B or fissile material packaging, after first use.

(b) The licensee shall submit a written report to the Commission of instances in which the conditions in the certificate of compliance were not followed during a shipment.

(c) *Written report*. Each licensee shall submit, in accordance with § 71.1, a written report required by paragraphs (a) or (b) of this section within 60 days of the event or discovery of the event. The licensee shall also provide a copy of each report submitted to the NRC to the applicable certificate holder. Written reports prepared pursuant to other regulations may be submitted to fulfill this requirement if the reports contain all the necessary information, and the appropriate distribution is made. These written reports must include the following:

(1) A brief abstract describing the major occurrences during the event, including all component or system failures that contributed to the event and significant corrective action taken or planned to prevent recurrence.

(2) A clear, specific, narrative description of the event that occurred so that knowledgeable readers conversant with the requirements of Part 71, but not familiar with the design of the packaging, can understand the complete event. The narrative description must include the following specific information as appropriate for the particular event.

(i) Status of components, or systems that were inoperable at the start of the event and that contributed to the event;

(ii) Dates and approximate times of occurrences;

(iii) The cause of each component or system failure or personnel error, if known;

(iv) The failure mode, mechanism, and effect of each failed component, if known;

(v) A list of systems or secondary functions that were also affected for failures of components with multiple functions;

(vi) The method of discovery of each component or system failure or procedural error;

(vii) For each human performance related root cause, a discussion of the cause(s) and circumstances;

(viii) The manufacturer and model number (or other identification) of each component that failed during the event; and

(ix) For events occurring during use of a packaging, the quantities and chemical and physical form(s) of the package contents.

(3) An assessment of the safety consequences and implications of the event. This assessment must include the availability of other systems or components that could have performed the same function as the components and systems that failed during the event.

(4) A description of any corrective actions planned as a result of the event, including the means employed to repair any defects, and actions taken to reduce the probability of similar events occurring in the future.

(5) Reference to any previous similar events involving the same packaging that are known to the licensee or certificate holder.

(6) The name and telephone number of a person within the licensee's organization who is knowledgeable about the event and can provide additional information.

(7) The extent of exposure of individuals to radiation or to radioactive materials without identification of individuals by name.

(d) *Report legibility*. The reports submitted by licensees and/or certificate holders under this section must be of sufficient quality to permit reproduction and micrographic processing.

15. In § 71.100, paragraph (b) is revised to read as follows:

§ 71.100 Criminal penalties.

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(b) The regulations in part 71 that are not issued under §§ 161b, 161i, or 161o for the purposes of § 223 are as follows: §§ 71.0, 71.2, 71.4, 71.6, 71.7, 71.10, 71.31, 71.33, 71.35, 71.37, 71.38, 71.39, 71.40, 71.41, 71.43, 71.45, 71.47, 71.51, 71.55, 71.59, 71.65, 71.71, 71.73, 71.74, 71.75, 71.77, 71.99, 71.100, and 71.151 through 71.169.

16. Subpart H to Part 71 is revised to read as follows:

Sec.

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. The licensee, certificate holder, and applicant for a CoC are responsible for the quality assurance requirements as they apply to design, fabrication, testing, and modification of packaging. Each licensee is responsible for the quality assurance provision which applies to its use of a packaging for the shipment of licensed material subject to this subpart.

(b) *Establishment of program*. Each licensee, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC shall execute the applicable criteria in a graded approach to an extent that is commensurate with the quality assurance requirement's importance to safety.

(c) *Approval of program*. (1) 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, in accordance with § 71.1, 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.

(2) Before the fabrication, testing, or modification of any package for the shipment of licensed material subject to this subpart, each licensee, certificate holder, or applicant for a CoC shall obtain Commission approval of its quality assurance program. Each certificate holder or applicant for a CoC shall, in accordance with § 71.1, 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.

(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 before 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 subpart H of this part, Appendix B of part 50 of this chapter, or subpart G of part 72 of this chapter, and that is established, maintained, and executed regarding transport packages, will be accepted as satisfying the requirements of paragraph (b) of this section. Before first use, the licensee, certificate holder, and applicant for a CoC shall notify the NRC, in accordance with § 71.1, of its intent to apply its previously approved subpart H, Appendix B, or subpart G quality assurance program to transportation activities. The licensee, certificate holder, and applicant for a CoC shall identify the program by date of submittal to the Commission, Docket Number, and date of Commission approval.

(g) *Radiography containers*. A program for transport container inspection and maintenance limited to radiographic exposure devices, source changers, or packages transporting these devices and meeting the requirements of § 34.31(b) of this chapter or equivalent Agreement State requirement, is deemed to satisfy the requirements of §§ 71.17(b) and 71.101(b).

§ 71.103 Quality assurance organization.

(a) The licensee,[20] certificate holder, and applicant for a CoC shall be responsible for the establishment and execution of the quality assurance program. The licensee, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC shall clearly establish and delineate, in writing, the authority and

[20] 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 before the time a package approval is issued.

duties of persons and organizations performing activities affecting the functions of structures, systems, and components that are important to safety. 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 functions that are important to safety have been correctly performed.

(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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC, 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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 they are executing.

§ 71.107 Package design control.

(a) The licensee, certificate holder, and applicant for a CoC shall establish measures to assure that applicable regulatory requirements and the package design, as specified in the license or CoC 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 functions of the materials, parts, and components of the packaging that are important to safety.

(b) The licensee, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC shall include suitable qualification testing of a prototype or sample unit under the most adverse design conditions. The licensee, certificate holder, and applicant for a CoC shall apply design control measures to 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, certificate holder, and applicant for a CoC 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 CoC require NRC prior approval.

§ 71.109 Procurement document control.

The licensee, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC or by its contractors or subcontractors. To the extent necessary, the licensee, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC shall retain, or have available, this documentary evidence for the life of the package to which it applies. The licensee, certificate holder, and applicant for a CoC shall assure that the evidence is sufficient to identify the specific requirements met by the purchased material and equipment.

(c) The licensee, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC shall retain these records for 3 years beyond the date when the licensee, certificate holder, and applicant for a CoC 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, certificate holder, and applicant for a CoC shall retain the superseded material for 3 years after it is superseded.

§ 71.137 Audits.

The licensee, certificate holder, and applicant for a CoC 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.

17. A new subpart I is added to Part 71 to read as follows:

Sec.

Subpart I - Type B(DP) Package Approval

§ 71.151 Procedures for applying for a Type B(DP) package approval.

(a) Spent fuel storage casks that have been issued a Certificate of Compliance (CoC) under subpart L of part 72 of this chapter may also be approved under this subpart as a Type B(DP) package for the transportation of spent fuel. A copy of the part 72 CoC issued for the cask, and any drawings and other documents referenced in the part 72 CoC, must be included with the application.

(b) An application for approval of a Type B(DP) package design must contain the information required by § 71.153 and be submitted in accordance with § 71.1.

(c) *Public inspection.* An application for the approval of a Type B(DP) package, or amendment of a Type B(DP) package, may be made available for public inspection under § 71.10.

(d) *Fees.* Fees for reviews and evaluations related to issuance of a Type B(DP) CoC and inspections related to package fabrication are those shown in § 170.31 of this chapter.

§ 71.153 Contents of application.

(a) An application for an approval of a Type B(DP) package under this subpart must include, for each proposed Type B(DP) packaging design, the following information:

(1) A package description as required by § 71.155;

(2) A package evaluation as required by § 71.157; and

(3) A quality assurance program description, as required by § 71.159, or a reference to a previously approved quality assurance program.

(b) A safety analysis report describing —

(1) The proposed Type B(DP) package design;

(2) How the package would be used to transport spent fuel safely;

(3) An analysis of potential accidents, package response to these potential accidents, and any consequences to the public; and

(4) How the package is suitable for the transportation of spent fuel for a period of at least 20 years.

(c) Except as provided in § 71.19, an application for modification of a Type B(DP) package design, whether for modification of the packaging or the authorized contents, must include sufficient information to demonstrate that the proposed design satisfies the Type B(DP) package standards in effect at the time the application is filed.

(d) 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.

§ 71.155 Package description.

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

(a) With respect to the packaging—

(1) Gross weight;

(2) Model number;

(3) Identification of the containment system;

(4) 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

(5) 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.

§ 71.157 Package evaluation.

The application submitted under § 71.151 must include the following:

(a) A demonstration that the Type B(DP) package satisfies the standards specified in subparts E and F of this part. The application need not address the requirements of §§ 71.61, 71.64, 71.74, 71.75, and 71.77;

(b) The number "N" for the Type B(DP) package as determined in accordance with § 71.59; and

(c) Any proposed special controls and precautions for transport, loading, unloading, and handling and any proposed special controls in case of an accident or delay.

§ 71.159 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 Type B(DP) package.

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

§ 71.161 Requirement for additional information.

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

§ 71.163 Issuance of an NRC certificate of compliance.

The NRC will issue a CoC for a Type B(DP) package on a finding that the requirements in §§ 71.151 through 71.159 are met. The term of a Type B(DP) CoC is to up to 20 years.

§ 71.165 Conditions for package reapproval.

(a) Except as provided in paragraph (b) of this section, each CoC for a Type B(DP) package or Quality Assurance Program Approval expires at the end of the day, in the month and year stated in the approval.

(b) *Timely renewal.* If a person holding a CoC for a Type B(DP) package or Quality Assurance Program Approval issued under this part has filed a proper application requesting renewal of either the CoC or the Quality Assurance Program Approval, then the CoC or Quality Assurance Program Approval is not considered to have expired until the Commission has taken final action on the application. The application must be submitted to the Commission not less than 2 years before the expiration of the CoC or the Quality Assurance Program Approval.

(c) In applying for renewal of an existing CoC for a Type B(DP) package 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.

(d) Applications for renewal of an existing CoC for a Type B(DP) package or Quality Assurance Program Approval must be submitted to the Commission in accordance with § 71.1.

§ 71.167 Application to amend a certificate of compliance.

A certificate holder desiring to amend its CoC for a Type B(DP) package — including a change to the terms, conditions, or specifications of the CoC — shall submit an application for amendment with the Commission, in accordance with § 71.1. The application must fully describe the changes desired and the reasons for these changes. The application should follow, as far as applicable, the form prescribed for an original application in § 71.151.

§ 71.169 Issuance of an amendment to a certificate of compliance.

In determining whether an amendment to a CoC for a Type B(DP) package will be issued to the applicant, the Commission will be guided by the considerations that govern the issuance of an initial CoC.

§ 71.171 Inspections and tests.

(a) The certificate holder and applicant for a CoC for a Type B(DP) package shall permit, and make provisions for, the NRC to inspect the premises and facilities where a Type B(DP) package is designed, fabricated, and tested.

(b) The certificate holder and applicant for a CoC for a Type B(DP) package shall make available to the NRC for inspection, upon reasonable notice, records kept by them pertaining to the design, fabrication, and testing of a Type B(DP) package.

(c) The certificate holder and applicant for a CoC for a Type B(DP) package shall perform, and make provisions that permit the NRC to perform tests that the Commission deems necessary or appropriate for the administration of the regulations in this part.

§ 71.173 Recordkeeping and reports.

(a) Each certificate holder or applicant shall maintain any records and produce any reports that may be required by the conditions of the CoC or by the rules, regulations, and orders of the NRC in effectuating the purposes of the Act.

