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Draft Environmental Assessment of Major Revision of 10 CFR Part 71

Proposed Rule

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ABSTRACT

This report presents the environmental assessment of the Nuclear Regulatory Commission's (NRC or Commission) rulemaking that would modify 10 CFR Part 71 requirements pertaining to the packaging and transport of radioactive materials, including fissile materials. The rulemaking is intended to: (1) harmonize transportation regulations found in 10 CFR Part 71 with the most recent transportation standards established by the International Atomic Energy Agency (IAEA), and the U.S. Department of Transportation's (DOT) requirements at 49 CFR; and (2) address the Commission's goals for risk-informed regulations and eliminating inconsistencies between Part 71 and other parts of 10 CFR. The purpose of this assessment is to evaluate the potential environmental, health, and safety impacts associated with the proposed regulatory changes as required by the National Environmental Policy Act (NEPA). This report includes: (1) a summary of the findings, (2) a discussion of the regulatory options analyzed, (3) an assessment of the estimated values and impacts identified for each regulatory option, (4) a rationale for the determination of the preferred option, and (5) supplementary information and analyses used in the development of this report. Based on this analysis, none of the 19 potential changes evaluated are expected to result in significant environmental impact.

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ABBREVIATIONS

ANI	Authorized Nuclear Inspector
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
Bq	Becquerel
CFR	Code of Federal Regulations
Ci	Curie
CoC	Certificate of Compliance
CRP	Coordinated Research Project
CSI	Criticality Safety Index
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
g	Gram
GSA	U.S. General Services Administration
HLW	High Level Waste
IAEA	International Atomic Energy Agency
ICC	Interstate Commerce Commission
INEEL	Idaho National Engineering and Environmental Laboratory
ISFSI	Independent Spent Fuel Storage Installation
LDM	Low Dispersible Material
LSA-III	Low Specific Activity
MOU	Memorandum of Understanding
NMSS	U.S. NRC Office of Nuclear Material Safety and Safeguards
NON	Notice of Non-compliance
NORM	Naturally Occurring Radioactive Material
NOV	Notice of Violation
NRC	U.S. Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Publication
ORNL	Oak Ridge National Laboratory
PE	Licensed Professional Engineer
PGE	Portland General Electric
PRM	Petition for Rulemaking
QA	Quality Assurance
Rem	Roentgen Equivalent Man
SI	Systeme` Internationale
SMAC	Shipment Mobility/Accountability Collection
SSC	Systems, Structures, and Components
Sv	Sievert
TI	Transport Index
TS-R-1	IAEA Safe Transportation Standards
μCi/g	Microcuries per gram
UF_6	Uranium Hexafluoride
U.S.	United States
USEC	United States Enrichment Company

GLOSSARY

A_1 means the maximum activity of special form radioactive material permitted in a Type A package. These values are listed in Appendix A or Table A-1 of 10 CFR Part 71 and may be derived in accordance with the procedure prescribed in Appendix A of 10 CFR Part 71.

A_2 means the maximum activity of radioactive material, other than special form, LSA and SCO material, permitted in a Type A package. These values are listed in Appendix A or Table A-1 of 10 CFR Part 71 and may be derived in accordance with the procedure prescribed in Appendix A of 10 CFR Part 71.

Becquerel means the special unit of activity in the SI system, equal to 1 disintegration per second.

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

Committed dose equivalent means the total dose equivalent (averaged over a given tissue) deposited over the 50-year period following the intake of a radionuclide.

Committed effective dose equivalent means the weighted sum of committed dose equivalents to specific organs and tissues, in analogy to the effective dose equivalent.

Consignee means any person, organization, or government which receives a consignment.

Consignment means any package or packages, or load of radioactive material, presented by a consignor for transport.

Consignor means any person, organization, or government which prepares a consignment for transport, and is named as consignor in the transport documents.

Conveyance means any vehicle for transport by road or rail, any vessel for transport by water, and any aircraft for transport by air.

Criticality Safety Index means a number which is used to provide control over the accumulation of packages, overpacks, or freight containers containing fissile material.

Curie means the unit of radioactivity, equal to the amount of a radioactive isotope that decays at the rate of $3.7x10^{10}$ disintegrations per second.

Dose equivalent means the product of the absorbed radiation dose, the quality factor for the particular kind of radioactivity absorbed, and any other modifying factors. The SI unit of dose equivalent is the sievert (Sv) and the English or conventional unit is the rem.

Effective dose equivalent means the sum over specified tissues of the products of the dose equivalent in a tissue or organ and the weighting factor for that tissue or organ.

Exclusive use means 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.

Exempt packages means packages exempt from the requirements of 10 CFR Part 71.

Fissile material means plutonium-238, plutonium-239, plutonium-241, uranium-233, uranium-235, or any combination of these radionuclides. Unirradiated natural uranium and depleted uranium, and natural uranium or depleted uranium that has been irradiated in thermal reactors only are not included in this definition. Certain exclusions from fissile material controls are provided in 10 CFR Part 71.53.

Licensed material means by-product, source, or special nuclear material received, possessed, used, or transferred under a general or specific license issued by NRC pursuant to 10 CFR Part 71.

Low dispersible radioactive material means either a solid radioactive material or a solid radioactive material in a sealed capsule, that has limited dispersibility and is not in powder form.

Low Specific Activity (LSA) material means radioactive material with limited specific activity that satisfies the descriptions and limits set forth in 10 CFR Part 71.4. Shielding materials surrounding the LSA material may not be considered in determining the estimated average specific activity of the package contents.

Non-special form (or normal form) radioactive material means radioactive material that has not been demonstrated to qualify as “special form radioactive material,” as defined below.

Q system is a series of models to consider radiation exposure routes to persons in the vicinity of a package involved in a hypothetical severe transport accident. The five models are for external photon does, external beta dose, inhalation dose, skin and ingestion dose due to contamination transfer, and submersion in gaseous isotopes dose.

Radioactive material means any material having a specific activity greater than 70 Bq per gram (0.002 microcurie per gram).

Radionuclide means the type of atom specified by its atomic number, atomic mass, and energy state that exhibits radioactivity.

Special arrangement means those provisions, approved by the competent authority, under which consignments which do not satisfy all the applicable requirements may be transported.

Special form radioactive material means either an indispersible solid radioactive material or a sealed capsule containing radioactive material.

Specific activity of a radionuclide means the activity 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 activity per unit mass of the material.

Surface contaminated object (SCO) means a solid object which is not itself radioactive, but which has radioactive material distributed on its surfaces.

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 TI is determined as specified in 10 CFR Part 71.4.

Type A package means a packaging that, together with its radioactive contents limited to A_1 or A_2 as appropriate, meets the requirements of 49 CFR 173.410 and 173.412, and is designed to retain the integrity of containment and shielding required by this part under normal conditions of transport.

Type B package means a Type B packaging together with its radioactive contents. A type B package design is designated by NRC as B(U) unless the package has a maximum normal operating pressure of more than 700 kPa (100 lb/in2) gauge or a pressure relief device that would allow the release of radioactive material to the environment under tests specified in 10 CFR Part 71.73, 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. To determine this distinction see DOT regulations in 49 CFR Part 173.

Type C package means a new package type described in IAEA's ST-1 that could withstand severe accident conditions in air transport without loss of containment or increase in external radiation.

EXECUTIVE SUMMARY

This document presents the Environmental Assessment of the U.S. Nuclear Regulatory Commission's (NRC's) proposed rulemaking that would modify Title 10 of the Code of Federal Regulations, Part 71 (10 CFR Part 71) requirements pertaining to the packaging and transport of radioactive materials, including fissile materials. The rulemaking is intended to:

(1) Harmonize transportation regulations found in 10 CFR Part 71 with the most recent transportation standards established by the IAEA (*Regulations for the Safe Transport of Radioactive Material*, IAEA Safety Standards Series No. TS-R-1, June 2000), and the DOT requirements at 49 CFR; and

(2) Address the Commission's goals for risk-informed regulations and eliminate inconsistencies between Part 71 and other parts of 10 CFR.

The intended effects of the regulatory action are to develop a level of consistency with other regulatory agencies, and to implement other NRC-initiated changes needed to simplify the regulations applicable to licensees shipping radioactive materials, while maintaining adequate protection of public health, safety, and the environment. The rulemaking would accomplish these objectives by adopting a number of requirements that are consistent with the transportation standards contained in IAEA's TS-R-1, implementing other non-IAEA related changes, and implementing a number of recommendations contained in NUREG/CR-5342 (*Assessment and Recommendations for Fissile-Material Packaging Exemptions and General Licenses Within 10 CFR Part 71,* Oak Ridge National Laboratory, July 1998). The proposed rulemaking addresses a total of 19 issues.

Table ES-1 provides a summary of the 19 individual issues described in Chapter 3 and evaluated in Chapter 4 of this document. For each issue, the expected net impact, both positive and negative, to public health, safety, and the environment, of the options is summarized. In the paragraphs that follow this table, further description of the expected impacts of the options is provided. Chapters 3 and 4 provide additional detail on the specific changes and associated public health, safety, and environmental impacts.

For the purpose of this analysis, these 19 different changes to Part 71 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 and are not considered further in this environmental assessment:

- Changes to Various Definitions in 10 CFR 71.4;
- Expansion of Part 71 Quality Assurance Requirements to Certificate of Compliance (CoC) Holders;
- Change Authority; and
- Modifications of Event Reporting Requirements.

Table ES-1: Summary of Expected Environmental Impacts

Technical Issue	Expected Environmental Impacts
1. Changing Part 71 to the International System of Units (SI) Only	No Negative Impacts - Slight Benefit
2. Radionuclide Exemption Values	Minor Impacts and Benefits
3. Revision of A_1 and A_2	No Negative Impacts - Slight Benefit
4. Uranium Hexafluoride Package Requirements	Slight Net Benefit
5. Introduction of the Criticality Safety Index Requirements	No Negative Impacts - Slight Benefit
6. Type C Packages and Low Dispersible Material	Minor Impacts and Benefits
7. Deep Immersion Test	Slight Net Benefit
8. Grandfathering Previously Approved Packages	No Negative Impacts - Slight Benefit
9. Changes to Various Definitions	Categorically Excluded
10. Crush Test for Fissile Material Package Design	Slight Net Benefit
11. Fissile Material Package Designs for Transport by Aircraft	Slight Net Benefit
12. Special Package Authorizations	No Negative Impacts - Slight Benefit
13. Expansion of Part 71 Quality Assurance Requirements to Certificate of Compliance (CoC) Holders	Categorically Excluded
14. Adoption of ASME Code	Slight Net Benefit
15. Change Authority	Categorically Excluded
16. Fissile Material Exemptions and General License Provisions (17 recommendations)	Mixed Impacts - Dependent on the specific recommendation
17. Double Containment of Plutonium (PRM-71-12)	Slight Net Benefit
18. Contamination Limits as Applied to Spent Fuel and High Level Waste (HLW) Packages	Not Evaluated
19. Modifications of Event Reporting Requirements	Categorically Excluded

None of the remaining 15 changes, which are described and evaluated in turn in the remainder of this report, are expected to cause a significant impact to human health, safety, or the environment, whether promulgated individually or together. 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 the latest international standards:

- Changing Part 71 to the International System of Units (SI) Only (see Sections 3.1.1 and 4.2.1);
- Revision of A_1 and A_2 (see Sections 3.1.3 and 4.2.3);
- A new requirement to display the Criticality Safety Index on shipping packages of fissile material (see Sections 3.1.5 and 4.2.5);
- A provision to "grandfather" older shipping packages under the Part 71 requirements in existence when their Certificates of Compliance (CoC) were issued (see Sections 3.1.8 and 4.2.8); and
- Procedures for approval of special arrangements for shipment of special packages (see Sections 3.2.1 and 4.3.1).

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

- Addition of uranium hexafluoride package requirements (see Sections 3.1.4 and 4.2.4);
- Strengthening the requirements in 10 CFR 71.61 to ensure package containment in deep submersion scenarios (see Sections 3.1.7 and 4.2.7);
- Adoption of the crush test for fissile material package design (see Sections 3.1.10 and 4.2.9);
- Adoption of fissile material package design requirements for transport by aircraft (see Sections 3.1.11 and 4.2.10); and
- Adoption of the ASME Code for spent fuel transportation casks (see Sections 3.2.3 and 4.3.2).

Radionuclide Exemption Values. As described in Sections 3.1.2 and 4.2.2, changing the existing 70 Bq/g (0.002 μCi/g) level in 10 CFR 71.10(a) for exempting any radionuclide from the Part 71 requirements to radionuclide-specific activity limits would result in 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 has judged that this change would not significantly increase the risk to individuals.

Type C Packages and Low Level Dispersible Material. The addition of the Type C package and low level dispersible material concepts (see Sections 3.1.6 and 4.2.6) 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.

Fissile Material Exemptions and General License Provisions. Changes to transportation regulations for fissile materials actually consist of 17 individual recommendations for revisions to Part 71, as discussed in Sections 3.2.5 and 4.3.3. 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.

Double Containment of Plutonium (PRM-71-12). Changes to the requirements for plutonium shipments in section 71.63 could result in a slight increase in the probability and consequences of accidental releases, primarily when and if plutonium is shipped in liquid form (see Sections 3.2.6 and 4.3.4). 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.

Contamination Limits Applied to Spent Fuel and High Level Waste (HLW) Packages. No options have been identified for the issue related to contamination limits as applied to spent fuel and high level waste. 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.

1. Introduction

The U.S. Nuclear Regulatory Commission (NRC or Commission) has initiated a proposed rulemaking to: (1) conform its transportation regulations in Title 10 of the Code of Federal Regulations (CFR), Part 71 ("Packaging and Transport of Radioactive Material") with the transportation regulations established by the International Atomic Energy Agency (IAEA) in TS-R-1; and (2) address the Commission's goals for risk-informed regulations and eliminating inconsistencies with other regulatory approaches.

This document presents ICF's Environmental Assessment of the regulatory options being considered by NRC. This document presents the Environmental Assessment of the regulatory options being considered by NRC. The purpose of this assessment is to evaluate the potential environmental, health, and safety impacts associated with the proposed regulatory changes as required by the National Environmental Policy Act (NEPA). The remainder of this introduction provides background information on the existing set of radioactive material transport regulations and outlines the organization of the document.

1.1 Background

As part of its mission to regulate the domestic use of byproduct, source, and special nuclear materials to ensure adequate protection of health and safety and the environment, the NRC is responsible for controlling the transport of radioactive materials. NRC shares responsibility for radioactive material transport with the U.S. Department of Transportation (DOT). DOT's regulations in 49 CFR Parts 171 through 180 (often called the "Hazmat Regulations") address packaging, shipper and carrier responsibilities, documentation, and radioactivity limits. In contrast, NRC's regulations in 10 CFR Part 71 are primarily concerned with special packaging requirements for large quantities of radioactive materials. A Memorandum of Understanding (MOU) published July 2, 1979 (44 FR 38690) specifies the roles of DOT and NRC in the regulation of the transportation of radioactive materials. The MOU outlines that DOT is responsible for regulating safety in transportation of all hazardous materials, including radioactive materials, whereas the NRC is responsible for regulating safety in receipt, possession, use, and transfer of byproduct, source, and special nuclear materials. This joint regulatory system protects health and safety and the environment by setting performance standards for the packages and by setting limits on the radioactive contents and radiation levels for packages and vehicles.

Before NRC and DOT began regulating the transportation of radioactive materials, the Interstate Commerce Commission (ICC) established the first regulations governing the safe shipment of radioactive materials, during the 1950s.[1] In 1961, partially based on regulations similar to those of the ICC, the International Atomic Energy Agency (IAEA) adopted regulations for the transport of radioactive materials. The IAEA recommended that these regulations, which appeared in Safety Series No. 6 (SS-6), be adopted by Member States and international organizations. After the initial harmonization of international and U.S. standards with the IAEA regulations, four comprehensive revisions to SS-6 were published in 1964, 1967, 1973, and 1985.

The revision of the IAEA transport regulations in 1967 led to a revision of the DOT Hazmat Regulations in 1968. This revision was also the basis for a major revision to the NRC's

[1] Grella, A. " Summary of the Regulations Governing Transport of Radioactive Materials in the USA." RAMTRANS, Volume 9. No.4, pp. 279-292 (1999).

transport regulations. In 1973, additional revisions were made to the international regulations to include a new system for classifying radionuclides. DOT and NRC adopted these revisions in 1983. In 1985, the IAEA issued a comprehensive revision of SS-6 that was later reprinted in 1990 with minor revisions.[2]

In 1995 (60 FR 50248, September 28, 1995), the NRC published a final rule amending the regulations in 10 CFR Part 71 in order to conform with the 1985 (as amended in 1990) revision of the IAEA transportation standards. The IAEA has since published a revised version of its regulations, "Regulations for the Safe Transport of Radioactive Materials, 1996 Edition, No. ST-1," in December 1996. NRC is currently working to harmonize 10 CFR Part 71 with the latest IAEA ST-1 transportation standards. At the same time, NRC is considering additional Part 71 changes to address other issues that have come up during the course of implementing the existing regulations.

On June 28, 2000, the Commission directed the staff in SRM-SECY-00-0117 to both use an enhanced-public-participation process (web-site and facilitated public meetings) to solicit public input in the 10 CFR Part 71 rulemaking; and also to publish, for public comment, the staff's Part 71 issue paper in the Federal Register (65 FR 44360, July 17, 2000). The issue paper discussed the NRC's plan to revise 10 CFR Part 71 and provided a summary of the changes being considered, both IAEA-related changes and Non-IAEA changes. The NRC published the Part 71 issue paper to begin an enhanced-public-participation process designed to solicit public input on the Part 71 upcoming changes. In addition to publication of the issue paper, this process included establishing an interactive web-site and holding three facilitated public meetings: a "roundtable" workshop with invited stakeholders and the general public at the NRC Headquarters, Rockville, MD, on August 10, 2000, and two "townhall" meetings, one in Atlanta, GA, on September 20, 2000, and one in Oakland, CA, on September 26,2000.

SRM-SECY-00-0117 also directed the staff to proceed, after completion of the public meetings, to develop a proposed rule for submittal to the Commission by March 1, 2001. Oral and written comments received from the public and invited stakeholders in the public meetings, and written comments received by mail, and electronic comments received on the NRC web site in response to the Issues Paper FRN, were considered in preparing this Environmental Assessment.

1.2 Document Organization

This document assesses the potential environmental, health, and safety impacts of the proposed regulatory changes, as required by NEPA. The rest of the document follows the basic outline for an Environmental Assessment specified in section 51.30(a) of the NRC's environmental protection regulations in 10 CFR Part 51. This outline includes a discussion of the need for the proposed action (Chapter 2), the proposed action and alternatives (Chapter 3), the environmental impacts of the proposed action and alternatives (Chapter 4), and a list of agencies and persons consulted and identification of sources used (Chapters 5 and 6,

[2] Ibid.

respectively). The discussion in these chapters is divided into two sections addressing, first, the changes being proposed to Part 71 to harmonize it with the latest IAEA standards, and second, other changes being proposed to Part 71 as part of the same rulemaking package.

2. Need For The Proposed Action

The proposed action can be organized into the following two major categories of changes to the NRC's radioactive material transportation regulations in 10 CFR Part 71:

- Changes to harmonize NRC's transportation regulations with other regulatory agencies (Department of Transportation, International Atomic Energy Agency); and

- Other NRC-initiated changes in order to simplify the regulations applicable to licensees shipping radioactive materials, while maintaining adequate protection of public health, safety, and the environment.

The need for these actions is discussed separately below.

<u>Harmonization of NRC's Transportation Regulations With Other Regulatory Agencies</u>

In general, the regulations in 10 CFR Part 71 are based on the safe transport standards developed by the IAEA, which are adopted by Member States, including the United States. As the IAEA periodically revises its transport standards, agencies that pattern their regulations after the IAEA standards make conforming changes, as discussed in Chapter 1.

On October 19, 1998, the Commission decided in SRM-SECY-98-168 to propose a rule to conform Part 71 with the latest revision of IAEA's safe transport standards, ST-1, published in December 1996. Accordingly, the NRC staff prepared a draft rulemaking plan to be supported by a Regulatory Analysis and an Environmental Assessment. These changes are needed to make the NRC's regulations consistent with international guidelines and DOT's regulations, which are also being revisited to conform to those guidelines.

<u>NRC-Initiated Changes</u>

Included within 10 CFR Part 71 are criteria that allow (1) exemptions from classification as a fissile material package and (2) general licenses for fissile material shipments.[3] Specifically, the regulations for fissile material exemptions are provided in section 71.53 and the regulations for general licenses are provided in sections 71.18, 71.20, 71.22, and 71.24. The exemptions and general licenses pertaining to requirements for packaging, preparation of shipments, transportation of licensed materials, and NRC approval of packaging and shipping procedures have not been significantly altered since their initial promulgation. Prevailing knowledge of radioactive material transport and historic practice indicated that little or no regulatory oversight was needed for the packaging or transport of certain quantities of fissile material that meet the criteria established in 10 CFR Part 71. Therefore, the fissile material exemptions and general license provisions allowed licensees to make shipments without first seeking approval from the NRC.

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[3] Section 71.4 currently defines fissile material as: "Plutonium-238, plutonium-239, plutonium-241, uranium-233, uranium-235, or any combination of these radionuclides. Unirradiated natural uranium and 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 section 71.53."

Before February 1997, section 71.53(d) exempted fissile material from the requirements in sections 71.55 and 71.59,[4] provided the package did not contain more than five grams of fissile material in any ten-liter (610-cubic inch) volume. The fissile material exemptions appearing in 10 CFR 71.53 were assumed to provide inherent criticality control for all practical cases in which fissile materials existed at or below the applicable regulatory limits (i.e., independent calculations would generally not be expected nor required). Thus, the fissile exemptions did not generally place limits on either the types of moderating/reflecting material present in fissile exempt packages or the number of fissile exempt packages that could be shipped in a single consignment. Also, these exemptions did not require the assignment of a transport index (TI) for criticality control.[5]

In February 1997, the NRC completed an emergency final rulemaking (62 FR 5907, February 10, 1997) to address newly encountered situations regarding the potential for inadequate criticality safety in certain shipments of exempted quantities of fissile material (beryllium oxide containing a low-concentration of high-enriched uranium). The emergency rule revised portions of 10 CFR Part 71 that limited the consignment mass for fissile material exemptions and restricted the presence of beryllium, deuterium, and graphite moderators.[6] Subsequent to its release, the NRC solicited public comments on the emergency rule. Five fuel cycle facility licensees and two other interested parties responded with comments that supported the need for the emergency rule but questioned whether some of the new restrictions were excessive. For example, some commenters noted that they had not encountered any problems shipping wastes that would have violated the emergency rule. Others stated that the new restrictions would at least double the number of waste shipments, thereby increasing costs, decreasing worker safety, and increasing the risk of accidents.

Based on these public comments and other relevant concerns, the NRC decided that further assessment was required, including a comprehensive assessment of all exemptions, general licenses, and other requirements pertaining to any fissile material shipment (i.e., not just fissile material shipments addressed by the emergency rulemaking). The NRC contracted Oak Ridge National Laboratory (ORNL) to conduct the assessment, and ORNL reviewed 10 CFR Part 71 (as modified by the emergency rule) in its entirety to assess its adequacy relative to the technical basis for assuring criticality safety. The results of the ORNL study were published as NUREG/CR-5342.[7] ORNL indicated that 10 CFR Part 71 needs updating, particularly to provide a simpler and more straightforward interpretation of the restrictions and criteria set in the regulations. Specific changes recommended in NUREG/CR-5342 are presented in Appendix A.

Based on the findings contained in NUREG/CR-5342, the NRC found it appropriate to evaluate other possible revisions to 10 CFR Part 71, with the objectives of:

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[4] These sections place additional requirements on fissile packages and shipments to preclude criticality.

[5] Transport index is defined in 10 CFR 71.4 as: “The dimensionless number (rounded up to the nearest tenth) placed on the label of a package to designate the degree of control to be exercised by the carrier during transportation.” See 10 CFR 71.4 for calculation criteria.

[6] For purposes of this report, the term “consignment mass” means the amount of fissile material offered by a consignor to a carrier for transport to a new location.

[7] NUREG/CR-5342, *Assessment and Recommendations for Fissile-Material Packaging Exemptions and General Licenses Within 10 CFR Part 71,* Oak Ridge National Laboratory, July 1998.

- simplifying the regulations applicable to licensees shipping fissile materials;

- relaxing restrictions on fissile material packages and shipments that are not justified based on plausible criticality concerns; and

- adequately addressing criticality safety for a number of newly considered plausible transportation and packaging situations.

In addition to the changes described above, the NRC has determined that there are other actions that can be taken efficiently as part of one rulemaking package. These other changes, which relate to several different SECY papers and a petition for rulemaking (PRM), include the following.

Packaging and Transportation

- SECY-97-161: Major on-going activities include: (1) a limited re-evaluation of the Commission's generic environmental impact statement on transportation (NUREG-0170) to address the impact of spent fuel shipments to a repository or central interim storage facility; (2) a joint DOT/NRC initiative to revise the IAEA process for adopting transportation regulations; and (3) development of standard review plans for both spent fuel and non-spent fuel applications.

- PRM-71-12 (International Energy Consultants): The petitioner requested that the NRC amend its regulations governing shipments of high-level waste under Part 71. The petitioner requested that paragraph 71.63(b), special requirements for plutonium shipments, be deleted in their entirety. This petition will be resolved as part of this rulemaking.

Other Regulations

- SECY-99-174: The objective is to revise 10 CFR 50.59 and 10 CFR 72.48 to clearly define those licensee procedural changes, tests, and experiments for which prior approval is required by the NRC.

- SECY-99-130: The objective is to expand the applicability of Part 71 to holders of, and applicants for, certificates of compliance (and also their contractors and subcontractors).

- SECY-99-100: The objective is to address commitments made by the Commission staff in SECY-98-138 to develop and implement a framework for risk-informed regulations in the Office of Nuclear Material Safety and Safeguards (NMSS).

- SECY-00-0117: The objective is to discuss the current IAEA standards for package surface removable contamination.

- SECY-00-0093: The objective is to review the reporting requirements contained in SECY-00-0093 to determine applicability to Part 71.
- Special Package Approval: The objective is to evaluate the need for revision to the current requirements for approval of special packages based on staff experience with recent exemption requests.

- Adoption of ASME Code: The objective is to evaluate the need for adoption into regulations of portions of the ASME code based on staff experience with spent fuel cask fabricators.

3. The Proposed Action and Alternatives

NRC is considering 19 changes to its radioactive material transportation regulations. Of these proposed changes, it was determined that four meet the NRC's categorical exclusion criteria as defined in 10 CFR 51.22. A categorical exclusion is a category of actions that do not result in a significant environmental impact and therefore do not require consideration in an environmental assessment. Therefore, this Environmental Assessment considers 15 independent proposed actions to change the radioactive material transportation regulations in 10 CFR Part 71. The first changes (see Section 3.1) are related to harmonizing the radioactive transportation regulations in 10 CFR Part 71 with the IAEA standards from "Regulations for the Safe Transport of Radioactive Materials," 1996 Edition, No. ST-1. The remaining changes (see Section 3.2) are modifications that could be considered by NRC to reduce paperwork and burden for licensees, while maintaining protection of public health, safety, and the environment. (In addition, one of these changes is based in part on the specific recommendations presented in NUREG/CR-5342.)

The proposed changes to 10 CFR Part 71 are summarized in Table 3-1 and described in more detail in the sections that follow (note that Table 3-1 also lists the four changes that meet the categorical exclusion criteria and are not considered further in this document). Each of these sections provide background information on the issue driving each change, describe the proposed action for resolving those issues, and outline what the no action alternative would entail.

