RULEMAKING ISSUE
NOTATION VOTE

March 2, 2001 SECY-01-0035

FOR: The Commissioners

FROM: William D. Travers
Executive Director for Operations

SUBJECT: PROPOSED RULE FOR REVISING 10 CFR PART 71 FOR COMPATIBILITY WITH IAEA TRANSPORTATION SAFETY STANDARDS [TS-R-1], AND FOR MAKING OTHER NRC-INITIATED CHANGES

PURPOSE:

To request Commission approval to: (1) publish a proposed rule, in the Federal Register (FR), that would amend 10 CFR Part 71; and (2) continue using an enhanced public-participation process (website and public meetings), during the proposed rule public comment period, to solicit maximum public input.

BACKGROUND:

The Commission directed the staff, in a Staff Requirements Memorandum (SRM), dated September 17, 1999 [SRM-SECY-99-200 (Attachment 1) - "Federal Register Notice Responding to Public Comments Received on an Emergency Final Rule for Fissile Material Shipments"], to prepare an overall rulemaking plan that addresses the need to make Part 71, "Packaging and Transportation of Radioactive Material," compatible with ST-1, the latest revision of the International Atomic Energy Agency (IAEA) transportation safety standards. The IAEA has been revising its transportation standards on approximately a 10-year cycle, with the last edition, ST-1, published in December 1996. This edition was revised in June 2000 and

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published as No. TS-R-1. The revisions in TS-R-1 were basically an errata redesignating ST-1 as TS-R-1. The Commission also directed the staff to address, as part of the overall rulemaking plan, the unintended economic impact of its emergency Part 71 final rule entitled, “Fissile Material Shipments and Exemptions,” (62 FR 5907; February 10, 1997).

On December 13, 1999, the Commission directed (M991109A, Attachment 2) the staff, after the Commission’s meeting on nuclear materials and waste activities with invited stakeholders, to improve stakeholder public participation in the Nuclear Regulatory Commission’s (NRC) activities, including rulemaking.

In SECY-00-0117, ”Rulemaking Process for Revising 10 CFR Part 71 for Compatibility with IAEA Transportation Safety Standards (ST-1), and to Make Other Changes”, dated May 30, 2000, the staff provided the Commission an issues paper that presented a summary of the changes being considered in the Part 71 rulemaking to solicit early public input on these changes. In SRM-00-0117 dated June 28, 2000 (Attachment 3), the Commission directed the staff to publish the Part 71 issues paper for public comment (65 FR 44360; July 17, 2000), and also approved the enhanced public-participation process. Subsequently, three public meetings were held; a roundtable workshop on August 10, 2000, at the NRC Headquarters, and two townhall meetings on September 20, 2000, in Atlanta, GA, and September 26, 2000, in Oakland, CA. Public participation in these meetings was broad, and included members of the public, environmental and public interest groups, state representatives, the Western Governor’s Association, the U.S. Department of Energy, the U.S. Department of Transportation (DOT), the Nuclear Energy Institute, representatives of the radioactive material shipping industry, the oil and gas industry, and the mineral industry. Transcripts of the meetings, as well as a summary of the comments, were provided to the Commission, and were also placed on the NRC rulemaking interactive website at http://ruleforum.llnl.gov. The public comment period on the issues paper closed on September 30, 2000. A total of 48 written comments were received. In addition, an NRC web page similar to that prepared for the control of solid materials was created at: http:\\www.nrc.gov\NMSS\IMNS\transport.html.

DOT is the lead agency for the regulation of transportation of hazardous material in the United States, and is a co-regulator with NRC of transportation of radioactive material. DOT is also the U.S. competent authority for interaction with the IAEA. Therefore, this revision to Part 71 is being coordinated with DOT, to ensure that consistent regulatory standards are maintained, between NRC’s Part 71 and DOT’s Hazardous Materials Regulations (in particular, 49 CFR Parts 171-178), and to ensure that both rules are published on approximately the same schedule. DOT’s proposed regulations are intended to be consistent with NRC’s proposed regulations. DOT published for public comment an Advance Notice of Proposed Rulemaking containing the ST-1 changes on December 28, 1999 (64 FR 72633). During the public meetings, a DOT representative made a presentation at the meetings and joined the NRC staff in responding to public comments as they relate to the DOT rulemaking. DOT committed to review public comments for applicability to their proposed changes.

DISCUSSION:

The NRC staff formed a working group (WG) that included staff from the Office of Nuclear Material Safety and Safeguards (NMSS), the Office of States and Tribal Programs (STP), the Office of the General Counsel (OGC), as well as a representative from DOT. Further, a steering group consisting of NMSS and OGC managers provided direction in the development of the proposed rule. Also, as directed by SRM M000211, dated March 9, 2000 (Attachment 4), the staff prepared a comparison between TS-R-1 and Part 71 (Attachment 5), to determine the major differences between these two regulations.

The issues paper presented 18 issues for comment. As directed by the Commission in SRM-SECY-00-0117, issue 18 was added prior to publication in the Federal Register, to discuss the current IAEA standards for package surface removable contamination. Subsequent to the publication of the issues paper, the staff has added issue 19: Event Reporting Requirements. The addition of this issue was an outgrowth of staff review of reporting requirements to be consistent with changes to other reporting requirements (SECY-00-0093, Rulemaking to Modify the Event Reporting in 10 CFR 50.72 and 50.73 and for Independent Spent Fuel Storage Installations (ISFSI) in 10 CFR 72.216).