(b) Records that are required by the regulations in this part or by conditions of the CoC must be maintained for the period specified by the appropriate regulation or the CoC conditions. If a retention period is not specified, the records must be maintained until the NRC terminates the CoC.

(c) Any record maintained under this part may be either the original or a reproduced copy by any state-of-the-art method provided that any reproduced copy is duly authenticated by authorized personnel and is capable of producing a clear and legible copy after storage for the period specified by NRC regulations.

(d) Each certificate holder shall maintain a record of each Type B(DP) package it has manufactured. The record must contain the following information:

(1) The package identification number;

(2) The package serial number;

(3) The date fabrication of the package was commenced; and

(4) The date fabrication of the package was completed.

§ 71.175 Changes.

(a) Definitions for the purposes of this section:

(1) *Change* means a modification or addition to, or removal from, a Type B(DP) package design or procedures that affect a design function, method of performing or controlling the function, or an evaluation that demonstrates that intended functions will be accomplished.

(2) *Departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses* means:

(i) Changing any of the elements of the method described in the FSAR (as updated) unless the results of the analysis are conservative or essentially the same; or

(ii) Changing from a method described in the FSAR to another method unless that method has been approved by NRC for the intended application.

(3) *A Type B(DP) package design as described in the Final Safety Analysis Report (FSAR) (as updated)* means:

(i) The structures, systems, and components (SSC) that are described in the FSAR (as updated),

(ii) The design and performance requirements for such SSCs described in the FSAR (as updated), and

(iii) The evaluations or methods of evaluation included in the FSAR (as updated) for such SSCs which demonstrate that their intended function(s) will be accomplished.

(4) *Final Safety Analysis Report (as updated)* means the Safety Analysis Report for a Type B(DP) package design as submitted, amended, and updated in accordance with § 71.177.

(5) *Procedures as described in the FSAR (as updated)* means those procedures that contain information described in the safety analysis report such as how SSCs are operated and controlled (including assumed operator actions and response times).

(b) This section applies to each holder of a CoC for Type B(DP) package issued under this subpart.

(c)(1) A certificate holder may make changes to a Type B(DP) package design, as described in the FSAR (as updated), and make changes in the procedures, as described in the FSAR (as updated), without obtaining a CoC amendment under § 71.167 if:

(i) A change in the terms, conditions, or specifications incorporated in the CoC is not required; and

(ii) The change does not meet any of the criteria in paragraph (c)(2) of this section.

(2) A certificate holder shall obtain a CoC amendment under § 71.167, before implementing a proposed change if the change would:

(i) Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR (as updated);

(ii) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a system, structure, or component (SSC) important to safety previously evaluated in the FSAR (as updated);

(iii) Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR (as updated);

(iv) Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR (as updated);

(v) Create a possibility for an accident of a different type than any previously evaluated in the FSAR (as updated);

(vi) Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR (as updated);

(vii) Result in a design basis limit for a fission product barrier as described in the FSAR (as updated) being exceeded or altered; or

(viii) Result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses.

(3) In implementing this paragraph, the FSAR (as updated) is considered to include FSAR changes resulting from evaluations performed under this section and analyses performed under § 71.161, since the last update of the FSAR as required by § 71.177.

(4) The provisions in this section do not apply to changes to procedures when the applicable regulations of this part establish more specific criteria for accomplishing such changes.

(d)(1) The certificate holder shall maintain records of changes to a Type B(DP) package and of changes in procedures made under paragraph (c) of this section. These records must include a written evaluation that provides the bases for the determination that the change does not require a CoC amendment under paragraph (c)(2) of this section.

(2) The certificate holder shall submit, as specified in § 71.1, a report containing a brief description of any changes, including a summary of the evaluation of each. A report must be submitted at intervals not to exceed 24 months.

(3) The records of changes in a Type B(DP) package design must be maintained until:

(i) The Commission terminates the CoC issued under this part; or

(ii) The package is permanently removed from service.

(4) The records of changes in procedures must be maintained for a period of 5 years.

(5) The holder of a Type B(DP) package design CoC, who permanently ceases operation, shall provide the records of changes to the new certificate holder or to the Commission, in accordance with § 71.1, as appropriate.

(6) A certificate holder shall provide a copy of the record for any changes to a Type B(DP) package design to any licensee using the package design within 60 days of implementing the change.

§ 71.177 Safety analysis report updating.

(a) Each certificate holder for a Type B(DP) package approved under this subpart shall update periodically, as provided in paragraph (b) of this section, the final safety analysis report

(FSAR) to assure that the information included in the report contains the latest information developed.

(1) Each certificate holder shall submit an original FSAR to the Commission, in accordance with § 71.1, within 90 days after the Type B(DP) package design has been approved under § 71.163.

(2) The original FSAR must be based on the safety analysis report submitted with the application and reflect any changes and applicant commitments developed during the Type B(DP) package design review process. The original FSAR must be updated to reflect any changes to requirements contained in the issued CoC.

(b) Each update must contain all the changes necessary to reflect information and analyses submitted to the Commission by the certificate holder or prepared by the certificate holder pursuant to Commission requirements since the submission of the original FSAR or, as appropriate, the last update to the FSAR under this section. The update must include the effects[21] of:

(1) All changes made in the dual-purpose spent fuel transportation package procedures as described in the FSAR;

(2) All safety analyses and evaluations performed by the certificate holder either in support of approved CoC amendments, or in support of conclusions that changes did not require a CoC amendment in accordance with § 71.175; and

(3) All analyses of new safety issues performed by or on behalf of the certificate holder at Commission request. The information shall be appropriately located within the updated FSAR.

[21] Effects of changes includes appropriate revisions of descriptions in the FSAR so that the FSAR (as updated) is complete and accurate.

(c)(1) The update of the FSAR must be filed in accordance with § 71.1, on a replacement-page basis;

(2) The update must include a list that identifies the current pages of the FSAR following page replacement;

(3) Each replacement page must include both a change indicator for the area changed, e.g., a bold line vertically drawn in the margin adjacent to the portion actually changed, and a page change identification (date of change or change number or both);

(4) The update must include:

(i) A certification by a duly authorized officer of the certificate holder that either the information accurately presents changes made since the previous submittal, or that no such changes were made; and

(ii) An identification of changes made by the certificate holder under the provisions of § 71.175, but not previously submitted to the Commission;

(5) The update must reflect all changes implemented up to a maximum of 6 months before the date of filing;

(6) Updates must be filed every 24 months from the date of issuance of the CoC;

(7) Updates must be filed within 90 days of issuance from the date of an amendment to the CoC; and

(8) The certificate holder shall provide a copy of the updated FSAR to each licensee who is using its Type B(DP) package design.

(d) The updated FSAR must be retained by the certificate holder until the Commission terminates the certificate.

(e) A certificate holder who permanently ceases operation shall provide the updated FSAR to the new certificate holder or to the Commission, in accordance with § 71.1, as appropriate.

18. Appendix A to Part 71 is revised to read as follows:

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 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.

II.(a) For individual radionuclides whose identities are known, but which are not listed in Table A-1, the A_1 and A_2 values contained in Table A-3 may be used. Otherwise, the licensee shall obtain prior Commission approval of the A_1 and A_2 values for radionuclides not listed in Table A-1, before shipping the material.

(b) For individual radionuclides whose identities are known, but which are not listed in Table A-2, the exempt material activity concentration and exempt consignment activity values contained in Table A-3 may be used. Otherwise, the licensee shall obtain prior Commission approval of the exempt material activity concentration and exempt consignment activity values, for radionuclides not listed in Table A-2, before shipping the material.

(c) The licensee shall submit requests for prior approval, described under paragraphs II(a) and II(b) of this Appendix, to the Commission, in accordance with § 71.1 of this part.

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 radionuclide has a half-life either longer than 10 days, or longer than that of the parent radionuclide, 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 radionuclide of that chain. In the case of radioactive decay chains in which any daughter radionuclide has a half-life either longer that 10 days, or greater than that of the parent radionuclide, the parent and those daughter radionuclides shall be considered as mixtures of different radionuclides.

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

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

$$\sum_{I} \frac{B(i)}{A_1(i)} \leq 1$$

Where B(i) is the activity of radionuclide I**,** and $A_1(i)$ is the A_1 value for radionuclide I.

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

$$\sum_{I} \frac{B(i)}{A_2(i)} \leq 1$$

Where B(i) is the activity of radionuclide I**,** and $A_2(i)$ is the A_2 value for radionuclide I.

(c) Alternatively, the 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 for radionuclide I in the mixture**,** and $A_1(i)$ is the appropriate A_1 value for radionuclide I.

(d) Alternatively, the 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 for radionuclide I in the mixture**,** and $A_2(i)$ is the appropriate A_2 value for radionuclide I.

(e) The exempt activity concentration for mixtures of nuclides may be determined as follows:

$$\text{Exempt activity concentration for mixture } = \frac{1}{\sum_I \frac{f(i)}{[A](i)}}$$

Where f(i) is the fraction of activity concentration of radionuclide I in the mixture, and [A] is the activity concentration for exempt material containing radionuclide I.

(f) The activity limit for an exempt consignment for mixtures of radionuclides may be determined as follows:

$$\text{Exempt consignment activity limit for mixture} = \frac{1}{\sum_{i} \frac{f(i)}{A(i)}}$$

Where f(i) is the fraction of activity of radionuclide I in the mixture, and A is the activity limit for exempt consignments for radionuclide 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)	Specific activity (Ci/g)
Ac-225 (a)	Actinium (89)	$8.0X10^{-1}$	$2.2X10^{1}$	$6.0X10^{-3}$	$1.6X10^{-1}$	$2.1X10^{3}$	$5.8X10^{4}$
Ac-227 (a)		$9.0X10^{-1}$	$2.4X10^{1}$	$9.0X10^{-5}$	$2.4X10^{-3}$	2.7	$7.2X10^{1}$
Ac-228		$6.0X10^{-1}$	$1.6X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$8.4X10^{4}$	$2.2X10^{6}$
Ag-105	Silver (47)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$1.1X10^{3}$	$3.0X10^{4}$
Ag-108m (a)		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$9.7X10^{-1}$	$2.6X10^{1}$
Ag-110m (a)		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.8X10^{2}$	$4.7X10^{3}$
Ag-111		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$5.8X10^{3}$	$1.6X10^{5}$
Al-26	Aluminum (13)	$1.0X10^{-1}$	2.7	$1.0X10^{-1}$	2.7	$7.0X10^{-4}$	$1.9X10^{-2}$
Am-241	Americium (95)	$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$1.3X10^{-1}$	3.4
Am-242m (a)		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$3.6X10^{-1}$	$1.0X10^{1}$
Am-243 (a)		5.0	$1.4X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$7.4X10^{-3}$	$2.0X10^{-1}$
Ar-37	Argon (18)	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$3.7X10^{3}$	$9.9X10^{4}$
Ar-39		$2.0X10^{1}$	$5.4X10^{2}$	$4.0X10^{1}$	$1.1X10^{3}$	1.3	$3.4X10^{1}$
Ar-41		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.5X10^{6}$	$4.2X10^{7}$
As-72	Arsenic (33)	$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$6.2X10^{4}$	$1.7X10^{6}$
As-73		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$8.2X10^{2}$	$2.2X10^{4}$
As-74		1.0	$2.7X10^{1}$	$9.0X10^{-1}$	$2.4X10^{1}$	$3.7X10^{3}$	$9.9X10^{4}$
As-76		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$5.8X10^{4}$	$1.6X10^{6}$
As-77		$2.0X10^{1}$	$5.4X10^{2}$	$7.0X10^{-1}$	$1.9X10^{1}$	$3.9X10^{4}$	$1.0X10^{6}$
At-211 (a)	Astatine (85)	$2.0X10^{1}$	$5.4X10^{2}$	$5.0X10^{-1}$	$1.4X10^{1}$	$7.6X10^{4}$	$2.1X10^{6}$
Au-193	Gold (79)	7.0	$1.9X10^{2}$	2.0	$5.4X10^{1}$	$3.4X10^{4}$	$9.2X10^{5}$
Au-194		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$1.5X10^{4}$	$4.1X10^{5}$