Table 3-1. List and Summary Description of Proposed Changes to 10 CFR Part 71

Technical Issue	Summary Description of Potential Requirements
IAEA-related changes	
1. Changing Part 71 to the International System of Units (SI) Only	Require the use of SI units exclusively in shipping papers and labels.
2. Radionuclide Exemption Values	Adopt IAEA's radionuclide-specific exemption values for some or all radionuclides.
3. Revision of A_1 and A_2	Change the A_1 and A_2 values promulgated in 10 CFR Part 71, and in standard review plans and guidance documents pertaining to 10 CFR Part 71, to the new values published in TS-R-1.
4. Uranium Hexafluoride Package Requirements	Incorporate the TS-R-1 language into Part 71.
5. Introduction of the Criticality Safety Index Requirements	The potential action would require labels indicating both the CSI and Transport Index (TI) for fissile material shipments.
6. Type C Packages and Low Dispersible Material	Incorporate provisions from TS-R-1 for Type C packages and low dispersible radioactive material.
7. Deep Immersion Test	Modify the requirements to state that a package for radioactive contents greater than 10^5 A_2 shall be designed to 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.
8. Grandfathering Previously Approved Packages	Modify Part 71 to subject all packages to regulations in place at the time a Certificate of Compliance was issued. The revised regulations would apply to all new packages, and existing packages after renewal of the Certificate of Compliance.

Technical Issue	Summary Description of Potential Requirements
9. Changes to various definitions*	Add a number of definitions to 10 CFR 71.4 to ensure compatibility with TS-R-1.
10. Crush test for fissile material package design*	Require crush test for fissile material package designs regardless of package activity.
11. Fissile Material Package Designs for Transport by Aircraft	Subject shipped-by-air fissile material packages with quantities greater than excepted amounts to additional criticality evaluation.
NRC-Initiated changes	
12. Special Package Authorizations	Incorporate requirements into Part 71 that address shipment of special packages and the demonstrated level of safety.
13. Expansion of Part 71 Quality Assurance Requirements to Certificate of Compliance (CoC) Holders	Subject cask certificate holders and applicants for a CoC to the requirements of Part 71.
14. Adoption of ASME Code	Adopt the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code Section III, Division 3, for spent fuel transportation casks in Part 71.
15. Change Authority*	Incorporate a new subpart in Part 71 that would allow licensees to make minimal changes to their packaging and transportation procedures, without license amendments (for dual purpose casks only).
16. Fissile Material Exemptions and General License Provisions	Modify Part 71 in numerous ways, as needed, to implement some or all of the 17 recommendations contained in NUREG/CR-5342.
17. Double Containment of Plutonium (PRM-71-12)	Remove the 10 CFR 71.63(b) requirements for plutonium shipments. Plutonium packaging requirements would be handled no differently than requirements for other nuclear material (i.e., the A_1/A_2 system), except that plutonium shipped in the U.S. would have to be shipped as a solid.
18. Contamination Limits as Applied to Spent Fuel and High Level Waste (HLW) Packages	For information only. No regulatory action taken. No regulatory analysis performed.
19. Modifications of Event Reporting Requirements*	Conform Part 71 to the revised requirements in Part 50 (65 FR 63769) for event notification.

* Subject to categorical exclusion.

For the changes to fissile material license provisions, the options are based in part on the specific recommendations presented in NUREG/CR-5342. Due to the complexity of the technical basis for the various recommendations posed in NUREG/CR-5342, this Environmental Assessment does not provide a detailed description of either the rationale for each recommendation or how the recommendation would be implemented in regulatory text (except where doing so is relatively simple). Consequently, the discussion assumes a familiarity with and understanding of NUREG/CR-5342.

3.1 Proposed Actions to Harmonize NRC Transportation Regulations with IAEA Safe Transport Standards

3.1.1 Changing Part 71 to the International System of Units (SI) Only

TS-R-1 uses the SI units exclusively. This change is stated in TS-R-1, Annex II, page 199. 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, the

TS-R-1 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).

On August 10, 1988, Congress passed the Omnibus Trade and Competitiveness Act (the Act), which amended the Metric Conversion Act of 1975. Section 5164 of the Act designates the metric system[8] as the preferred system of weights and measures for U.S. trade and commerce. Congress noted that use of the metric system would improve the competitive position of U.S. products in international markets because world trade is increasingly conducted in metric units. In an effort to have an orderly change to metric units, the Act also requires that all Federal agencies convert to the metric system of measurement in their procurements, grants, and other business-related activities by the end of fiscal year 1992, unless this was impractical or likely to cause significant efficiencies or loss of markets to U.S. firms.

In order to implement the Congressional designation of the metric system as the preferred system of weights and measures for U.S. trade and commerce, Presidential Executive Order 12770 of July 25, 1991, designated the Secretary of Commerce to direct and coordinate metric conversion efforts by all Federal departments and agencies. Executive Order 12770 also directed all executive branch departments and agencies of the U.S. Government to establish an effective process for a policy-level and program-level review of potential exceptions to metric usage. The transition to use of metric units in Government publications would be made as publications are revised on normal schedules or new publications are developed, or as metric publications are required in support of metric usage.

In response to the Act and Executive Order 12770, as well as concerns of certain NRC licensees and other interested parties, NRC, on February 10, 1992, issued a proposed policy statement on metrication for public comment (57 FR 4891). After reviewing public comments, the NRC issued its policy on metrication on October 7, 1992 (57 FR 46202). The metrication policy stated that, after three years, the NRC was to assess the state of metric use by the licensed nuclear industry in the U.S. to determine whether the metrication policy should be modified.

In accordance with the NRC's policy statement of October 7, 1992, the NRC issued a request for public comment on its existing metrication policy on September 27, 1995 (60 FR 49928). After contacting various industrial, standards, and governmental organizations to determine their view of the policy and reviewing comments submitted in response to the request for public comment, the NRC issued its final Statement of Policy on Conversion to the Metric System on June 19, 1996 (61 FR 31169). The NRC considers its metrication policy to be final, and its conversion to the metric system complete.

Metrication Policy

The metrication policy, which affects NRC licensees and applicants, was designed to allow for response to market forces in determining the extent and timing for the use of the metric system of measurement. The policy also affects the Commission in that the NRC will adhere to the Federal Acquisition Regulations and the General Service Administration (GSA) metrication program for its own purchases.

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[8] The term "metric system" refers to the International System of Units as established by the General Conference of Weights and Measures in 1960 as interpreted or modified for the U.S. by the Secretary of Commerce.

The NRC's metrication policy commits the Commission to work with licensees and applicants and with national, international, professional, and industry standards-setting bodies (e.g., ANSI, ASTM, ASME) to ensure metric-compatible regulations and regulatory guidance. Through its metrication policy, the NRC encourages its licensees and applicants to employ the metric system of measurement wherever and whenever its use is not potentially detrimental to public health and safety or is uneconomic. The NRC did not want to 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, there is a mix of licensees and applicants using both the metric and the customary systems of measurement.[9]

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
- all written communications directed to the public.

The metrication policy also states that, in dual-unit documents, the first unit presented will be in the International System of Units with the customary unit shown in parenthesis. In addition, documents specific to a licensee, such as inspection reports and docketed material dealing with a particular licensee, will be in the system of units employed by the licensee.

It should be noted that, currently, NRC requires shipping papers and labels to be completed according to DOT regulations in 49 CFR Part 172. In its regulations, DOT does not specify the unit of measurement in which shipping papers used in the transportation of radioactive materials have to be completed (49 CFR 172.203(d)(4)). Further DOT regulations do not specify the units of measurement for labels used in the packaging and transportation of radioactive materials (49 CFR 172.403(g)(2)).

Option 1: No-Action Alternative

The No-Action alternative (Option 1) would not modify Part 71 regarding the use of SI units exclusively. With this option, the NRC adheres to its policy of dual units.

Option 2: Proposed Action

Under Option 2, NRC would amend Part 71 to make it compatible with TS-R-1 by requiring the use of SI units only. This would mean requiring a single system of units for both domestic and international shipments.

3.1.2 Radionuclide Exemption Values

[9] Based on telephone conversations with Mr. Felix Killar, NEI on August 30, 1999 and Ms. Lynette Hendricks, NEI on August 31, 1999.

NRC currently uses one specific activity limit for exemption of any type of radionuclide from its packaging and transportation regulations. Specifically, 10 CFR 71.10(a) states "[a] licensee is exempt from all requirements of this part with respect to shipment or carriage of a package containing radioactive material having a specific activity not greater than 70 Bq/g (0.002 μCi/g)." Similarly, DOT regulations in 49 CFR 173.403 define radioactive material as "any material having a specific activity greater than 70 Bq/g (0.002 μCi/g)."

TS-R-1, Table I, has been revised to include new, radionuclide-specific values for exempt materials. The IAEA activity concentrations for exempt material range from 1 x 10^{-1} to 1 x 10^{7} Bq/g. TS-R-1 also provides a formula to be used to determine the exemption of mixtures of radionuclides. The radionuclide-specific concentration limits are based on IAEA's Basic Safety Standards No. 115 (SS-115, entitled "International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources"), which applies to those natural materials or ores that are part of the nuclear fuel cycle or that will be processed in order to use their radioactive properties.

The general principles for the IAEA exemptions are:

- The radiation risks to individuals caused by the exempted practice or source be sufficiently low as to be of no regulatory concern;

- The collective radiological impact of the exempted practice or source is sufficiently low as not to warrant regulatory control under the prevailing circumstances; and

- The exempted practices and sources are inherently safe, with no appreciable likelihood of scenarios that could lead to a failure to meet the first two criteria.

IAEA exemption values have been derived in SS-115 on the following basis:

- An individual effective dose of 10 μSv per year for normal conditions;
- A collective dose of 1 person-Sv per year of practice for normal conditions;
- An individual effective dose of 1 mSv for accidental conditions; and
- An individual dose to the skin of 50 mSv for both normal and accidental conditions.

These levels were derived for SS-115 using scenarios that did not explicitly address the transport of radioactive material. Additional derivations were performed by IAEA for transport-specific scenarios, and the results were found to be similar to those in SS-115. Therefore, the exemption levels of SS-115 were adopted in TS-R-1.

The nature of the potential change makes it difficult to quantify the values or impacts. The most significant impact would be on shippers of materials which are not currently subject to the regulations (i.e., less than 70 Bq/g) and which would become subject to them (for example, NORM [Naturally Occurring Radioactive Materials] in natural ores and minerals, or piping, drilling equipment, or drilling waste products from the oil & gas industry). There is no known reliable information on the nature and amounts of materials which would be so affected.

This change would conform Part 71 to DOT's recommended change in its proposed rule. To determine whether Part 71 amendments are appropriate, the following two alternatives were considered:

<u>Option 1: No-Action Alternative</u>

Under the No-Action alternative (Option 1), NRC would continue to use one specific activity limit for exemption of any type of radionuclide.

Option 2: Proposed Action

Under Option 2, NRC would adopt, in 10 CFR Part 71, IAEA's radionuclide-specific exemption values for all radionuclides.

3.1.3 Revision of A_1 and A_2

TS-R-1 includes numerous revisions to the individual A_1 and A_2 values for radionuclides. The A_1 and A_2 values are used for determining what type of package must be used for the transportation of radioactive material. The A_1 values are the maximum activity of special form material allowed in a Type A package. The A_2 values are the maximum activity of "other than special" form material allowed in a Type A package. A_1 and A_2 values are also used for several other packaging limits throughout TS-R-1, such as specifying Type B package activity leakage limits, low-specific activity limits, and excepted package contents limits. (These specified values are included in Part 71 - Appendix A.)

The basic radiological criteria for determining A_1 and A_2 values are:

- The effective or committed effective dose to a person exposed in the vicinity of a transport package following an accident should not exceed a reference dose of 50 mSv (5 rem).

- The dose or committed equivalent dose received by individual organs, including the skin, of a person involved in the accident should not exceed 0.5 Sv (50 rem), or in the special case of the lens of the eye, 0.15 Sv (15 rem). A person is unlikely to remain at 1 m from the damaged package for more than 30 minutes.

The IAEA revised A_1 and A_2 values in TS-R-1 based on an analysis technique that includes improved dosimetric models that use the Q System (see Appendix D for the values contained in TS-R-1). The Q System includes consideration of a broader range of specific exposure pathways than the earlier A_1 and A_2 calculations. The five Q models are for external photon dose, external beta dose, inhalation dose, skin and ingestion dose due to contamination transfer, and dose from submersion in gaseous isotopes. The value of A_1 is determined from the most restrictive of the photon and beta doses, and the value of A_2 is determined from the most restrictive of the A_1 value and remaining Q model values.

The impact of these analyses is that the radionuclides have now been subjected to a more realistic assessment concerning exposure to an individual should a Type A transport package of radioactive material encounter an accident condition during transport. The new A_1 and A_2 values reflect that assessment.

During the enhanced public participation process, commenters requested that NRC and DOT retain the current exceptions of A_1 and A_2 for two radionuclides - ^{99}Mo and ^{252}Cf.

Option 1: No-Action Alternative

Under the No-Action alternative (Option 1), NRC would retain the current A_1 and A_2 values promulgated in 10 CFR Part 71.

Option 2: Proposed Action

Under Option 2, NRC would revise Part 71 to incorporate the TS-R-1 A_1 and A_2 values, while maintaining the current exceptions for ^{252}Cf and ^{99}Mo.

3.1.4 Uranium Hexafluoride (UF_6) Package Requirements

Uranium hexafluoride is generated as a result of uranium processing to prepare enriched uranium for use in nuclear power plants. Natural uranium ore is mined and milled to produce an intermediate product known as yellow cake. Yellow cake is then converted into UF_6. This UF_6 is sent to an enrichment facility in Paducah, Kentucky to increase the relative abundance of the fissile isotope ^{235}U from its natural abundance of 0.711 percent by weight to greater than one percent. It is then sent to another enrichment plant in Portsmouth, Ohio where it is further enriched. The enriched UF_6 is then sent to private fuel fabricators where it is converted to uranium oxide for use in nuclear power plants. Both of the existing enrichment facilities (in Portsmouth and Paducah) are run by the United States Enrichment Corporation (USEC), and produce depleted UF_6 as a waste. This depleted UF_6, which contains less than the natural abundance of ^{235}U, is stored in large cylinders in outdoor storage yards. Additionally, DOE operates the K-25 site at Oak Ridge, Tennessee, which in the past had been an enrichment facility and at which there are also cylinders of depleted UF_6 stored in outdoor yards. Depleted UF_6 is usually stored in Type 48 cylinders, while enriched UF_6 is transported in smaller Type 30 cylinders with overpacks.[10] Type 48 cylinders, which can contain either 10 or 14 short tons, are usually 9 to 12 feet long and 4 feet in diameter, while the Type 30 cylinders, which can contain 2.5 short tons, are usually about 7 feet long and 2.5 feet in diameter. Smaller amounts of UF_6 are occasionally shipped in smaller cylinders, such as for laboratory analysis. These smaller cylinders are usually overpacked.

The enrichment facility in Paducah receives about seven Type 48 cylinders a day of UF_6 from the private conversion facilities.[11] Because the UF_6 leaving Paducah and destined for Portsmouth is enriched, it is typically sent in Type 30 cylinders that are overpacked. As reported in the *Cost Analysis Report for the Long Term Management of Depleted Uranium Hexafluoride*, the stockpiles of depleted UF_6 cylinders at the USEC and DOE sites are extensive: Paducah had 28,351 cylinders, Portsmouth had 13,388 cylinders, and K-25 had 4,683 cylinders as of May 1997. In addition, between the two operating sites, approximately 2,000 and 2,500 new cylinders are generated per year for storage. DOE recently issued a record of decision outlining the plan for future management of these cylinders,[12] which involves building at least one conversion facility at either Paducah or Portsmouth to convert the depleted UF_6 back to uranium oxide, which is a more stable form. Another possibility being considered is that a conversion facility will be built at both of these sites.

Current regulation of UF_6 packaging and transportation is a combination of NRC and DOT requirements. The DOT regulations contain provisions which govern many aspects of packaging and shipment preparation, including a requirement that the material be packaged in

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[10] Overpacks are enclosures used by a single consigner to provide protection or convenience in handling a package or to consolidate two or more packages.

[11] Personal communication with Randy Reynolds, Bectel Jacobs Energy Systems, September 1998.

[12] *Record of Decision for Long-Term Management and Use of Depleted Uranium Hexafluoride*, U.S. Department of Energy, August 3, 1999, http://web.ead.anl.gov/uranium/new/index.cfm.

cylinders that meet the ANSI N14.1 standard. The NRC regulates fissile and Type B packaging designs for all materials, including the fissile UF_6.

Previous editions of the IAEA regulations did not specifically address UF_6, but TS-R-1 contains detailed requirements for UF_6 packages designed for more than 0.1 Kg UF_6. First, TS-R-1 requires the use of an international standard, ISO 7195 Packaging of Uranium Hexafluoride for Transport, instead of the ANSI N14.1 standard with the condition that approval by all countries involved in the shipment is obtained (i.e., multilateral approval, (Para 629)). 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 packages from utilizing pressure relief devices (Para 631). Fourth, TS-R-1 includes a new exception for UF_6 packages, regarding the evaluation of a single package. The new provision (Para 677(b)) allows UF_6 packages to be evaluated without considering the in-leakage of water into the containment system. This provision means that a single fissile UF_6 package does not have to be subcritical assuming that water leaks into the containment system. This provision only applies when: (1) there is no contact of the cylinder under hypothetical accident tests and the valve remains leak-tight, and (2) when there is a high degree of quality control in the manufacture, maintenance and repair of packagings coupled with tests to demonstrate closure of each package before each shipment.

<u>Option 1: No-Action Alternative</u>

Under the No-Action alternative (Option 1), NRC would not modify Part 71 to incorporate the TS-R-1 UF_6 requirements.

<u>Option 2: Proposed Action</u>

Under Option 2, NRC would revise Part 71 to incorporate the TS-R-1 UF_6 packaging requirement by promulgating new section 71.55(g), while restricting use of the exception to a maximum enrichment of 5 weight percent ^{235}U.

3.1.5 Introduction of the Criticality Safety Index Requirements

In current NRC and DOT regulations, the Transport Index (TI) is defined as follows:

> *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 determined as follows:*
>
> *(1) For nonfissile material packages, the number determined by multiplying the maximum radiation level in millisievert (mSv) per hour at one meter (3.3 feet) from the external surface of the package by 100 (equivalent to the maximum radiation level in millirem per hour at one meter (3.3 feet)); or*
>
> *(2) For fissile material packages, the number determined by multiplying the maximum radiation level in millisievert per hour at one meter (3.3 feet) from any external surface of the package by 100 (equivalent to the maximum radiation level in millirem per hour at one meter (3.3 feet)) or, for criticality control purposes, the number obtained by dividing*

50 by the allowable number of packages which may be transported together, whichever number is larger.

TS-R-1 has a requirement (paragraphs 541, 544, and 545) that a Criticality Safety Index (CSI) (paragraph 218), as well as the TI, be posted on packages of fissile material. The CSI assigned to a package, overpack, or freight container containing fissile material shall mean a number that is used to provide control over the accumulation of such containers containing fissile material. Previously, the IAEA regulations used a TI that used one number to accommodate both radiological safety and criticality safety.

The CSI for packages would be determined by using a formula provided by TS-R-1, which is the same as the formula for the TI for criticality control purposes found in NRC and DOT regulations. The CSI for each consignment would be determined as the sum of the CSIs of all the packages in that consignment. In addition, TS-R-1 states that the CSI of any package or overpack should not exceed 50, except for exclusive use consignments.

In order to make NRC regulations consistent with TS-R-1, a definitions for CSI would have to be added, and the CSI component would need to be removed from the current definition of TI.

<u>Option 1: No-Action Alternative</u>

Under the No-Action alternative (Option 1), NRC would not require labels or modify definitions for CSI and would retain the current TI label requirement.

<u>Option 2: Proposed Action</u>

Under Option 2, NRC would revise 10 CFR Part 71 to include a definition of CSI for fissile material packages and revise the existing TI definition.

3.1.6 Type C Packages and Low Dispersible Material

Analogous to a Type B package, IAEA has devised the concept of a Type C package that could withstand severe accident conditions in air transport without loss of containment or increase in external radiation (see TS-R-1 paragraphs 230, 667-670, 730, and 734-737). However, the design-basis accident conditions are somewhat different.

- One of the potential post-crash environments that a Type C package is more likely to see than a Type B package is burial. If a package whose contents generate heat becomes buried, an increase in package temperature and internal pressure could result. Therefore, Type C packages are required to meet heat-up and corrosion tests to which Type B packages are not subject.

- Type C packages are more likely to end up in deep water after an accident, so all Type C packages, no matter the design curie content, are required to undergo deep immersion testing.

- Puncture/tearing tests are required to account for the possibility of rigid parts of the air craft damaging the package.

- Since aircraft carry much more fuel than trucks, Type C packages are subjected for 60 minutes to a thermal test similar to the 30-minute Type B package test.

- Since aircraft travel at higher speeds than surface vehicles, the impact test is done at 90 m/s.

- Tests for Type C packages are not sequential because of the velocities and the space involved in aircraft accidents reduce the likelihood of a cask receiving high levels of multiple stresses.

U.S. regulations have no Type C package requirements, but have specific requirements for the air transport of plutonium. In addition to meeting Type B package requirements, to be certified for the air transport of plutonium, a package must withstand:

- an impact velocity of 129 m/sec;
- a compressive load of 31,800 kg;
- impact of a 227 kg dropped weight (small packages);
- impact of a structural steel angle falling from a height of 46 m;
- a 60 minute fire;
- a terminal velocity impact test; and
- deep submersion to 4 MPa (600 lbs/in^2).

The Type C package tests in IAEA's TS-R-1 are less rigorous than the U.S. tests for air transport of plutonium.

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). 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.

<u>Option 1: No-Action Alternative</u>

Under the No-Action alternative (Option 1), NRC would not adopt Type C packages or the "low dispersible radioactive material" concepts into 10 CFR Part 71.

<u>Option 2: Proposed Action</u>

Under Option 2, NRC would revise 10 CFR Part 71 to incorporate the Type C Package and low dispersible radioactive material concepts for air transportation but retain section 71.74, the accident conditions for air transport of plutonium.

3.1.7 Deep Immersion Test

The NRC currently requires a deep immersion test for some packages of irradiated nuclear fuel. This requirement is contained in 10 CFR 71.61 and states that "a package for irradiated nuclear fuel with activity greater than 37 PBq (10^6 Ci) must be so designed that its undamaged containment system can withstand an external water pressure of 2 MPa (290 psi) for a period of not less than one hour without collapse, buckling, or inleakage of water."

The revised IAEA requirement in TS-R-1 (paragraphs 657 and 730) no longer specifically states that it applies only to packages of irradiated fuel, but instead applies to all Type B(U) and B(M) packages containing more than 10^5 A_2Option, as well as Type C packages. In addition, TS-R-1 states only that the containment system can not fail, and does not require that the containment system not buckle or allow inleakage of water. ST-2 (para. 730.3) states that some degree of buckling or deformation is acceptable provided that there is no rupture. ST-2 (para. 657.5) also states that it is recognized that leakage into and out of the package is possible, and the aim is to ensure that only dissolved activity is released.

This expansion in the types of materials required to meet this requirement in TS-R-1 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 radiation exposure as a result of an accident.

The pressure requirement of 2 MPa (which is equivalent to 200 m of water submersion) corresponds approximately to the continental shelf and the depths where some studies indicated that radiological impacts could be important. Recovery of a package from this depth would be possible and salvage would be facilitated if the containment system did not rupture.

Currently, there are no Type C packages licensed for use in the U.S. If a Type C package design was developed and certified, it would need to pass the enhanced deep immersion test. Type C packages are addressed further in Section 2.1.6.

Option 1: No-Action Alternative

Under Option 1, the No-Action alternative, NRC would not require design of a package with radioactive contents greater than 10^5 A_2Option or irradiated nuclear fuel with activity greater than 37 PBq to withstand external water pressure of 2 MPa for a period of one hour or more without rupture of the system.

Option 2: Proposed Action

Under Option 2, the NRC would revise Part 71 to require 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.

3.1.8 Grandfathering Previously Approved Packages

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 which are in the process of being fabricated or which 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.

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) or 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) or 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 or 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) or 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.

Option 1: No-Action Alternative

Under the No-Action alternative (Option 1), NRC would not adopt the new grandfathering provisions contained in TS-R-1.

Option 2: Proposed Action

Under Option 2, NRC would modify section 71.13 to phase out packages approved under Safety Series 6. This Option would include a 3-year transition period for the grandfathering provision on packages approved under Safety Series 6 (1967). This period will provide industry the opportunity to phase out old packages and phase in new ones. In addition, packages approved under Safety Series 6 (1985) would not be allowed to be fabricated after December 31, 2006.

3.1.9 Changes to Various Definitions

The changes contemplated by NRC in this proposed rulemaking would require changes to various definitions in order to improve consistency with IAEA safe transportation standards contained in TS-R-1.

Option 1: No-Action Alternative

Under the No-Action alternative (Option 1), NRC would not adopt any new definitions, nor modify any existing definitions concurrent with the potential modifications addressed in the proposed rule.

Option 2: Proposed Action

Under Option 2, NRC would add various definitions to 10 CFR 71.4 and modify existing definitions to both ensure compatibility with definitions found in TS-R-1 and to improve clarity in NRC regulations. Specifically, the following definitions would be added or modified:

- Criticality Safety Index
- Certificate of Compliance
- Department of Transportation
- Deuterium
- A_1
- A_2
- LSA-III
- Fissile Material
- Graphite
- Package
- Spent Nuclear Fuel/Spent Fuel
- Structures, Systems, and Components Important to Safety (SSCs)
- Transport Index

3.1.10 Crush Test for Fissile Material Package Design

IAEA's TS-R-1 broadened the crush test requirements to apply to fissile material package designs (regardless of package activity). [IAEA Safety Series 6 and Part 71 have previously required the crush test for certain Type B packages.] This was done in recognition that the crush environment was a potential accident force which 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-1 682(b) requires tests specified in paragraphs 719-724 followed by whichever of the following is the more limiting: (1) the tests specified in paragraph 727(b) (drop test onto a bar) and, either paragraph 727(c) (crush test) for packages having a mass not greater than 500 kg and an overall density not greater than 1,000 kg/m^3 based on external dimensions, or paragraph 727(a) (nine meter drop test) for all other packages; or (2) the test specified in paragraph 729 (water immersion test).

Safety Series 6 (paragraph 548) required and 10 CFR Part 71 (71.73) presently requires the crush test for packages having: (1) a mass not greater than 500 kg and an overall density not greater than 1,000 kg/m^3 based on external dimensions, and (2) radioactive contents greater than 1000 A_2 not as special form radioactive material. Under TS-R-1, the radioactive contents greater than 1,000 A_2 criterion 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.

To be consistent with TS-R-1, the NRC would have to revise 10 CFR Part 71 wording to recognize that the 1,000 A_2 criterion does not apply to fissile material package designs.

Option 1: No-Action Alternative

Under the No-Action alternative (Option 1), the NRC would not modify Part 71 to incorporate the crush test requirement for fissile material packages.

Option 2: Proposed Action

Under Option 2, the NRC staff would revise section 71.73(c)(2) wording to agree with TS-R-1 and extend the crush test requirement to fissile material package designs.