In developing the staff positions on the 19 issues in this proposed rule, the staff considered the existing Commission policy statements (e.g., NRC's metrication policy); the NRC's previous positions on some of these issues (e.g., on the IAEA radionuclide exemption values, the Type C package, and uranium hexafluoride packaging requirements); technical considerations from the NRC staff's latest experience with the 19 issues, either in the United States, or from the latest staff interactions at the IAEA meetings in Vienna; and, finally, the public comments received in the meetings, by mail, and on the NRC web site. The attached draft Federal Register Notice (FRN) (Attachment 6), contains the Part 71 proposed rule with an analysis and proposed staff position for each issue. Summary and categorization of public comments is enclosed as Attachment 7. A draft Regulatory Analysis and draft Environmental Assessment are also attached (Attachments 8 and 9, respectively).

The following is the list of Part 71 issues discussed in the FRN:

A. TS-R-1 or ST1 Compatibility Issues
  - Issue 1: Changing Part 71 to the International System of Units (SI) Only
  - Issue 2: Radionuclide Exemption Values
  - Issue 3: Revision of A_1 and A_2
  - Issue 4: Uranium Hexafluoride Package Requirements
  - Issue 5: Introduction of the Criticality Safety Index Requirements
  - Issue 6: Type C Package and Low Dispersible Material
  - Issue 7: Deep Immersion Test
  - Issue 8: Grandfathering Previously Approved Packages
  - Issue 9: Changes to Various Definitions
  - Issue 10: Crush Test for Fissile Material Package Design
  - Issue 11: Fissile Material Package Design for Transport by Aircraft

B. NRC-Initiated Issues
  - Issue 12: Special Package Approvals
  - Issue 13: Expansion of Part 71 Quality Assurance Requirements to Holders of, and Applicants for, a Certificate of Compliance
  - Issue 14: Adoption of American Society of Mechanical Engineers (ASME) Code
  - Issue 15: Change Authority for Part 71 Certificate Holders
  - Issue 16: Fissile Material Exemptions and General License Provisions

Issue 17: Double Containment of Plutonium (PRM-71-12)
Issue 18: Contamination Limits as Applied to Spent Fuel and High-Level Waste (HLW) Packages
Issue 19: Modifications of Event Reporting Requirements

On the TS-R-1 compatibility issues the staff is recommending that NRC adopt the TS-R-1 position on issues 2, 3,4, 5, 7, 8, 9, 10, and 11, either fully or with some modification. The staff is also recommending that NRC not adopt the TS-R-1 position on issues 1 and 6. On the NRC-initiated issues, the staff is recommending adoption of new requirements on issues 12, 13, 15, 16, 17, and 19. The staff is also recommending that NRC not adopt the ASME Code for storage and transportation cask fabrication (issue 14), and not adopt surface contamination-limits for large packages (spent fuel and HLW packages)[issue 18]. These recommendations are discussed in the FRN. DOT is in agreement with these recommendations.

The staff notes that several of the issues in this proposed rule engendered a high level of interest and discussion in public and staff meetings and comments on the NRC website on the issues paper. Further, the comments indicate a wide range of views on the appropriate action to resolve four of these issues. These issues are: issue 2, "Radionuclide Exemption Values," issue 12, "Special Package Approvals," issue 15, "Change Authority for Part 71 Certificate Holders," and issue 17, "Double Containment of Plutonium (PRM-71-12)." These four issues are summarized below. A detailed discussion is contained in the attached FRN (Attachment 6).

1. *Issue 2:* This issue presents the NRC with the challenge of whether to adopt a uniform-dose IAEA standard (and a radionuclide-specific Table of exemption values) versus using the current simpler single-value activity of 70 Bq/g limit for all radionuclides. The IAEA standard contains a provision to allow 10 times the specified exemption values for natural material and ores containing naturally occurring radioisotopes provided those ores/material are not intended to be processed for use of their isotopes. The staff notes that adopting the IAEA exemption provisions to harmonize with the IAEA standards results in an inconsistent level of protection, and a different regulatory treatment of these natural material and ores shipments based on the end use of the shipments, i.e., processing for extraction of radioisotopes versus processing for extraction of minerals, or disposal. However, the staff recommends adoption of the use of radionuclide-specific exemption values.

2. *Issue 12:* This issue is whether the NRC should propose Part 71 amendments to provide a standard for review of large-object packages, such as the Trojan Reactor Vessel, rather than reviewing each request on a case-by-case basis via the current exemption process, each of which requires Commission approval. The staff recommends the establishment of standards for approving these large objects as packages.

3. *Issue 15:* This issue discussed possible authority for certificate holders to safely make limited changes to the design of a transportation package — just as reactor and spent fuel storage facilities can safely make changes to their facilities (under 10 CFR 50.59 and 72.48). The staff recommends the extension of change authority to Part 71 certificate holders, but only for domestic dual-purpose spent fuel storage and transportation packages, i.e., for systems approved for both the transportation and storage of spent fuel.

4. *Issue 17:* This issue is whether the current, single-containment-barrier, Type B package standards would provide adequate accident protection when applied to packages

transporting plutonium, versus the existing double containment requirements. There is no comparable IAEA requirement for double containment of plutonium. The staff believes that a single Type B containment barrier is adequate for all transportation packages.