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)	Specific activity (Ci/g)
Au-195	Gold (79)	$1.0X10^{1}$	$2.7X10^{2}$	6.0	$1.6X10^{2}$	$1.4X10^{2}$	$3.7X10^{3}$
Au-198		1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$9.0X10^{3}$	$2.4X10^{5}$
Au-199		$1.0X10^{1}$	$2.7X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$7.7X10^{3}$	$2.1X10^{5}$
Ba-131 (a)	Barium (56)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$3.1X10^{3}$	$8.4X10^{4}$
Ba-133		3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	9.4	$2.6X10^{2}$
Ba-133m		$2.0X10^{1}$	$5.4X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.2X10^{4}$	$6.1X10^{5}$
Ba-140 (a)		$5.0X10^{-1}$	$1.4X10^{1}$	$3.0X10^{-1}$	8.1	$2.7X10^{3}$	$7.3X10^{4}$
Be-7	Beryllium (4)	$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{1}$	$5.4X10^{2}$	$1.3X10^{4}$	$3.5X10^{5}$
Be-10		$4.0X10^{1}$	$1.1X10^{3}$	$6.0X10^{-1}$	$1.6X10^{1}$	$8.3X10^{-4}$	$2.2X10^{-2}$
Bi-205	Bismuth (83)	$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$1.5X10^{-3}$	$4.2X10^{4}$
Bi-206		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$3.8X10^{3}$	$1.0X10^{5}$
Bi-207		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	1.9	$5.2X10^{1}$
Bi-210		1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$4.6X10^{3}$	$1.2X10^{5}$
Bi-210m (a)		$6.0X10^{-1}$	$1.6X10^{1}$	$2.0X10^{-2}$	$5.4X10^{-1}$	$2.1X10^{-5}$	$5.7X10^{-4}$
Bi-212 (a)		$7.0X10^{-1}$	$1.9X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$5.4X10^{5}$	$1.5X10^{7}$
Bk-247	Berkelium (97)	8.0	$2.2X10^{2}$	$8.0X10^{-4}$	$2.2X10^{-2}$	$3.8X10^{-2}$	1.0
Bk-249 (a)		$4.0X10^{1}$	$1.1X10^{3}$	$3.0X10^{-1}$	8.1	$6.1X10^{1}$	$1.6X10^{3}$
Br-76	Bromine (35)	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$9.4X10^{4}$	$2.5X10^{6}$
Br-77		3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$2.6X10^{4}$	$7.1X10^{5}$
Br-82		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{4}$	$1.1X10^{6}$
C-11	Carbon (6)	1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.1X10^{7}$	$8.4X10^{8}$
C-14		$4.0X10^{1}$	$1.1X10^{3}$	3.0	$8.1X10^{1}$	$1.6X10^{-1}$	4.5

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)	Specific activity (Ci/g)
Ca-41	Calcium (20)	Unlimited	Unlimited	Unlimited	Unlimited	$3.1X10^{-3}$	$8.5X10^{-2}$
Ca-45		$4.0X10^{1}$	$1.1X10^{3}$	1.0	$2.7X10^{1}$	$6.6X10^{2}$	$1.8X10^{4}$
Ca-47 (a)		3.0	$8.1X10^{1}$	$3.0X10^{-1}$	8.1	$2.3X10^{4}$	$6.1X10^{5}$
Cd-109	Cadmium (48)	$3.0X10^{1}$	$8.1X10^{2}$	2.0	$5.4X10^{1}$	$9.6X10^{1}$	$2.6X10^{3}$
Cd-113m		$4.0X10^{1}$	$1.1X10^{3}$	$5.0X10^{-1}$	$1.4X10^{1}$	8.3	$2.2X10^{2}$
Cd-115 (a)		3.0	$8.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.9X10^{4}$	$5.1X10^{5}$
Cd-115m		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$9.4X10^{2}$	$2.5X10^{4}$
Ce-139	Cerium (58)	7.0	$1.9X10^{2}$	2.0	$5.4X10^{1}$	$2.5X10^{2}$	$6.8X10^{3}$
Ce-141		$2.0X10^{1}$	$5.4X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.1X10^{3}$	$2.8X10^{4}$
Ce-143		$9.0X10^{-1}$	$2.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.5X10^{4}$	$6.6X10^{5}$
Ce-144 (a)		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$1.2X10^{2}$	$3.2X10^{3}$
Cf-248	Californium (98)	$4.0X10^{1}$	$1.1X10^{3}$	$6.0X10^{-3}$	$1.6X10^{-1}$	$5.8X10^{1}$	$1.6X10^{3}$
Cf-249		3.0	$8.1X10^{1}$	$8.0X10^{-4}$	$2.2X10^{-2}$	$1.5X10^{-1}$	4.1
Cf-250		$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{-3}$	$5.4X10^{-2}$	4.0	$1.1X10^{2}$
Cf-251		7.0	$1.9X10^{2}$	$7.0X10^{-4}$	$1.9X10^{-2}$	$5.9X10^{-2}$	1.6
Cf-252 (h)		$1.0X10^{-1}$	2.7	$1.0X10^{-3}$	$2.7X10^{-2}$	$2.0X10^{1}$	$5.4X10^{2}$
Cf-253 (a)		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{-2}$	1.1	$1.1X10^{3}$	$2.9X10^{4}$
Cf-254		$1.0X10^{-3}$	$2.7X10^{-2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$3.1X10^{2}$	$8.5X10^{3}$
Cl-36	Chlorine (17)	$1.0X10^{1}$	$2.7X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.2X10^{-3}$	$3.3X10^{-2}$
Cl-38		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$4.9X10^{6}$	$1.3X10^{8}$
Cm-240	Curium (96)	$4.0X10^{1}$	$1.1X10^{3}$	$2.0X10^{-2}$	$5.4X10^{-1}$	$7.5X10^{2}$	$2.0X10^{4}$
Cm-241		2.0	$5.4X10^{1}$	1.0	$2.7X10^{1}$	$6.1X10^{2}$	$1.7X10^{4}$

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)	Specific activity (Ci/g)
Cm-242	Curium (96)	$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-2}$	$2.7X10^{-1}$	$1.2X10^{2}$	$3.3X10^{3}$
Cm-243		9.0	$2.4X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$1.9X10^{-3}$	$5.2X10^{1}$
Cm-244		$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{-3}$	$5.4X10^{-2}$	3.0	$8.1X10^{1}$
Cm-245		9.0	$2.4X10^{2}$	$9.0X10^{-4}$	$2.4X10^{-2}$	$6.4X10^{-3}$	$1.7X10^{-1}$
Cm-246		9.0	$2.4X10^{2}$	$9.0X10^{-4}$	$2.4X10^{-2}$	$1.1X10^{-2}$	$3.1X10^{-1}$
Cm-247 (a)		3.0	$8.1X10^{1}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$3.4X10^{-6}$	$9.3X10^{-5}$
Cm-248		$2.0X10^{-2}$	$5.4X10^{-1}$	$3.0X10^{-4}$	$8.1X10^{-3}$	$1.6X10^{-5}$	$4.2X10^{-3}$
Co-55	Cobalt (27)	$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$1.1X10^{5}$	$3.1X10^{6}$
Co-56		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.1X10^{3}$	$3.0X10^{4}$
Co-57		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$3.1X10^{2}$	$8.4X10^{3}$
Co-58		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$1.2X10^{3}$	$3.2X10^{4}$
Co-58m		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$2.2X10^{5}$	$5.9X10^{6}$
Co-60		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$4.2X10^{1}$	$1.1X10^{3}$
Cr-51	Chromium (24)	$3.0X10^{1}$	$8.1X10^{2}$	$3.0X10^{1}$	$8.1X10^{2}$	$3.4X10^{3}$	$9.2X10^{4}$
Cs-129	Cesium (55)	4.0	$1.1X10^{2}$	4.0	$1.1X10^{2}$	$2.8X10^{4}$	$7.6X10^{5}$
Cs-131		$3.0X10^{1}$	$8.1X10^{2}$	$3.0X10^{1}$	$8.1X10^{2}$	$3.8X10^{3}$	$1.0X10^{5}$
Cs-132		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$5.7X10^{3}$	$1.5X10^{5}$
Cs-134		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$4.8X10^{1}$	$1.3X10^{3}$
Cs-134m		$4.0X10^{1}$	$1.1X10^{3}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.0X10^{5}$	$8.0X10^{6}$
Cs-135		$4.0X10^{1}$	$1.1X10^{3}$	1.0	$2.7X10^{1}$	$4.3X10^{-5}$	$1.2X10^{-3}$
Cs-136		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$2.7X10^{3}$	$7.3X10^{4}$
Cs-137 (a)		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	3.2	$8.7X10^{1}$

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)	Specific activity (Ci/g)
Cu-64	Copper (29)	6.0	$1.6X10^{2}$	1.0	$2.7X10^{1}$	$1.4X10^{5}$	$3.9X10^{6}$
Cu-67		$1.0X10^{1}$	$2.7X10^{2}$	$7.0X10^{-1}$	$1.9X10^{1}$	$2.8X10^{4}$	$7.6X10^{5}$
Dy-159	Dysprosium (66)	$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{1}$	$5.4X10^{2}$	$2.1X10^{2}$	$5.7X10^{3}$
Dy-165		$9.0X10^{-1}$	$2.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.0X10^{5}$	$8.2X10^{6}$
Dy-166 (a)		$9.0X10^{-1}$	$2.4X10^{1}$	$3.0X10^{-1}$	8.1	$8.6X10^{3}$	$2.3X10^{5}$
Er-169	Erbium (68)	$4.0X10^{1}$	$1.1X10^{3}$	1.0	$2.7X10^{1}$	$3.1X10^{3}$	$8.3X10^{4}$
Er-171		$8.0X10^{-1}$	$2.2X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$9.0X10^{4}$	$2.4X10^{6}$
Eu-147	Europium (63)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$1.4X10^{3}$	$3.7X10^{4}$
Eu-148		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$6.0X10^{2}$	$1.6X10^{4}$
Eu-149		$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{1}$	$5.4X10^{2}$	$3.5X10^{2}$	$9.4X10^{3}$
Eu-150 (short lived)		2.0	$5.4X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$6.1X10^{4}$	$1.6X10^{6}$
Eu-150 (long lived)		2.0	$5.4X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$6.1X10^{4}$	$1.6X10^{6}$
Eu-152		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	6.5	$1.8X10^{2}$
Eu-152m		$8.0X10^{-1}$	$2.2X10^{1}$	$8.0X10^{-1}$	$2.2X10^{1}$	$8.2X10^{4}$	$2.2X10^{6}$
Eu-154		$9.0X10^{-1}$	$2.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	9.8	$2.6X10^{2}$
Eu-155		$2.0X10^{1}$	$5.4X10^{2}$	3.0	$8.1X10^{1}$	$1.8X10^{1}$	$4.9X10^{2}$
Eu-156		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$2.0X10^{3}$	$5.5X10^{4}$
F-18	Fluorine (9)	1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.5X10^{6}$	$9.5X10^{7}$
Fe-52 (a)	Iron (26)	$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$2.7X10^{5}$	$7.3X10^{6}$
Fe-55		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$8.8X10^{1}$	$2.4X10^{3}$
Fe-59		$9.0X10^{-1}$	$2.4X10^{1}$	$9.0X10^{-1}$	$2.4X10^{1}$	$1.8X10^{3}$	$5.0X10^{4}$
Fe-60 (a)		$4.0X10^{1}$	$1.1X10^{3}$	$2.0X10^{-1}$	5.4	$7.4X10^{-4}$	$2.0X10^{-2}$