3.1.11 Fissile Material Package Designs for Transport by Aircraft

The IAEA's TS-R-1 introduced new requirements for fissile material package designs that are intended to be transported aboard aircraft (paragraph 680). TS-R-1 requires that shipped-by-air fissile material packages with quantities greater than excepted amounts (which would include all the NRC certified fissile packages) be subjected to an additional criticality evaluation. Specifically, TS-R-1 paragraph 680 requires that packages must remain subcritical, assuming 20 centimeters of water reflection but not inleakage (i.e., moderation) when subjected to the tests for Type C packages[13]. 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 test conditions followed by a water immersion test. "Special features" generally mean features that could prevent water inleakage (and therefore could be taken credit for in criticality analyses) under the hypothetical accident conditions. Special features are permitted under current 10 CFR 71.55(c).

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

- Type A fissile package by air must:

  (A) withstand incident-free conditions of transport with respect to release, shielding, and maintaining subcriticality (single package and 5xN array),

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

[13] The ST-1 imposition of Type C and LDM requirements (see 3.1.6) were 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. Since 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.

(C) comply with para 680 with respect to maintaining subcriticality. (single package).

- Type B fissile package by air must:

  (A) withstand incident-free conditions of transport <u>and</u> 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 para 680 with respect to maintaining subcriticality. (single package)

- Type C fissile material package must withstand:

  (A) Incident-free conditions of transport (single package and 5xN array), Type B tests (single package and 2xN array), and Type C tests (single package) with respect to release, shielding, and maintaining subcriticality.

The draft advisory material for the IAEA transport regulations (ST-2) indicates that the requirement "...is provided to preclude a rapid approach to criticality that may arise from potential geometrical changes in a single package..." ST-2 also indicates that "...Where the condition of the package following the tests cannot be demonstrated, worst case assumptions regarding the geometric arrangement of the package and contents should be made taking into account all moderating and structural components of the packaging."

There are no provisions in TS-R-1 for "grandfathering" 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 prior to their use.

<u>Option 1: No-Action Alternative</u>

Under the No-Action alternative (Option 1), the NRC would not modify Part 71 to incorporate the TS-R-1 requirements contained in paragraph 680.

<u>Option 2: Proposed Action</u>

Under Option 2, the NRC would include this new TS-R-1 requirement for an additional criticality evaluation, in a new paragraph 71.55(f), that only applies to air transport.

3.2 NRC-Specific Changes

3.2.1 Special Package Authorizations

IAEA's TS-R-1 establishes procedures for demonstrating the level of safety for shipment of packages under special arrangements. Paragraphs 312 and 824 through 826 of TS-R-1 address approval of shipments under special arrangement and are provided verbatim below:

> *312. 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 a 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.

824. Each consignment transported internationally under special arrangement shall require multilateral approval.

825. An application for approval of shipments under special arrangement shall include all the information necessary to satisfy the competent authority that the overall level of safety in transport is at least equivalent to that which would be provided if all the applicable requirements of these Regulations had been met. The application shall also include:

A statement of the respects in which, and of the reasons why, the consignment cannot be made in full accordance with the applicable requirements; and

A statement of any special precautions or special administrative or operational controls which are to be employed during transport to compensate for the failure to meet the applicable requirements.

826. Upon approval of shipments under special arrangement, the competent authority shall issue an approval certificate.

A Memorandum of Understanding (MOU) published July 2, 1979 (44 FR 38690) specifies the roles of DOT and NRC in the regulation of the transportation of radioactive materials. The MOU outlines that DOT is responsible for regulating safety in transportation of all hazardous materials, including radioactive materials, whereas the NRC is responsible for regulating safety in receipt, possession, use, and transfer of byproduct, source, and special nuclear materials. Thus DOT serves the role of U.S. Competent National Authority and NRC certifies packages for domestic transport of radioactive material. Consequently, a shipper of radioactive materials must first obtain an NRC Certificate of Compliance for the package. Before the package may be exported the shipper must apply for and receive a competent authority certificate from DOT.

According to statistics compiled by the Nuclear Energy Institute, 31 states have operating nuclear reactors with a total of 103 operating reactors. After a nuclear power plant is closed and removed from service it must be decommissioned. As explained in NUREG-1628, *Staff Responses to Frequently Asked Questions Concerning Decommissioning of Nuclear Power Reactors*, decommissioning a nuclear power plant requires the licensee to reduce radioactive material on site. This effort to terminate the NRC license entails removal and disposal of all radioactive components and materials at each site, including the reactor.

Current NRC practice is to grant exemptions for package approval on special arrangement shipments, as the Commission did for the Portland General Electric (PGE) Trojan Reactor Vessel. 10 CFR 71.8 states:

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 the life or property nor the common defense and security.

In October 1998, the NRC staff used this provision to grant a request for approval from PGE to transport the Trojan reactor vessel to a disposal site at the Hanford Nuclear Reservation near Richland, Washington. Specifically, PGE was exempted from 10 CFR 71.71(c)(7), which requires transport packages to be capable of surviving a 30-foot drop, and 71.73(c)(1), which requires the integrity of transport packages to be tested by a one-foot drop onto a flat, unyielding surface prior to shipment. PGE requested these exemptions in order to ship the reactor vessel and internals via barge and land transport to the disposal facility. This scenario was preferred to the alternative separate disposal of the reactor vessel and internals because it resulted in lower radiation exposures to the general public and workers, a shortened decommissioning schedule, and lower overall costs.

Although approval of designs for packages to be used for the transportation of licensed materials qualifies for a categorical exclusion, the exception from preparing an environmental assessment or an environmental impact statement (10 CFR 51.22(c)(13)) does not apply to NRC packages authorized under an exemption. Consequently, the Trojan shipment was authorized for transport only after an Environmental Assessment and Finding of No Significant Impact had been published in the *Federal Register*. Additionally, PGE was required to apply for an exemption from DOT regulations governing radioactive material shipments that do not recognize packages approved under an NRC exemption.

NUREG-1628 reports that as of January 1998, three NRC-licensed power reactors had completed decommissioning. In addition to the Trojan plant, five other nuclear power reactors are now in various stages of dismantlement and decontamination. Because decommissioning is a condition for obtaining a 40-year NRC nuclear power operating license, further decommissioning efforts of the nuclear power reactors can be anticipated for the future.

<u>Option 1: No-Action Alternative</u>

Under the No-Action alternative (Option 1), NRC would continue to address approval of special packages using exemptions under 10 CFR 71.8.

<u>Option 2: Proposed Action</u>

Under Option 2, the NRC would incorporate new requirements in 10 CFR Part 71 that address approval for shipment of special packages and that demonstrate an acceptable level of safety. These requirements would be based on paragraph 312 of TS-R-1, but would also address regulatory and environmental conditions and requirements that are characteristic to the nuclear industry in the U.S.

3.2.2 Expansion of Part 71 Quality Assurance Requirements Certificate of Compliance (CoC) Holders

NRC has determined that 10 CFR Part 71 is not clear when addressing the issue of applicability of the regulations contained therein (i.e., who is covered by and must comply with the regulations). In fiscal year 1996, NRC staff identified several instances of nonconformance by CoC holders and their contractors. Nonconformance was observed in the following areas: design, design control, fabrication, and corrective actions. Due to the fact that these problems are typically addressed under a quality assurance program, the proposed rulemaking focuses on amending regulations in Subpart H of Part 71, Quality Assurance. The regulations contained in Subpart H will explicitly include CoC holders and CoC applicants. Recordkeeping and reporting requirements for these entities will also be established.

The following citation discusses the applicability of Part 71:

> *10 CFR Part 71.0(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.*

CoC holders and CoC applicants appear to be outside the applicability of 10 CFR Part 71.0(c). As noted above, the regulations in Part 71 apply only to NRC licensees. CoC holders are not necessarily NRC licensees. In fact, a CoC holder must only abide by the requirements of Part 71, Subpart D to obtain a CoC.

Because CoC holders and CoC applicants would be subject to the regulations contained in 10 CFR Part 71 under the potential action, they would also be subject to NRC enforcement actions if they fail to comply with the regulations. Currently, CoC holders and CoC applicants are only subject to administrative Notices of Noncompliance (NONs). Adding these entities to the applicability of Part 71 would allow NRC to issue Notices of Violation (NOVs), which assign graduated severity levels to violations. The issuance of an NOV performs the following functions: (1) conveys to the entity violating the requirement and to the public that a violation of a legally binding requirement has occurred; (2) uses graduated severity levels to convey the severity level of the violation; and (3) shows that NRC has concluded that a potential risk to public health and safety could exist. The evidence gathered to formulate an NOV can then be used to support the issuance of further enforcement sanctions such as NRC orders.

<u>Option 1: No-Action Alternative</u>

Under the No-Action alternative (Option 1), NRC would not subject CoC holders or CoC applicants to the requirements contained in 10 CFR Part 71.

<u>Option 2: Proposed Action</u>

Under Option 2, NRC would explicitly subject CoC holders and CoC applicants to the requirements contained in 10 CFR Part 71. NRC would also add recordkeeping and reporting requirements for CoC holders and CoC applicants.

3.2.3 Adoption of ASME Code

Currently, licensees are responsible for implementing and describing a quality assurance (QA) plan as part of the package approval process. The following citation discusses quality assurance:

> *10 CFR Part 71.37(a) The applicant shall describe the quality assurance program [...] for the design, fabrication, assembly, testing, maintenance, repair, modification, and use of the proposed package.*

In addition to licensee QA programs, NRC inspects licensee and licensee contractor operations from time-to-time. NRC inspections of vendor/fabricator shops have uncovered, over the past several years, QA problems with the production of transportation and storage casks. In some instances, QA problems have persisted in spite of repeated NRC deficiency findings. Implementation of the QA provisions set forth in Subpart H of 10 CFR Part 71 is the responsibility of the individual licensees. Because a specific ASME code was not available for spent fuel containers in the past, only portions of various ASME pressure vessel codes were employed in their design and construction. Many QA procedures employed as part of ASME code implementation were therefore not implemented by container designers and fabricators. ASME recently issued "Containment Systems and Transport Packages for Spent Fuel and High Level Radioactive Waste," Boiler and Pressure Vessel Code, Division 3 Section III. Fabricators manufacturing transportation cask containment systems subject to this specific ASME code would therefore be permitted to stamp components. ASME is also developing a code which, if approved, would allow the stamping of the confinement component for storage casks.

Three principal QA activities are employed when conforming to the ASME code:

- Preparation for and passing of an ASME Survey of each shop and field site involved in fabrication;

- Preparation of a Design Report certified by a licensed professional engineer (PE); and

- Introduction of a full-time Authorized Nuclear Inspector (ANI) on site during fabrication.

The most important aspect of the ASME QA program is the on-site presence of the ANI. The ANI is an independent professional capable of reporting QA issues to the management of the licensee/fabricator, and to the NRC. This on-site expert presence would alleviate the need for NRC inspections of licensee and fabrication facilities.

Implementation of the ASME Code would be consistent with the National Technology Transfer and Advancement Act of 1995, Public Law 104-113, Section 12(d), which requires governmental agency adoption of consensus technical standards. Government agencies are required to adopt these standards unless doing so would be inconsistent with other laws or would be impractical to implement. The proposed rule implementing the ASME consensus technical standards will conform to NRC's "Interim Guidance on the Use of Government-Unique and Voluntary Consensus Standards," May 3, 1999.

<u>Option 1: No-Action Alternative</u>

Under the No-Action alternative (Option 1), NRC would retain the current QA provisions for the package approval process so that the on-site presence of the ANI would not be required and NRC inspections of licensee and fabrication facilities would continue.

<u>Option 2: Proposed Action</u>

Under Option 2, NRC would adopt the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code Section III, Division 3, for spent fuel transportation casks in 10 CFR Part 71. This action would currently apply to spent fuel transportation cask containments. The industry is in the process of revising Division 3 to include storage casks and when re-issued (2 to 5 years), would broaden its current scope to include spent fuel storage canisters and internals, in addition to transportation casks containment and internals. The action would also apply to dual-purpose casks.

3.2.4 Change Authority

Part 71 currently contains no regulations that would: 1) provide a Part 71 certificate holder (for a transportation cask) with the authority to make changes, tests, and experiments equivalent to Part 72.48, or (2) instruct a Part 71 certificate holder on how to apply to amend the Part 71 CoC equivalent to Part 72.244. Part 71 also does not require the user to have a copy of the safety analysis report or other documents that describe the design of the package. In addition, Part 71, Subpart D, currently uses the terminology submission of a "package description" in an application, rather than the terminology submission of a "safety analysis report." Lastly, Part 71 currently contains no regulations that would require an update of a FSAR — reflecting any changes made under a Part 71.48 — equivalent to Part 72.248.

The NRC has recently issued a final rule in 10 CFR Part 72 to allow licensees and cask certificate holders to perform minor changes, tests and experiments relative to an Independent Spent Fuel Storage Installation (ISFSI) or spent fuel storage cask design or to conduct tests and experiments — without prior NRC approval — if certain conditions are met. The NRC staff initially considered, based on: (1) public comment received on the Part 72 proposed rule, (2) the staff's discussions of technical issues in SECY-99-130, and (3) subsequent Commission's approval, to extend the approach used in the Part 72 final rule to Part 71 for domestic dual-purpose casks (i.e., casks used for both transportation and storage of spent nuclear fuel).

Subsequently, NRC staff have determined that the regulatory structure of Part 71 does not lend itself to implementing a parallel change with Part 72. The result could be 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 acceptable for use under section 71.87.

Option 1: No-Action Alternative

Under the No-Action alternative (Option 1), licensees or cask certificate holders would still be required to gain NRC approval for changes to procedures, or cask designs, through license amendments.

Option 2: Proposed Action

Under Option 2, NRC would revise 10 CFR Part 71 to add a new section regulating dual-purpose transportation packages (i.e., casks designed for both shipment and storage of spent nuclear fuel) used for domestic purposes only. In addition to providing a new process for approving dual purpose transportation packages, the new requirements would provide for the authority for CoCs to make changes to a dual purpose package design without prior NRC approval. The section would also include new requirements for submitting and updating a final safety analysis report describing the package's design.

3.2.5 Fissile Material Exemptions and General License Provisions

Included within 10 CFR Part 71 are criteria that allow exemptions from classification as a fissile material package and general licenses for fissile material shipments:

1. Subpart B -- Exemptions
    - Exemption for low-level material (section 71.10)

2. Subpart C -- General Licenses
   - Fissile material, limited quantity per package (section 71.18)
   - Fissile material, limited moderator per package (section 71.20)
   - Fissile material, limited quantity, controlled shipment (section 71.22)
   - Fissile material, limited moderator, controlled shipment (section 71.24)

3. Subpart E -- Package Approval Standards
   - Fissile material exemptions (section 71.53)

Since their initial promulgation, the exemptions and general licenses pertaining to requirements for packaging, preparation of shipments, transportation of licensed materials, and NRC approval of packaging and shipping procedures have not been significantly altered. Available knowledge on radioactive materials transportation and historic practices confirmed the need for little or no regulatory oversight of packaging or shipment of fissile materials meeting the criteria established in 10 CFR Part 71. The fissile material exemptions and general license provisions allowed licensees to prepare and send shipments of such fissile materials without obtaining specific approval from NRC.

Before February 1997, section 71.53(d) exempted fissile material from the requirements in sections 71.55 and 71.59, provided the package did not contain more than 5 grams of fissile material in any 10-liter (610-cubic inch) volume. The fissile exemptions appearing in 10 CFR 71.53 were assumed to provide inherent criticality control for all practical cases in which fissile materials existed at or below the applicable regulatory limits (i.e., independent calculations would generally not be expected nor required). Thus, the fissile exemptions did not generally place limits on either the types of moderating/reflecting material present in fissile exempt packages or the number of fissile exempt packages that could be shipped in a single consignment. Also, these exemptions did not require the assignment of a transport index for criticality control.

In February 1997, NRC completed an emergency final rulemaking (62 FR 5907, February 10, 1997) to address newly-encountered situations regarding the potential for inadequate criticality safety in certain shipments of exempted quantities of fissile material (beryllium oxide containing a low-concentration of high-enriched uranium). The emergency rule revised portions of 10 CFR Part 71 that limited the consignment mass for fissile material exemptions and restricted the presence of beryllium, deuterium, and graphite moderators. Subsequent to its release, NRC solicited public comments on the emergency rule. Five NRC fuel cycle facility licensees and two other interested parties responded with comments that supported the need for the emergency rule, but argued that the restrictions imposed therein were excessive. For example, several commenters noted that they had shipped wastes that violated the emergency rule in the past without any problems and that the new restrictions would at least double the number of waste shipments, thereby increasing costs, decreasing worker safety, and increasing the risk of accidents.

Based on these public comments and other relevant concerns, NRC decided that further assessment was required, including a comprehensive assessment of all exemptions, general licenses, and other requirements pertaining to *any* fissile material shipment (i.e., not just fissile material shipments addressed by the emergency rulemaking). NRC contracted Oak Ridge National Laboratory (ORNL) to conduct the assessment, and ORNL reviewed 10 CFR Part 71 (as modified by the emergency rule) in its entirety to assess its adequacy relative to the technical basis for assuring criticality safety. Specifically, ORNL:

- documented perceived deficiencies in the technical or licensing bases that might be incapable of maintaining subcriticality under normal conditions of transport and hypothetical accident conditions;

- identified areas where regulatory wording could cause confusion among licensees and potentially lead to subsequent safety concerns;

- studied and identified the practical aspects of transportation and licensing that could mitigate, justify, or provide a historical basis for any identified potential deficiency; and

- developed recommendations for revising the current regulations to minimize operational and economic impacts on licensees, while maintaining safe practices and correcting licensing deficiencies.

The results of the ORNL study (NUREG/CR-5342) indicated that the fissile material exemptions and general licenses need updating, particularly to provide a simpler and more straightforward interpretation of the restrictions and criteria set in the regulations. The regulatory options are based on the recommendations contained in NUREG/CR-5342.

<u>Option 1: No-Action Alternative</u>

Under the No-Action alternative (Option 1), NRC would not modify 10 CFR Part 71 to implement the 17 recommendations contained in NUREG/CR-5342, but would continue to use the modified regulations promulgated under 10 CFR Part 71, RIN 3150-AF58, Fissile Material Shipments and Exemptions, final rule. This alternative involves amendments of regulations for the shipment of exempt quantities of fissile material and the shipment of fissile material under a general license through the restriction of the use of beryllium and other special moderating materials in the shipment of fissile materials and the consignment of limits on fissile exempt shipments.

<u>Option 2: Proposed Action</u>

Under Option 2, NRC would modify the 10 CFR Part 71 regulations in numerous ways, as needed, to implement the entire set of 17 recommendations contained in NUREG/CR-5342. These recommendations and the potential changes to Part 71, which are summarized in Table 3-2 below, involve the exemption of fissile material from shipment as radioactive material; the shipment of fissile material under general licenses; and the shipment of fissile material classified as exempt.

3.2.6 Double Containment of Plutonium (PRM-71-12)

NRC's regulations in section 71.63 include the following special requirements for plutonium shipments:

> *§71.63 Special requirements for plutonium shipments.*
>
> *(a) Plutonium in excess of 0.74 TBq (20 Ci) per package must be shipped as a solid.*
>
> *(b) Plutonium in excess of 0.74 TBq (20 Ci) per package must be packaged in a separate inner container placed within outer packaging that meets the requirements of Subparts E and F of this part for packaging of material in normal form. If the entire*

package is subjected to the tests specified in §71.71 ("Normal conditions of transport"), the separate inner container must not release plutonium as demonstrated to a sensitivity of 10^{-6} A_2/h. If the entire package is subjected to the tests specified in §71.73 ("Hypothetical accident conditions"), the separate inner container must restrict the loss of plutonium to not more than A_2 in 1 week. Solid plutonium in the following forms is exempt from the requirements of this paragraph:

Table 3-2. Proposed Recommendations and Changes Related to Fissile Material Packaging Exemptions and General Licenses

NUREG/CR-5342 Recommendation	Summary of Recommended Action
1. Revise the definitions in §71.4 and other text in 10 CFR Part 71 (perhaps considering relationships between 49 CFR Part 173 and IAEA No. TS-R-1) to ensure consistency and to clarify any intended distinctions between words/phrases such as: - exemption, exception, and exclusion - manifest, consignment, shipment, and conveyance - consignment, consignor, and shipper - controlled shipment, exclusive use, etc.	Amend definitions and phrases in 10 CFR Part 71 to ensure consistency between 10 CFR Part 71, IAEA safe transportation standards in TS-R-1, and DOT requirements contained in 49 CFR Part 173.
2. Revise the definition of “fissile material” in §71.4 and other text in 10 CFR Part 71 to (1) eliminate the nuclide ^{238}Pu from the definition, and (2) clarify whether “fissile material” consists of fissile nuclides or of materials containing fissile nuclides.	Amend 10 CFR 71.4 by revising the definitions of “fissile material,” “package,” and “transportation index.” 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, not materials containing fissile nuclides, and redesignating the reference to exclusions from the fissile material controls from §71.53 to new §71.11. The definition of “package” would be revised by redefining “Type A packages” in accordance with DOT regulations contained in 49 CFR Part 173. The definition of “transport index” (TI) would be revised to provide greater clarity on the two different bases for the TI: radiation safety and criticality safety, and to clarify where equations for calculating the TI are located within the regulations.
3. Revise §71.11 so that, if the radioactive material contains fissile material, the exemption applies only if the specific activity is not greater than 43 Bq/g.	Amend 10 CFR 71.11 to exempt radioactive material containing fissile material if 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.
4. Revise the §71.10(b) exemption so that it does not include fissile material that should meet a packaging requirement.	Revise paragraph (b) by redesignating the reference to fissile material exemption standards from §71.53 to new §71.11.
5. Move the §71.53 fissile material exemptions to Subpart B of Part 71, from Subpart E.	Redesignate §71.53 as §71.11 and relocate these requirements to Subpart B with the other Part 71 exemptions. This section would also be amended by adding new paragraphs to provide mass-based limits in classifying fissile material. In addition, the concentration or consignment based limits currently described in §71.53 would be removed with the exception of the 15 g limit; and a new ratio of fissile to non-fissile material would be added.

Table 3-2. Proposed Recommendations and Changes Related to Fissile Material Packaging Exemptions and General Licenses (Continued)

NUREG/CR-5342 Recommendation	Summary of Recommended Action
6. Establish at NRC or DOE a fissile shipment database to help NRC better understand fissile shipments and make more informed regulatory determinations in the future. This recommendation would probably require regulatory changes to either or both of §71.91 ("Records) and §71.95 ("Reports"), depending on how shipment information would be obtained.	Add new reporting and recordkeeping requirements to §71.19 to track information pertaining to fissile material shipments.
7. Create a separate general license for Pu-Be sources, revise the quantity of plutonium allowed to be shipped as Pu-Be neutron sources, and/or provide packaging requirements that prevent challenges to the basis for criticality safety.	Replace existing §71.20 with a new section to provide regulations on the shipment of Pu-Be special form material, consolidating regulations contained in §§71.18 and 71.22. The overall effect of the potential change to be to permit shipments of Pu-Be sealed sources containing between 24 and 240 g of fissile Pu on exclusive use shipments. Shipments containing less than 240 g could be made under the potential revisions to §71.18 and on exclusive or non-exclusive use conveyances. Shipment of Pu-Be sealed sources containing greater than 240 g fissile Pu would be made in Type B packages on an exclusive use conveyance.
8. Simplify the general license provisions and make them consistent with §71.59 by (1) merging sections addressing general licenses for controlled shipments (§71.22 and §71.24) along with sections addressing general licenses for limited quantity/moderator per package (§71.18 and §71.20), and (2) specifying the aggregate transport index (TI) allowed for non-exclusive use and exclusive use.	Remove §§71.22 and 71.24. 10 CFR 71.59 would be revised to use the term "criticality safety index" consistently between §§71.59, 71.18 and 71.20. The potential action will also be revised such that packages shipped under these sections should use the criticality control transport index determined by those sections. The potential action would revise the phrase "[n]ot in excess of 10" to be "[l]ess than or equal to 10.0." In addition, the section will be revised to provide guidance when the criticality control transport index is exactly 10.0.
9. Revise §71.20 and §71.24 to use bounding non-uniform quantities of ^{235}U rather than to distinguish between uniform and non-uniform distributions. Alternatively, add a definition of "non-uniform distribution" that can be clearly interpreted by licensees to §71.4.	Remove the requirements contained in §§ 71.20 and 71.24 and incorporated into the new §71.18 - General license: Fissile material.
10. Delete/revise §71.18(e) and §71.22(e), which address the shipment under general licenses of fissile materials containing Be, C, and D_2O, to remove the Be, C, and D_2O quantity restrictions, except to note that these materials should not be present as a reflector material (limiting the quantity of these materials to 500g per package should eliminate any concern relative to their effectiveness as a reflector).	See recommended action for Recommendation 8.
11. Revise the mass control in 10 CFR 71.18(d) and the mass restriction in 10 CFR 71.20(c)(4) for moderators having a hydrogen density greater than water to apply (only) whenever such high-density hydrogenous moderator exceeds 15 percent of the mass of hydrogenous moderator in the package.	Revise the gram limits for fissile material mixed with material having a hydrogen density greater than water and place them in new Table 71-1.

Table 3-2. Proposed Recommendations and Changes Related to Fissile Material Packaging Exemptions and General Licenses (Continued)

NUREG/CR-5342 Recommendation	Summary of Recommended Action
12. Specify minimum package requirements as provided by §71.43 and §71.45 for shipments under the general licenses to help ensure good shipping practices for fissile materials with low specific activity.	Specify that fissile material shipped under the general license provisions of new §71.18 would be contained in a Type A package.
13. Given the implementation of Recommendation 12, increase the package mass limits allowed by §71.18 and §71.20 to provide similar safety equivalence as certified packages defined under §71.55 and §71.59.	See recommended action for Recommendation 12.
14. Revision to mass-limited exemptions. Provide criteria based on a ratio of the mass of fissile material per mass of nonfissile material that is non-combustible, insoluble in water, and not Be, C or D_20. Alternatively, incorporate into §71.53 a conveyance control based on a TI of 100. Given one of the above, remove the restriction on Be, C, and D_2O from §71.53 except for §71.53(b).	Provide mass-based limits in classifying fissile material. The recommended action would allow for increasing quantities of fissile material to be shipped; however, there would be additional restrictions in the form of ratios of the mass of the fissile material to non-fissile material present in the package. The mass of moderating materials would not be allowed in the mass of the package when calculating the ratio of fissile to non-fissile material.
15. Revise §71.53(a), (c), and (d) by deleting restrictions on Be, C, and D_2O.	The current restrictions on Be, C, and D_2O would be removed as licensees would be allowed to us a mass-ratio rather than a mass-limit.
16. Revise §71.53(c) by adding the minimum packaging standard at §71.43 to the exemption for uranyl nitrite solutions transport.	Amend the current requirement to clarify that the nitrogen to uranium atomic ratio for shipments of liquid uranyl nitrate must be greater than or equal to 2.0 Further, a requirement specifying the use of Type A packages would be added.
17. Revise §71.53(b) by removing the requirement that the fissile material be distributed homogeneously throughout the package contents and that the material not form a lattice arrangement within the package. (Maintain the moderator criteria restricting the mass of Be, C, and D_2O to less than 0.1 percent of the fissile material mass.)	Revise the requirement in §71.53(b) to provide that beryllium, graphite, and hydrogenous material enriched in deuterium, constitute less than 0.1 percent of the fissile material mass.