The staff plans to conduct a number of facilitated public meetings during the public comment period to discuss the proposed rule to enhance public input. These meetings will be held in locations across the country. The staff will provide the Commission with summaries of each of the public meetings. The proposed rule will be posted on the NRC's website and the staff plan to maintain the website dedicated for Part 71 up to date. The staff notes that, since publication of the Part 71 issues paper in July 2000, it made supporting information and documents available to the public, both on the Part 71 website and in the public document room.

The staff is particularly seeking stakeholders' comments related to quantitative information on the costs and benefits resulting from the proposed requirements, and operational data on exposures that might result (or be reduced) from implementing these proposed requirements. The staff hopes that stakeholder comments will help to quantify the potential impact of these proposed changes and will assist NRC in developing a risk-informed final rule.

The draft FRN was provided to the Agreement States for comment. The Agreement States' input was considered in the development of this draft proposed rule.

In developing its positions on each of the issues, the staff considered the four performance goals in the NRC Strategic Plan. For all issues, the performance goal of maintaining safety, protection of the environment, and the common defense and security, has been met. The remaining three performance goals: increasing public confidence; making the NRC activities and decisions more effective, efficient, and realistic; and reducing unnecessary regulatory burden on stakeholders, were addressed to varying degrees throughout the issues. For example, the staff believes that harmonizing Part 71 with the IAEA International regulations (issues 1 to 11) will both increase regulatory efficiency and effectiveness, and reduce unnecessary regulatory burden on licensees by eliminating the need to satisfy different regulatory requirements depending on whether the package is shipped domestically or internationally. The staff considered comments on the issues paper concerning any reduction in regulatory requirements such as the revised A_1 and A_2 values and addition of a change authority for certificate holders and believes that, on balance, public confidence will be increased because more accurate modeling of the dose will be used and safety will be maintained. In addition, the staff believes that overall public confidence will be increased as a result of this rulemaking because of the addition of regulatory requirements such as the criticality safety index and expansion of quality assurance requirements to certificate holders. The staff will further consider the relationship of this rulemaking and the performance goals in the NRC Strategic Plan as a result of comments on this proposed rule.

SCHEDULE:

Because of the complexity and size of this rulemaking, the staff is recommending a 90-day public comment period (typically 75 days) for the proposed rule during which time the public meetings would be held. The NRC also needs to ensure that the proposed and final rules on amending Part 71 and DOT's companion regulations are published concurrently. Thus, the staff proposes changing the current due date for delivery of a final rule to the Commission from the current date of June 30, 2002, to a date of one year after the close of public comment period. The staff will coordinate with DOT to publish the final rule concurrent with DOT's final rule.

COORDINATION:

OGC has reviewed this proposed rule and has no legal objection. The Office of the Chief Financial Officer has reviewed the Commission paper for resource impacts and has no objections. The Office of the Chief Information Officer has reviewed the Commission paper for information technology and information management implications and concurs in it. New information requirements resulting from this proposed rule will be submitted to the Office of Management and Budget at the same time the rule is forwarded to the Federal Register for publication. This paper has been coordinated with the Office of Enforcement.

RECOMMENDATION:

That the Commission:

1. Approve for publication in the Federal Register the proposed amendments to Part 71.

2. Note:

   a. That the proposed amendments will be published in the Federal Register allowing 90 days for public comment.

   b. That the proposed amendments will be published in a timeframe compatible with the proposed amendments from the U.S. Department of Transportation's associated rulemaking.

   c. That the staff will use an enhanced public participation process during the public comment period. A separate notice will be published in the Federal Register, subsequent to the publication of the proposed amendments, identifying the dates and locations of the public meetings on this rulemaking.

   d. That the Chief Counsel for Advocacy of the Small Business Administration will be informed of the certification and the reasons for it, as required by the Regulatory Flexibility Act, 5 U.S.C. 605(b).

   e. That a draft Regulatory Analysis has been prepared for this rulemaking (Attachment 7).

   f. That a draft Environmental Assessment has been prepared for this rulemaking (Attachment 8).

   g. The appropriate Congressional Committees will be informed of this action.

   h. That a press release will be issued by the Office of Public Affairs when the proposed rulemaking is filed with the Office of the Federal Register.

   i. That OMB review is required and a clearance package will be forwarded to OMB no later than the date the proposed rule is submitted to the Office of the Federal Register for publication.

j. That the resources to complete and implement this rulemaking are included in the current budget.

k. That the staff will deliver a final rule to the Commission one year after the close of the public comment period, or concurrent with DOT's final rule.

/RA/

William D. Travers
Executive Director
for Operations

Attachments:
1. SRM-SECY-99-200
2. SRM-M991109A
3. SRM-SECY-00-0117
4. SRM M000211
5. Comparison Between TS-R-1 and 10 CFR Part 71
6. Proposed Rule FRN
7. Summary and Categorization of public comments
8 Draft Regulatory Analysis
9. Draft Environmental Assessment

September 17, 1999

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Annette Vietti-Cook, Secretary /s/

SUBJECT: STAFF REQUIREMENTS - SECY-99-200 - FEDERAL REGISTER NOTICE RESPONDING TO PUBLIC COMMENTS RECEIVED ON AN EMERGENCY FINAL RULE FOR FISSILE MATERIAL SHIPMENTS

The Commission has approved the proposed Federal Register notice (FRN) subject to the following comments and the changes noted in the attachment.