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)	Specific activity (Ci/g)
Ga-67	Gallium (31)	7.0	$1.9X10^{2}$	3.0	$8.1X10^{1}$	$2.2X10^{4}$	$6.0X10^{5}$
Ga-68		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$1.5X10^{6}$	$4.1X10^{7}$
Ga-72		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.1X10^{5}$	$3.1X10^{6}$
Gd-146 (a)	Gadolinium (64)	$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$6.9X10^{2}$	$1.9X10^{4}$
Gd-148		$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{-3}$	$5.4X10^{-2}$	1.2	$3.2X10^{1}$
Gd-153		$1.0X10^{1}$	$2.7X10^{2}$	9.0	$2.4X10^{2}$	$1.3X10^{2}$	$3.5X10^{3}$
Gd-159		3.0	$8.1X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.9X10^{4}$	$1.1X10^{6}$
Ge-68 (a)	Germanium (32)	$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$2.6X10^{2}$	$7.1X10^{3}$
Ge-71		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$5.8X10^{3}$	$1.6X10^{5}$
Ge-77		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.3X10^{5}$	$3.6X10^{6}$
Hf-172 (a)	Hafnium (72)	$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$4.1X10^{1}$	$1.1X10^{3}$
Hf-175		3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$3.9X10^{2}$	$1.1X10^{4}$
Hf-181		2.0	$5.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$6.3X10^{2}$	$1.7X10^{4}$
Hf-182		Unlimited	Unlimited	Unlimited	Unlimited	$8.1X10^{-6}$	$2.2X10^{-4}$
Hg-194 (a)	Mercury (80)	1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$1.3X10^{-1}$	3.5
Hg-195m (a)		3.0	$8.1X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$1.5X10^{4}$	$4.0X10^{5}$
Hg-197		$2.0X10^{1}$	$5.4X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$9.2X10^{3}$	$2.5X10^{5}$
Hg-197m		$1.0X10^{1}$	$2.7X10^{2}$	$4.0X10^{-1}$	$1.1X10^{1}$	$2.5X10^{4}$	$6.7X10^{5}$
Hg-203		5.0	$1.4X10^{2}$	1.0	$2.7X10^{1}$	$5.1X10^{2}$	$1.4X10^{4}$
Ho-166	Holmium (67)	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$2.6X10^{4}$	$7.0X10^{5}$
Ho-166m		$6.0X10^{-1}$	$1.6X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$6.6X10^{-2}$	1.8

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)	Specific activity (Ci/g)
I-123	Iodine (53)	6.0	$1.6X10^{2}$	3.0	$8.1X10^{1}$	$7.1X10^{4}$	$1.9X10^{6}$
I-124		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$9.3X10^{3}$	$2.5X10^{5}$
I-125		$2.0X10^{1}$	$5.4X10^{2}$	3.0	$8.1X10^{1}$	$6.4X10^{2}$	$1.7X10^{4}$
I-126		2.0	$5.4X10^{1}$	1.0	$2.7X10^{1}$	$2.9X10^{3}$	$8.0X10^{4}$
I-129		Unlimited	Unlimited	Unlimited	Unlimited	$6.5X10^{-6}$	$1.8X10^{-4}$
I-131		3.0	$8.1X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$4.6X10^{3}$	$1.2X10^{5}$
I-132		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$3.8X10^{5}$	$1.0X10^{7}$
I-133		$7.0X10^{-1}$	$1.9X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$4.2X10^{4}$	$1.1X10^{6}$
I-134		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$9.9X10^{5}$	$2.7X10^{7}$
I-135 (a)		$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.3X10^{5}$	$3.5X10^{6}$
In-111	Indium (49)	3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$1.5X10^{4}$	$4.2X10^{5}$
In-113m		4.0	$1.1X10^{2}$	2.0	$5.4X10^{1}$	$6.2X10^{5}$	$1.7X10^{7}$
In-114m (a)		$1.0X10^{1}$	$2.7X10^{2}$	$5.0X10^{-1}$	$1.4X10^{1}$	$8.6X10^{2}$	$2.3X10^{4}$
In-115m		7.0	$1.9X10^{2}$	1.0	$2.7X10^{1}$	$2.2X10^{5}$	$6.1X10^{6}$
Ir-189 (a)	Iridium (77)	$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$1.9X10^{3}$	$5.2X10^{4}$
Ir-190		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$2.3X10^{3}$	$6.2X10^{4}$
Ir-192		1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.4X10^{2}$	$9.2X10^{3}$
Ir-194		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$3.1X10^{4}$	$8.4X10^{5}$
K-40	Potassium (19)	$9.0X10^{-1}$	$2.4X10^{1}$	$9.0X10^{-1}$	$2.4X10^{1}$	$2.4X10^{-7}$	$6.4X10^{-6}$
K-42		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$2.2X10^{5}$	$6.0X10^{6}$
K-43		$7.0X10^{-1}$	$1.9X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.2X10^{5}$	$3.3X10^{6}$

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)	Specific activity (Ci/g)
Kr-81	Krypton (36)	4.0×10^{1}	1.1×10^{3}	4.0×10^{1}	1.1×10^{3}	7.8×10^{-4}	2.1×10^{-2}
Kr-85		1.0×10^{1}	2.7×10^{2}	1.0×10^{1}	2.7×10^{2}	1.5×10^{1}	3.9×10^{2}
Kr-85m		8.0	2.2×10^{2}	3.0	8.1×10^{1}	3.0×10^{5}	8.2×10^{6}
Kr-87		2.0×10^{-1}	5.4	2.0×10^{-1}	5.4	1.0×10^{6}	2.8×10^{7}
La-137	Lanthanum (57)	3.0×10^{1}	8.1×10^{2}	6.0	1.6×10^{2}	1.6×10^{-3}	4.4×10^{-2}
La-140		4.0×10^{-1}	1.1×10^{1}	4.0×10^{-1}	1.1×10^{1}	2.1×10^{4}	5.6×10^{5}
Lu-172	Lutetium (71)	6.0×10^{-1}	1.6×10^{1}	6.0×10^{-1}	1.6×10^{1}	4.2×10^{3}	1.1×10^{5}
Lu-173		8.0	2.2×10^{2}	8.0	2.2×10^{2}	5.6×10^{1}	1.5×10^{3}
Lu-174		9.0	2.4×10^{2}	9.0	2.4×10^{2}	2.3×10^{1}	6.2×10^{2}
Lu-174m		2.0×10^{1}	5.4×10^{2}	1.0×10^{1}	2.7×10^{2}	2.0×10^{2}	5.3×10^{3}
Lu-177		3.0×10^{1}	8.1×10^{2}	7.0×10^{-1}	1.9×10^{1}	4.1×10^{3}	1.1×10^{5}
Mg-28 (a)	Magnesium (12)	3.0×10^{-1}	8.1	3.0×10^{-1}	8.1	2.0×10^{5}	5.4×10^{6}
Mn-52	Manganese (25)	3.0×10^{-1}	8.1	3.0×10^{-1}	8.1	1.6×10^{4}	4.4×10^{5}
Mn-53		Unlimited	Unlimited	Unlimited	Unlimited	6.8×10^{-5}	1.8×10^{-3}
Mn-54		1.0	2.7×10^{1}	1.0	2.7×10^{1}	2.9×10^{2}	7.7×10^{3}
Mn-56		3.0×10^{-1}	8.1	3.0×10^{-1}	8.1	8.0×10^{5}	2.2×10^{7}
Mo-93	Molybdenum (42)	4.0×10^{1}	1.1×10^{3}	2.0×10^{1}	5.4×10^{2}	4.1×10^{-2}	1.1
Mo-99 (a) (h)		1.0	2.7×10^{1}	7.4×10^{-1}	2.0×10^{1}	1.8×10^{4}	4.8×10^{5}
N-13	Nitrogen (7)	9.0×10^{-1}	2.4×10^{1}	6.0×10^{-1}	1.6×10^{1}	5.4×10^{7}	1.5×10^{9}
Na-22	Sodium (11)	5.0×10^{-1}	1.4×10^{1}	5.0×10^{-1}	1.4×10^{1}	2.3×10^{2}	6.3×10^{3}
Na-24		2.0×10^{-1}	5.4	2.0×10^{-1}	5.4	3.2×10^{5}	8.7×10^{6}

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)	Specific activity (Ci/g)
Nb-93m	Niobium (41)	$4.0X10^{1}$	$1.1X10^{3}$	$3.0X10^{1}$	$8.1X10^{2}$	8.8	$2.4X10^{2}$
Nb-94		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$6.9X10^{-3}$	$1.9X10^{-1}$
Nb-95		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$1.5X10^{3}$	$3.9X10^{4}$
Nb-97		$9.0X10^{-1}$	$2.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$9.9X10^{5}$	$2.7X10^{7}$
Nd-147	Neodymium (60)	6.0	$1.6X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.0X10^{3}$	$8.1X10^{4}$
Nd-149		$6.0X10^{-1}$	$1.6X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$4.5X10^{5}$	$1.2X10^{7}$
Ni-59	Nickel (28)	Unlimited	Unlimited	Unlimited	Unlimited	$3.0X10^{-3}$	$8.0X10^{-2}$
Ni-63		$4.0X10^{1}$	$1.1X10^{3}$	$3.0X10^{1}$	$8.1X10^{2}$	2.1	$5.7X10^{1}$
Ni-65		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$7.1X10^{5}$	$1.9X10^{7}$
Np-235	Neptunium (93)	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$5.2X10^{1}$	$1.4X10^{3}$
Np-236 (short-lived)		$2.0X10^{1}$	$5.4X10^{2}$	2.0	$5.4X10^{1}$	$4.7X10^{-4}$	$1.3X10^{-2}$
Np-236 (long-lived)		$2.0X10^{1}$	$5.4X10^{2}$	2.0	$5.4X10^{1}$	$4.7X10^{-4}$	$1.3X10^{-2}$
Np-237		$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{-3}$	$5.4X10^{-2}$	$2.6X10^{-5}$	$7.1X10^{-4}$
Np-239		7.0	$1.9X10^{2}$	$4.0X10^{-1}$	$1.1X10^{1}$	$8.6X10^{3}$	$2.3X10^{5}$
Os-185	Osmium (76)	1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$2.8X10^{2}$	$7.5X10^{3}$
Os-191		$1.0X10^{1}$	$2.7X10^{2}$	2.0	$5.4X10^{1}$	$1.6X10^{3}$	$4.4X10^{4}$
Os-191m		$4.0X10^{1}$	$1.1X10^{3}$	$3.0X10^{1}$	$8.1X10^{2}$	$4.6X10^{4}$	$1.3X10^{6}$
Os-193		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.0X10^{4}$	$5.3X10^{5}$
Os-194 (a)		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.1X10^{1}$	$3.1X10^{2}$
P-32	Phosphorus (15)	$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$1.1X10^{4}$	$2.9X10^{5}$
P-33		$4.0X10^{1}$	$1.1X10^{3}$	1.0	$2.7X10^{1}$	$5.8X10^{3}$	$1.6X10^{5}$