(1) Reactor fuel elements;

(2) Metal or metal alloy; and

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

The NRC received a petition for rulemaking on behalf of International Energy Consultants, Inc. dated September 25, 1997. In its petition, the petitioner requested that section 71.63(b) be deleted. The petitioner believed that provisions stated in this regulation 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. The petitioner further stated that the requirement that a Type B package must be used whenever package content exceeds an A_2 quantity should be applied consistently for any radionuclide. The petitioner believed that if a Type B package is sufficient for a quantity of a radionuclide X which exceeds A_2, then a Type B package should be sufficient for a quantity of radionuclide Y which exceeds A_2, and this should be similarly so for every other radionuclide.

The petitioner stated that while, for the most part, the regulations embrace this simple logical congruence, the congruence fails under section 71.63(b) because packages containing plutonium must include a separate inner container for quantities of plutonium having an activity exceeding 0.74 TBq (20 Ci). The petitioner believed 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 radioactivity limit for every 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 requirements 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.

The petitioner believed that the continuing presence of section 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.” The petitioner further stated that “...excessively high costs occur in some transport campaigns,” and that one example “... of damage to our national budget is in the transport of transuranic wastes.” Because large numbers of transuranic waste drums must be shipped in packages that have a “separate inner container” to comply with the existing rule, the petitioner believed that large savings would accrue without this rule. Therefore, the petitioner believed that elimination of section 71.63(b) would resolve these regulatory “defects.”

As a corollary to the primary petition, the petitioner believed that an option to eliminate section 71.63(a) as well as section 71.63(b) should also be considered. This option would have the effect of totally eliminating section 71.63. The petitioner believed that the arguments propounded to support the elimination section 71.63(b) also support the elimination of section 71.63(a).

By letter dated April 30, 1999, the NRC informed the petitioner that it had considered the petition and the public comments and decided to defer final action on the petition. The NRC informed the petitioner of its development of the current Part 71 rulemaking and that the subject matter of the petition and elements of the rulemaking address similar issues, and that resolution of the petition would be conducted with the rulemaking action.

The NRC anticipated in 1974 that a large number of shipments of plutonium nitrate liquids could result from spent nuclear fuel reprocessing and revised its regulations to require that plutonium in excess of 0.74 TBq be shipped in solid form. The NRC did so because shipment of plutonium liquids is susceptible to leakage (if the shipping package is improperly or not tightly sealed). The value of 0.74 TBq (20 Ci) was chosen because it was equal to a large quantity of plutonium as defined in 10 CFR Part 71 in effect in 1974. Although this definition no longer appears in 10 CFR Part 71, the value as applied to double containment of plutonium has been retained. The concern about leakage of liquids arose because of the potential for a large number of packages (probably of more complex design) to be shipped due to reprocessing and the increased possibility of human error resulting from handling this expanded shipping load.

The NRC treats dispersible plutonium oxide powder in the same way because it also is susceptible to leakage if packages are improperly sealed. Plutonium oxide powder was of particular concern because it was the most likely alternative form (as opposed to plutonium nitrate liquids) for shipment in a fuel reprocessing economy. To address the concern with dispersible powder, the NRC required that plutonium not only must be in solid form, but also that solid plutonium be shipped in packages requiring double containment. Moreover, the NRC stated that the additional inner containment requirements are intended to take into account that the plutonium may be in a respirable form and that solid forms that are essentially nonrespirable, such as reactor fuel elements, are suitable for exemption from the double containment requirement.

The Commission further stated:

> *Since the double containment provision compensates for the fact that the plutonium may not be in a "nonrespirable" form, solid forms of plutonium that are essentially nonrespirable should be exempted from the double containment requirement. Therefore, it appears appropriate to exempt from the double containment requirements reactor fuel elements, metal or metal alloy, and other plutonium bearing solids that the commission determines suitable for such exemption. The latter category provides a means for the Commission to evaluate, on a case-by-case basis, requests for exemption of other solid material where the quantity and form of the material permits a determination that double containment is unnecessary.*

Placing the 1974 decision in the context of the times, in a document dated June 17, 1974, titled "Environmental Impact Appraisal Concerning Proposed Amendments to 10 CFR Part 71 Pertaining to the Form of Plutonium for Shipment" the following statements were made:

> *Using the present criteria and requirements of Part 71, hundreds of packages containing plutonium nitrate solutions have been shipped with no reported instances of plutonium leaks from the containment vessel.*
>
> *The present situation with respect to the quantity and specific activity (radioactivity per unit mass) of plutonium involved in transportation is expected to change significantly over the next several years. Increasingly large quantities of plutonium shipped and the number of shipments made are expected to increase. For example, the amount of plutonium available for recovery was estimated to be about 500 kg in 1974 as compared to 20,000 kg in 1980. In addition, the specific activity of the plutonium will increase with higher reactor fuel burn-up, resulting in higher gamma and neutron radiation levels, greater heat generation, and greater potential for pressure generation (through radiolysis) in shipping packages containing plutonium nitrate solutions.*
>
> *Because of expected changes in the quantities and characteristics of plutonium to be transported and because of the inherent susceptibility of liquids to leakage, the Commission believes that safety would be enhanced if the physical form of plutonium for shipment was restricted to a solid, except for packages containing less than 20 curies.*

Further, in SECY-R-74-5, dated July 6, 1973, it was acknowledged by NRC that:

> *The arguments for requiring a solid form of plutonium for shipment are largely subjective, in that there is no hard evidence on which to base statistical probabilities or to assess quantitatively the incremental increase in safety which is expected. The discussion in the Regulatory staff paper, SECY-R-702, is not intended to be a technical argument which incontrovertibly leads to the 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 increasing assurance against the problem developing.*

On November 30, 1993, the DOE petitioned the Commission to amend section 71.63 to add a provision that would specifically remove canisters containing plutonium-bearing vitrified waste from the packaging requirement for double containment. DOE's main arguments were that the canistered vitrified waste provided a comparable level of protection to reactor fuel elements, that the plutonium concentrations in the vitrified waste will be lower than in spent nuclear fuel, and that the vitrified waste is in an essentially nonrespirable form. The Commission published a notice of receipt for the petition, docketed as PRM-71-11, in the *Federal Register* on February 18, 1994, requesting public comment by May 4, 1994. The public comment period was subsequently extended to June 3, 1994, at the request of the Idaho National Engineering and Environmental Laboratory (INEEL) Oversight Program of the State of Idaho.

On June 1, 1995, the NRC staff met with the DOE in a public meeting to discuss the petitioner's request and the possible alternative of requesting an NRC determination under section 71.63(b)(3) to exempt vitrified high level waste from the double containment requirement. The DOE informed the NRC in a letter dated January 25, 1996, of its intent to seek this exemption and the NRC received DOE's request on July 16, 1996. The original petition for rulemaking was requested to be held in abeyance until a decision was reached on the exemption request.

In response to DOE's request, the NRC staff prepared a Commission paper (SECY-96-215, dated October 8, 1996) outlining and requesting Commission approval of the NRC staff's proposed approach for making a determination under section 71.63(b)(3). The determination would have been the first made after the promulgation of the original rule, "Packaging of Radioactive Material for Transport and Transportation of Radioactive Materials Under Certain Conditions," published on June 17, 1974 (39 FR 20960). In a staff requirements memorandum dated October 31, 1996, the Commission disapproved the NRC staff's plan and directed that this policy issue be addressed by rulemaking.

In response, the NRC staff reactivated the DOE petition and developed a proposed rule. On June 15, 1998, the final rule was noticed in the *Federal Register*. In summary, the NRC amended its regulations to add vitrified high level waste, contained in a sealed canister designed to maintain waste containment during handling activities associated with transport, to the forms of plutonium which are exempt from the double containment packaging requirements for transportation of plutonium.

In a October 31, 1996, SRM for SECY-96-215 (dealing with the vitrified waste issue) the Commission directed the staff to "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, dated September 29, 1997, the Commission was informed that "the staff believes the technical bases for 10 CFR 71.63 remain valid and that the provisions provide adequate flexibility for current and future technologies. The staff believes it is desirable to retain those provisions of 10 CFR 71.63 that are not being covered by a separate rulemaking currently underway." The rulemaking underway referred to the DOE petition regarding transport of vitrified high level waste containing plutonium. In the discussion section of SECY-97-218, the staff again admitted that the special provisions (of 10 CFR 71.63) were not based on quantitative evidence of statistical analysis. Instead, subjective arguments regarding experience with shipment and design of packages were used as the basis to support the conclusion.

It should be noted that in press release No. 97-070, dated May 8, 1997, announcing the change in the regulations to allow shipment of plutonium-bearing vitrified waste, the NRC stated:

> *When the existing rule was published, the NRC anticipated that a large number of shipments of plutonium nitrate liquids or plutonium oxide powder could result from spent fuel reprocessing. However, the anticipated large number of shipments has not occurred, because commercial reprocessing is currently not taking place in this country for policy and economic reasons.*

Option 1: No-Action Alternative

Under the No-Action alternative (Option 1), NRC would retain the section 71.63 special requirements for plutonium shipments, which would place increased plutonium shipping requirements in the U.S. compared to the IAEA requirements.

Option 2: Proposed Action

Under Option 2, NRC would adopt, in part, the recommended action of PRM-71-12. Specifically, the NRC would remove the double containment requirement of section 71.63(b). However, the NRC would retain the package contents requirement in section 71.63(a) — for shipments whose contents contain greater than 0.74 TBq (20 Ci) of plutonium must be made with the contents in solid form.

3.2.7 Contamination Limits as Applied to Spent Fuel and High Level Waste (HLW) Packages

TS-R-1 contains contamination limits for all packages of 4.0 Bq/cm^2 (22,000 dpm/100 cm^2) for beta and gamma and low toxicity alpha emitting radionuclides, and 0.4 Bq/cm^2 (2,200 dpm/100 cm^2) for all other alpha emitting radionuclides. Although TS-R-1 uses the term limit, IAEA considers these to be guidance values, or derived limits, above which appropriate action should be considered. In the case of contamination, that action is to decontaminate to within the limits.

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 section 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 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 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.

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 Member States. IAEA will then consider CRP report and any further actions or remedies that may be warranted at periodic meetings (at TRANSAC).

The NRC is proposing no regulatory change at this time. Therefore, the Agency has not identified any regulatory options. The above discussion is for information purposes only.

3.2.8 Modifications of Event Reporting Requirements

The current regulations in section 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 is in use to transport radioactive material, (2) details of any defects with safety significance found after first use of the cask, and (3) failure to comply with conditions of the certificate of compliance (CoC) during use.

The Commission recently issued a final rule to revise the event reporting requirements in 10 CFR Part 50 (see 65 FR 63769). This final rule revised the verbal and written event notification requirements for power reactor licensees in 10 CFR 50.72 and 50.73. In SECY-99-181,[14] 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 staff also reviewed the Part 71 event reporting requirements in 10 CFR 71.95 and concluded that conforming changes should be made to the Part 71 event report requirements. NRC staff also concluded that this proposed rule was the appropriate vehicle to consider such changes.

The NRC staff 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, the Commission recently reduced the reporting burden on reactor licensees in the Part 50 final rule from submitting written reports in 30 days to 60 days.

<u>Option 1: No-Action Alternative</u>

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[14] 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.

Under the No-Action alternative (Option 1), NRC would not modify section 71.95 and would continue to 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 safety significance found after first use of the cask, and (3) failure to comply with conditions of the certificate of compliance (CoC) during use.

Option 2: Proposed Action

Under Option 2, NRC would revise section 71.95 to require that the licensee and certificate holder jointly submit a written report for the criteria in new subparagraphs (a)(1) and (a)(2). The NRC also would add new paragraphs (c) and (d) to section 71.95 which would provide guidance on the content of these written reports. This new requirement is consistent with the written report requirements for Part 50 and 72 licensees (i.e., sections 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. Additionally, the NRC would remove the specific location for submission of written reports from section 71.95(c) and instead require that reports be submitted "in accordance with section 71.1." Lastly, the NRC would reduce the regulatory burden for licensees by lengthening the report submission period from 30 to 60 days.

4. Potential Environmental, Health, and Safety Impacts of Alternatives Considered

This chapter characterizes the potential environmental, health, and safety impacts expected to result from NRC's proposed rulemaking. It is divided into three main sections. Section 4.1 outlines the impact assessment methodology. Section 4.2 characterizes the potential impacts associated with the proposed actions to harmonize the NRC's transportation regulations with the IAEA's latest safety standards. Finally, Section 4.3 discusses the potential impacts associated with the NRC-specific proposed actions.

4.1 Methodology

This Environmental Assessment was prepared in conjunction with a Regulatory Analysis, which appears in a separate document ("Regulatory Analysis of Major Revision of 10 CFR Part 71, Draft Final Report," February 2000). As part of this combined effort, ICF undertook a significant data collection effort. The first step in the data collection was to determine data needs to support the analysis of potential impacts for each of the proposed actions outlined in Chapter 3. Specifically, ICF identified the following types of information necessary to develop the value-impact analysis:

<u>Baseline Information</u>

- Number of exempt packages
- Number of non-exempt packages
- Number of exempt shipments
- Number of non-exempt shipments
- Average number of packages per exempt shipments
- Average number of packages per non-exempt shipment

<u>Information for Each Proposed Action</u>

- Change in occupational person-rems per year from exposure due to criticality accidents
- Change in public person-rems per year from exposure due to criticality accidents
- Change in occupational person-rems per year from exposure due to traffic accidents
- Change in public person-rems per year from exposure due to traffic accidents
- Change in occupational person-rems per year from routine radiological exposures
- Change in number of exempt packages
- Change in number of non-exempt packages
- Change in number of exempt shipments
- Change in number of non-exempt shipments
- Average number of packages per exempt shipment
- Average number of packages per non-exempt shipment
- Change in time required for record-keeping/reporting
- Change in time for regulatory determinations/calculations
- Change in time for regulatory review

ICF conducted numerous initial searches of existing literature using several databases. For example, ICF reviewed information contained in DOE's Shipment Mobility/Accountability Collection (SMAC) database in an attempt to identify technical information on exempted shipments of fissile materials and fissile material shipments of exempted quantities, or those made under a general license. In addition, extensive searches were conducted via the Internet. Each search was targeted at obtaining specific information related to a proposed change.

Further, for the NUREG/CR-5342 recommendations to change the fissile material requirements, ICF conducted a survey of licensees that currently ship fissile materials to identify the potential change in the number of packages/shipments and associated costs for each of the proposed actions. The questions developed for this survey are listed in Appendix B. ICF, however, received only one survey response. While the information was useful, it did not provide nearly the level of detail necessary to assist the Commission in developing a quantitative value-impact analysis for the proposed actions for fissile materials.

Given the lack of quantitative information, ICF conducted a qualitative impact analysis. To provide a framework for this analysis, ICF developed a process flow that encompasses the many steps involved during the shipment of nuclear materials under 10 CFR Part 71. This process flow, in which materials originate with a shipper and terminate with a receiver, is illustrated in Exhibit 4-1. Each proposed action reviewed for this Environmental Assessment was evaluated based on which steps in the process flow it affects. For example, specific activities within the shipment process that were evaluated for potential environmental effects were (1) shipper planning, (2) shipper packaging, (3) shipper inspection, (4) shipper loading, (5) shipping, and (6) receiver unloading. The assessment also considered inspection and unpackaging of the material by the receiver. These activities take place within three general locations, and present three separate accident scenarios, as shown in Exhibit 4-1: planning, packaging, inspection and loading all take place within the shipper environment (A); shipping takes place external to both the shipper and receiver environments (B); and unloading, inspection, and unpackaging also take place within the internal receiver environment (C).

All proposed actions were analyzed within each accident location for indication of changes in accident frequency and changes in accident consequence. Proposed actions were subsequently evaluated for impact on each activity within the shipment process. Key indicators for activity-related impacts that were considered are outlined below:

1. Planning
    a. Procedures required prior to shipment

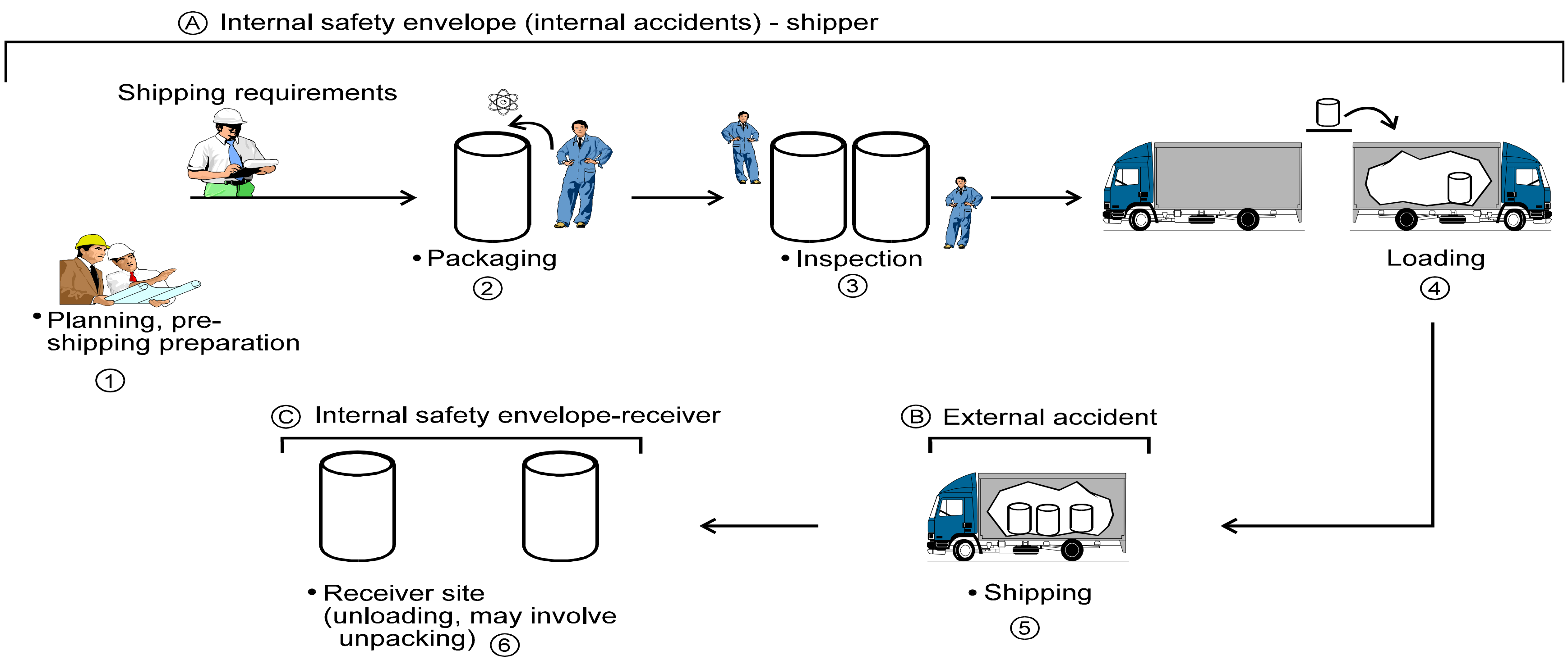
2. Packaging
    a. Changes in the number of loads
    b. Changes in length of time for packing a load
    c. Changes in worker exposure for packing a load

3. Inspection
    a. Changes in the number of inspections
    b. Changes in the length of time for each inspection
    c. Changes in worker exposure for conducting an inspection

4. Loading
    a. Changes in the number of loads
    b. Changes in the length of time for loading
    c. Changes in worker exposure for loading

Exhibit 4-1

Process Flow for Nuclear Transportation



5. Shipping
    a. Changes in the number of shipments
    b. Changes in the quantity per shipment
    c. Changes in the length of time for shipping
    d. Changes in worker/public exposure per shipment

6. Unloading
    a. Changes in the number of loads
    b. Changes in the length of time for unloading
    c. Change in worker exposure for unloading

4.2 Environmental Impacts of Proposed Actions to Harmonize 10 CFR Part 71 with IAEA ST-1

4.2.1 Changing Part 71 to the International System of Units (SI) Only

<u>Impacts of the Proposed Action</u>

It is expected that the proposed change would have negligible effects on the inspection, loading, or receiving of packages. However, the proposed change would require, in some instances, conversion from English units to SI units in order to satisfy Part 71 requirements. Industry sectors currently using English units (e.g., companies who ship spent fuel, regular fuel, and/or low-specific activity material to destination sites within the United States) would have to modify some of their administrative and pre-shipment preparation activities to include SI units (e.g., preparing shipping papers, labeling). It should be noted, however, that the NRC's shipping papers currently require that most of the information be completed in SI units. In cases where unit conversions are needed, there is a small chance that accident frequencies may change during normal packaging and transportation operations as a result of possible errors made in the conversion from English units to SI units. Changes in accident frequencies, however, would not be expected to impact accident consequences, as the proposed rule would not affect the manner in which material is protected in accordance with current packaging and transportation requirements. Any potential changes in accident frequencies associated with conversion from English units to SI units would be primarily restricted to a minimal increased risk of radiation exposure to the public and workers.

It is expected that there would be a negligible effect on emergency responders because they typically do not have to make unit conversions. At any type of accident and possible release, the emergency responders (i.e., firefighters or HazMat team) would examine markings on the vehicle, markings on the shipping containers, and shipping papers (e.g., the bill of lading, MSDS sheets) to determine: (1) the hazardous materials involved; (2) the amount of material; and (3) the risk/effect to life, health, property, and the environment. In cases where accidents or releases involve radioactive materials, emergency responders usually contact Chemtrec or the NRC about the incident and request assistance from the shipper or producer before taking any action. Overall, emergency response capabilities and effectiveness would not change if markings and papers used SI units rather than English units.

<u>Impacts of the No-Action Alternative</u>

Under the No-Action alternative, NRC licensees and applicants would continue to use their preferred system of measurement for complying with reporting requirements in 10 CFR Part 71. Licensees submitting documentation in English units would not have to convert their data into

SI units. Thus, an increase in the current number of flawed conversions or accident rates within the U.S. is not expected. At the same time, there would continue to be some instances of confusion, possibly resulting in mishandling or accidents, when packages are received from or shipped to international locations that all use SI units only.

4.2.2 Radionuclide Exemption Values

Impacts of the Proposed Action

The nature of the proposed change makes it difficult to quantify the safety impacts or benefits. Because exempt packages are not subject to the reporting requirements for NRC and DOT-regulated packages, there are no data on the number or frequency of exempt packages shipped in the U.S.

In order to gain some insight into how the proposed change could affect regulated packages, ICF examined a Sandia report titled "Transport of Radioactive Material in the United States: Results of a Survey to Determine the Magnitude and Characteristics of Domestic, Unclassified Shipments of Radioactive Materials" (SAND84-7174). This report presents the estimated number of packages shipped, organized by radionuclide. The six radionuclides that comprised the largest number of shipments were identified and compared to the corresponding exemption amount in IAEA's ST-1. The results are shown in Table 4-1 below.

Table 4-1. Radionuclide Shipments

Radionuclide[1]	Number of Packages[1]	Annual Curies Shipped[2]	IAEA Exemption Level (Bq/g)
Am-241	395,000	60,300	1
Co-60	283,000	2,430,000	10
Tc-99m	570,000	69,900	100
Mo-99	219,000	1,210,000	100
Ir-192	80,500	4,930,000	10
Cs-137	196,000	48,600	10

[1] - From SAND84-7174
[2] - Derived from SAND84-7174

Of the six radionuclides examined, two (Tc-99m and Mo-99) would have a higher exemption level than the current 70 Bq/g, while the other four would have a lower exemption value. For the purpose of discussion, changing the 70 Bq/g level to either 1 Bq/g, 10 Bq/g, or 100 Bq/g will have an impact too small to measure. In general, higher exemption levels could lead to an increase in the number of exempted shipments and lower exemption levels could lead to a decrease in the number of exempted shipments. IAEA has judged that the exemption levels that are less restrictive (i.e., higher) than current NRC values do not cause a significant risk to individuals.

The above mentioned isotopes, as most others in normal commerce, are shipped in highly purified forms. Typically, they are shipped in Type-B quantities from initial production at a reactor or accelerator, and then distributed in small quantities to medical and/or industrial users. Since these shipments contain highly purified forms, the change to the exemption limit will not have a significant effect on the total number of shipments or impacts of commercially shipping these items (in other words, these radionuclides will continue to be shipped in relatively high concentrations regardless of the exemption limits). Additionally, based on a review of the entire list of radionuclides with new exemption limits in IAEA's ST-1, most exemption limits would only change from 70 Bq/g to either 100 Bq/g or 10 Bq/g. These changes would not affect how the material is handled, since it is generally at or near a level that would affect contaminated waste handling, not product distribution.

The following isotopes have new IAEA exemption limits of 1,000 Bq/g or higher: Ag-111, Ar-37, Ar-39, As-73, As-77, At-211, Be-10, C-14, Ca-41, Ca-45, Co-58m, Cs-134m, Cs-135, Eu-150, Fe-55, Ge-71, Ho-166, Kr-81, Kr-85, Lu-177, Mn-53, Ni-59, Ni-63, Np-235, Np-236, Os-191m, P-33, Pb-205, Pd-107, Pm-147, Pm-149, Pt-193, Pr-143, Pt-197, Rb-87, Rb(nat), Re-187, Re(nat), Rb-103m, S-35, Se-79, Si-31, Si-32, Sn-119m, Sn-121m, Sn-123, Sr-89, Ta-179, Tb-157, Tc-96m, Tc-97, Tc-97m, Th-231, Th-234, Tl-204, Tm-170, Tm-171, V-49, W-181, W-185, Xe-127, Xe-131m, Xe-133, Xe-135, Y-90, Y-91, Yb-175, Zn-69, and Zr-93. Of these isotopes, the only ones that contribute 0.01 percent or more of the total curie amount transported are Ni-63 (0.01 percent) and Xe-133 (0.49 percent). Both of these are generally found only in fission products, and are shipped as spent fuel or high-level waste. Therefore, the change should not impact the package used or the number of shipments.

The following isotopes have new IAEA exemption limits of 1 Bq/g or lower: Ac-277, Am-241, Am-242m, Am-243, Bk-247, Cf-249, Cf-251, Cf-254, Cm-243, Cm-245, Cm-246, Cm-247, Cm-248, Np-237, Pa-231, Pu-238, Pu-239, Pu-240, Pu-242, and U-232. Of these, the isotopes that contribute 0.01 percent or more of the total curie amount transported are the americium, neptunium, and plutonium isotopes. The impacts of americium shipments are discussed in the paragraphs above and in Section 4.2.3. No significant change in the impacts of americium shipments would be expected. The lowering of the plutonium and neptunium limits from 70 Bq/g to 1 Bq/g might have an impact on transporting low-level wastes from DOE facilities. In particular, packages containing between 1 and 69 Bq/g which used to qualify for an exemption would now be subject to the reporting requirements for NRC and DOT-regulated packages. This change would result in a decrease in the number of these shipments and/or some level of improved protection for the shipments that continue to be made.

The DOE Waste Management EIS was reviewed to determine if significant amounts of radioisotopes would be transported under exemptions. No such shipments were mentioned in the EIS. Since most waste shipments would be using Type A packages and most impacts were attributed to the smaller number of Type B packages that would be shipped, the change in regulation would have little or no impact on DOE site clean-up activities.

In summary, the impacts of adopting the ST-1 radionuclide-specific exemption limits would be as follows:

1. Planning and preshipment would be more difficult with radionuclide-specific exemption limits because package contents would have to be examined and compared to the limit for each and every radionuclide. Additional effort to characterize the material being shipped would increase occupational exposure.

2. More rigorous packaging for shipments containing small concentrations of plutonium and neptunium may be required. However, it is believed that all shipments of these isotopes already meet the existing stringent packaging requirements.