1. The staff should take the following actions to address the potentially significant unintended economic impact on the shipment of fissile material without special moderators (the unintended impact issue) that was imposed by the emergency rule :

   a. Revise the FRN to indicate that the unintended impact issue will be addressed as part of the rulemaking to make Part 71 compatible with IAEA transportation standards.

   b. Revise the FRN to note the NRC need for data to document the unintended impact of the emergency rule. This approach will place the industry on notice that the data is needed.

   (EDO) (SECY Suspense (items 1.a & 1.b): 10/22/99)

   c. Prepare an overall rulemaking plan for Part 71 that addresses the need for changes to make NRC regulations compatible with IAEA transportation standards and also addresses the unintended impacts of the emergency rule. This should be completed by May 31, 2000. If the rulemaking activities associated with IAEA compatibility are delayed, the staff should prepare a separate plan that addresses the unintended economic impacts of the emergency rule. This should also be completed by May 31, 2000.

   d. If, based on further analysis of the unintended economic impact of the emergency rule, the staff concludes that those impacts are far less than characterized in SECY 99-200, the staff should inform the Commission, with some description of the supporting basis, that rulemaking is not warranted, in lieu of preparing a rulemaking plan for this issue.

2. On page 16 of the FRN, a commenter stated that the wording of the rule is not clear with respect to application of a limit to deuterium. The staff response clarifies the intent of

the rule but does not state whether staff intends to amend the existing rule language to remove the ambiguity. The response should be modified to indicate that this will be done at the time the corrective amendment is proposed.

(EDO) (SECY Suspense (items 1.c, 1.d, & 2): 5/31/00)

Attachment: Editorial Changes to the FRN in SECY-99-200

cc: Chairman Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
OGC
CIO
CFO
OCA
OIG
OPA
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR
DCS

Attachment

Editorial Changes to the FRN in SECY-99-200

1. On page 1, in the summary paragraph, line 7, delete the italics so that ‘Administrative Procedure Act’ appear in normal text.

2. On page 6, last paragraph, line 4, delete the italics so that ‘Administrative Procedure Act’ appear in normal text.

3. On page 6, last paragraph, revise the last sentence to end after ‘March 12, 1997' and insert the following after the period: ‘Following publication of the emergency rule and receipt of public comments, the staff sought to study the technical issues raised by the public comments, and to perform an independent evaluation of Part 71 regulations relating to the fissile material exemption and general license limits. The NRC awarded a contract to the Oak Ridge National Laboratory (ORNL). The results of the ORNL study were published by NRC in July 1998 and noticed in the Federal Register on August 13, 1998.’ Revise the remainder of the sentence to read ‘This notice responds to the comments received on the rule and the results of the Oak Ridge National Laboratory study.’

4. On page 10, 1st full paragraph, revise line 9 to read ‘ ... Commission has not, as yet, been able to obtain specific information regarding ~~substantiate~~ the burden claim ....’

5. On page 12, paragraph 2, revise line 8 to read ‘ ... the staff has not been able to obtain specific information regarding ~~substantiate~~ the burden ....’ In line 9, delete the comma after ‘consider’.

6. On page 13, 4th full paragraph, line 1, delete the comma after ‘rule’.

7. On page 15, line 13 from the top, delete ‘with NEI’.

8. On page 16, top paragraph, delete the last sentence (The National Technology Transfer ... or otherwise impractical.).

IN RESPONSE, PLEASE
REFER TO: M991109A

December 13, 1999

MEMORANDUM TO: William D. Travers
Executive Director for Operations

Stuart Reiter
Acting Chief Information Officer

FROM: Annette L. Vietti-Cook, Secretary /s/

SUBJECT: STAFF REQUIREMENTS - MEETING ON NRC INTERACTIONS WITH STAKEHOLDERS ON NUCLEAR MATERIALS AND WASTE ACTIVITIES, 9:00 A.M., TUESDAY, NOVEMBER 9, 1999, AUDITORIUM, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission met with invited stakeholders representing the nuclear materials industry, a public interest group, the Organization of Agreement States, the National Congress of American Indians, and the NRC staff to conduct a discussion on improving stakeholder participation processes used in the nuclear materials and waste activities.

The Commission appreciates the support and the time spent by the various individuals and organizations that participated in the stakeholder meeting. The broad representation of views provides the Commission with necessary insights for improving its interactions with all stakeholders.

The Commission commends the staff's past efforts to improve materials stakeholder participation and encourages the staff to consider the comments and discussion provided during the course of the meeting in order to improve interactions with stakeholders, and to incorporate the suggestions, as appropriate, into current NRC activities and plans. If problems are identified as additional improvements are made, the staff should request Commission guidance as necessary.

The following areas for improvement were highlighted by stakeholders:

1. **Communication Skills of NRC Representatives** - NRC representatives were often process oriented, legalistic, and reluctant to communicate about issues.

   Action: NRC management should clearly communicate the Commission's expectation for timely, open, accurate communications with all stakeholders. Additional training in this area should be provided as needed to the staff.

2. **Responsiveness to Stakeholder Comments and Concerns** - Stakeholders were often unaware of the disposition of their comments and concerns that were provided to

the NRC staff.