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)	Specific activity (Ci/g)
Pa-230 (a)	Protactinium (91)	2.0	$5.4X10^{1}$	$7.0X10^{-2}$	1.9	$1.2X10^{3}$	$3.3X10^{4}$
Pa-231		4.0	$1.1X10^{2}$	$4.0X10^{-4}$	$1.1X10^{-2}$	$1.7X10^{-3}$	$4.7X10^{-2}$
Pa-233		5.0	$1.4X10^{2}$	$7.0X10^{-1}$	$1.9X10^{1}$	$7.7X10^{2}$	$2.1X10^{4}$
Pb-201	Lead (82)	1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$6.2X10^{4}$	$1.7X10^{6}$
Pb-202		$4.0X10^{1}$	$1.1X10^{3}$	$2.0X10^{1}$	$5.4X10^{2}$	$1.2X10^{-4}$	$3.4X10^{-3}$
Pb-203		4.0	$1.1X10^{2}$	3.0	$8.1X10^{1}$	$1.1X10^{4}$	$3.0X10^{5}$
Pb-205		Unlimited	Unlimited	Unlimited	Unlimited	$4.5X10^{-6}$	$1.2X10^{-4}$
Pb-210 (a)		1.0	$2.7X10^{1}$	$5.0X10^{-2}$	1.4	2.8	$7.6X10^{1}$
Pb-212 (a)		$7.0X10^{-1}$	$1.9X10^{1}$	$2.0X10^{-1}$	5.4	$5.1X10^{4}$	$1.4X10^{6}$
Pd-103 (a)	Palladium (46)	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$2.8X10^{3}$	$7.5X10^{4}$
Pd-107		Unlimited	Unlimited	Unlimited	Unlimited	$1.9X10^{-5}$	$5.1X10^{-4}$
Pd-109		2.0	$5.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$7.9X10^{4}$	$2.1X10^{6}$
Pm-143	Promethium (61)	3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$1.3X10^{2}$	$3.4X10^{3}$
Pm-144		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$9.2X10^{1}$	$2.5X10^{3}$
Pm-145		$3.0X10^{1}$	$8.1X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	5.2	$1.4X10^{2}$
Pm-147		$4.0X10^{1}$	$1.1X10^{3}$	2.0	$5.4X10^{1}$	$3.4X10^{1}$	$9.3X10^{2}$
Pm-148m (a)		$8.0X10^{-1}$	$2.2X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$7.9X10^{2}$	$2.1X10^{4}$
Pm-149		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.5X10^{4}$	$4.0X10^{5}$
Pm-151		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.7X10^{4}$	$7.3X10^{5}$
Po-210	Polonium (84)	$4.0X10^{1}$	$1.1X10^{3}$	$2.0X10^{-2}$	$5.4X10^{-1}$	$1.7X10^{2}$	$4.5X10^{3}$
Pr-142	Praseodymium (59)	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$4.3X10^{4}$	$1.2X10^{6}$
Pr-143		3.0	$8.1X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.5X10^{3}$	$6.7X10^{4}$

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)	Specific activity (Ci/g)
Pt-188 (a)	Platinum (78)	1.0	$2.7X10^{1}$	$8.0X10^{-1}$	$2.2X10^{1}$	$2.5X10^{3}$	$6.8X10^{4}$
Pt-191		4.0	$1.1X10^{2}$	3.0	$8.1X10^{1}$	$8.7X10^{3}$	$2.4X10^{5}$
Pt-193		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	1.4	$3.7X10^{1}$
Pt-193m		$4.0X10^{1}$	$1.1X10^{3}$	$5.0X10^{-1}$	$1.4X10^{1}$	$5.8X10^{3}$	$1.6X10^{5}$
Pt-195m		$1.0X10^{1}$	$2.7X10^{2}$	$5.0X10^{-1}$	$1.4X10^{1}$	$6.2X10^{3}$	$1.7X10^{5}$
Pt-197		$2.0X10^{1}$	$5.4X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.2X10^{4}$	$8.7X10^{5}$
Pt-197m		$1.0X10^{1}$	$2.7X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.7X10^{5}$	$1.0X10^{7}$
Pu-236	Plutonium (94)	$3.0X10^{1}$	$8.1X10^{2}$	$3.0X10^{-3}$	$8.1X10^{-2}$	$2.0X10^{1}$	$5.3X10^{2}$
Pu-237		$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{1}$	$5.4X10^{2}$	$4.5X10^{2}$	$1.2X10^{4}$
Pu-238		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$6.3X10^{-1}$	$1.7X10^{1}$
Pu-239		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$2.3X10^{-3}$	$6.2X10^{-2}$
Pu-240		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$8.4X10^{-3}$	$2.3X10^{-1}$
Pu-241 (a)		$4.0X10^{1}$	$1.1X10^{3}$	$6.0X10^{-2}$	1.6	3.8	$1.0X10^{2}$
Pu-242		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$1.5X10^{-4}$	$3.9X10^{-3}$
Pu-244 (a)		$4.0X10^{-1}$	$1.1X10^{1}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$6.7X10^{-7}$	$1.8X10^{-5}$
Ra-223 (a)	Radium (88)	$4.0X10^{-1}$	$1.1X10^{1}$	$7.0X10^{-3}$	$1.9X10^{-1}$	$1.9X10^{3}$	$5.1X10^{4}$
Ra-224 (a)		$4.0X10^{-1}$	$1.1X10^{1}$	$2.0X10^{-2}$	$5.4X10^{-1}$	$5.9X10^{3}$	$1.6X10^{5}$
Ra-225 (a)		$2.0X10^{-1}$	5.4	$4.0X10^{-3}$	$1.1X10^{-1}$	$1.5X10^{3}$	$3.9X10^{4}$
Ra-226 (a)		$2.0X10^{-1}$	5.4	$3.0X10^{-3}$	$8.1X10^{-2}$	$3.7X10^{-2}$	1.0
Ra-228 (a)		$6.0X10^{-1}$	$1.6X10^{1}$	$2.0X10^{-2}$	$5.4X10^{-1}$	$1.0X10^{1}$	$2.7X10^{2}$

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)	Specific activity (Ci/g)
Rb-81	Rubidium (37)	2.0	$5.4X10^{1}$	$8.0X10^{-1}$	$2.2X10^{1}$	$3.1X10^{5}$	$8.4X10^{6}$
Rb-83 (a)		2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$6.8X10^{2}$	$1.8X10^{4}$
Rb-84		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$1.8X10^{3}$	$4.7X10^{4}$
Rb-86		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$3.0X10^{3}$	$8.1X10^{4}$
Rb-87		Unlimited	Unlimited	Unlimited	Unlimited	$3.2X10^{-9}$	$8.6X10^{-8}$
Rb(nat)		Unlimited	Unlimited	Unlimited	Unlimited	$6.7X10^{6}$	$1.8X10^{8}$
Re-184	Rhenium (75)	1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$6.9X10^{2}$	$1.9X10^{4}$
Re-184m		3.0	$8.1X10^{1}$	1.0	$2.7X10^{1}$	$1.6X10^{2}$	$4.3X10^{3}$
Re-186		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$6.9X10^{3}$	$1.9X10^{5}$
Re-187		Unlimited	Unlimited	Unlimited	Unlimited	$1.4X10^{-9}$	$3.8X10^{-8}$
Re-188		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$3.6X10^{4}$	$9.8X10^{5}$
Re-189 (a)		3.0	$8.1X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.5X10^{4}$	$6.8X10^{5}$
Re(nat)		Unlimited	Unlimited	Unlimited	Unlimited	0.0	$2.4X10^{-8}$
Rh-99	Rhodium (45)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$3.0X10^{3}$	$8.2X10^{4}$
Rh-101		4.0	$1.1X10^{2}$	3.0	$8.1X10^{1}$	$4.1X10^{1}$	$1.1X10^{3}$
Rh-102		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$4.5X10^{1}$	$1.2X10^{3}$
Rh-102m		2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$2.3X10^{2}$	$6.2X10^{3}$
Rh-103m		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$1.2X10^{6}$	$3.3X10^{7}$
Rh-105		$1.0X10^{1}$	$2.7X10^{2}$	$8.0X10^{-1}$	$2.2X10^{1}$	$3.1X10^{4}$	$8.4X10^{5}$
Rn-222 (a)	Radon (86)	$3.0X10^{-1}$	8.1	$4.0X10^{-3}$	$1.1X10^{-1}$	$5.7X10^{3}$	$1.5X10^{5}$

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)	Specific activity (Ci/g)
Ru-97	Ruthenium (44)	5.0	$1.4X10^{2}$	5.0	$1.4X10^{2}$	$1.7X10^{4}$	$4.6X10^{5}$
Ru-103 (a)		2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$1.2X10^{3}$	$3.2X10^{4}$
Ru-105		1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.5X10^{5}$	$6.7X10^{6}$
Ru-106 (a)		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$1.2X10^{2}$	$3.3X10^{3}$
S-35	Sulphur (16)	$4.0X10^{1}$	$1.1X10^{3}$	3.0	$8.1X10^{1}$	$1.6X10^{3}$	$4.3X10^{4}$
Sb-122	Antimony (51)	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.5X10^{4}$	$4.0X10^{5}$
Sb-124		$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$6.5X10^{2}$	$1.7X10^{4}$
Sb-125		2.0	$5.4X10^{1}$	1.0	$2.7X10^{1}$	$3.9X10^{1}$	$1.0X10^{3}$
Sb-126		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$3.1X10^{3}$	$8.4X10^{4}$
Sc-44	Scandium (21)	$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$6.7X10^{5}$	$1.8X10^{7}$
Sc-46		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$1.3X10^{3}$	$3.4X10^{4}$
Sc-47		$1.0X10^{1}$	$2.7X10^{2}$	$7.0X10^{-1}$	$1.9X10^{1}$	$3.1X10^{4}$	$8.3X10^{5}$
Sc-48		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$5.5X10^{4}$	$1.5X10^{6}$
Se-75	Selenium (34)	3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$5.4X10^{2}$	$1.5X10^{4}$
Se-79		$4.0X10^{1}$	$1.1X10^{3}$	2.0	$5.4X10^{1}$	$2.6X10^{-3}$	$7.0X10^{-2}$
Si-31	Silicon (14)	$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.4X10^{6}$	$3.9X10^{7}$
Si-32		$4.0X10^{1}$	$1.1X10^{3}$	$5.0X10^{-1}$	$1.4X10^{1}$	3.9	$1.1X10^{2}$
Sm-145	Samarium (62)	$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$9.8X10^{1}$	$2.6X10^{3}$
Sm-147		Unlimited	Unlimited	Unlimited	Unlimited	$8.5X10^{-1}$	$2.3X10^{-8}$
Sm-151		$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{1}$	$2.7X10^{2}$	$9.7X10^{-1}$	$2.6X10^{1}$
Sm-153		9.0	$2.4X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.6X10^{4}$	$4.4X10^{5}$