3. No significant changes to inspection efforts would be anticipated.

4. No significant changes to the loading process would be anticipated.

5. During shipping, the occasional use of more rigorous packaging would reduce the already low chance and level of exposure due to packages being damaged during normal conditions of transport.

6. No significant changes to package receipt would be anticipated.

Impacts of No-Action Alternative

The No-Action alternative is to keep the current U.S. exemption value of 70 Bq/g (0.002 μCi/g). This would make U.S. standards inconsistent with countries who adopt the international standards. A package being imported into the U.S. carrying an isotope that has an exemption limit greater than 70 Bq/g (20 Ci) could be violating U.S. laws. A package being exported from the U.S. carrying an isotope that has an exemption limit less than 70 Bq/g (20 Ci) could be in violation of another country's laws. However, since most import/export shipments contain highly purified and/or highly radioactive isotopes, these scenarios would rarely, if ever, occur.

4.2.3 Revision of A_1 and A_2

Impacts of the Proposed Action

The A_1 and A_2 values were revised in ST-1 based on an analysis technique that includes improved dosimetric models. The models include consideration of external photon dose, external beta dose, inhalation dose, skin and ingestion dose due to contamination transfer, and dose from submersion in gaseous isotopes. The revised A_1 and A_2 values are based on the same dose standards as the current Part 71 values, which are:

- The effective or committed effective dose to a person exposed in the vicinity of a transport package following an accident should not exceed a reference dose of 50 mSv (5 rem).

- The dose or committed equivalent dose received by individual organs, including the skin, of a person involved in the accident should not exceed 0.5 Sv (50 rem), or in the special case of the lens of the eye, 0.15 Sv (15 rem). A person is unlikely to remain at 1 m from the damaged package for more than 30 minutes.

Because the dose standards underlying the A_1 and A_2 values have not changed, the proposed changes are not expected to have any net effect on the planning, packaging, inspection, loading, shipping, or receiving of radioactive materials. There is expected to be no net impact on occupational or public health, or any environmental effects.

Impacts of the No-Action Alternative

Because the dose standards underlying the A_1 and A_2 values have not changed, the proposed changes are not expected to have any net effect on the planning, packaging, inspection, loading, shipping, or receiving of radioactive materials. No net impact is expected on occupational or public health, or any environmental effects.

4.2.4 Uranium Hexafluoride (UF_6) Package Requirements

Impacts of the Proposed Action

The environmental impacts of the proposed action, which could include both increased radiological exposure to workers, and decreased radiological exposure to truck operators, the public, off-site property, and the environment along shipping routes, depend on two factors: (1) whether older cylinders that were manufactured and put into use before the effective date of this rulemaking would be grandfathered (i.e., would these cylinders be allowed to be shipped in accordance with the current regulations or would they need to be upgraded to comply with the new regulations?); and (2) whether the competent national authority (DOT) will waive the thermal test requirements for cylinders containing more than 9,000 kg UF_6.

- If DOT waived the thermal test requirement for cylinders containing more than 9,000 kg UF_6 (regardless of whether older cylinders were grandfathered), there would be no environmental impact. This is because it was determined that there is no substantial difference between the ANSI N14.1 standard and the ISO 7195 standard for UF_6 packaging. Therefore, the only cylinders that would be affected in this scenario would be cylinders containing less than 9,000 kg UF_6. These smaller cylinders are either (1) not typically used for natural or depleted UF_6, or (2) used for enriched UF_6, which is considered to be fissile, and therefore already subject to the thermal test requirement.

- If older cylinders were grandfathered but DOT did not waive the thermal test requirement for new cylinders containing more than 9,000 kg UF_6, between 2,000 and 2,500 new cylinders a year would need to be overpacked in the course of normal operations. (Type 48 cylinders are assumed to fail the thermal test unless these cylinders are overpacked.)

- Finally, if older cylinders were not grandfathered and DOT did not waive the thermal test requirement for cylinders containing more than 9,000 kg UF_6, between 2,000 and 2,500 cylinders a year would need to be overpacked in the course of normal operations. In addition, at some point in the future when a conversion facility or facilities are built to process the stockpiled depleted UF_6, between 4,683 and more than 50,000 cylinders could be affected. These cylinders would also need to be overpacked to pass the thermal test.

Assuming a shipping process that includes planning and preshipment, packaging, inspection, loading, shipping, and receiving, the following environmental impacts might occur in these last two cases:

1. Planning and preshipment would not affect radiologic exposure.

2. Packaging would require overpacks which in turn would increase worker radiological exposure while placing the cylinder in the overpack, due to the increased length of time involved.

3. Inspection of the overpacked cylinder might also result in increased worker radiological exposure due to the addition of an inspection step to verify the overpack was used correctly. At the same time, this increase in time would be offset to some degree by the fact that the cylinder is overpacked.

4. The loading process should have lower worker radiological exposure, based on a similar loading time and the increased safety provided by the overpack.

5. During shipping, there is likely to be a reduction in radiological exposure due to the overpack. That is, while the accident frequency will not be reduced, the amount of radiological exposure should be reduced by the overpack. Consequently, truck operators, the public, off-site property, and the environment along the shipping route will all have a lower risk of exposure to radiation in the event of a fire following a vehicular accident.

6. At the receiving site, there will be a decrease in worker radiological exposure during unloading, but a possible increase in worker radiological exposure while inspecting and removing the overpack, similar to the increases and decreases in worker radiological exposure at the loading site as described above.

Impacts of No-Action Alternative

The No-Action alternative would not result in any change to the current level of radiological exposure consistent with the NRC's policy to maintain radiation exposure to workers and the public as low as reasonably achievable.

4.2.5 Introduction of the Criticality Safety Index Requirements

<u>Impacts of Proposed Action</u>

This issue only affects fissile material packages, and does not affect the accident or incident free radiation doses. Since there are no notification or reporting requirements for fissile material packages, the number of packages affected cannot be estimated. However, Babcock and Wilcox provided an estimate of the annual number of shipments of fissile material. Some quantitative insight can be derived from their analysis, but it cannot be generalized to cover the entire industry. The following environmental impacts might occur if the additional label is required:

1. Planning and preshipment would not be affected because both the CSI and TI are calculated.

2. Packaging would not be affected.

3. Inspectors would have to ensure that the additional labels were correctly placed and correctly labeled. However, since they have to walk around the vehicle whether or not the regulation is changed, the additional inspection time and dose would be negligible.

4. The loading process would not be affected.

5. The incident free dose during shipping would not be affected. In the unlikely event of an accident that requires emergency response, the responders would be better informed as to the contents of the vehicle. It is unlikely that their response actions would be different as a result of the second placard.

6. The receiving process would not be affected.

Impacts of No-Action Alternative

The No-Action alternative would not result in any change to the current level of radiological exposure consistent with the NRC's policy to maintain radiation exposure to workers and the public as low as reasonably achievable.

4.2.6 Type C Packages and Low Dispersible Material

Impacts of Proposed Action

Two potential uses for Type C containers were identified. First, if Type C package regulations were available in the U.S., DOE may consider flying several shipments per year of spent foreign research reactor fuel to the continental U.S. for eventual shipment to the Savannah River Site. Currently, some spent nuclear fuel is at remote reactor sites. Because the highways and railways in some countries are not adequate for long distance transportation, DOE has shipped some spent fuel via air. DOE has loaded the fuel onto a truck, driven it to the airport nearest the reactor site, flown it to a port city in a foreign country, loaded it onto a truck, driven from the airport to the sea port, and loaded the fuel onto a ship. The ship has off loaded the fuel at a U.S. port, and DOE has shipped it to the Savannah River Site using both trucks and trains. The process could be simplified, if, once airborne, the plane was allowed to fly to the U.S., and load the fuel onto a truck in the freight handling area of an airport.

The second use would be for the shipment of fresh mixed oxide (MOX) reactor fuel. Over the next several decades, there may be limited amounts of MOX fuel shipped internationally. For example, DOE's Fast Flux Test Facility may use German MOX fuel (64 FR 178)[15]. Air transport of MOX fuel is not considered a likely alternative to truck shipments for any domestic transportation. Unlike uranium fuel, MOX is normally shipped in Type B quantities. Since MOX fuel contains plutonium, it would be subject to air transport of plutonium regulations. The origin or destination for these shipments would almost certainly be a DOE facility.

For each use, the air transport in a Type C package would basically replace the shipboard transport leg in a more complicated transportation plan.

The following environmental impacts might occur under the scenarios described above:

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[15] 64 FR 178, *Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development in Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility (DOE/EIS-0310),* September 15, 1999.

1. Planning and preshipment would be simplified, but no significant change in environmental impacts would result from these changes.

2. Packaging would be about the same, since a Type C package could be the same size and of about the same construction as a Type B package.

3. The inspection effort at the origin and destination would be about the same for either an air shipment or a sea shipment.

4. The loading process would vary from package-to-package. Typically, exposures while loading packages onto trucks, planes, or ships are low. However, loading an airplane would generally require people to be closer to a package for a longer period than loading a truck. In turn, loading a truck would require more time near a package than loading a ship. Based on the analysis of unloading casks with exposure rates equal to the regulatory limit, the total exposure (to handlers, crane operators, truck drivers, observers, and inspectors) for a cask unloading is on the order of one thousandth ($1x10^{-3}$) of a person-rem (DOE/EIS-0218F)[16]. Exposure from off loading a plane or a truck may be slightly higher, but still less than double that for a ship or less than $2x10^{-3}$ person-rem per operation. In general, the loading/unloading doses are higher for scenarios in which the lack of a Type C package requires an extra handling evolution. If the same number of handlings are needed, the loading/unloading doses would be higher when a Type C package is shipped by air.

5. Shipping impacts are divided into incident free doses and accident risks. For the purpose of analysis, two reasonable destinations were selected to estimate the impacts associated with shipping a Type C package in the air: a DOE facility in the Eastern U.S. and a DOE facility in the Western U.S. For each destination, two shipping schemes were analyzed: (1) air travel to an airport near the facility, followed by trucking to the facility, and (2) ship travel to an east coast port, followed by trucking to the facility.

   Incident free doses during shipping are higher than doses during loading, so they will drive the overall workers' exposure. A review of the various scenarios in DOE/EIS-0218F shows that about one person-rem is expected per cask shipped from either Europe or Asia to the appropriate U.S. coast. Because of the speed of an aircraft, the doses to crew would be less than 0.1 person-rem for air travel to either the eastern or western U.S. DOE/EIS-0218F calculates about 0.3 person-rem to the truck crew and 0.7 person-rem to the public for a cross-country trip. Shorter trips from seaports or airports result in proportionately less exposure. For each destination, the air shipment resulted in less crew and public exposure.

   The accident risk can be higher or lower for air transportation of a Type C package, depending on the destination of the cargo. The package was assumed to come from Europe. The data used are from NUREG-0170, and the metric used was the probability of occurrence of an accident severity category VI, VII, or VIII. For the eastern DOE site, the air shipment results in a higher public risk, and for the western DOE facility, the air shipment results in a lower public risk.

[16] DOE/EIS-0218F, *Final EIS on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel,* February 1996.

Therefore, incident free exposures would be lower if the U.S. had regulations allowing Type C packages. However, the change in accident risks cannot be conclusively estimated.

6. The discussion in item 4 above concerning loading applies equally to environmental impacts associated with unloading under the proposed action.

Impacts of No-Action Alternative

The No-Action alternative would not result in any change to the current level of radiological exposure consistent with the NRC's policy to maintain radiation exposure to workers and the public as low as reasonably achievable.

4.2.7 Deep Immersion Test

Impacts of Proposed Action

It is expected that the proposed action would have negligible effects on the inspection, loading, or receiving of packages. However, the proposed action may affect the planning, packaging, and shipping of material with respect to human health and environmental effects.

The proposed action would have some effects on the planning and packaging of shipments. Shippers would need to develop procedures for determining whether the material being shipped should be placed in a package that would meet the deep submersion test. Procedures would also need to be developed for the packaging of the materials in a proper package. However, these effects are expected to have minimal effects on human health or environmental protection.

The proposed action may also have some small benefit by preventing the rupture of package containment at deeper depths, thereby preventing possible contamination of the marine environment. However, the number of packages shipped over deep water with a high enough activity level to be subject to the deep immersion test is expected to be very small; therefore, the reduction in environmental impacts would also be small.

The proposed action may have some effect on the shipping of packages by reducing the likelihood of release in the case of an accident. The package would be able to withstand the pressure at increased depths without rupturing, 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 action was implemented.

The proposed action could also decrease occupational exposure in the event of an accident in which the package was submersed in water at a depth of less than 200 m (660 ft). The package would be able to withstand the pressure at this depth and not rupture, thereby keeping the radioactive materials enclosed. The deep immersion test would be for packages containing activity of more than 10^5 A_2Option, so as to ensure that the containment system does not fail and create a radiation hazard or inflict environmental harm. If such a package were lost in water less than 200 m deep, it is likely that the package would be recovered.

The occupational dose from the recovery operation of a ruptured spent fuel cask that has a dose rate at the regulatory limit has been estimated to be approximately 410 person-mrem[17]. This estimate is still considered to be valid, although somewhat conservative since shielding effects of water were not considered and the package may in fact be well below the dose rate limits.

The proposed action would affect the accident consequences of a package being lost in water of less than 200 m in depth. This type of scenario may result from severe accidents involving truck or rail transportation over or near coastal areas, rivers, or lakes. A scenario in which a severe accident takes place near or over deep water, resulting in the package being rolled or dropped into the water, is an extremely unlikely event and possibly beyond reasonable credibility.

Another applicable accident scenario would be the sinking or capsizing of a ship or barge while at sea over the continental shelf, near port in a bay channel or river, or in port. The probability of the loss of a vessel has been approximated to be 0.001 per trans-Pacific trip[18]. It is assumed that approximately 100 such shipments would occur each year. The probability of 0.001 accidents per trip multiplied by 100 shipments per year results in an annual probability of a deep immersion accident of 0.1 per year. This annual probability combined with the estimated 410 person-mrem dose results in an expected annual radiological exposure of 41 person-mrem/yr, or 0.041 person-rem/yr. Therefore, the proposed action would be expected to result in the savings of 0.041 person-rem/yr by preventing the rupture of the containment system of a package lost in deep water.

Impacts of No-Action Alternative

The No-Action alternative would not result in any change to the current level of radiological exposure consistent with the NRC's policy to maintain radiation exposure to workers and the public as low as reasonably achievable.

4.2.8 Grandfathering Previously Approved Packages

Impacts of Proposed Action

Under the proposed change, packages would be subject to existing regulations in 10 CFR Part 71 after renewal of the existing Certificate of Compliance, when the proposed regulations would apply. The existing and proposed regulations are believed to be equally protective of human health and the environment. Thus, an increase in potential environmental, human health, and safety impacts as a result of the proposed change is not expected.

Impacts of No-Action Alternative

Under the No-Action alternative, all packages would be subject to proposed regulations in 10 CFR Part 71 on the effective date of the rule. The proposed regulations are believed to be protective of human health and the environment. Thus, an increase in potential environmental, human health, and safety impacts as a result of the No-Action alternative is not expected.

[17] NRC, 1994, Regulatory Analysis of Changes to 10 CFR Part 71 – NRC Regulations on Packaging and Transportation of Radioactive Material, Division of Safeguards & Transportation, Office of Nuclear Material Safety & Safeguards, Washington, DC, August.

[18] Ibid.

4.2.9 Crush Test for Fissile Material Package Design

Impacts of the Proposed Action

It is expected that the proposed action would have negligible effects on the planning, packaging, inspection, loading, shipping, or receiving of packages. Analysis of the shipping process reveals that the proposed action will not affect planning and pre-shipment preparation activities. While the packaging requirements for fissile material packages may result in the requirement for crush testing of previously exempted packages, this is not expected to result in any increase in occupational exposure. Likewise, inspection, loading, shipping and receiving activities would not deviate from those required without this proposed rulemaking.

Impacts of the No-Action Alternative

The No-Action alternative would not result in any change to the current level of radiological exposure consistent with the NRC's policy to maintain radiation exposure to workers and the public as low as reasonably achievable.

4.2.10 Fissile Material Package Designs for Transport by Aircraft

Impacts of the Proposed Action

It is expected that the proposed action would have negligible effects on the planning, packaging, inspection, loading, shipping, or receiving of packages. Analysis of the shipping process reveals that the proposed action will not affect planning and pre-shipment preparation activities. The adoption of the additional criticality evaluation is not expected to result in any increase in occupational exposure. To the contrary, the additional requirement for criticality evaluation is likely to result in a decrease in exposure from fissile materials in the case of an accident during transport by aircraft. Inspection, loading, shipping and receiving activities would not deviate from those required without this proposed rulemaking.

Impacts of the No-Action Alternative

The No-Action alternative would not result in any change to the current level of radiological exposure consistent with the NRC's policy to maintain radiation exposure to workers and the public as low as reasonably achievable.

4.3 Environmental Impacts of NRC-Specific Proposed Actions

4.3.1 Special Package Authorizations

Impacts of Proposed Action

This proposed action is not expected to result in any increased or decreased radiological exposure relative to current requirements. Shipments under special arrangement are expected to continue to be a preferred method of shipment based on lower radiation exposures to the general public and workers as well as reductions in costs and decommissioning timeframes. NRC standardization of safety information collection requirements will not impact the number of shipments authorized under special arrangement.

Analysis of the shipping process reveals that the proposed action will not affect planning and pre-shipment preparation activities. Although the demonstrated level of safety required of the shipper is to be standardized, the impact to the shipper in the pre-shipment stage can be assumed to be negligible. Similarly, the packaging requirements for special arrangement shipments will not be affected. An increased number of special arrangement shipments may be anticipated in the future, due to further decommissioning efforts of the nation's nuclear power reactors. This increase in the number of shipments, however, remains unrelated to the outcome of this proposed action. Likewise, inspection, loading, shipping and receiving activities would not deviate from those required without this proposed rulemaking.

Such shipments involve no irreversible or irretrievable commitments of resources and continued approval of them will result in a negligible change in radiological exposure relative to current requirements. Demonstration of safety for special arrangement shipments ensures that the safety of each shipment is consistent with the NRC's policy to maintain radiation exposure to workers and the public as low as reasonably achievable.

Impacts of No-Action Alternative

The No-Action alternative would not result in any change to the current level of radiological exposure consistent with the NRC's policy to maintain radiation exposure to workers and the public as low as reasonably achievable.

4.3.2 Adoption of ASME Code

Impacts of Proposed Action

The full-time presence of the ANI would likely prevent fabrication errors that might otherwise not be identified. Because licensee and contractor QA plans are not currently subject to full-time on-site verification by the NRC or another outside auditor, NRC has limited assurances that all licensees have implemented a competent QA plan. The ANI is an independent professional capable of reporting QA issues to the management of the licensee/fabricator, and to the NRC. The shop/field surveys and preparation of a PE-certified design report ensure that the design of the container, and the fabrication area meet ASME Code standards. Without surveys and PE design approval, there is no assurance that the fabrication area and container designs meet the NRC safety standards. The presence of a full-time ANI in the fabrication shop would substantially decrease the likelihood of flawed cask/container production.

Implementation of the proposed action is not expected to affect shipment planning activities or shipping requirements. In the case of an accident during packaging, inspection, loading, shipping, unloading, or receiving, the marginally safer casks that are produced as a result of ASME code implementation would result in a very slightly increased level of safety for workers and emergency responders. Shipping casks were found to exhibit a satisfactory level of safety in the December 1977 NRC EIS *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes.* The accident frequency during the transportation of shipping casks is projected to be very low (there has never been an accident involving a cask), and the casks are considered safe without currently implementing the ASME QA/QC procedures. It is difficult to quantify the increased level of safety that enhanced QA/QC procedures through full code implementation would achieve. However, the marginal improvement in safety due to ASME code implementation is not expected to significantly decrease the consequences of accidents. It is therefore expected that implementation of this measure will have a negligible positive effect on the environment.

Impacts of No-Action Alternative

If the ASME code is not implemented for spent fuel casks and dual-purpose casks, the current inconsistent system of licensee QA procedures would remain in place. NRC and the licensees would be responsible for ensuring that adequate QA procedures are followed. NRC does not have the staffing capability to engage in full-time fabricator supervision. Licensees and contractors would therefore continue to self-certify that they are implementing a competent QA plan, and continue their own QA procedures. The marginal improvement in cask safety obtained through implementation of the ASME code would therefore not be achieved.

4.3.3 Fissile Material Revisions

Impacts of Proposed Action

The main purpose of the transportation regulations for fissile materials is to ensure that subcriticality can be maintained under both normal and hypothetical accident conditions. The regulations are formulated to ensure subcriticality by specifying requirements for packages containing fissile material and implementing operational controls for its shipment. The package requirements are intended to ensure that the chemical, physical, and material conditions of the package necessary for subcriticality are always maintained. Further, the implementation of operational controls (e.g., TI) provides straightforward procedures for the safe handling of packages by transportation workers.

> The principal parameters of concern in controlling the criticality safety (maintaining subcriticality) of transportation packages are:
>
> - type, mass, and form of fissile material;
> - moderator-to-fissile material ratio;
> - amount and distribution of moderator and absorber materials;
> - package geometry; and
> - reflector effectiveness.

The fissile material exemptions and general licenses of 10 CFR Part 71 provide no requirements for packaging assessments relative to criticality safety. Hence, controls provided by package geometry or absorber/moderator materials cannot be relied upon in the assessment of regulatory specifications. In addition, the abundance of water in nature and its effectiveness as a reflector would limit the controlling parameters to type, mass, form and moderator-to-fissile material ratio for ensuring subcriticality of the shipments containing fissile material in packages that are exempt from a criticality safety assessment.

Table 4-2 summarizes the various criteria provided within the revised (current) 10 CFR Part 71 under the general licenses sections (sections 71.18, 71.20, 71.22, and 71.24) and the fissile exemptions section (section 71.53) for transport of fissile material and provides various limit values for comparison. These criteria were developed to control the transport of less than Type A[19] quantities of fissile material by specifying mass limits. Only NRC licensees with an approved quality assurance program can ship fissile materials using a general license. These shipments are controlled either via use of a TI for each package (sections 71.18 and 71.20) or DOT shipment requirements that prevent commingling with other fissile material shipments (sections 71.22 and 71.24). The latter sections (sections 71.22 and 71.24) allow for an increased quantity of fissile material within a controlled shipment (e.g., an exclusive-use shipment), apparently perceiving controlled shipments as providing an added safety margin. The fissile material exemptions allow packages that meet the content specifications of section 71.53 to exclude the standards and controls requirements of sections 71.55 and 71.59 regarding fissile material packages.

Table 4-2. Comparison of Allowable Limits and Requirements Under the General Licenses and Fissile Exemptions

Provisions (Sections of 10 CFR Part 71)	Mass limits per package	Mass limits Non-exclusive use shipment [a]	Mass limits Exclusive use shipment [b]	Methods of control per package
§71.18(c)	up to 40 g of ^{235}U, or up to 30 g of ^{233}U, or up to 25 g of ^{239}Pu	up to 200 g of ^{235}U, or up to 150 g of ^{233}U, or up to 125 g of ^{239}Pu	up to 400 g of ^{235}U, or up to 300 g of ^{233}U, or up to 250 g of ^{239}Pu	TI of 10 (criticality)
§71.18(d) -- mixed with substances having a hydrogen density > water	up to 29 g of ^{235}U, or up to 18 g of ^{233}U, or up to 18 g of ^{239}Pu	up to 145 g of ^{235}U, or up to 90 g of ^{233}U, or up to 90 g of ^{239}Pu	up to 290 g of ^{235}U, or up to 180 g of ^{233}U, or up to 180 g of ^{239}Pu	TI of 10 (criticality)
§71.18(c)(3) and §71.18(f)(2)	A_1 quantity of encapsulated Pu-Be neutron source in special form: up to 400 g of ^{239}Pu	up to 2,000 g of ^{239}Pu	up to 4,000 g of ^{239}Pu	TI of 10 (criticality)
§71.20	up to 40 g of ^{235}U (for enrichment > 24%)	up to 200 g of ^{235}U	up to 400 g of ^{235}U	TI of 10 (criticality)
§71.22(d)(1)	Not Applicable	Not Applicable	up to 500 g of ^{235}U, or up to 300 g of others[c]	Exclusive Use[d], TI of 100
§71.22(c) and §71.22(d)(2)	up to 400 g ^{239}Pu in Pu-Be neutron source	Not Applicable	up to 2,500 g of ^{239}Pu	

[19] Section 71.4 defines Type A quantity as: "A quantity of radioactive material, the aggregate of which does not exceed A_1 for special form radioactive material, or A_2, for normal form radioactive material. The values of A_1 and A_2 are given in [Table A-1 of 10 CFR Part 71]."

Table 4-2. Comparison of Allowable Limits and Requirements Under the General Licenses and Fissile Exemptions (continued)

Provisions (Sections of 10 CFR Part 71)	Mass limits per package	Mass limits Non-exclusive use shipment [a]	Mass limits Exclusive use shipment [b]	Methods of control per package
§71.22(d) -- mixed with substances having a hydrogen density > water	Not Applicable	Not Applicable	up to 290 g of ^{235}U, or up to 180 g of others[c]	Exclusive Use, TI of 100[d]
§71.24(b)(6) -- <1% ^{233}U in the package	Not Applicable	Not Applicable	up to 520 g of ^{235}U (for enrichment >20%)	Exclusive Use, TI of 100[d]
§71.24(b)(7) -- >1% ^{233}U in the package	Not Applicable	Not Applicable	up to 400 g of ^{235}U, or up to 225 g of ^{233}U, or up to 250 g of ^{239}Pu	Exclusive Use, TI of 100[d]
§71.53(a)	up to 15 g of fissiles, or up to 5 g of fissiles per any 10 liter volume	up to 400 g of ^{235}U, or up to 250 g of others	up to 400 g of ^{235}U, or up to 250 g of others	Consignment Mass
§71.53(a) -- mixed with substances having a hydrogen density > water	up to 15 g of fissiles, or up to 5 g of fissiles per any 10 liter volume	up to 290 g of ^{235}U, or up to 180 g of others	up to 290 g of ^{235}U, or up to 180 g of others	Consignment Mass

[a] Maximum TI of 50 for the shipments under general licenses [per §71.59(c)(1)].
[b] Maximum TI of 100 for shipments under general licenses [per §71.59(c)(1)].
[c] Others mean the sum of other fissile material (e.g., ^{233}U and ^{239}Pu).
[d] Sum of TIs of all packages.

There are several inconsistencies within the criteria provided in Table 4-2 relative to shipment requirements and allowed fissile masses. For example, there is a mass inconsistency between an exclusive-use shipment made under section 71.18 (or section 71.20) versus that made under section 71.22 (or section 71.24). The public comments and NRC staff concerns with respect to these inconsistencies led NRC to contract with ORNL to further assess the revised 10 CFR Part 71 exemptions and general licenses. In most cases, the ORNL study documented in NUREG/CR-5342 concluded that the quantities of fissile material allowed in a shipment under any of the general licenses and fissile material exemptions have a sound technical basis related to (1) information on minimum critical masses of water-reflected, water-moderated systems, and (2) that the minimum critical mass would always occur for a hydrogenous-moderated system.