Action: NRC management and staff should ensure that, before comments are solicited, stakeholders are informed how their comments will be used. Stakeholders should also be informed how the NRC responses to their comments can be obtained, if comments are being addressed individually. If not, the staff should inform stakeholders about any summary documents that discuss the resolution of comments.

3. **Increased Use of Dedicated Teams, Working Groups, Published Activity Plans, Scoping Meetings, Regional and Facilitated Meetings, and Participatory Workshops** - Stakeholders noted that when these techniques are used, NRC has been successful in obtaining and considering stakeholder input.

   Action: NRC management should continue the use of these techniques in all major NRC activities. The staff should maintain and improve, as appropriate, the use of those techniques that have proven to be effective in ensuring adequate stakeholder input. As part of the improvement process, the staff should internally document the pros and cons of each technique as part of a lessons learned process.

4. **Stakeholder Representation** - Public Interest Groups were concerned that they were underrepresented at the meeting.

   Action: To the extent practicable, the Commission should ensure that a diversity of stakeholder groups are represented at its meetings. The staff should work with SECY to ensure that a wide range of public interest groups are adequately included.

5. **NRC Web Page and Access to Information** - The NRC web page is difficult to navigate to find information. Information should be available on the web in a timely way. The NRC should use technology that facilitates information exchange with stakeholders, including use of "list servers."

   Also, the NRC should be sensitive to the large number of public stakeholders that do not have Internet access. The NRC should ensure information is provided via other means.

   Action: The staff should take action in the following areas: (1) make the agency web site easier to navigate for ease of locating information while ensuring that information is available in a timely way, (2) facilitate information exchange with agency stakeholders, and (3) ensure that agency information is provided by other means for the large number of stakeholders without Internet access. In all instances, the OCIO is the lead organization.

   (1) Improving NRC's Web Site

   - In conjunction with current efforts underway to improve the NRC web site, review other major web sites (such as but not limited to OSHA, EPA, cadc.uscourts.gov, and thomas.loc.gov web pages) as models for possible improvements of the NRC site.

- Solicit the views of stakeholders who are frequent users of NRC's and other web sites as well as the views of others with experience retrieving information from the Web, such as members of the general public, researchers, and representatives of the library community. The staff should actively meet and hold discussions with these individuals in the development of an improved NRC web site.

- Work with the new NRC Communications Manager to identify and implement goals for the public site to support NRC's strategic plan.

- Ensure that the public site is compliant with the Americans with Disabilities Act.

(2) Facilitating Information Exchange with Agency Stakeholders

- Work with NMSS and OSP to explore the costs for the re-establishment of the "list servers" to reach out actively to stakeholders.

(3) Facilitating Information Availability for Stakeholders without Internet Access

- Review, update, and consider expanding the range of information sources currently available (e.g., PDR 800 number, the "Citizen's Guide to NRC Information") to improve awareness of access to agency information for stakeholders without Internet access.

- Consider, for example, enhanced outreach to public librarians to enlist their support in helping those who are not Internet-aware or who do not have Web access in their homes.

The staff is to provide a status report, including a completion plan, to the Commission on the above 3 items.

(CIO/EDO) (SECY Suspense: 2/15/00)

6. **Compatibility Requirements** - Carefully weigh more stringent compatibility requirements for Agreement States so as to assure greater consistency in regulatory requirements involving interstate commerce. Setting compatibility high minimizes variations in regulations between the NRC and the Agreement States. These variations can be costly and confusing, and do not improve safety. Sometimes, the Agreement State rules have consequences that were not expected and affected parties may not be adequately informed about proposed or final revisions.

   Action: In conjunction with current efforts to improve materials regulations, evaluate the appropriate compatibility levels for new regulations, including the public comment process in Agreement States, so as to balance the benefits of uniformity in regulations that have transboundary implications against the benefits of providing flexibility to the Agreement States. Additionally, provide an evaluation of the feasibility of NRC creating and maintaining a web page serving as a bulletin board for Agreement State rulemaking activities. This bulletin board may consist of simply a link to all the appropriate NRC and

Agreement State web sites or something different. As part of this evaluation, the staff should provide the Commission with pros and cons of the proposal addressing issues such as (1) would the web site be cost effective, (2) proposed methods of financing the web site, (3) should Agreement State participation be voluntary or mandatory, (4) legal implications of developing the web site, and (5) potential alternative methods than the proposed web site for ensuring copies of Agreement States' proposed and final rules are readily accessible to persons in other States.

(EDO/CIO) (SECY Suspense: 5/30/00)

cc: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
OGC
CFO
OCA
OIG
OPA
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR - Advance
DCS - P1-17

June 28, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Annette L. Vietti-Cook, Secretary **/RA/**

SUBJECT: STAFF REQUIREMENTS - SECY-00-0117 - RULEMAKING PROCESS FOR REVISING 10 CFR PART 71 FOR COMPATIBILITY WITH IAEA TRANSPORTATION SAFETY STANDARDS [ST-1], AND TO MAKE OTHER CHANGES

The Commission has approved the staff's recommendation to use an enhanced-public-participation process (web-site and public meetings) in the 10 CFR Part 71 rulemaking; and to publish, for public comment, an issues paper in the Federal Register that discusses NRC's plan to revise 10 CFR Part 71 and provides a summary of the changes being considered, subject to the attached changes to the Federal Register notice.