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)	Specific activity (Ci/g)
Sn-113 (a)	Tin (50)	4.0	$1.1X10^{2}$	2.0	$5.4X10^{1}$	$3.7X10^{2}$	$1.0X10^{4}$
Sn-117m		7.0	$1.9X10^{2}$	$4.0X10^{-1}$	$1.1X10^{1}$	$3.0X10^{3}$	$8.2X10^{4}$
Sn-119m		$4.0X10^{1}$	$1.1X10^{3}$	$3.0X10^{1}$	$8.1X10^{2}$	$1.4X10^{2}$	$3.7X10^{3}$
Sn-121m (a)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-1}$	$2.4X10^{1}$	2.0	$5.4X10^{1}$
Sn-123		$8.0X10^{-1}$	$2.2X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.0X10^{2}$	$8.2X10^{3}$
Sn-125		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{3}$	$1.1X10^{5}$
Sn-126 (a)		$6.0X10^{-1}$	$1.6X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.0X10^{-3}$	$2.8X10^{-2}$
Sr-82 (a)	Strontium (38)	$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$2.3X10^{3}$	$6.2X10^{4}$
Sr-85		2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$8.8X10^{2}$	$2.4X10^{4}$
Sr-85m		5.0	$1.4X10^{2}$	5.0	$1.4X10^{2}$	$1.2X10^{6}$	$3.3X10^{7}$
Sr-87m		3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$4.8X10^{5}$	$1.3X10^{7}$
Sr-89		$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.1X10^{3}$	$2.9X10^{4}$
Sr-90 (a)		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	5.1	$1.4X10^{2}$
Sr-91 (a)		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.3X10^{5}$	$3.6X10^{6}$
Sr-92 (a)		1.0	$2.7X10^{1}$	$3.0X10^{-1}$	8.1	$4.7X10^{5}$	$1.3X10^{7}$
T(H-3)	Tritium (1)	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$3.6X10^{2}$	$9.7X10^{3}$
Ta-178 (long-lived)	Tantalum (73)	1.0	$2.7X10^{1}$	$8.0X10^{-1}$	$2.2X10^{1}$	$4.2X10^{6}$	$1.1X10^{8}$
Ta-179		$3.0X10^{1}$	$8.1X10^{2}$	$3.0X10^{1}$	$8.1X10^{2}$	$4.1X10^{1}$	$1.1X10^{3}$
Ta-182		$9.0X10^{-1}$	$2.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$2.3X10^{2}$	$6.2X10^{3}$
Tb-157	Terbium (65)	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$5.6X10^{-1}$	$1.5X10^{1}$
Tb-158		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$5.6X10^{-1}$	$1.5X10^{1}$
Tb-160		1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$4.2X10^{2}$	$1.1X10^{4}$

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)	Specific activity (Ci/g)
Tc-95m (a)	Technetium (43)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$8.3X10^{2}$	$2.2X10^{4}$
Tc-96		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.2X10^{4}$	$3.2X10^{5}$
Tc-96m (a)		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.4X10^{6}$	$3.8X10^{7}$
Tc-97		Unlimited	Unlimited	Unlimited	Unlimited	$5.2X10^{-5}$	$1.4X10^{-3}$
Tc-97m		$4.0X10^{1}$	$1.1X10^{3}$	1.0	$2.7X10^{1}$	$5.6X10^{2}$	$1.5X10^{4}$
Tc-98		$8.0X10^{-1}$	$2.2X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$3.2X10^{-5}$	$8.7X10^{-4}$
Tc-99		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-1}$	$2.4X10^{1}$	$6.3X10^{-4}$	$1.7X10^{-2}$
Tc-99m		$1.0X10^{1}$	$2.7X10^{2}$	4.0	$1.1X10^{2}$	$1.9X10^{5}$	$5.3X10^{6}$
Te-121	Tellurium (52)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$2.4X10^{3}$	$6.4X10^{4}$
Te-121m		5.0	$1.4X10^{2}$	3.0	$8.1X10^{1}$	$2.6X10^{2}$	$7.0X10^{3}$
Te-123m		8.0	$2.2X10^{2}$	1.0	$2.7X10^{1}$	$3.3X10^{2}$	$8.9X10^{3}$
Te-125m		$2.0X10^{1}$	$5.4X10^{2}$	$9.0X10^{-1}$	$2.4X10^{1}$	$6.7X10^{2}$	$1.8X10^{4}$
Te-127		$2.0X10^{1}$	$5.4X10^{2}$	$7.0X10^{-1}$	$1.9X10^{1}$	$9.8X10^{4}$	$2.6X10^{6}$
Te-127m (a)		$2.0X10^{1}$	$5.4X10^{2}$	$5.0X10^{-1}$	$1.4X10^{1}$	$3.5X10^{2}$	$9.4X10^{3}$
Te-129		$7.0X10^{-1}$	$1.9X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$7.7X10^{5}$	$2.1X10^{7}$
Te-129m (a)		$8.0X10^{-1}$	$2.2X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.1X10^{3}$	$3.0X10^{4}$
Te-131m (a)		$7.0X10^{-1}$	$1.9X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$3.0X10^{4}$	$8.0X10^{5}$
Te-132 (a)		$5.0X10^{-1}$	$1.4X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.1X10^{4}$	$8.0X10^{5}$
Th-227	Thorium (90)	$1.0X10^{1}$	$2.7X10^{2}$	$5.0X10^{-3}$	$1.4X10^{-1}$	$1.1X10^{3}$	$3.1X10^{4}$
Th-228 (a)		$5.0X10^{-1}$	$1.4X10^{1}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$3.0X10^{1}$	$8.2X10^{2}$
Th-229		5.0	$1.4X10^{2}$	$5.0X10^{-4}$	$1.4X10^{-2}$	$7.9X10^{-3}$	$2.1X10^{-1}$
Th-230		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$7.6X10^{-4}$	$2.1X10^{-2}$

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)	Specific activity (Ci/g)
Th-231	Thorium (90)	$4.0X10^{1}$	$1.1X10^{3}$	$2.0X10^{-2}$	$5.4X10^{-1}$	$2.0X10^{4}$	$5.3X10^{5}$
Th-232		Unlimited	Unlimited	Unlimited	Unlimited	$4.0X10^{-9}$	$1.1X10^{-7}$
Th-234 (a)		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$8.6X10^{2}$	$2.3X10^{4}$
Th(nat)		Unlimited	Unlimited	Unlimited	Unlimited	$8.1X10^{-9}$	$2.2X10^{-7}$
Ti-44 (a)	Titanium (22)	$5.0X10^{-1}$	$1.4X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	6.4	$1.7X10^{2}$
Tl-200	Thallium (81)	$9.0X10^{-1}$	$2.4X10^{1}$	$9.0X10^{-1}$	$2.4X10^{1}$	$2.2X10^{4}$	$6.0X10^{5}$
Tl-201		$1.0X10^{1}$	$2.7X10^{2}$	4.0	$1.1X10^{2}$	$7.9X10^{3}$	$2.1X10^{5}$
Tl-202		2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$2.0X10^{3}$	$5.3X10^{4}$
Tl-204		$1.0X10^{1}$	$2.7X10^{2}$	$7.0X10^{-1}$	$1.9X10^{1}$	$1.7X10^{1}$	$4.6X10^{2}$
Tm-167	Thulium (69)	7.0	$1.9X10^{2}$	$8.0X10^{-1}$	$2.2X10^{1}$	$3.1X10^{3}$	$8.5X10^{4}$
Tm-170		3.0	$8.1X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.2X10^{2}$	$6.0X10^{3}$
Tm-171		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$
U-230 (fast lung absorption) (a)(d)	Uranium (92)	$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-1}$	2.7	$1.0X10^{3}$	$2.7X10^{4}$
U-230 (medium lung absorption) (a)(e)		$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-1}$	2.7	$1.0X10^{3}$	$2.7X10^{4}$
U-230 (slow lung absorption) (a)(f)		$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-1}$	2.7	$1.0X10^{3}$	$2.7X10^{4}$
U-232 (fast lung absorption) (d)		$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-2}$	$2.7X10^{-1}$	$8.3X10^{-1}$	$2.2X10^{1}$
U-232 (medium lung absorption) (e)		$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-2}$	$2.7X10^{-1}$	$8.3X10^{-1}$	$2.2X10^{1}$
U-232 (slow lung absorption) (f)		$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-2}$	$2.7X10^{-1}$	$8.3X10^{-1}$	$2.2X10^{1}$

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)	Specific activity (Ci/g)
U-233 (fast lung absorption) (d)	Uranium (92)	$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-2}$	2.4	$3.6X10^{-4}$	$9.7X10^{-3}$
U-233 (medium lung absorption) (e)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-2}$	2.4	$3.6X10^{-4}$	$9.7X10^{-3}$
U-233 (slow lung absorption) (f)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-2}$	2.4	$3.6X10^{-4}$	$9.7X10^{-3}$
U-234 (fast lung absorption) (d)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-2}$	2.4	$2.3X10^{-4}$	$6.2X10^{-3}$
U-234 (medium lung absorption) (e)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-2}$	2.4	$2.3X10^{-4}$	$6.2X10^{-3}$
U-234 (slow lung absorption) (f)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-2}$	2.4	$2.3X10^{-4}$	$6.2X10^{-3}$
U-235 (all lung absorption types) (a),(d),(e),(f)		Unlimited	Unlimited	Unlimited	Unlimited	$8.0X10^{-8}$	$2.2X10^{-6}$
U-236 (fast lung absorption) (d)		Unlimited	Unlimited	Unlimited	Unlimited	$2.4X10^{-6}$	$6.5X10^{-5}$
U-236 (medium lung absorption) (e)		Unlimited	Unlimited	Unlimited	Unlimited	$2.4X10^{-6}$	$6.5X10^{-5}$
U-236 (slow lung absorption) (f)		Unlimited	Unlimited	Unlimited	Unlimited	$2.4X10^{-6}$	$6.5X10^{-5}$
U-238 (all lung absorption types) (d),(e),(f)		Unlimited	Unlimited	Unlimited	Unlimited	$1.2X10^{-8}$	$3.4X10^{-7}$
U (nat)		Unlimited	Unlimited	Unlimited	Unlimited	$2.6X10^{-8}$	$7.1X10^{-7}$
U (enriched to 20% or less)(g)		Unlimited	Unlimited	Unlimited	Unlimited	N/A	N/A
U (dep)		Unlimited	Unlimited	Unlimited	Unlimited	0.0	(See Table A-3)
V-48	Vanadium (23)	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$6.3X10^{3}$	$1.7X10^{5}$
V-49		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$3.0X10^{2}$	$8.1X10^{3}$