Table 4-3 summarizes the critical and subcritical minimum mass values calculated for selected moderators and fissile material. As shown, subcriticality ($K_{eff} \leq 0.95$) is readily maintained with a water-moderated fissile-material mass value (614 g of ^{235}U, 437 g of ^{233}U, and 379 g of ^{239}Pu) greater than that allowed by the general license provisions of section 71.18 (400 g of ^{235}U, 300 g of ^{233}U, and 250 g of ^{239}Pu) and section 71.22 (500 g of ^{235}U, 300 g of others [^{233}U, and

Table 4-3. Critical and Subcritical Minimum Mass Values Calculated for Selected Moderators

Moderator (Density: g/cm^3)	Fissile Material	Calculated Minimum Fissile Mass Values (g)		Moderator Mass (g) at Minimum Value		Subcritical Dimension [b] (cm)
		Subcritical $k_{eff} \le 0.95$	Critical $k_{eff} = 1.0$	Subcritical $k_{eff} \le 0.95$	Critical $k_{eff} = 1.0$	
H_2O (0.996)	^{235}U	614	820[a]	11,760	15,700	14.03
	^{233}U	437	600[a]	7,600	10,000	14.5
	^{239}Pu	379	510[a]	12,840	18,000	12.2
CH_2 (0.96)	^{235}U	N.C.	527	N.C.	7,394	12.3
	^{233}U	N.C.	N.C.	N.C.	N.C.	N.C.
	^{239}Pu	N.C.	N.C.	N.C.	N.C.	N.C.
SiO_2 (1.6)	^{235}U	147,000	N.C.	43,162,000	N.C.	186.5
	^{233}U	61,616	N.C.	17,453,000	N.C.	199.2
	^{239}Pu	72,688	N.C.	52,919,000	N.C.	137.6
C (2.1)	^{235}U	2,186	N.C.	2,792,000	N.C.	68.2
	^{233}U	1,722	N.C.	1,951,000	N.C.	67.3
	^{239}Pu	1,212	N.C.	2,677,000	N.C.	60.54
Be (1.85)	^{235}U	765	N.C.	351,600	N.C.	35.6
	^{233}U	605	N.C.	233,700	N.C.	35.1
	^{239}Pu	424	N.C.	335,300	N.C.	31.1
D_2O (1.1)	^{235}U	1,044	N.C.	444,300	N.C.	45.8
	^{233}U	851	N.C.	219,000	N.C.	43.4
	^{239}Pu	602	N.C.	378,000	N.C.	36.2

[a](Paxton and Pruvost, 1986).
[b]The radius of a fully-water reflected sphere of a homogeneous fissile material and the selected moderator.
N.C. = Not calculated.
Source: NUREG/CR-5342.

^{239}Pu]). The referenced critical mass values for similar systems are 820 g of ^{235}U, 600 g of ^{233}U, and 510 g of ^{239}Pu (Paxton and Pruvost, 1986). The subcritical mass values were calculated considering a fully-water reflected sphere of homogeneous fissile material and water and other select moderators. Also, the study evaluated the potential for criticality arising from the accumulation of ^{235}U with select moderators in 208-liter (55-gallon) drums, that could be in five public highway transport vehicles (each vehicle pulling two tandem trailers), arranged in a fully-water reflected near-cubic array with optimum pitch geometry. The results of these evaluations indicated that a sufficient margin of subcriticality would be maintained. In other words, fissile material masses far in excess of those currently limited by the exemptions are required to reach criticality.

The ORNL study identified two provisions where sufficient margin of safety could not be ensured:

1. The general licenses provisions in sections 71.18(c)(3) and 71.18(f)(2) allow up to 400 g of ^{239}Pu in an encapsulated plutonium-beryllium neutron source in special form to be present in a package (see Table 4-3). This amount of plutonium is close to its subcritical mass limit with beryllium as a moderator in an idealized configuration (see Table 4-3). Unless there are provisions that specify limiting materials of construction and packaging to those that would ensure subcriticality, the current packaging under the

general licenses cannot be relied on in the criticality assessment. Therefore, the shipment quantities as given in Table 4-2 have a potential for criticality.[20]

2. The exemption for low-level materials criterion, as given in section 71.10(a), could lead to a criticality situation if the package limiting specific activity of 70 Bq/g (0.002 µCi/g) were to be all from fissile material (i.e., ^{235}U). Even though 70 Bq/g (0.002 µCi/g) of highly enriched uranium (i.e., 93% ^{235}U) per gram of material or 0.029 g of highly enriched uranium per liter of water is far below the minimum critical concentration of 12 g per liter (Paxton and Pruvost 1986), it would exceed an idealized infinite media subcritical concentration value if heavy water were the moderator. The infinite media subcritical concentration value in heavy water is 0.0192 g ^{235}U per liter.

Except for the conditions stated above, the results of the ORNL study generally indicated that for all shipments, the likelihood of accumulating sufficient fissile material to achieve criticality is highly improbable; such an occurrence would require the complete loss of packaging and an idealized spherical configuration under normal and/or accident conditions.

As stated earlier, the specified regulations in 10 CFR Part 71 are formulated to ensure subcriticality during transport of waste and fissile material packages. The ORNL study concluded, with two exceptions, that the specified regulations provide sufficient safety margin (subcritical margin) to make a criticality condition highly improbable. Any potential for criticality during normal conditions of transport and/or hypothetical accident conditions is considered unacceptable by NRC, and would require immediate enactment of regulatory revisions to preclude criticality. Therefore, the analysis of potential impacts to human health and the environment, particularly as it pertains to the transportation of fissile materials packages from implementation of the alternatives, is primarily focused on criticality safety.

NRC's emergency rulemaking for 10 CFR Part 71 referenced the Commission's generic environmental impact statement (NUREG-0170), which analyzed radioactive material transportation by various modalities (e.g., road, rail, air, and water). That document found the overall transportation risk for all radioactive materials acceptable from a regulatory standpoint. Further, for a given year, NUREG-0170 estimated approximately 100 million hazardous materials packages (flammables, explosives, poisons, and radioactive materials) are shipped in the United States. Of those shipments, fewer than 5 percent contained radioactive materials.[21] Although NUREG-0170 did not state the number of limited quantity, fissile material shipments containing special moderating materials, it did estimate that 50,000 fissile material packages

[20] It should be noted that the shipment quantities used in the analysis are based upon the upper limit value per package that could potentially be present considering pure ^{239}Pu. Historically, plutonium-beryllium neutron sources have been used by universities and DOE laboratories. Currently, such sources are being returned to the Los Alamos National Laboratory for treatment and disposal. Those that have been returned so far have had ^{239}Pu masses in the range of 16-32 g (non-exclusive shipments with TIs of 2 to 7), and a few have had a maximum mass of 160 g (exclusive shipments with TIs of 10-12). These shipments were made using a detailed operational control in packaging and limiting a maximum TI of 50 for both the exclusive and non-exclusive shipments (LANL, 1998). The limiting TI for these sources is from the neutron radiation, not criticality.

[21] The most recent study of the transport of radioactive materials captured data on the shipment of radioactive materials for the 1982 calendar year and concluded that approximately 2 million shipments of radioactive materials are made each year (SNL, 1984). These 2 million shipments constitute about 2.79 million packages of radioactive materials. The 2 million radioactive materials shipments account for only 3 % of the total number of hazardous materials transported each year in the United States.

(used for larger quantities of, and/or more highly enriched, fissile materials) were shipped in 1985.

In its finding of no significant impact (FONSI) for the emergency rule, NRC concluded that the overall transportation risk estimated in NUREG-0170 bounds the potential impacts associated with the proposed fissile material changes for 10 CFR Part 71 (62 FR 5907, February 10, 1997). In addition, NRC argued that the number of shipments affected by the emergency rule was a small fraction of the 50,000 fissile material packages addressed in NUREG-0170. Therefore, because fissile material packages containing special moderating materials are less common than those containing moderately enriched fissile materials, NRC concluded that the transportation risk for these shipments was smaller still.

As discussed previously, beyond the data presented in NUREG-0170 (including its 1985 update), the literature contains no more recent studies that estimate either the number of fissile material shipments or the number of fissile material shipments containing special moderating materials. Although a credible transportation baseline for these shipments cannot be established, even if the number of shipments of fissile materials significantly increases or decreases as a result of the proposed rulemaking, as documented in NUREG-0170, public exposures from routine shipments of this type are negligible.

Table 4-4 presents the qualitative definitions of potential impacts used in this assessment.

Table 4-4. Qualitative Definitions of Impacts Used to Signify the Importance of Each Recommendation

None	No significant effect on the quality of the human environment.
Small	The effects on the quality of the human environment are not detectable or are so minor that they would neither destabilize nor noticeably alter any resource.
Medium	The effects on the quality of the human environment are sufficient to alter noticeably, but not to destabilize, any resource.
Large	The effects on the quality of the human environment are clearly noticeable and sufficient to destabilize any resource.

(Adapted from 10 CFR Part 51)

Table 4-5 summarizes the Proposed Action’s recommendations and their potential impacts, in qualitative incremental changes (positive impact for increase in consequences and negative impact for decrease in consequences), as compared to those described in the No-Action alternative.

Impacts of No-Action Alternative

The No-Action alternative is the continued use of modified regulations issued under emergency order as currently codified in 10 CFR Part 71. As explained earlier and detailed in the ORNL study, the current regulations on general licenses need to be revised to provide consistent criteria related to shipments and fissile material masses, and at least two of the

Table 4-5. Recommended Changes to 10 CFR Part 71 and Their Qualitative Impacts

Category	#	Recommendation	Qualitative Impacts
General	1	Clarifications of the definitions in 10 CFR Part 71	None: This recommendation only enhances the definitions; thus, environment, health and safety are not impacted.
	2	Clarification of the "fissile material" definition	None: This recommendation reduces the regulatory burden to licensees and makes the requirements consistent with those promulgated by the IAEA.
	3	Revision to exemptions for low-level material	Large: This recommendation precludes the potential for criticality. Shipments of radioactive material with known quantities of fissile material would no longer be exempt from the §71.53 requirements. Previously, for example, there was no limit on the number of fissile exempt packages that could be shipped in a single consignment. By taking away this exemption, the concern over inadequate criticality safety in exempted quantities of fissile material would be lessened.
General (continued)	4	Placement of fissile material exemption under Subpart B	None: This recommendation consolidates the fissile material exemptions under one heading.
	5	Modification to §71.10(b)	None: This recommendation consolidates the fissile material exemptions under one heading.
	6	Establishment of a shipment database	None: This recommendation only provides for future quantitative evaluations of impacts.
General Licenses	7	Removal, or modification, of provisions related to the shipment of Pu-Be neutron source	Large: This recommendation precludes criticality potential. The current amount of plutonium (in an encapsulated Pu-Be neutron source) allowed to be shipped is not technically justified based on available information and is close to its subcritical mass limit. Unless there are provisions that specify limiting materials of construction and packaging to those that would ensure subcriticality, the current packaging under the general licenses cannot be relied on in the criticality assessment. Thus, removing or modifying the Pu-Be neutron source provisions would greatly enhance criticality avoidance.
	8	Consolidation of general licenses for controlled shipment and for limited quantity per package	Small: This recommendation simplifies the general license provisions and eliminates confusion by making them consistent with §71.59. This would involve merging sections addressing general licenses for controlled shipments (§71.22 and §71.24) with sections addressing general licenses for limited quantity/moderator per package (§71.18 and §71.20). Consolidating all of these regulations would act to streamline the licensing process. In addition, the section would be revised to provide guidance on the criticality control transport index.
	9	Elimination of ^{235}U distribution distinctions	None: This recommendation simplifies regulations.
	10	Clarification of General Licenses select moderator restrictions	None: This recommendation simplifies regulations.
	11	Maintenance of mass control for moderators with a hydrogen density greater than water	None: This recommendation simplifies regulations.

Table 4-5. Recommended Changes to 10 CFR Part 71 and Their Qualitative Impacts (continued)

Category	#	Recommendation	Qualitative Impacts
General Licenses (continued)	12	Specification for minimum package requirements	Small: This recommendation provides assurance for safe and secure transport of fissile material.
	13	Increase of package mass limits for general licenses	None: This recommendation reduces confusion.
Fissile Material Exemptions	14	Revision to mass-limited exemptions and removal of restrictions on Be, C, and D_2O[2]	Small: This recommendation allows a consistent mass limit within various sections, and reduces number of packages under §71.18. This approach would add enhanced assurance in preventing a potential transport situation that could provide a criticality safety concern, and maintain flexibility for regulators, licensees, and operators by precluding the need to prescribe and use a TI for transport control.
	15	Deletion of requirements in §71.53(a), (c), and (d); restrictions on Be, C, and D_2O	Small: (See #14.)
	16	Addition of minimum packaging standard for §71.53(c)	Small: (See #12.)
	17	Removal of homogeneity requirements in §71.53(b)	None: This recommendation simplifies regulations

provisions (i.e., sections 71.18(c)(2) and 71.10(a)) need to be modified to preclude a potential for adverse criticality safety under any hypothetical condition. Therefore, the No-Action alternative, as it stands, could lead to a criticality condition, the consequences of which are unacceptable from a regulatory standpoint.

4.3.4 Double Containment of Plutonium (PRM-71-12)

<u>Impacts of Proposed Action</u>

DOE is required to follow NRC regulations when shipping plutonium. Most plutonium shipments will be made by DOE in association with:

- Surplus Plutonium Disposition;
- Plutonium Residue and Scrub Alloy;
- Plutonium 238 Supply; and
- Waste Isolation Pilot Plant Disposal.

DOE prepared EISs for each of these projects. The EISs included public and occupational health impacts for each of the projects. None of the EISs appear to adjust the impacts of accidents for the increased level of safety associated with the double-containment of plutonium. However, based on the information in these EISs and a review of the existing packaging requirements, it was concluded that the proposal to delete the section 71.63 special requirements for plutonium shipments would result in the following impacts.

1. Planning and preshipment would not be affected.

2. Workers currently receive additional exposure while sealing the second layer of packaging. Eliminating this step and the associated radiation exposure could result in a reduction of 0.004 latent cancer fatalities per year. However, most of DOE's plutonium is normally stored in a "storage" package that would act as an inner container for shipment. Much of DOE's plutonium is in, or will be moved to, containers that meet DOE-STD-3013-96, *Criteria for Preparing and Packaging Plutonium Metals and Oxides for Long-Term Storage*. Steps are in progress to ship DOE's transuranic waste in TRUPACTs, which provide double-containment. Several other double containment packaging systems are also in use.

3. Most conceivable plutonium transportation, whether under double containment regulations or not, would use sealed inner containers. Therefore, no change to inspection efforts is anticipated.

4. Since the additional container does not provide significant shielding against the high energy gamma rays associated with plutonium and americium (a daughter product of plutonium), there would be no significant difference in loading risks.

5. Removing a layer of packaging (protection) increases the probability and consequences of accidents that can breech the Type B package. The total risk of plutonium transportation is less than 0.1 latent cancer fatalities per year (depending on the alternatives chosen by DOE). None of the EISs take explicit credit for the double containment of plutonium, and plutonium is only released in the most severe accidents hypothesized. No detailed technical analysis has been located, but removing the requirement for double containment could add as much as 0.05 latent cancer fatalities per year. Deletion of section 71.63(a) could increase an individual shipment's accident risk, primarily if and when plutonium is shipped in liquid form. Given the unlikely occurrence of a severe accident, approximately 100 times more liquid plutonium would be released compared to solid plutonium subjected to the same accident. However, most plutonium shipments are either related to disposition of plutonium wastes or to production of MOX. Neither process would create a need to ship a liquid plutonium solution.

6. Since the plutonium will most likely be left in the inner container, no change is expected at the receiving site.

<u>Impacts of No-Action Alternative</u>

The No-Action alternative would not result in any change to the current level of radiological exposure consistent with the NRC's policy to maintain radiation exposure to workers and the public as low as reasonably achievable.

4.3.5 Contamination Limits as Applied to Spent Fuel and High Level Waste (HLW) Packages

Impacts of Proposed Action

DOT's regulations in 49 CFR 173 provide two sets of limits for surface contamination: one for wiping and one that is ten times higher for other types of appropriate contamination testing. The wipe limits are ten times lower because it is assumed that wiping has an efficiency of 10 percent; therefore, if the wipe limits are multiplied by ten, they are the same as the limits given for other contamination assessments.

The proposed action would not change the basic limit for surface contamination of packages being transported, which is 4 Bq/cm2 (10-4 µCi/cm2) for beta and gamma emitters and low toxicity alpha emitters and 0.4 Bq/cm2 (10-5 µCi/cm2) for all other alpha emitters. Because the limits for surface contamination would not change, the proposed action would not result in any human health or environmental impacts from the planning, packaging, inspection, loading, shipping, or receiving of packages of radioactive material.

Impacts of No-Action Alternative

The No-Action alternative would not result in any change to the current level of radiological exposure consistent with the NRC's policy to maintain radiation exposure to workers and the public as low as reasonably achievable.

5. Agencies and Persons Consulted

Babcock and Wilcox, Naval Nuclear Fuel Division, Preston L. Foster

Los Alamos National Laboratory, S. Jones

Oak Ridge National Laboratory, Richard Rawl

U.S. Department of Transportation, Fred Feratti

U.S. Nuclear Regulatory Agency, John Cook

U.S. Nuclear Regulatory Agency, Philip Brochman

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APPENDIX A

APPENDIX A
NUREG/CR-5342 Recommendations

The bases for and clarity of the general licenses for fissile material and the exemptions for fissile material in 10 CFR Part 71 have become increasingly obfuscated with adjustments and accommodations of the regulations over time, as well as with shipper (consignor) interpretations and applications. Any proposed revision of these portions of the regulations should seek to provide clear, unambiguous, and straightforward specifications. The regulations should specify simplified bounding requirements that provide fissile material general licenses and exemptions with a near equivalency in safety as that applied to packages certified to transport fissile material.

This section provides and discusses a consistent set of recommendations that are judged to be the most straightforward and effective for consideration in any future rule making process.

A.1 General Recommendations

- Consistency in definition and stated intent needs to be provided to the extent possible. It is recommended that definitions for "consignment," "consignor," and "shipper" be provided. Furthermore, the licensee is subject to possible confusion because of the differences between the wording used in 49 CFR 173 and 10 CFR Part 71. Even within 10 CFR Part 71 there are instances where no guidance or definition of words is provided to help clearly identify or explain the required specifications. For example, the regulations need to eliminate the wording "controlled shipment" or distinguish it from "exclusive-use shipment."

- The definition of fissile material should be simplified and made technically correct by eliminating the nuclide ^{238}Pu from the definition. The impracticality of obtaining a large enough mass required for criticality (6 kg) and the high decay heat rate prevent any conceived consequences of this change that are adverse to criticality safety. Similarly, the usage of the words "fissile material" in the regulations needs to be clarified; sometimes it is used to specify fissile nuclides, while other times it is used to imply material containing fissile nuclides.

- The criteria for exempting fissile material from consideration as radioactive material regulated by 10 CFR Part 71 [e.g., section 71.10(a)] should be revised to not allow material with known quantities of fissile material from being included in the radioactive material exemption. This is the simplest and most straightforward approach. An alternative would be to lower the exemption concentration such that an infinite system would be subcritical. These criteria correspond to a value of 43 Bq/g (0.001 µCi/g) and are judged to be sufficiently limiting for all materials. An infinite medium subcritical concentration is sufficiently small, and the associated volume for criticality so large, that a change in concentration associated with the required volume for criticality is not deemed probable in a practical system.

- Although not discussed previously in the assessment, it is also recommended that 71.10(b) be modified to ensure that exemptions are not provided to fissile material that should meet some packaging requirements (e.g., section 71.53(d)). The recommendations under Section A.3 include some additional packaging requirements for selected fissile-material exemptions.

- The fissile-material exemptions should be moved to Subpart B – "Exemptions." Placement of the fissile-material exemptions under Subpart B would be more consistent with the placement of other exemptions of 10 CFR Part 71.

- The NRC or DOT should consider keeping a database of shipments made under fissile-material exemptions and general license(s). The database should include a description of material shipped; the mass of fissile material in the consignment or shipment; the TI of the shipment, if applicable; the exemption criteria satisfied, if applicable; and the package description, if applicable. The database would be used to provide the NRC with historical information to better understand the type of material being shipped under the fissile-material exemptions and general licenses so that a more informed decision can be made relative to the impacts of any future changes to these portions of the regulations.

A.2 Recommendations for General Licenses

- The provisions related to shipment of Pu-Be sources should be removed from the general licenses. It may be possible to develop a separate general license for Pu-Be sources. The quantity of plutonium currently allowed to be shipped as Pu-Be sources is not technically justified based on available information and the lack of packaging requirements provided in the current regulations. Any new section that is developed should revise the quantity of plutonium allowed to be shipped as Pu-Be neutron sources and/or provide packaging requirements that prevent challenges to the basis for criticality safety.

- The general licenses for controlled shipments (sections 71.22 and 71.24) should be merged with the general licenses for limited quantity per package (sections 71.18 and 71.20) to provide a single general license paragraph that consolidates the needed technical criteria and operational controls. This merger, together with a clear specification of the aggregate TI allowed for nonexclusive use and exclusive use, should provide consistency with the approach of section 71.59 and simplify the regulations.

- The distinction between quantities of ^{235}U that can be shipped as a uniform distribution and nonuniform distribution should be eliminated. The bounding nonuniform quantities should be used. This change is recommended because the simplicity offered by this solution outweighs the complexity and confusion that would result from trying to develop a comprehensive definition for "nonuniform," which is currently lacking in the regulations.

- Restrictions on quantities of Be, C, and D_2O should be removed from the general licenses, except perhaps to indicate these materials should not be present as a reflector material. Restricting its presence in quantities that might provide reflection of neutrons should be fairly simple and would be prudent since these packages are not under regulatory review. Limiting the quantity of these materials to 500 g per package should eliminate any concern relative to their effectiveness as a reflector.

- Maintaining a separate mass control (e.g., section 71.18) or restriction (e.g., section 71.20) for moderators having a hydrogen density greater than water is recommended. Where separate mass limits are provided, the fissile mass limit associated with moderators having hydrogen density greater than water should be used whenever such a high-density hydrogenous moderator exceeds 15% of the mass of hydrogenous moderator in the package.

- Minimum package requirements as provided by section 71.43 should be specified for shipments under the general licenses. The intent is to include good practice that an NRC licensee should have in place under a quality assurance program that handles shipment of fissile material with low specific activity.

- The package mass limits currently allowed by sections 71.18 and 71.20 should be increased to provide similar safety equivalence provided by certified packages per the criteria of sections 71.55 and 71.59. Justification for these increases is based partly on the implementation of an improved minimum packaging standard (section 71.43), as discussed above. The recommended mass values are provided in Tables A-1 and A-2. The values in Table A-1 were obtained by raising the mass limits to just under the mass values that ensure subcriticality ($k_{eff} \leq 0.95$) based on the information of Table 3. The fissile-material mass values for systems with moderators having a hydrogen density greater than water were subsequently obtained by using a scaling factor based on the ^{235}U critical mass values for a water-moderated system (820 g) and a system moderated by high-density polyethylene (527 g). The values of Table A-3 were obtained using a scaling factor based on the ratio of the new water-moderated ^{235}U limit shown in Table A-2 (60 g) and the existing value of section 71.18 (40 g).

A.3 Recommendations for Fissile-Material Exemptions

- The mass-limited exemptions of section 71.53(a) should be revised to provide criteria based on a ratio of the mass of fissile material per mass of nonfissile material. The nonfissile material considered in the ratio determination should be insoluble-in-water and noncombustible. It may be necessary to provide a definition and/or criteria for such material. Mass quantities of Be, C, and D_2O should be excluded from consideration as nonfissile material for the purposes of determining the ratio value. This approach would:

  1. add enhanced assurance in preventing a potential transport situation that could provide a criticality safety concern; and

  2. maintain flexibility for regulators, licensees, and operators by precluding the need to prescribe and use a TI for transport control.

  Mass ratios are often easier for licensees to determine than values related to volumetric concentration, and they can be defined to provide sufficient control under hypothetical accident conditions (i.e., assurance that desired volumes are maintained during hypothetical accident conditions is much more difficult than assurance that mass values are maintained). The recommended ratios of fissile-to-nonfissile mass for the various exemption considerations are provided in Table A-3. If the approach using mass ratios is not acceptable, then conveyance control based on a TI should be incorporated into the fissile exemptions.

Table A-1. Mass Limits for General-license Packages Containing Mixed Quantities of Fissile Material or ^{235}U of Unknown Enrichment

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

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

Table A-2. Mass Limits for General-license Packages Containing ^{235}U of Known Enrichment

Uranium enrichment in weight percent of ^{235}U not exceeding	Permissible maximum grams of ^{235}U per package (X)
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

Table A-3. Proposed Fissile-Exempt Mass Ratios to Replace Criteria of Section 71.53(a)

Package fissile material limit	Ratio: Fissile-to-nonfissile
15 g	1:200
350 g	1:2000
350 g	1:200[a]

[778a]Packaging required to satisfy standards for normal transport condition.

- The restriction on Be, C, and D_2O in sections 71.53(a), 71.53(c), and 71.53(d) should be removed if either approach (defined mass ratios or TI) discussed in the previous bullet is adopted.

- The exemption for uranyl nitrate solutions should be revised to incorporate packaging standards of section 71.43.

- The exemption for uranium enriched to less than 1 wt % ^{235}U should be modified to remove the requirement for homogeneity and prevention of a lattice arrangement. Instead, the moderator criteria restricting the mass of Be, C, or D_2O to less than 0.1% of the fissile mass should be maintained. This change removes the need to provide definitions which are difficult to define and to apply practically, such as “homogeneous”and “lattice arrangement.”

APPENDIX B

APPENDIX B
Questions Developed for Survey of Fissile Material Licensees

Packages

- How many packages of exempted and general licensed fissile materials does your firm typically prepare each year?
- How much does it cost your firm to prepare these fissile material packages?
- Which factors (e.g., labor, material, manifest, insurance, etc.) contribute to this cost?
- What is the typical dose rate at one meter from the surface for these fissile material packages?

Shipments

- How many shipments of exempted and general licensed fissile materials does your firm typically make each year?
- How much does it cost your firm to make these fissile material shipments?
- Which factors (e.g., labor, material, manifest, insurance, etc.) contribute to this cost?
- What is the average number of exempted and general license fissile material packages in a single shipment?
- What is the most common destination for these shipments, or the average distance shipped? (Please distinguish between truck and rail shipments, if applicable).

Material Characterization

- Which other radioactive materials (please specify by radionuclide, activity, and volume) are included in the packages containing fissile material?

Recommendations in NUREG/CR-5342 (provide separate information for each recommendation)

- How many more (less) fissile material packages will your firm prepare each year?
- What is the basis for this increase (decrease) in fissile material packages?
- Would your firm expect any increase (decrease) in worker or driver dose from shipping and handling? (If so, then how much increase [decrease] is expected?)

- What will be the average number of fissile material packages in a single shipment?
- Will your firm experience a change in the time required for recordkeeping or reporting?
- Will your firm experience a change in the time required for regulatory determinations or calculations?