The staff should have informal communication with the Commissioners' Technical Assistants to provide feedback to the Commission on the public meetings scheduled for this summer on the 10 CFR Part 71 rulemaking and the comments received, the staff's progress on the 10 CFR Part 71 rulemaking, and the status of DOT's rulemaking effort. The Commission office principal points of contact for periodic briefings are:

Ron Zelac, Office of Chairman Meserve
Joe Olencz, Office of Commissioner Dicus
Diane Flack, Office of Commissioner Diaz
Janet Schlueter, Office of Commissioner McGaffigan
John Thoma, Office of Commissioner Merrifield

After conduct of the public meetings scheduled for this summer, the staff should proceed directly to develop a proposed rule for submittal to the Commission by March 1, 2001.

Attachment: Changes to the Federal Register Notice

cc: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
OGC
CIO
CFO
OCA
OIG
OPA
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR

Attachment

Changes to the Federal Register Notice

1. The Federal Register notice should be revised to include a schedule with key milestones for development of a revised 10 CFR Part 71 consistent with the staff plan to submit a final rule to the Commission for approval in June 2002, concurrent with the timing of a final Department of Transportation (DOT) rule.

2. Appendix A of the Federal Register notice should be revised to include the referenced Tables I and II and Figures 2, 3 and 4 from ST-1 to enhance the public's participation in the rulemaking process.

3. The Federal Register notice should be revised to state that, contrary to NRC's rulemaking process under the Administrative Procedure Act, development of the International Atomic Energy Agency's (IAEA) Safety Series No. ST-1 for the transport of radioactive material did not directly involve the public or include a cost-benefit analysis to our knowledge. In contrast, NRC is bound to consider costs and benefits in its regulatory analyses, and is prepared to differ from the ST-1 standards, at least for domestic purposes, to the extent the standards can not be justified from a cost-benefit perspective.

4. The staff should revise the issues paper prior to its release to:

   a. add a new issue eighteen to discuss the current IAEA standard for package surface removable contamination (i.e., 4 Becquerel per centimeter squared) applied to spent fuel and high-level waste (HLW) containers; and

   b. modify Issue 2, "Radionuclide Exemption Values," which allows certain packages containing radioactive material to be shipped without being labeled as or considered radioactive, to capture the possibility of unintended consequences in implementing ST-1's concentration values in areas outside of transportation and to request stakeholder help in assessing those consequences. The current discussion in the issues paper should be expanded to more clearly discuss the fact that the DOT current exempt material standard of 2000 picoCurie per gram (2000 pCi/gm), based on previous IAEA transportation standards, has application by cross reference outside the domain of transportation. Therefore, the staff should engage the industries, organizations, and State and Federal agencies most likely to be potentially impacted from adopting the new IAEA values to ensure that all stakeholders have an opportunity to provide input on this matter.

The staff should be prepared at the public meetings to explain its or the Commission's previous positions on the issues and to discuss the staff's current views, subject to acknowledgment that the staff's and the Commission's final views have not been determined, and to seek public comment on items 4.a and 4.b (above) at the public meeting.

Action: Kane, NMSS

Cys: Travers
Paperiello
Miraglia
Norry
Blaha
Schroll, SECY

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SECRETARY

March 9, 2000

IN RESPONSE, PLEASE REFER TO: M000211

MEMORANDUM TO: William D. Travers
Executive Director for Operations

William M. Beecher, Director
Office of Public Affairs

FROM: Annette L. Vietti-Cook, Secretary Annette Vietti-Cook

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON STATUS OF NMSS PROGRAMS, PERFORMANCE, AND PLANS, 9:30 A.M., FRIDAY, FEBRUARY 11, 2000, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the NRC staff on the programs, performance, and plans in the Office of Nuclear Materials Safety and Safeguards (NMSS). The Commission commends NMSS for its accomplishments over the past year and its contribution to maintaining public health and safety.

Due to the broad range of programs and diverse stakeholders, NMSS has a complex challenge in communicating risk to the public. Consistent with the SRM from the meeting with the stakeholders concerning nuclear materials and waste activities (M991109A), the staff should provide its plans for ensuring that NRC staff, including inspection and licensing staff, are appropriately trained in communication skills.

(~~EDO~~/OPA) (SECY suspense: ~~9/30/00~~) 200000040
(NMSS/ 9/23/00

As part of the rulemaking on 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," the staff should perform a comparison of NRC transportation requirements and IAEA standards and their potential impacts on NRC licensees.

(NMSS) 199800008

A Comparison of Part 71 and TS-R-1
Prepared for the Nuclear Regulatory Commission

Transportation Technologies Group
Oak Ridge National Laboratory
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Introduction

The Nuclear Regulatory Commission (NRC) is undertaking rulemaking actions to keep its transport safety regulations (Part 71) harmonized with those of the International Atomic Energy Agency (IAEA). The IAEA's "Regulations for the Safe Transport of Radioactive Material", also known as TS-R-1, are periodically revised to keep them current with developing transportation technologies and the latest radiation protection principles. The latest revision of the IAEA regulations was published in 1996. Member States and international organizations are now working to implement regulations which are harmonized with TS-R-1.