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)	Specific activity (Ci/g)
W-178 (a)	Tungsten (74)	9.0	$2.4X10^{2}$	5.0	$1.4X10^{2}$	$1.3X10^{3}$	$3.4X10^{4}$
W-181		$3.0X10^{1}$	$8.1X10^{2}$	$3.0X10^{1}$	$8.1X10^{2}$	$2.2X10^{2}$	$6.0X10^{3}$
W-185		$4.0X10^{1}$	$1.1X10^{3}$	$8.0X10^{-1}$	$2.2X10^{1}$	$3.5X10^{2}$	$9.4X10^{3}$
W-187		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.6X10^{4}$	$7.0X10^{5}$
W-188 (a)		$4.0X10^{-1}$	$1.1X10^{1}$	$3.0X10^{-1}$	8.1	$3.7X10^{2}$	$1.0X10^{4}$
Xe-122 (a)	Xenon (54)	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$4.8X10^{4}$	$1.3X10^{6}$
Xe-123		2.0	$5.4X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$4.4X10^{5}$	$1.2X10^{7}$
Xe-127		4.0	$1.1X10^{2}$	2.0	$5.4X10^{1}$	$1.0X10^{3}$	$2.8X10^{4}$
Xe-131m		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$3.1X10^{3}$	$8.4X10^{4}$
Xe-133		$2.0X10^{1}$	$5.4X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$6.9X10^{3}$	$1.9X10^{5}$
Xe-135		3.0	$8.1X10^{1}$	2.0	$5.4X10^{1}$	$9.5X10^{4}$	$2.6X10^{6}$
Y-87 (a)	Yttrium (39)	1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$1.7X10^{4}$	$4.5X10^{5}$
Y-88		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$5.2X10^{2}$	$1.4X10^{4}$
Y-90		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$2.0X10^{4}$	$5.4X10^{5}$
Y-91		$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$9.1X10^{2}$	$2.5X10^{4}$
Y-91m		2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$1.5X10^{6}$	$4.2X10^{7}$
Y-92		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$3.6X10^{5}$	$9.6X10^{6}$
Y-93		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.2X10^{5}$	$3.3X10^{6}$
Yb-169	Ytterbium (79)	4.0	$1.1X10^{2}$	1.0	$2.7X10^{1}$	$8.9X10^{2}$	$2.4X10^{4}$
Yb-175		$3.0X10^{1}$	$8.1X10^{2}$	$9.0X10^{-1}$	$2.4X10^{1}$	$6.6X10^{3}$	$1.8X10^{5}$

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)	Specific activity (Ci/g)
Zn-65	Zinc (30)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$3.0X10^{2}$	$8.2X10^{3}$
Zn-69		3.0	$8.1X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.8X10^{6}$	$4.9X10^{7}$
Zn-69m (a)		3.0	$8.1X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.2X10^{5}$	$3.3X10^{6}$
Zr-88	Zirconium (40)	3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$6.6X10^{2}$	$1.8X10^{4}$
Zr-93		Unlimited	Unlimited	Unlimited	Unlimited	$9.3X10^{-5}$	$2.5X10^{-3}$
Zr-95 (a)		2.0	$5.4X10^{1}$	$8.0X10^{-1}$	$2.2X10^{1}$	$7.9X10^{2}$	$2.1X10^{4}$
Zr-97 (a)		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$7.1X10^{4}$	$1.9X10^{6}$

NOTES

(a) A1 and/or A2 values include contributions from daughter nuclides with half-lives less than 10 days

(b) Parent nuclides and their progeny included in secular equilibrium are listed in the following:

Sr-90	Y-90
Zr-93	Nb-93m
Zr-97	Nb-97
Ru-106	Rh-106
Cs-137	Ba-137m
Ce-134	La-134
Ce-144	Pr-144
Ba-140	La-140
Bi-212	Tl-208 (0.36), Po-212 (0.64)
Pb-210	Bi-210, Po-210
Pb-212	Bi-212, Tl-208 (0.36), Po-212 (0.64)
Rn-220	Po-216
Rn-222	Po-218, Pb-214, Bi-214, Po-214
Ra-223	Rn-219, Po-215, Pb-211, Bi-211, Tl-207
Ra-224	Rn-220, Po-216, Pb-212, Bi-212, Tl-208 (0.36), Po-212 (0.64)
Ra-226	Rn-222, Po-218, Pb-214, Bi-214, Po-214, Pb-210, Bi-210, Po-210
Ra-228	Ac-228
Th-226	Ra-222, Rn-218, Po-214
Th-228	Ra-224, Rn-220, Po-216, Pb212, Bi-212, Tl208 (0.36), Po-212 (0.64)
Th-229	Ra-225, Ac-225, Fr-221, At-217, Bi-213, Po-213, Pb-209
Th-nat	Ra-228, Ac-228, Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208 (0.36), Po-212 (0.64)
Th-234	Pa-234m
U-230	Th-226, Ra-222, Rn-218, Po-214
U-232	Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208 (0.36), Po-212 (0.64)
U-235	Th-231

U-238	Th-234, Pa-234m
U-nat	Th-234, Pa-234m, U-234, Th-230, Ra-226, Rn-222, Po-218, Pb-214, Bi-214, Po-214,
U-240	Np-240m
Np-237	Pa-233
Am-242m	Am-242
Am-243	Np-239

(c) The quantity may be determined from a measurement of the rate of decay or a measurement of the radiation level at a prescribed distance from the source.

(d) These values apply only to compounds of uranium that take the chemical form of UF6, UO2F2 and UO2(NO3)2 in both normal and accident conditions of transport.

(e) These values apply only to compounds of uranium that take the chemical form of UO3, UF4, UCl4 and hexavalent compounds in both normal and accident conditions of transport.

(f) These values apply to all compounds of uranium other than those specified in (d) and (e) above.

(g) These values apply to unirradiated uranium only.

(h) These values apply to domestic transport only. For international transport use the values in the table below.

A_1 AND A_2 VALUES FOR RADIONUCLIDES FOR INTERNATIONAL SHIPMENTS

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Cf-252	Californium (98)	$5.0X10^{-2}$	1.4	$3.0X10^{-3}$	$8.1X10^{-2}$	$2.0X10^{1}$	$5.4X10^{2}$
Mo-99 (a)	Molybdenum (42)	1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.8X10^{4}$	$4.8X10^{5}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Ac-225 (a)	Actinium (89)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Ac-227 (a)		$1.0X10^{-1}$	$2.7X10^{-12}$	$1.0X10^{3}$	$2.7X10^{-8}$
Ac-228		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ag-105	Silver (47)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ag-108m (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ag-110m (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ag-111		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Al-26	Aluminum (13)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Am-241	Americium (95)	1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Am-242m (a)		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Am-243 (a)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Ar-37	Argon (18)	$1.0X10^{6}$	$2.7X10^{-5}$	$1.0X10^{8}$	$2.7X10^{-3}$
Ar-39		$1.0X10^{7}$	$2.7X10^{-4}$	$1.0X10^{4}$	$2.7X10^{-7}$
Ar-41		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{9}$	$2.7X10^{-2}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
As-72	Arsenic (33)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
As-73		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
As-74		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
As-76		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
As-77		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
At-211 (a)	Astatine (85)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Au-193	Gold (79)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Au-194		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Au-195		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Au-198		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Au-199		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ba-131 (a)	Barium (56)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ba-133		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ba-133m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ba-140 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Be-7	Beryllium (4)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Be-10		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{6}$	$2.7X10^{-5}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Bi-205	Bismuth (83)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Bi-206		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Bi-207		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Bi-210		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Bi-210m (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Bi-212 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Bk-247	Berkelium (97)	1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Bk-249 (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Br-76	Bromine (35)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Br-77		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Br-82		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
C-11	Carbon (6)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
C-14		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Ca-41	Calcium (20)	$1.0X10^{5}$	$2.7X10^{-6}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ca-45		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ca-47 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cd-109	Cadmium (48)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cd-113m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cd-115 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cd-115m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ce-139	Cerium (58)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ce-141		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ce-143		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ce-144 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cf-248	Californium (98)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cf-249		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Cf-250		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cf-251		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Cf-252		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cf-253 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Cf-254		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Cl-36	Chlorine (17)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cl-38		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cm-240	Curium (96)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cm-241		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cm-242		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cm-243		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cm-244		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cm-245		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Cm-246		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Cm-247 (a)		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cm-248		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Co-55	Cobalt (27)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Co-56		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Co-57		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Co-58		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Co-58m		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Co-60		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cr-51	Chromium (24)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Cs-129	Cesium (55)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cs-131		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cs-132		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cs-134		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cs-134m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cs-135		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Cs-136		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cs-137 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cu-64	Copper (29)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cu-67		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Dy-159	Dysprosium (66)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Dy-165		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Dy-166 (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Er-169	Erbium (68)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Er-171		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Eu-147	Europium (63)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-148		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-149		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Eu-150 (short lived)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-150 (long lived)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-152		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-152 m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-154		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-155		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Eu-156		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
F-18	Fluorine (9)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Fe-52 (a)	Iron (26)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Fe-55		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{6}$	$2.7X10^{-5}$
Fe-59		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Fe-60 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES

Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Ga-67	Gallium (31)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ga-68		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ga-72		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Gd-146 (a)	Gadolinium (64)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Gd-148		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Gd-153		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Gd-159		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ge-68 (a)	Germanium (32)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ge-71		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{8}$	$2.7X10^{-3}$
Ge-77		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Hf-172 (a)	Hafnium (72)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hf-175		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hf-181		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hf-182		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Hg-194 (a)	Mercury (80)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hg-195m (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hg-197		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Hg-197m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hg-203		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ho-166	Holmium (67)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ho-166m		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
I-123	Iodine (53)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
I-124		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
I-125		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
I-126		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
I-129		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
I-131		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
I-132		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
I-133		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
I-134		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
I-135 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
In-111	Indium (49)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
In-113m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
In-114m (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
In-115m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ir-189 (a)	Iridium (77)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ir-190		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ir-192		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Ir-194		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
K-40	Potassium (19)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
K-42		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
K-43		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Kr-81	Krypton (36)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Kr-85		$1.0X10^{5}$	$2.7X10^{-6}$	$1.0X10^{4}$	$2.7X10^{-7}$
Kr-85m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{10}$	$2.7X10^{-1}$
Kr-87		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{9}$	$2.7X10^{-2}$
La-137	Lanthanum (57)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
La-140		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Lu-172	Lutetium (71)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Lu-173		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Lu-174		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Lu-174m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Lu-177		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Mg-28 (a)	Magnesium (12)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Mn-52	Manganese (25)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Mn-53		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{9}$	$2.7X10^{-2}$
Mn-54		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Mn-56		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Mo-93	Molybdenum (42)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{8}$	$2.7X10^{-3}$
Mo-99 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
N-13	Nitrogen (7)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{9}$	$2.7X10^{-2}$
Na-22	Sodium (11)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Na-24		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Nb-93m	Niobium (41)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Nb-94		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Nb-95		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Nb-97		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Nd-147	Neodymium (60)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Nd-149		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ni-59	Nickel (28)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{8}$	$2.7X10^{-3}$
Ni-63		$1.0X10^{5}$	$2.7X10^{-6}$	$1.0X10^{8}$	$2.7X10^{-3}$
Ni-65		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Np-235	Neptunium (93)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Np-236 (short-lived)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Np-236 (long-lived)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Np-237		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Np-239		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Os-185	Osmium (76)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Os-191		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Os-191m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Os-193		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Os-194 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
P-32	Phosphorus (15)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{5}$	$2.7X10^{-6}$
P-33		$1.0X10^{5}$	$2.7X10^{-6}$	$1.0X10^{8}$	$2.7X10^{-3}$
Pa-230 (a)	Protactinium (91)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pa-231		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Pa-233		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pb-201	Lead (82)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pb-202		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pb-203		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pb-205		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pb-210 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Pb-212 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Pd-103 (a)	Palladium (46)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{8}$	$2.7X10^{-3}$
Pd-107		$1.0X10^{5}$	$2.7X10^{-6}$	$1.0X10^{8}$	$2.7X10^{-3}$
Pd-109		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pm-143	Promethium (61)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pm-144		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pm-145		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pm-147		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pm-148m (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pm-149		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pm-151		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Po-210	Polonium (84)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Pr-142	Praseodymium (59)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Pr-143		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{6}$	$2.7X10^{-5}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Pt-188 (a)	Platinum (78)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pt-191		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pt-193		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pt-193m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pt-195m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pt-197		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pt-197m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pu-236	Plutonium (94)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Pu-237		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pu-238		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Pu-239		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Pu-240		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Pu-241 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Pu-242		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Pu-244 (a)		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Ra-223 (a)	Radium (88)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ra-224 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ra-225 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ra-226 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Ra-228 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Rb-81	Rubidium (37)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Rb-83 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Rb-84		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Rb-86		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Rb-87		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Rb(nat)		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Re-184	Rhenium (75)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Re-184m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Re-186		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Re-187		$1.0X10^{6}$	$2.7X10^{-5}$	$1.0X10^{9}$	$2.7X10^{-2}$
Re-188		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Re-189 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Re(nat)		$1.0X10^{6}$	$2.7X10^{-5}$	$1.0X10^{9}$	$2.7X10^{-2}$
Rh-99	Rhodium (45)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Rh-101		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Rh-102		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Rh-102m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Rh-103m		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{8}$	$2.7X10^{-3}$
Rh-105		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Rn-222 (a)	Radon (86)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{8}$	$2.7X10^{-3}$
Ru-97	Ruthenium (44)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ru-103 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ru-105		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ru-106 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
S-35	Sulphur (16)	$1.0X10^{5}$	$2.7X10^{-6}$	$1.0X10^{8}$	$2.7X10^{-3}$
Sb-122	Antimony (51)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{4}$	$2.7X10^{-7}$
Sb-124		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sb-125		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sb-126		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Sc-44	Scandium (21)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Sc-46		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sc-47		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sc-48		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Se-75	Selenium (34)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Se-79		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Si-31	Silicon (14)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Si-32		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sm-145	Samarium (62)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Sm-147		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Sm-151		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{8}$	$2.7X10^{-3}$
Sm-153		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Sn-113 (a)	Tin (50)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Sn-117m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sn-119m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Sn-121m (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Sn-123		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sn-125		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Sn-126 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Sr-82 (a)	Strontium (38)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Sr-85		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sr-85m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Sr-87m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sr-89		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sr-90 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{4}$	$2.7X10^{-7}$
Sr-91 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Sr-92 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
T(H-3)	Tritium (1)	$1.0X10^{6}$	$2.7X10^{-5}$	$1.0X10^{9}$	$2.7X10^{-2}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Ta-178 (long-lived)	Tantalum (73)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ta-179		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ta-182		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Tb-157	Terbium (65)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Tb-158		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tb-160		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tc-95m (a)	Technetium (43)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tc-96		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tc-96m (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Tc-97		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{8}$	$2.7X10^{-3}$
Tc-97m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Tc-98		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tc-99		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Tc-99m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Te-121	Tellurium (52)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Te-121m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Te-123m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Te-125m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Te-127		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Te-127m (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Te-129		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Te-129m (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Te-131m (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Te-132 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Th-227	Thorium (90)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Th-228 (a)		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Th-229		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Th-230		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Th-231		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Th-232		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Th-234 (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{5}$	$2.7X10^{-6}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Th (nat)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Ti-44 (a)	Titanium (22)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Tl-200	Thallium (81)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tl-201		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tl-202		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tl-204		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{4}$	$2.7X10^{-7}$
Tm-167	Thulium (69)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tm-170		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tm-171		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{8}$	$2.7X10^{-3}$
U-230 (fast lung absorption) (a)(d)	Uranium (92)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
U-230 (medium lung absorption) (a)(e)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
U-230 (slow lung absorption) (a)(f)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
U-232 (fast lung absorption) (d)	Uranium (92)	1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
U-232 (medium lung absorption) (e)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
U-232 (slow lung absorption) (f)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
U-233 (fast lung absorption) (d)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-233 (medium lung absorption) (e)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-233 (slow lung absorption) (f)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-234 (fast lung absorption) (d)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-234 (medium lung absorption) (e)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-234 (slow lung absorption) (f)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-235 (all lung absorption types) (a),(d),(e),(f)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
U-236 (fast lung absorption) (d)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-236 (medium lung absorption) (e)	Uranium (92)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-236 (slow lung absorption) (f)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-238 (all lung absorption types) (d),(e),(f)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U (nat)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
U (enriched to 20% or less)(g)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
U (dep)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
V-48	Vanadium (23)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
V-49		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
W-178 (a)	Tungsten (74)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
W-181		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
W-185		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
W-187		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
W-188 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Xe-122 (a)	Xenon (54)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{9}$	$2.7X10^{-2}$
Xe-123		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{9}$	$2.7X10^{-2}$
Xe-127		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{5}$	$2.7X10^{-6}$
Xe-131m		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{4}$	$2.7X10^{-7}$
Xe-133		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{4}$	$2.7X10^{-7}$
Xe-135		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{10}$	$2.7X10^{-1}$
Y-87 (a)	Yttrium (39)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Y-88		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Y-90		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{5}$	$2.7X10^{-6}$
Y-91		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Y-91m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Y-92		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Y-93		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Yb-169	Ytterbium (79)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Yb-175		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Zn-65	Zinc (30)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Zn-69		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{6}$	$2.7X10^{-5}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Zn-69m (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Zr-88	Zirconium (40)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Zr-93		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Zr-95 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Zr-97 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$

NOTES

(a) A1 and/or A2 values include contributions from daughter nuclides with half-lives less than 10 days

(b) Parent nuclides and their progeny included in secular equilibrium are listed in the following:

Sr-90	Y-90
Zr-93	Nb-93m
Zr-97	Nb-97
Ru-106	Rh-106
Cs-137	Ba-137m
Ce-134	La-134
Ce-144	Pr-144
Ba-140	La-140
Bi-212	Tl-208 (0.36), Po-212 (0.64)
Pb-210	Bi-210, Po-210
Pb-212	Bi-212, Tl-208 (0.36), Po-212 (0.64)
Rn-220	Po-216
Rn-222	Po-218, Pb-214, Bi-214, Po-214
Ra-223	Rn-219, Po-215, Pb-211, Bi-211, Tl-207
Ra-224	Rn-220, Po-216, Pb-212, Bi-212, Tl-208 (0.36), Po-212 (0.64)
Ra-226	Rn-222, Po-218, Pb-214, Bi-214, Po-214, Pb-210, Bi-210, Po-210
Ra-228	Ac-228
Th-226	Ra-222, Rn-218, Po-214
Th-228	Ra-224, Rn-220, Po-216, Pb212, Bi-212, Tl208 (0.36), Po-212 (0.64)
Th-229	Ra-225, Ac-225, Fr-221, At-217, Bi-213, Po-213, Pb-209
Th-nat	Ra-228, Ac-228, Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208 (0.36), Po-212 (0.64)
Th-234	Pa-234m
U-230	Th-226, Ra-222, Rn-218, Po-214
U-232	Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208 (0.36), Po-212 (0.64)

U-235	Th-231
U-238	Th-234, Pa-234m
U-nat	Th-234, Pa-234m, U-234, Th-230, Ra-226, Rn-222, Po-218, Pb-214, Bi-214, Po-214,
U-240	Np-240m
Np-237	Pa-233
Am-242m	Am-242
Am-243	Np-239

(c) The quantity may be determined from a measurement of the rate of decay or a measurement of the radiation level at a prescribed distance from the source.

(d) These values apply only to compounds of uranium that take the chemical form of UF6, UO2F2 and UO2(NO3)2 in both normal and accident conditions of transport.

(e) These values apply only to compounds of uranium that take the chemical form of UO3, UF4, UCl4 and hexavalent compounds in both normal and accident conditions of transport.

(f) These values apply to all compounds of uranium other than those specified in (d) and (e) above.

(g) These values apply to unirradiated uranium only.

TABLE A-3: GENERAL VALUES FOR A_1 AND A_2

Contents	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limits for exempt consignments (Bq)	Activity limits for exempt consignments (Ci)
Only beta or gamma emitting radionuclides are known to be present	1×10^{-1}	2.7×10^{0}	2×10^{-2}	5.4×10^{-1}	1×10^{1}	2.7×10^{-10}	1×10^{4}	2.7×10^{-7}
Only alpha emitting radionuclides are known to be present	2×10^{-1}	5.4×10^{0}	9×10^{-5}	2.4×10^{-3}	1×10^{-1}	2.7×10^{-12}	1×10^{3}	2.7×10^{-8}
No relevant data are available	1×10^{-3}	2.7×10^{-2}	9×10^{-5}	2.4×10^{-3}	1×10^{-1}	2.7×10^{-12}	1×10^{3}	2.7×10^{-8}

TABLE A-4: ACTIVITY-MASS RELATIONSHIPS FOR URANIUM

Uranium Enrichment[1] wt % U-235 present	Specific Activity	
	TBq/g	Ci/g
0.45	1.8×10^{-8}	5.0×10^{-7}
0.72	2.6×10^{-8}	7.1×10^{-7}
1.0	2.8×10^{-8}	7.6×10^{-7}
1.5	3.7×10^{-8}	1.0×10^{-6}
5.0	1.0×10^{-7}	2.7×10^{-6}
10.0	1.8×10^{-7}	4.8×10^{-6}
20.0	3.7×10^{-7}	1.0×10^{-5}
35.0	7.4×10^{-7}	2.0×10^{-5}
50.0	9.3×10^{-7}	2.5×10^{-5}
90.0	2.2×10^{-6}	2.8×10^{-5}
93.0	2.6×10^{-6}	7.0×10^{-5}
95.0	3.4×10^{-6}	9.1×10^{-5}

[1] The figures for uranium include representative values for the activity of the uranium-234 that is concentrated during the enrichment process.

Dated at Rockville, Maryland, this _________day of _____________, 2001.

For the Nuclear Regulatory Commission.

____
Annette L. Vietti-Cook,
Secretary for the Commission