APPENDIX C

APPENDIX C
Comparison of A_1 and A_2 Values in TS-R-1 and Part 71

Symbol of radionuclide	Element and atomic number	A1 TS-R-1 (TBq)	A1 PART 71 (TBq)	Δ A1 (TS-R-1-Pt 71)	Δ A1 (%)	A2 TS-R-1 (TBq)	A2 PART 71 (TBq)	Δ A2 (TS-R-1-Pt 71)	Δ A2 (%)
Ac-225 (a)	Actinium (89)	8.0×10^{-1}	6.0×10^{-1}	2.0×10^{-1}	33%	6.0×10^{-3}	1.0×10^{-2}	4.0×10^{-3}	40%
Ac-227 (a)		9.0×10^{-1}	4.0×10^{1}	3.9×10^{1}	98%	9.0×10^{-5}	2.0×10^{-5}	7.0×10^{-5}	350%
Ac-228		6.0×10^{-1}	6.0×10^{-1}	0.0×10^{0}	0%	5.0×10^{-1}	4.0×10^{-1}	1.0×10^{-1}	25%
Ag-105	Silver (47)	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%
Ag-108m (a)		7.0×10^{-1}	6.0×10^{-1}	1.0×10^{-1}	17%	7.0×10^{-1}	6.0×10^{-1}	1.0×10^{-1}	17%
Ag-110m (a)		4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%	4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%
Ag-111		2.0×10^{0}	6.0×10^{-1}	1.4×10^{0}	233%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Al-26	Aluminum (13)	1.0×10^{-1}	4.0×10^{-1}	3.0×10^{-1}	75%	1.0×10^{-1}	4.0×10^{-1}	3.0×10^{-1}	75%
Am-241	Americium (95)	1.0×10^{1}	2.0×10^{0}	8.0×10^{0}	400%	1.0×10^{-3}	2.0×10^{-4}	8.0×10^{-4}	400%
Am-242m (a)		1.0×10^{1}	2.0×10^{0}	8.0×10^{0}	400%	1.0×10^{-3}	2.0×10^{-4}	8.0×10^{-4}	400%
Am-243 (a)		5.0×10^{0}	2.0×10^{0}	3.0×10^{0}	150%	1.0×10^{-3}	2.0×10^{-4}	8.0×10^{-4}	400%
Ar-37	Argon (18)	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%
Ar-39		2.0×10^{1}	2.0×10^{1}	0.0×10^{0}	0%	4.0×10^{1}	2.0×10^{1}	2.0×10^{1}	100%
Ar-41		3.0×10^{-1}	6.0×10^{-1}	3.0×10^{-1}	50%	3.0×10^{-1}	6.0×10^{-1}	3.0×10^{-1}	50%
As-72	Arsenic (33)	3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%	3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%
As-73		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%
As-74		1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%	9.0×10^{-1}	5.0×10^{-1}	4.0×10^{-1}	80%
As-76		3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%	3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%
As-77		2.0×10^{1}	2.0×10^{1}	0.0×10^{0}	0%	7.0×10^{-1}	5.0×10^{-1}	2.0×10^{-1}	40%
At-211 (a)	Astatine (85)	2.0×10^{1}	3.0×10^{1}	1.0×10^{1}	33%	5.0×10^{-1}	2.0×10^{0}	1.50×10^{0}	75%
Au-193	Gold (79)	7.0×10^{0}	6.0×10^{0}	1.0×10^{0}	17%	2.0×10^{0}	6.0×10^{0}	4.0×10^{0}	67%
Au-194		1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%	1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%
Au-195		1.0×10^{1}	1.0×10^{1}	0.0×10^{0}	0%	6.0×10^{0}	1.0×10^{1}	4.0×10^{0}	40%
Au-198		1.0×10^{0}	3.0×10^{0}	2.0×10^{0}	67%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Au-199		1.0×10^{1}	1.0×10^{1}	0.0×10^{0}	0%	6.0×10^{-1}	9.0×10^{-1}	3.0×10^{-1}	33%
Ba-131 (a)	Barium (56)	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%
Ba-133		3.0×10^{0}	3.0×10^{0}	0.0×10^{0}	0%	3.0×10^{0}	3.0×10^{0}	0.0×10^{0}	0%
Ba-133m		2.0×10^{1}	1.0×10^{1}	1.0×10^{1}	100%	6.0×10^{-1}	9.0×10^{-1}	3.0×10^{-1}	33%
Ba-140 (a)		5.0×10^{-1}	4.0×10^{-1}	1.0×10^{-1}	25%	3.0×10^{-1}	4.0×10^{-1}	1.0×10^{-1}	25%
Be-7	Beryllium (4)	2.0×10^{1}	2.0×10^{1}	0.0×10^{0}	0%	2.0×10^{1}	2.0×10^{1}	0.0×10^{0}	0%
Be-10		4.0×10^{1}	2.0×10^{1}	2.0×10^{1}	100%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Bi-205	Bismuth (83)	7.0×10^{-1}	6.0×10^{-1}	1.0×10^{-1}	17%	7.0×10^{-1}	6.0×10^{-1}	1.0×10^{-1}	17%
Bi-206		3.0×10^{-1}	3.0×10^{-1}	0.0×10^{0}	0%	3.0×10^{-1}	3.0×10^{-1}	0.0×10^{0}	0%
Bi-207		7.0×10^{-1}	7.0×10^{-1}	0.0×10^{0}	0%	7.0×10^{-1}	7.0×10^{-1}	0.0×10^{0}	0%
Bi-210		1.0×10^{0}	6.0×10^{-1}	4.0×10^{-1}	67%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Bi-210m (a)		6.0×10^{-1}	3.0×10^{-1}	3.0×10^{-1}	100%	2.0×10^{-2}	3.0×10^{-2}	1.0×10^{-2}	33%
Bi-212 (a)		7.0×10^{-1}	3.0×10^{-1}	4.0×10^{-1}	133%	6.0×10^{-1}	3.0×10^{-1}	3.0×10^{-1}	100%
Bk-247	Berkelium (97)	8.0×10^{0}	2.0×10^{0}	6.0×10^{0}	300%	8.0×10^{-4}	2.0×10^{-4}	6.0×10^{-4}	300%
Bk-249 (a)		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	3.0×10^{-1}	8.0×10^{-2}	2.2×10^{-1}	275%
Br-76	Bromine (35)	4.0×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	33%	4.0×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	33%
Br-77		3.0×10^{0}	3.0×10^{0}	0.0×10^{0}	0%	3.0×10^{0}	3.0×10^{0}	0.0×10^{0}	0%
Br-82		4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%	4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%
C-11	Carbon (6)	1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
C-14		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	3.0×10^{0}	2.0×10^{0}	1.0×10^{0}	50%
Ca-41	Calcium (20)	Unlimited	4.0×10^{1}	NA	NA	Unlimited	4.0×10^{1}	NA	NA
Ca-45		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	1.0×10^{0}	9.0×10^{-1}	1.0×10^{-1}	11%
Ca-47 (a)		3.0×10^{0}	9.0×10^{-1}	2.1×10^{0}	233%	3.0×10^{-1}	5.0×10^{-1}	2.0×10^{-1}	40%

APPENDIX C
Comparison of A_1 and A_2 Values in TS-R-1 and Part 71 (continued)

Symbol of radionuclide	Element and atomic number	A1 TS-R-1 (TBq)	A1 PART 71 (TBq)	Δ A1 (TS-R-1-Pt 71)	Δ A1 (%)	A2 TS-R-1 (TBq)	A2 PART 71 (TBq)	Δ A2 (TS-R-1-Pt 71)	Δ A2 (%)
Cd-109	Cadmium (48)	3.0×10^{1}	4.0×10^{1}	1.0×10^{1}	25%	2.0×10^{0}	1.0×10^{0}	1.0×10^{0}	100%
Cd-113m		4.0×10^{1}	2.0×10^{1}	2.0×10^{1}	100%	5.0×10^{-1}	9 x 10	NA	NA
Cd-115 (a)		3.0×10^{0}	4.0×10^{0}	1.0×10^{0}	25%	4.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Cd-115m		5.0×10^{-1}	3.0×10^{-1}	2.0×10^{-1}	67%	5.0×10^{-1}	3.0×10^{-1}	2.0×10^{-1}	67%
Ce-139	Cerium (58)	7.0×10^{0}	6.0×10^{0}	1.0×10^{0}	17%	2.0×10^{0}	6.0×10^{0}	4.0×10^{0}	67%
Ce-141		2.0×10^{1}	1.0×10^{1}	1.0×10^{1}	100%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Ce-143		9.0×10^{-1}	6.0×10^{-1}	3.0×10^{-1}	50%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Ce-144 (a)		2.0×10^{-1}	2.0×10^{-1}	0.0×10^{0}	0%	2.0×10^{-1}	2.0×10^{-1}	0.0×10^{0}	0%
Cf-248	Californium (98)	4.0×10^{1}	3.0×10^{1}	1.0×10^{1}	33%	6.0×10^{-3}	3.0×10^{-3}	3.0×10^{-3}	100%
Cf-249		3.0×10^{0}	2.0×10^{0}	1.0×10^{0}	50%	8.0×10^{-4}	2.0×10^{-4}	6.0×10^{-4}	300%
Cf-250		2.0×10^{1}	5.0×10^{0}	1.5×10^{1}	300%	2.0×10^{-3}	5.0×10^{-4}	1.5×10^{-3}	300%
Cf-251		7.0×10^{0}	2.0×10^{0}	5.0×10^{0}	250%	7.0×10^{-4}	2.0×10^{-4}	5.0×10^{-4}	250%
Cf-252		5.0×10^{-2}	1.0×10^{-1}	5.0×10^{-2}	50%	3.0×10^{-3}	1.0×10^{-3}	2.0×10^{-3}	200%
Cf-253 (a)		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	4.0×10^{-2}	$6.0 \times 10^{+2}$	$6.0 \times 10^{+2}$	100%
Cf-254		1.0×10^{-3}	3.0×10^{-3}	2.0×10^{-3}	67%	1.0×10^{-3}	6.0×10^{-4}	4.0×10^{-4}	67%
Cl-36	Chlorine (17)	1.0×10^{1}	2.0×10^{1}	1.0×10^{1}	50%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Cl-38		2.0×10^{-1}	2.0×10^{-1}	0.0×10^{0}	0%	2.0×10^{-1}	2.0×10^{-1}	0.0×10^{0}	0%
Cm-240	Curium (96)	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	2.0×10^{-2}	2.0×10^{-2}	0.0×10^{0}	0%
Cm-241		2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	1.0×10^{0}	9.0×10^{-1}	1.0×10^{-1}	11%
Cm-242		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	1.0×10^{-2}	1.0×10^{-2}	0.0×10^{0}	0%
Cm-243		9.0×10^{0}	3.0×10^{0}	6.0×10^{0}	200%	1.0×10^{-3}	3.0×10^{-4}	7.0×10^{-4}	233%
Cm-244		2.0×10^{1}	4.0×10^{0}	1.6×10^{1}	400%	2.0×10^{-3}	4.0×10^{-4}	1.6×10^{-3}	400%
Cm-245		9.0×10^{0}	2.0×10^{0}	7.0×10^{0}	350%	9.0×10^{-4}	2.0×10^{-4}	7.0×10^{-4}	350%
Cm-246		9.0×10^{0}	2.0×10^{0}	7.0×10^{0}	350%	9.0×10^{-4}	2.0×10^{-4}	7.0×10^{-4}	350%
Cm-247 (a)		3.0×10^{0}	2.0×10^{0}	1.0×10^{0}	50%	1.0×10^{-3}	2.0×10^{-4}	8.0×10^{-4}	400%
Cm-248		2.0×10^{-2}	4.0×10^{-2}	2.0×10^{-2}	50%	3.0×10^{-4}	5.0×10^{-5}	2.5×10^{-4}	500%
Co-55	Cobalt (27)	5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%	5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%
Co-56		3.0×10^{-1}	3.0×10^{-1}	0.0×10^{0}	0%	3.0×10^{-1}	3.0×10^{-1}	0.0×10^{0}	0%
Co-57		1.0×10^{1}	8.0×10^{0}	2.0×10^{0}	25%	1.0×10^{1}	8.0×10^{0}	2.0×10^{0}	25%
Co-58		1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%	1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%
Co-58m		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%
Co-60		4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%	4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%
Cr-51	Chromium (24)	3.0×10^{1}	3.0×10^{1}	0.0×10^{0}	0%	3.0×10^{1}	3.0×10^{1}	0.0×10^{0}	0%
Cs-129	Cesium (55)	4.0×10^{0}	4.0×10^{0}	0.0×10^{0}	0%	4.0×10^{0}	4.0×10^{0}	0.0×10^{0}	0%
Cs-131		3.0×10^{1}	4.0×10^{1}	1.0×10^{1}	25%	3.0×10^{1}	4.0×10^{1}	1.0×10^{1}	25%
Cs-132		1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%	1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%
Cs-134		7.0×10^{-1}	6.0×10^{-1}	1.0×10^{-1}	17%	7.0×10^{-1}	5.0×10^{-1}	2.0×10^{-1}	40%
Cs-134m		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	6.0×10^{-1}	9.0×10^{0}	8.4×10^{0}	93%
Cs-135		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	1.0×10^{0}	9.0×10^{-1}	1.0×10^{-1}	11%
Cs-136		5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%	5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%
Cs-137 (a)		2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Cu-64	Copper (29)	6.0×10^{0}	5.0×10^{0}	1.0×10^{0}	20%	1.0×10^{0}	9.0×10^{-1}	1.0×10^{-1}	11%
Cu-67		1.0×10^{1}	9.0×10^{0}	1.0×10^{0}	11%	7.0×10^{-1}	9.0×10^{-1}	2.0×10^{-1}	22%
Dy-159	Dysprosium (66)	2.0×10^{1}	2.0×10^{1}	0.0×10^{0}	0%	2.0×10^{1}	2.0×10^{1}	0.0×10^{0}	0%
Dy-165		9.0×10^{-1}	6.0×10^{-1}	3.0×10^{-1}	50%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Dy-166 (a)		9.0×10^{-1}	3.0×10^{-1}	6.0×10^{-1}	200%	3.0×10^{-1}	3.0×10^{-1}	0.0×10^{0}	0%
Er-169	Erbium (68)	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	1.0×10^{0}	9.0×10^{-1}	1.0×10^{-1}	11%
Er-171		8.0×10^{-1}	6.0×10^{-1}	2.0×10^{-1}	33%	5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%

APPENDIX C
Comparison of A_1 and A_2 Values in TS-R-1 and Part 71 (continued)

Symbol of radionuclide	Element and atomic number	A1 TS-R-1 (TBq)	A1 PART 71 (TBq)	Δ A1 (TS-R-1-Pt 71)	Δ A1 (%)	A2 TS-R-1 (TBq)	A2 PART 71 (TBq)	Δ A2 (TS-R-1-Pt 71)	Δ A2 (%)
Eu-147	Europium (63)	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%
Eu-148		5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%	5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%
Eu-149		2.0×10^{1}	2.0×10^{1}	0.0×10^{0}	0%	2.0×10^{1}	2.0×10^{1}	0.0×10^{0}	0%
Eu-150 (short lived)		2.0×10^{0}	7.0×10^{-1}	1.3×10^{0}	186%	7.0×10^{-1}	7.0×10^{-1}	0.0×10^{0}	0%
Eu-150 (long lived)		2.0×10^{0}	7.0×10^{-1}	1.3×10^{0}	186%	7.0×10^{-1}	7.0×10^{-1}	0.0×10^{0}	0%
Eu-152		1.0×10^{0}	9.0×10^{-1}	1.0×10^{-1}	11%	1.0×10^{0}	9.0×10^{-1}	1.0×10^{-1}	11%
Eu-152m		8.0×10^{-1}	6.0×10^{-1}	2.0×10^{-1}	33%	8.0×10^{-1}	5.0×10^{-1}	3.0×10^{-1}	60%
Eu-154		9.0×10^{-1}	8.0×10^{-1}	1.0×10^{-1}	13%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Eu-155		2.0×10^{1}	2.0×10^{1}	0.0×10^{0}	0%	3.0×10^{0}	2.0×10^{0}	1.0×10^{0}	50%
Eu-156		7.0×10^{-1}	6.0×10^{-1}	1.0×10^{-1}	17%	7.0×10^{-1}	5.0×10^{-1}	2.0×10^{-1}	40%
F-18	Fluorine (9)	1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Fe-52 (a)	Iron (26)	3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%	3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%
Fe-55		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%
Fe-59		9.0×10^{-1}	8.0×10^{-1}	1.0×10^{-1}	13%	9.0×10^{-1}	8.0×10^{-1}	1.0×10^{-1}	13%
Fe-60 (a)		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	2.0×10^{-1}	2.0×10^{-1}	0.0×10^{0}	0%
Ga-67	Gallium (31)	7.0×10^{0}	6.0×10^{0}	1.0×10^{0}	17%	3.0×10^{0}	6.0×10^{0}	3.0×10^{0}	50%
Ga-68		5.0×10^{-1}	3.0×10^{-1}	2.0×10^{-1}	67%	5.0×10^{-1}	3.0×10^{-1}	2.0×10^{-1}	67%
Ga-72		4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%	4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%
Gd-146 (a)	Gadolinium (64)	5.0×10^{-1}	4.0×10^{-1}	1.0×10^{-1}	25%	5.0×10^{-1}	4.0×10^{-1}	1.0×10^{-1}	25%
Gd-148		2.0×10^{1}	3.0×10^{0}	1.7×10^{1}	567%	2.0×10^{-3}	3.0×10^{-4}	1.7×10^{-3}	567%
Gd-153		1.0×10^{1}	1.0×10^{1}	0.0×10^{0}	0%	9.0×10^{0}	5.0×10^{0}	4.0×10^{0}	80%
Gd-159		3.0×10^{0}	4.0×10^{0}	1.0×10^{0}	25%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Ge-68 (a)	Germanium (32)	5.0×10^{-1}	3.0×10^{-1}	2.0×10^{-1}	67%	5.0×10^{-1}	3.0×10^{-1}	2.0×10^{-1}	67%
Ge-71		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%
Ge-77		3.0×10^{-1}	3.0×10^{-1}	0.0×10^{0}	0%	3.0×10^{-1}	3.0×10^{-1}	0.0×10^{0}	0%
Hf-172 (a)	Hafnium (72)	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%	6.0×10^{-1}	3.0×10^{-1}	3.0×10^{-1}	100%
Hf-175		3.0×10^{0}	3.0×10^{0}	0.0×10^{0}	0%	3.0×10^{0}	3.0×10^{0}	0.0×10^{0}	0%
Hf-181		2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	5.0×10^{-1}	9.0×10^{-1}	4.0×10^{-1}	44%
Hf-182		Unlimited	4.0×10^{0}	NA	NA	Unlimited	3.0×10^{-2}	NA	NA
Hg-194 (a)	Mercury (80)	1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%	1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%
Hg-195m (a)		3.0×10^{0}	5.0×10^{0}	2.0×10^{0}	40%	7.0×10^{-1}	5.0×10^{0}	4.3×10^{0}	86%
Hg-197		2.0×10^{1}	1.0×10^{1}	1.0×10^{1}	100%	1.0×10^{1}	1.0×10^{1}	0.0×10^{0}	0%
Hg-197m		1.0×10^{1}	1.0×10^{1}	0.0×10^{0}	0%	4.0×10^{-1}	9.0×10^{-1}	5.0×10^{-1}	56%
Hg-203		5.0×10^{0}	4.0×10^{0}	1.0×10^{0}	25%	1.0×10^{0}	9.0×10^{-1}	1.0×10^{-1}	11%
Ho-166	Holmium (67)	4.0×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	33%	4.0×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	33%
Ho-166m		6.0×10^{-1}	6.0×10^{-1}	0.0×10^{0}	0%	5.0×10^{-1}	3.0×10^{-1}	2.0×10^{-1}	67%
I-123	Iodine (53)	6.0×10^{0}	6.0×10^{0}	0.0×10^{0}	0%	3.0×10^{0}	6.0×10^{0}	3.0×10^{0}	50%
I-124		1.0×10^{0}	9.0×10^{-1}	1.0×10^{-1}	11%	1.0×10^{0}	9.0×10^{-1}	1.0×10^{-1}	11%
I-125		2.0×10^{1}	2.0×10^{1}	0.0×10^{0}	0%	3.0×10^{0}	2.0×10^{0}	1.0×10^{0}	50%
I-126		2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	1.0×10^{0}	9.0×10^{-1}	1.0×10^{-1}	11%
I-129		Unlimited	Unlimited	NA	NA	Unlimited	Unlimited	NA	NA
I-131		3.0×10^{0}	3.0×10^{0}	0.0×10^{0}	0%	7.0×10^{-1}	5.0×10^{-1}	2.0×10^{-1}	40%
I-132		4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%	4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%
I-133		7.0×10^{-1}	6.0×10^{-1}	1.0×10^{-1}	17%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
I-134		3.0×10^{-1}	3.0×10^{-1}	0.0×10^{0}	0%	3.0×10^{-1}	3.0×10^{-1}	0.0×10^{0}	0%
I-135 (a)		6.0×10^{-1}	6.0×10^{-1}	0.0×10^{0}	0%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%

APPENDIX C
Comparison of A_1 and A_2 Values in TS-R-1 and Part 71 (continued)

Symbol of radionuclide	Element and atomic number	A1 TS-R-1 (TBq)	A1 PART 71 (TBq)	Δ A1 (TS-R-1-Pt 71)	Δ A1 (%)	A2 TS-R-1 (TBq)	A2 PART 71 (TBq)	Δ A2 (TS-R-1-Pt 71)	Δ A2 (%)
In-111	Indium (49)	3.0×10^{0}	2.0×10^{0}	1.0×10^{0}	50%	3.0×10^{0}	2.0×10^{0}	1.0×10^{0}	50%
In-113m		4.0×10^{0}	4.0×10^{0}	0.0×10^{0}	0%	2.0×10^{0}	4.0×10^{0}	2.0×10^{0}	50%
In-114m (a)		1.0×10^{1}	3.0×10^{-1}	9.7×10^{0}	3233%	5.0×10^{-1}	3.0×10^{-1}	2.0×10^{-1}	67%
In-115m		7.0×10^{0}	6.0×10^{0}	1.0×10^{0}	17%	1.0×10^{0}	9.0×10^{-1}	1.0×10^{-1}	11%
Ir-189 (a)	Iridium (77)	1.0×10^{1}	1.0×10^{1}	0.0×10^{0}	0%	1.0×10^{1}	1.0×10^{1}	0.0×10^{0}	0%
Ir-190		7.0×10^{-1}	7.0×10^{-1}	0.0×10^{0}	0%	7.0×10^{-1}	7.0×10^{-1}	0.0×10^{0}	0%
Ir-192		1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Ir-194		3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%	3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%
K-40	Potassium (19)	9.0×10^{-1}	6.0×10^{-1}	3.0×10^{-1}	50%	9.0×10^{-1}	6.0×10^{-1}	3.0×10^{-1}	50%
K-42		2.0×10^{-1}	2.0×10^{-1}	0.0×10^{0}	0%	2.0×10^{-1}	2.0×10^{-1}	0.0×10^{0}	0%
K-43		7.0×10^{-1}	1.0×10^{0}	3.0×10^{-1}	30%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Kr-81	Krypton (36)	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%
Kr-85		1.0×10^{1}	2.0×10^{1}	1.0×10^{1}	50%	1.0×10^{1}	1.0×10^{1}	0.0×10^{0}	0%
Kr-85m		8.0×10^{0}	6.0×10^{0}	2.0×10^{0}	33%	3.0×10^{0}	6.0×10^{0}	3.0×10^{0}	50%
Kr-87		2.0×10^{-1}	2.0×10^{-1}	0.0×10^{0}	0%	2.0×10^{-1}	2.0×10^{-1}	0.0×10^{0}	0%
La-137	Lanthanum (57)	3.0×10^{1}	4.0×10^{1}	1.0×10^{1}	25%	6.0×10^{0}	2.0×10^{0}	4.0×10^{0}	200%
La-140		4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%	4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%
Lu-172	Lutetium (71)	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Lu-173		8.0×10^{0}	8.0×10^{0}	0.0×10^{0}	0%	8.0×10^{0}	8.0×10^{0}	0.0×10^{0}	0%
Lu-174		9.0×10^{0}	8.0×10^{0}	1.0×10^{0}	13%	9.0×10^{0}	4.0×10^{0}	5.0×10^{0}	125%
Lu-174m		2.0×10^{1}	2.0×10^{1}	0.0×10^{0}	0%	1.0×10^{1}	8.0×10^{0}	2.0×10^{0}	25%
Lu-177		3.0×10^{1}	3.0×10^{1}	0.0×10^{0}	0%	7.0×10^{-1}	9.0×10^{-1}	2.0×10^{-1}	22%
Mg-28 (a)	Magnesium (12)	3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%	3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%
Mn-52	Manganese (25)	3.0×10^{-1}	3.0×10^{-1}	0.0×10^{0}	0%	3.0×10^{-1}	3.0×10^{-1}	0.0×10^{0}	0%
Mn-53		Unlimited	Unlimited	NA	NA	Unlimited	Unlimited	NA	NA
Mn-54		1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%	1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%
Mn-56		3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%	3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%
Mo-93	Molybdenum (42)	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	2.0×10^{1}	7.0×10^{0}	1.3×10^{1}	186%
Mo-99 (a)		1.0×10^{0}	6.0×10^{-1}	4.0×10^{-1}	67%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
N-13	Nitrogen (7)	9.0×10^{-1}	6.0×10^{-1}	3.0×10^{-1}	50%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Na-22	Sodium (11)	5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%	5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%
Na-24		2.0×10^{-1}	2.0×10^{-1}	0.0×10^{0}	0%	2.0×10^{-1}	2.0×10^{-1}	0.0×10^{0}	0%
Nb-93m	Niobium (41)	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	3.0×10^{1}	6.0×10^{0}	2.4×10^{1}	400%
Nb-94		7.0×10^{-1}	6.0×10^{-1}	1.0×10^{-1}	17%	7.0×10^{-1}	6.0×10^{-1}	1.0×10^{-1}	17%
Nb-95		1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%	1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%
Nb-97		9.0×10^{-1}	6.0×10^{-1}	3.0×10^{-1}	50%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Nd-147	Neodymium (60)	6.0×10^{0}	4.0×10^{0}	2.0×10^{0}	50%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Nd-149		6.0×10^{-1}	6.0×10^{-1}	0.0×10^{0}	0%	5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%
Ni-59	Nickel (28)	Unlimited	4.0×10^{1}	NA	NA	Unlimited	4.0×10^{1}	NA	NA
Ni-63		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	3.0×10^{1}	3.0×10^{1}	0.0×10^{0}	0%
Ni-65		4.0×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	33%	4.0×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	33%
Np-235	Neptunium (93)	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%
Np-236 (short-lived)		2.0×10^{1}	7.0×10^{0}	1.3×10^{1}	186%	2.0×10^{0}	1.0×10^{-3}	2.0×10^{0}	199900%
Np-236 (long-lived)		2.0×10^{1}	7.0×10^{0}	1.3×10^{1}	186%	2.0×10^{0}	1.0×10^{-3}	2.0×10^{0}	199900%
Np-237		2.0×10^{1}	2.0×10^{0}	1.8×10^{1}	900%	2.0×10^{-3}	2.0×10^{-4}	1.8×10^{-3}	900%
Np-239		7.0×10^{0}	6.0×10^{0}	1.0×10^{0}	17%	4.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%

APPENDIX C
Comparison of A_1 and A_2 Values in TS-R-1 and Part 71 (continued)