As part of the analysis of possible issues related to harmonizing with TS-R-1, NMSS/IMNS requested that ICF, Inc., in collaboration with Oak Ridge National Laboratory, develop a comparison of the TS-R-1 and Part 71 regulations. This comparison was prepared under the technical direction of SFPO. Rather than preparing a simple text-by-text comparison, it was decided that an explanation of the comparability between the two sets of regulations would be most helpful. These explanations are provided on a paragraph and subparagraph basis since many of the requirements in the regulations are quite detailed. The comparison table presents a summary of each paragraph and subparagraph in TS-R-1, lists any corresponding Part 71 citations and provides an explanation their comparability.

Radioactive materials transportation safety regulatory responsibilities in the United States are shared by the NRC and the Department of Transportation (DOT). These responsibilities have been coordinated and are delineated in a Memorandum of Understanding between the two agencies. Accordingly, the regulations promulgated by each agency reflect these areas of responsibility. Since the TS-R-1 regulations cover all aspects of radioactive materials transportation safety, they span the areas of responsibility of both NRC and DOT. In order to present an accurate and complete description of the regulatory requirements in the US, the comparison table indicates this by noting where TS-R-1 requirements are reflected in the DOT requirements. Since a majority of the TS-R-1 regulations address topics that are the responsibility of DOT, there are a large number of TS-R-1 paragraphs that are not reflected in Part 71 (these are noted in the table). This does not reflect a shortcoming in Part 71, just the division of responsibilities between NRC and DOT.

Organization of the Comparison Table

The table contains the following columns:

- Column 1 - identifies the relevant TS-R-1 paragraph or subparagraph
- Column 2 - identifies the relevant Part 71 paragraph or subparagraph
- Column 3 - contains the regulatory text from TS-R-1, or, where the regulatory text is very long, the TS-R-1 requirements are summarized

- Column 4 - provides a comparison of the TS-R-1 and Part 71 requirements

When Part 71 does not have a requirement corresponding to a TS-R-1 requirement, this is generally noted by a blank in column 2 and an explanation in column 4. When Part 71 contains a requirement which is not reflected in TS-R-1 this is generally noted by a blank in column 1 and an explanation in column 4. The table is sorted by TS-R-1 paragraph numbers (column 1). This results in the Part 71 requirements which do not have corresponding TS-R-1 requirements appearing “en block” at the end of the table.

The radioactive materials transport safety regulations are very detailed and there are many degrees to which multiple sets of regulations may be consistent. This degree of consistency ranges from being identical to being very different. In order to provide a measure of understanding of how close the regulations are to each other, the following terms are used in the table with their associated meanings:

- Identical — exactly the same
- Essentially the same — minor wording differences
- Similar requirement — wording and content difference for a **mandatory statement**
- No similar requirement — no corresponding mandatory statement
- Similar provision — wording and content difference for a **permissive statement**
- No similar provision — no corresponding permissive statement

The table identifies all of the significant differences between the US and IAEA regulations, both those which are historical in nature and those introduced by TS-R-1. In this latter case, it is noted in column 4 (the comparison), including noting where there is:

- No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking
- No similar requirement, however, this topic is within DOT’s area of responsibility
- No similar provision; this is a new provision in TS-R-1 and is being considered in the Part 71 rulemaking
- No similar provision, however, this topic is within DOT’s area of responsibility

Since Part 71 was revised earlier to be harmonized with the previous edition of the IAEA regulations (Safety Series No. 6, 1985 (as amended 1990)), most of the current differences between Part 71 and TS-R-1 stem from changes introduced by TS-R-1. However, some historical differences between Part 71 and the IAEA regulations continue to exist for various reasons, such as: the NRC’s license- and certificate-based regulatory approach; requirements implemented in response to unique US situations (such as the plutonium by air requirements); and variations in the level of detail specified in some topics (such as quality assurance). It should be noted that it is not unusual for IAEA Member States to have national variations in their domestic regulations.

The major differences between Part 71 and TS-R-1, both historical and those resulting from changes in the IAEA regulations, have been addressed as “Issues” in the Part 71 rulemaking. The comparison did not identify other major issues which are not presently covered in the rulemaking. Additional differences (which are not major) are identified in column 4 of the table.

TS-R-1	10 CFR Part 71	Safety Series TS-R-1 Summary	TS-R-1 to 10 CFR Part 71 Comparison
101		**SECTION I — INTRODUCTION** BACKGROUND (101-103) Defines the purpose of the regulations as establishing standards of safety and levels of control for radiological hazards for people, property, and the environment.	Similar to 71.0, Purpose and scope, but Part 71 does not build on the IAEA Basic Safety Standards, SS No. 115 and the "'Radiation Protection and Safety of Radiation Sources", SS No. 120.
102		Indicates that this Safety Standard is supplemented by a hierarchy of Safety Guidelines and Safety Practices, including ST-2 and Safety Series Nos. 87, 112, and 113.	No comparable paragraph, it describes the IAEA guidance documents.
103	71.0 (c)	It is each government's prerogative to assign responsibility to specific persons for carrying out particular regulatory actions.	71.0 (c) makes part 71 applicable to licensees "if the license delivers that material to a carrier for transport..."
104		OBJECTIVE (104-105) States that the objective of these regulations is to protect persons, property and environment from the effects of radiation during the transport of radioactive material. Prescribes the general requirements necessary to achieve this protection. (a) containment of the radioactive contents' (b) control of external radiation levels; (c) prevention of criticality; and (d) preventionof damage caused by hear	This is explanatory text . No similar general statement in Part 71.
105		States that in the transport of radioactive material that the safety of persons, who are either members of the public or workers, is assured when these Regulations are complied with. Confidence is achieved through quality assurance and compliance assurance programs.	This is explanatory text . No similar general statement in Part 71.
106	71.0 (c), (e), and (f)	SCOPE (106-109) Specifies that these Regulations apply to the transport of radioactive material by all modes on land, water, or in the air, including transport which is incidental to the use of the radioactive material. Transport comprises all operations and conditions associated with and involved in the movement of radioactive material; these include the design, manufacture, maintenance, and repair of packaging, and the preparation, consigning, loading, carriage including in-transit storage, unloading and receipt at the final destination of loads of ration material and packages. A graded approach is applied to the performance standards in these Regulations that is characterized by three general severity levels:	Similar to 71.0 (c), (e) and (f).