Symbol of radionuclide	Element and atomic number	A1 TS-R-1 (TBq)	A1 PART 71 (TBq)	Δ A1 (TS-R-1-Pt 71)	Δ A1 (%)	A2 TS-R-1 (TBq)	A2 PART 71 (TBq)	Δ A2 (TS-R-1-Pt 71)	Δ A2 (%)
Os-185	Osmium (76)	1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%	1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%
Os-191		1.0×10^{1}	1.0×10^{1}	0.0×10^{0}	0%	2.0×10^{0}	9.0×10^{-1}	1.1×10^{0}	122%
Os-191m		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	3.0×10^{1}	4.0×10^{1}	1.0×10^{1}	25%
Os-193		2.0×10^{0}	6.0×10^{-1}	1.4×10^{0}	233%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Os-194 (a)		3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%	3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%
P-32	Phosphorus (15)	5.0×10^{-1}	3.0×10^{-1}	2.0×10^{-1}	67%	5.0×10^{-1}	3.0×10^{-1}	2.0×10^{-1}	67%
P-33		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	1.0×10^{0}	9.0×10^{-1}	1.0×10^{-1}	11%
Pa-230 (a)	Protactinium (91)	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	7.0×10^{-2}	1.0×10^{-1}	3.0×10^{-2}	30%
Pa-231		4.0×10^{0}	6.0×10^{-1}	3.4×10^{0}	567%	4.0×10^{-4}	6.0×10^{-5}	3.4×10^{-4}	567%
Pa-233		5.0×10^{0}	5.0×10^{0}	0.0×10^{0}	0%	7.0×10^{-1}	9.0×10^{-1}	2.0×10^{-1}	22%
Pb-201	Lead (82)	1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%	1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%
Pb-202		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	2.0×10^{1}	2.0×10^{0}	1.8×10^{1}	900%
Pb-203		4.0×10^{0}	3.0×10^{0}	1.0×10^{0}	33%	3.0×10^{0}	3.0×10^{0}	0.0×10^{0}	0%
Pb-205		Unlimited	Unlimited	NA	NA	Unlimited	Unlimited	NA	NA
Pb-210 (a)		1.0×10^{0}	6.0×10^{-1}	4.0×10^{-1}	67%	5.0×10^{-2}	9.0×10^{-3}	4.1×10^{-2}	456%
Pb-212 (a)		7.0×10^{-1}	3.0×10^{-1}	4.0×10^{-1}	133%	2.0×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	33%
Pd-103 (a)	Palladium (46)	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%
Pd-107		Unlimited	Unlimited	NA	NA	Unlimited	Unlimited	NA	NA
Pd-109		2.0×10^{0}	6.0×10^{-1}	1.4×10^{0}	233%	5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%
Pm-143	Promethium (61)	3.0×10^{0}	3.0×10^{0}	0.0×10^{0}	0%	3.0×10^{0}	3.0×10^{0}	0.0×10^{0}	0%
Pm-144		7.0×10^{-1}	6.0×10^{-1}	1.0×10^{-1}	17%	7.0×10^{-1}	6.0×10^{-1}	1.0×10^{-1}	17%
Pm-145		3.0×10^{1}	3.0×10^{1}	0.0×10^{0}	0%	1.0×10^{1}	7.0×10^{0}	3.0×10^{0}	43%
Pm-147		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	2.0×10^{0}	9.0×10^{-1}	1.1×10^{0}	122%
Pm-148m (a)		8.0×10^{-1}	5.0×10^{-1}	3.0×10^{-1}	60%	7.0×10^{-1}	5.0×10^{-1}	2.0×10^{-1}	40%
Pm-149		2.0×10^{0}	6.0×10^{-1}	1.4×10^{0}	233%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Pm-151		2.0×10^{0}	3.0×10^{0}	1.0×10^{0}	33%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Po-210	Polonium (84)	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	2.0×10^{-2}	2.0×10^{-2}	0.0×10^{0}	0%
Pr-142	Praseodymium (59)	4.0×10^{-1}	2.0×10^{-1}	2.0×10^{-1}	100%	4.0×10^{-1}	2.0×10^{-1}	2.0×10^{-1}	100%
Pr-143		3.0×10^{0}	4.0×10^{0}	1.0×10^{0}	25%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Pt-188 (a)	Platinum (78)	1.0×10^{0}	6.0×10^{-1}	4.0×10^{-1}	67%	8.0×10^{-1}	6.0×10^{-1}	2.0×10^{-1}	33%
Pt-191		4.0×10^{0}	3.0×10^{0}	1.0×10^{0}	33%	3.0×10^{0}	3.0×10^{0}	0.0×10^{0}	0%
Pt-193		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%
Pt-193m		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	5.0×10^{-1}	9.0×10^{0}	8.5×10^{0}	94%
Pt-195m		1.0×10^{1}	1.0×10^{1}	0.0×10^{0}	0%	5.0×10^{-1}	2.0×10^{0}	1.5×10^{0}	75%
Pt-197		2.0×10^{1}	2.0×10^{1}	0.0×10^{0}	0%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Pt-197m		1.0×10^{1}	1.0×10^{1}	0.0×10^{0}	0%	6.0×10^{-1}	9.0×10^{-1}	3.0×10^{-1}	33%
Pu-236	Plutonium (94)	3.0×10^{1}	7.0×10^{0}	2.3×10^{1}	329%	3.0×10^{-3}	7.0×10^{-4}	2.3×10^{-3}	329%
Pu-237		2.0×10^{1}	2.0×10^{1}	0.0×10^{0}	0%	2.0×10^{1}	2.0×10^{1}	0.0×10^{0}	0%
Pu-238		1.0×10^{1}	2.0×10^{0}	8.0×10^{0}	400%	1.0×10^{-3}	2.0×10^{-4}	8.0×10^{-4}	400%
Pu-239		1.0×10^{1}	2.0×10^{0}	8.0×10^{0}	400%	1.0×10^{-3}	2.0×10^{-4}	8.0×10^{-4}	400%
Pu-240		1.0×10^{1}	2.0×10^{0}	8.0×10^{0}	400%	1.0×10^{-3}	2.0×10^{-4}	8.0×10^{-4}	400%
Pu-241 (a)		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	6.0×10^{-2}	1.0×10^{-2}	5.0×10^{-2}	500%
Pu-242		1.0×10^{1}	2.0×10^{0}	8.0×10^{0}	400%	10×10^{-3}	2.0×10^{-4}	8.0×10^{-4}	400%
Pu-244 (a)		4.0×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	33%	1.0×10^{-3}	2.0×10^{-4}	8.0×10^{-4}	400%
Ra-223 (a)	Radium (88)	4.0×10^{-1}	6.0×10^{-1}	2.0×10^{-1}	33%	7.0×10^{-3}	3.0×10^{-2}	2.3×10^{-2}	77%
Ra-224 (a)		4.0×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	33%	2.0×10^{-2}	6.0×10^{-2}	4.0×10^{-2}	67%
Ra-225 (a)		2.0×10^{-1}	6.0×10^{-1}	4.0×10^{-1}	67%	4.0×10^{-3}	2.0×10^{-2}	1.6×10^{-2}	80%
Ra-226 (a)		2.0×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	33%	3.0×10^{-3}	2.0×10^{-2}	1.7×10^{-2}	85%
Ra-228 (a)		6.0×10^{-1}	6.0×10^{-1}	0.0×10^{0}	0%	2.0×10^{-2}	4.0×10^{-2}	2.0×10^{-2}	50%
Rb-81	Rubidium (37)	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	8.0×10^{-1}	9.0×10^{-1}	1.0×10^{-1}	11%
Rb-83 (a)		2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%

APPENDIX C
Comparison of A_1 and A_2 Values in TS-R-1 and Part 71 (continued)

Symbol of radionuclide	Element and atomic number	A1 TS-R-1 (TBq)	A1 PART 71 (TBq)	Δ A1 (TS-R-1-Pt 71)	Δ A1 (%)	A2 TS-R-1 (TBq)	A2 PART 71 (TBq)	Δ A2 (TS-R-1-Pt 71)	Δ A2 (%)
Rb-84		1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%	1.0×10^{0}	9.0×10^{-1}	1.0×10^{-1}	11%
Rb-86		5.0×10^{-1}	3.0×10^{-1}	2.0×10^{-1}	67%	5.0×10^{-1}	3.0×10^{-1}	2.0×10^{-1}	67%
Rb-87		Unlimited	Unlimited	NA	NA	Unlimited	Unlimited	NA	NA
Rb(nat)		Unlimited	Unlimited	NA	NA	Unlimited	Unlimited	NA	NA
Re-184	Rhenium (75)	1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%	1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%
Re-184m		3.0×10^{0}	3.0×10^{0}	0.0×10^{0}	0%	1.0×10^{0}	3.0×10^{0}	2.0×10^{0}	67%
Re-186		2.0×10^{0}	4.0×10^{0}	2.0×10^{0}	50%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Re-187		Unlimited	Unlimited	NA	NA	Unlimited	Unlimited	NA	NA
Re-188		4.0×10^{-1}	2.0×10^{-1}	2.0×10^{-1}	100%	4.0×10^{-1}	2.0×10^{-1}	2.0×10^{-1}	100%
Re-189 (a)		3.0×10^{0}	4.0×10^{0}	1.0×10^{0}	25%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Re(nat)		Unlimited	Unlimited	NA	NA	Unlimited	Unlimited	NA	NA
Rh-99	Rhodium (45)	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%
Rh-101		4.0×10^{0}	4.0×10^{0}	0.0×10^{0}	0%	3.0×10^{0}	4.0×10^{0}	1.0×10^{0}	25%
Rh-102		5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%	5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%
Rh-102m		2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	2.0×10^{0}	9.0×10^{-1}	1.1×10^{0}	122%
Rh-103m		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%
Rh-105		1.0×10^{1}	1.0×10^{1}	0.0×10^{0}	0%	8.0×10^{-1}	9.0×10^{-1}	1.0×10^{-1}	11%
Rn-222 (a)	Radon (86)	3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%	4.0×10^{-3}	4.0×10^{-3}	0.0×10^{0}	0%
Ru-97	Ruthenium (44)	5.0×10^{0}	4.0×10^{0}	1.0×10^{0}	25%	5.0×10^{0}	4.0×10^{0}	1.0×10^{0}	25%
Ru-103 (a)		2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	2.0×10^{0}	9.0×10^{-1}	1.1×10^{0}	122%
Ru-105		1.0×10^{0}	6.0×10^{-1}	4.0×10^{-1}	67%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Ru-106 (a)		2.0×10^{-1}	2.0×10^{-1}	0.0×10^{0}	0%	2.0×10^{-1}	2.0×10^{-1}	0.0×10^{0}	0%
S-35	Sulphur (16)	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	3.0×10^{0}	2.0×10^{0}	1.0×10^{0}	50%
Sb-122	Antimony (51)	4.0×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	33%	4.0×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	33%
Sb-124		6.0×10^{-1}	6.0×10^{-1}	0.0×10^{0}	0%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Sb-125		2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	1.0×10^{0}	9.0×10^{-1}	1.0×10^{-1}	11%
Sb-126		4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%	4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%
Sc-44	Scandium (21)	5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%	5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%
Sc-46		5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%	5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%
Sc-47		1.0×10^{1}	9.0×10^{0}	1.0×10^{0}	11%	7.0×10^{-1}	9.0×10^{-1}	2.0×10^{-1}	22%
Sc-48		3.0×10^{-1}	3.0×10^{-1}	0.0×10^{0}	0%	3.0×10^{-1}	3.0×10^{-1}	0.0×10^{0}	0%
Se-75	Selenium (34)	3.0×10^{0}	3.0×10^{0}	0.0×10^{0}	0%	3.0×10^{0}	3.0×10^{0}	0.0×10^{0}	0%
Se-79		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%
Si-31	Silicon (14)	6.0×10^{-1}	6.0×10^{-1}	0.0×10^{0}	0%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Si-32		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	5.0×10^{-1}	2.0×10^{-1}	3.0×10^{-1}	150%
Sm-145	Samarium (62)	1.0×10^{1}	2.0×10^{1}	1.0×10^{1}	50%	1.0×10^{1}	2.0×10^{1}	1.0×10^{1}	50%
Sm-147		Unlimited	Unlimited	NA	NA	Unlimited	Unlimited	NA	NA
Sm-151		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	1.0×10^{1}	4.0×10^{0}	6.0×10^{0}	150%
Sm-153		9.0×10^{0}	4.0×10^{0}	5.0×10^{0}	125%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Sn-113 (a)	Tin (50)	4.0×10^{0}	4.0×10^{0}	0.0×10^{0}	0%	2.0×10^{0}	4.0×10^{0}	2.0×10^{0}	50%
Sn-117m		7.0×10^{0}	6.0×10^{0}	1.0×10^{0}	17%	4.0×10^{-1}	2.4×10^{1}	2.4×10^{1}	98%
Sn-119m		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	3.0×10^{1}	4.0×10^{1}	1.0×10^{1}	25%
Sn-121m (a)		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	9.0×10^{-1}	9.0×10^{-1}	0.0×10^{0}	0%
Sn-123		8.0×10^{-1}	6.0×10^{-1}	2.0×10^{-1}	33%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Sn-125		4.0×10^{-1}	2.0×10^{-1}	2.0×10^{-1}	100%	4.0×10^{-1}	2.0×10^{-1}	2.0×10^{-1}	100%
Sn-126 (a)		6.0×10^{-1}	3.0×10^{-1}	3.0×10^{-1}	100%	4.0×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	33%

APPENDIX C
Comparison of A_1 and A_2 Values in TS-R-1 and Part 71 (continued)

Symbol of radionuclide	Element and atomic number	A1 TS-R-1 (TBq)	A1 PART 71 (TBq)	Δ A1 (TS-R-1-Pt 71)	Δ A1 (%)	A2 TS-R-1 (TBq)	A2 PART 71 (TBq)	Δ A2 (TS-R-1-Pt 71)	Δ A2 (%)
Sr-82 (a)	Strontium (38)	2.0×10^{-1}	2.0×10^{-1}	0.0×10^{0}	0%	2.0×10^{-1}	2.0×10^{-1}	0.0×10^{0}	0%
Sr-85		2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%
Sr-85m		5.0×10^{0}	5.0×10^{0}	0.0×10^{0}	0%	5.0×10^{0}	5.0×10^{0}	0.0×10^{0}	0%
Sr-87m		3.0×10^{0}	3.0×10^{0}	0.0×10^{0}	0%	3.0×10^{0}	3.0×10^{0}	0.0×10^{0}	0%
Sr-89		6.0×10^{-1}	6.0×10^{-1}	0.0×10^{0}	0%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Sr-90 (a)		3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%	3.0×10^{-1}	1.0×10^{-1}	2.0×10^{-1}	200%
Sr-91 (a)		3.0×10^{-1}	3.0×10^{-1}	0.0×10^{0}	0%	3.0×10^{-1}	3.0×10^{-1}	0.0×10^{0}	0%
Sr-92 (a)		1.0×10^{0}	8.0×10^{-1}	2.0×10^{-1}	25%	3.0×10^{-1}	5.0×10^{-1}	2.0×10^{-1}	40%
T(H-3)	Tritium (1)	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%
Ta-178 (long-lived)	Tantalum (73)	1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%	8.0×10^{-1}	1.0×10^{0}	2.0×10^{-1}	20%
Ta-179		3.0×10^{1}	3.0×10^{1}	0.0×10^{0}	0%	3.0×10^{1}	3.0×10^{1}	0.0×10^{0}	0%
Ta-182		9.0×10^{-1}	8.0×10^{-1}	1.0×10^{-1}	13%	5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%
Tb-157	Terbium (65)	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	4.0×10^{1}	1.0×10^{1}	3.0×10^{1}	300%
Tb-158		1.0×10^{0}	1.0×10^{0}	0.0×10^{0}	0%	1.0×10^{0}	7.0×10^{-1}	3.0×10^{-1}	43%
Tb-160		1.0×10^{0}	9.0×10^{-1}	1.0×10^{-1}	11%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Tc-95m (a)	Technetium (43)	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%
Tc-96		4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%	4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%
Tc-96m (a)		4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%	4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%
Tc-97		Unlimited	Unlimited	NA	NA	Unlimited	Unlimited	NA	NA
Tc-97m		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	1.0×10^{0}	4.0×10^{1}	3.9×10^{1}	98%
Tc-98		8.0×10^{-1}	7.0×10^{-1}	1.0×10^{-1}	14%	7.0×10^{-1}	7.0×10^{-1}	0.0×10^{0}	0%
Tc-99		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	9.0×10^{-1}	9.0×10^{-1}	0.0×10^{0}	0%
Tc-99m		1.0×10^{1}	8.0×10^{0}	2.0×10^{0}	25%	4.0×10^{0}	8.0×10^{0}	4.0×10^{0}	50%
Te-121	Tellurium (52)	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%
Te-121m		5.0×10^{0}	5.0×10^{0}	0.0×10^{0}	0%	3.0×10^{0}	5.0×10^{0}	2.0×10^{0}	40%
Te-123m		8.0×10^{0}	7.0×10^{0}	1.0×10^{0}	14%	1.0×10^{0}	7.0×10^{0}	6.0×10^{0}	86%
Te-125m		2.0×10^{1}	3.0×10^{1}	1.0×10^{1}	33%	9.0×10^{-1}	9.0×10^{0}	8.1×10^{0}	90%
Te-127		2.0×10^{1}	2.0×10^{1}	0.0×10^{0}	0%	7.0×10^{-1}	5.0×10^{-1}	2.0×10^{-1}	40%
Te-127m (a)		2.0×10^{1}	2.0×10^{1}	0.0×10^{0}	0%	5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%
Te-129		7.0×10^{-1}	6.0×10^{-1}	1.0×10^{-1}	17%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Te-129m (a)		8.0×10^{-1}	6.0×10^{-1}	2.0×10^{-1}	33%	4.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Te-131m (a)		7.0×10^{-1}	7.0×10^{-1}	0.0×10^{0}	0%	5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%
T-132 (a)		5.0×10^{-1}	4.0×10^{-1}	1.0×10^{-1}	25%	4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%
Th-227	Thorium (90)	1.0×10^{1}	9.0×10^{0}	1.0×10^{0}	11%	5.0×10^{-3}	1.0×10^{-2}	5.0×10^{-3}	50%
Th-228 (a)		5.0×10^{-1}	3.0×10^{-1}	2.0×10^{-1}	67%	1.0×10^{-3}	4.0×10^{-4}	6.0×10^{-4}	150%
Th-229		5.0×10^{0}	3.0×10^{-1}	4.7×10^{0}	1567%	5.0×10^{-4}	3.0×10^{-5}	4.7×10^{-4}	1567%
Th-230		1.0×10^{1}	2.0×10^{0}	8.0×10^{0}	400%	1.0×10^{-3}	2.0×10^{-4}	8.0×10^{-4}	400%
Th-231		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	2.0×10^{-2}	9.0×10^{-1}	8.8×10^{-1}	98%
Th-232		Unlimited	Unlimited	NA	NA	Unlimited	Unlimited	NA	NA
Th-234 (a)		3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%	3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%
Th(nat)		Unlimited	Unlimited	NA	NA	Unlimited	Unlimited	NA	NA
Ti-44 (a)	Titanium (22)	5.0×10^{-1}	5.0×10^{-1}	0.0×10^{0}	0%	4.0×10^{-1}	2.0×10^{-1}	2.0×10^{-1}	100%
Tl-200	Thallium (81)	9.0×10^{-1}	8.0×10^{-1}	1.0×10^{-1}	13%	9.0×10^{-1}	8.0×10^{-1}	1.0×10^{-1}	13%
Tl-201		1.0×10^{1}	1.0×10^{1}	0.0×10^{0}	0%	4.0×10^{0}	1.0×10^{1}	6.0×10^{0}	60%
Tl-202		2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%
Tl-204		1.0×10^{1}	4.0×10^{0}	6.0×10^{0}	150%	7.0×10^{-1}	5.0×10^{-1}	2.0×10^{-1}	40%
Tm-167	Thulium (69)	7.0×10^{0}	7.0×10^{0}	0.0×10^{0}	0%	8.0×10^{-1}	7.0×10^{0}	6.2×10^{0}	89%
Tm-170		3.0×10^{0}	4.0×10^{0}	1.0×10^{0}	25%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Tm-171		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	4.0×10^{1}	1.0×10^{1}	3.0×10^{1}	300%

APPENDIX C
Comparison of $A_1$and A_2 Values in TS-R-1 and Part 71 (continued)

Symbol of radionuclide	Element and atomic number	A1 TS-R-1 (TBq)	A1 PART 71 (TBq)	Δ A1 (TS-R-1-Pt 71)	Δ A1 (%)	A2 TS-R-1 (TBq)	A2 PART 71 (TBq)	Δ A2 (TS-R-1-Pt 71)	Δ A2 (%)
U-230 (fast lung absorption)(a)(d)	Uranium (92)	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	1.0×10^{-1}	1.0×10^{-2}	9.0×10^{-2}	900%
U-230 (medium lung absorption)(a)(e)		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	1.0×10^{-1}	1.0×10^{-2}	9.0×10^{-2}	900%
U-230 (slow lung absorption)(a)(f)		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	1.0×10^{-1}	1.0×10^{-2}	9.0×10^{-2}	900%
U-232 (fast lung absorption)(d)		4.0×10^{1}	3.0×10^{0}	3.7×10^{1}	1233%	1.0×10^{-2}	3.0×10^{-4}	9.7×10^{-3}	3233%
U-232 (medium lung absorption)(e)		4.0×10^{1}	3.0×10^{0}	3.7×10^{1}	1233%	1.0×10^{-2}	3.0×10^{-4}	9.7×10^{-3}	3233%
U-232 (slow lung absorption)(f)		4.0×10^{1}	3.0×10^{0}	3.7×10^{1}	1233%	1.0×10^{-2}	3.0×10^{-4}	9.7×10^{-3}	3233%
U-233 (fast lung absorption)(d)		4.0×10^{1}	1.0×10^{1}	3.0×10^{1}	300%	9.0×10^{-2}	1.0×10^{-3}	8.9×10^{-2}	8900%
U-233 (medium lung absorption)(e)		4.0×10^{1}	1.0×10^{1}	3.0×10^{1}	300%	9.0×10^{-2}	1.0×10^{-3}	8.9×10^{-2}	8900%
U-233 (slow lung absorption)(f)		4.0×10^{1}	1.0×10^{1}	3.0×10^{1}	300%	9.0×10^{-2}	1.0×10^{-3}	8.9×10^{-2}	8900%
U-234 (fast lung absorption)(d)		4.0×10^{1}	1.0×10^{1}	3.0×10^{1}	300%	9.0×10^{-2}	1.0×10^{-3}	8.9×10^{-2}	8900%
U-234 (medium lung absorption)(e)		4.0×10^{1}	1.0×10^{1}	3.0×10^{1}	300%	9.0×10^{-2}	1.0×10^{-3}	8.9×10^{-2}	8900%
U-234 (slow lung absorption)(f)		4.0×10^{1}	1.0×10^{1}	3.0×10^{1}	300%	9.0×10^{-2}	1.0×10^{-3}	8.9×10^{-2}	8900%
U-235 (all lung absorption types)(a),(d),(e),(f)		Unlimited	Unlimited	NA	NA	Unlimited	Unlimited	NA	NA
U-236 (fast lung absorption)(d)		Unlimited	1.0×10^{1}	NA	NA	Unlimited	1.0×10^{-3}	NA	NA
U-236 (medium lung absorption)(e)		Unlimited	1.0×10^{1}	NA	NA	Unlimited	1.0×10^{-3}	NA	NA
U-236 (slow lung absorption)(f)		Unlimited	1.0×10^{1}	NA	NA	Unlimited	1.0×10^{-3}	NA	NA
U-238 (all lung absorption types)(d),(e),(f)		Unlimited	Unlimited	NA	NA	Unlimited	Unlimited	NA	NA
U (nat)		Unlimited	Unlimited	NA	NA	Unlimited	Unlimited	NA	NA

APPENDIX C
Comparison of A_1 and A_2 Values in TS-R-1 and Part 71 (continued)

Symbol of radionuclide	Element and atomic number	A1 TS-R-1 (TBq)	A1 PART 71 (TBq)	Δ A1 (TS-R-1-Pt 71)	Δ A1 (%)	A2 TS-R-1 (TBq)	A2 PART 71 (TBq)	Δ A2 (TS-R-1-Pt 71)	Δ A2 (%)
U (enriched to 20% or less)(g)		Unlimited	#N/A	NA	NA	Unlimited	#N/A	NA	NA
U (dep)		Unlimited	Unlimited	NA	NA	Unlimited	Unlimited	NA	NA
V-48	Vanadium (23)	4.0×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	33%	4.0×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	33%
V-49		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%
W-178 (a)	Tungsten (74)	9.0×10^{0}	1.0×10^{0}	8.0×10^{0}	800%	5.0×10^{0}	1.0×10^{0}	4.0×10^{0}	400%
W-181		3.0×10^{1}	3.0×10^{1}	0.0×10^{0}	0%	3.0×10^{1}	3.0×10^{1}	0.0×10^{0}	0%
W-185		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	8.0×10^{-1}	9.0×10^{-1}	1.0×10^{-1}	11%
W-187		2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
W-188 (a)		4.0×10^{-1}	2.0×10^{-1}	2.0×10^{-1}	100%	3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%
Xe-122 (a)	Xenon (54)	4.0×10^{-1}	2.0×10^{-1}	2.0×10^{-1}	100%	4.0×10^{-1}	2.0×10^{-1}	2.0×10^{-1}	100%
Xe-123		2.0×10^{0}	2.0×10^{-1}	1.8×10^{0}	900%	7.0×10^{-1}	2.0×10^{-1}	5.0×10^{-1}	250%
Xe-127		4.0×10^{0}	4.0×10^{0}	0.0×10^{0}	0%	2.0×10^{0}	4.0×10^{0}	2.0×10^{0}	50%
Xe-131m		4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%	4.0×10^{1}	4.0×10^{1}	0.0×10^{0}	0%
Xe-133		2.0×10^{1}	2.0×10^{1}	0.0×10^{0}	0%	1.0×10^{1}	2.0×10^{1}	1.0×10^{1}	50%
Xe-135		3.0×10^{0}	4.0×10^{0}	1.0×10^{0}	25%	2.0×10^{0}	4.0×10^{0}	2.0×10^{0}	50%
Y-87 (a)	Yttrium (39)	1.0×10^{0}	2.0×10^{0}	1.0×10^{0}	50%	1.0×10^{0}	2.0×10^{0}	1.0×10^{0}	50%
Y-88		4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%	4.0×10^{-1}	4.0×10^{-1}	0.0×10^{0}	0%
Y-90		3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%	3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%
Y-91		6.0×10^{-1}	3.0×10^{-1}	3.0×10^{-1}	100%	6.0×10^{-1}	3.0×10^{-1}	3.0×10^{-1}	100%
Y-91m		2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%
Y-92		2.0×10^{-1}	2.0×10^{-1}	0.0×10^{0}	0%	2.0×10^{-1}	2.0×10^{-1}	0.0×10^{0}	0%
Y-93		3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%	3.0×10^{-1}	2.0×10^{-1}	1.0×10^{-1}	50%
Yb-169	Ytterbium (79)	4.0×10^{0}	3.0×10^{0}	1.0×10^{0}	33%	1.0×10^{0}	3.0×10^{0}	2.0×10^{0}	67%
Yb-175		3.0×10^{1}	2.0×10^{0}	2.8×10^{1}	1400%	9.0×10^{-1}	2.0×10^{0}	1.1×10^{0}	55%
Zn-65	Zinc (30)	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%	2.0×10^{0}	2.0×10^{0}	0.0×10^{0}	0%
Zn-69		3.0×10^{0}	4.0×10^{0}	1.0×10^{0}	25%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Zn-69m (a)		3.0×10^{0}	2.0×10^{0}	1.0×10^{0}	50%	6.0×10^{-1}	5.0×10^{-1}	1.0×10^{-1}	20%
Zr-88	Zirconium (40)	3.0×10^{0}	3.0×10^{0}	0.0×10^{0}	0%	3.0×10^{0}	3.0×10^{0}	0.0×10^{0}	0%
Zr-93		Unlimited	4.0×10^{1}	NA	NA	Unlimited	2.0×10^{-1}	NA	NA
Zr-95 (a)		2.0×10^{0}	1.0×10^{0}	1.0×10^{0}	100%	8.0×10^{-1}	9.0×10^{-1}	1.0×10^{-1}	11%
Zr-97 (a)		4.0×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	33%	4.0×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	33%