106 (a)		routine conditions of transport (incident free);	No similar provision.
1 0 6 (b)		normal condition of transport (minor mishaps);	No similar provision.
106 (c)		accident conditions of transport.	No similar provision.
107 (a)		Specifies that regulations do not apply to radioactive material that is an integral part of the means of transport.	No similar provision in Part 71.
1 0 7 (b)	71.0 (c)	Specifies that regulations do no apply to radioactive material moved within an establishment which is subject to appropriate safety regulations in force in the establishment and where the movement does not involve public roads or railways.	71.0 (c) is similar.
107 (c)		Specifies that regulations do not apply to radioactive material implanted or incorporated into a person or live animal for diagnosis or treatment.	No similar provision, but the 71.5(a) reference to Title 49 includes 173.401 (b) which is similar.
1 0 7 (d)		Specifies that the transport regulations do not apply to radioactive material in consumer products which have received regulatory approval, following their sale to the end user.	No similar provision, however, this topic is within DOT's area of responsibility.
107 (e)		States that the regulations do not apply to natural occurring radionuclides which are not intended to be processed for use of these radionuclides provided the activity concentration of the material does not exceed 10 times the values specified in paras. 401-406.	No similar provision, however, this topic is within DOT's area of responsibility.
108		States that these regulations do not specify controls such as routing or physical protection which may be instituted for reasons other than radiological safety.	Routing is covered by Title 49 and physical protection is covered in other Parts of Title 10.
109		States that for radioactive material having subsidiary risks, the relevant regulations for each country in which transportation takes place applies in addition to these Regulations.	No similar provision, however, this topic is within DOT's area of responsibility.
110		STRUCTURE (110) Discusses the organizational structure of TS-R-1.	Not applicable.
201	71.4	**SECTION II - DEFINITIONS** A1 & A2 shall mean the activity value of radioactive material which is listed in Table 1 or derived in Section IV and is used to determine the activity limits for the requirements of these regulations.	Essentially the same, but Part 71 contains reference to "...permitted in a Type A package..." which is not in TS-R-1.
202		Cargo aircraft shall mean any aircraft, other than a passenger aircraft, which is carrying goods or property.	No similar defined term in Part 71. See 49 CFR 171.8.
203		Passenger aircraft, shall mean an aircraft that carries any person other than a crew member, a carrier's employee in an official capacity, an	No similar defined term in Part 71. See 49 CFR 171.8.

		authorized representative of an appropriate national authority, or a person accompanying a consignment.	
204		Multilateral approval shall mean approval by the relevant competent authority both of the country of origin of the design or shipment and of each country through or into which the consignment is to be transported. The term "through or into" specifically excludes "over", i.e., the approval and notification requirements shall not apply to a country over which radioactive material is carried in an aircraft, provided that there is no scheduled stop in that country.	No similar defined term in Part 71. See 49 CFR 171.403 (similar to TS-R-1).
205		Unilateral approval shall mean an approval of a design which is required to be given by the competent authority of the country of origin of the design only.	No similar defined term in Part 71. See 49 CFR 171.403.
206	71.4	Carrier shall mean any person, organization or government undertaking the carriage of radioactive material by any means of transport. The term includes both carriers for hire or reward (known as common or contract carriers in some countries) and carriers on own account (known as private carriers in some countries).	Similar wording in Part 71.
207		Competent authority shall mean any national or international regulatory body or authority designed or otherwise recognized as such for any purpose in connection with these Regulations.	No similar defined term in Part 71. See 49 CFR 171.8.
208		Compliance assurance shall mean a systematic programme of measures applied by a competent authority which is aimed at ensuring that the provisions of these Regulation are met in practice.	No similar defined term in Part 71.
209		Confinement system shall mean the assembly of fissile material and packaging components specified by the designer and agreed to by the competent authority as intended to preserve criticality safety.	No similar defined term in Part 71.
210		Consignee shall mean any person, organization or government which receives a consignment.	No similar defined term in Part 71.
211		Consignment shall mean any package or packages, or load of radioactive material, presented by a consignor for transport.	No similar defined term in Part 71.
212		Consignor shall mean any person, organization or government which prepares a consignment for transport, and is named as consignor in the transport documents.	No similar defined term in Part 71.

213	71.4	Containment system shall mean the assembly of components of the packaging specified by the designer as intended to retain the radioactive material during transport.	Similar wording in Part 71.
214		Contamination shall mean the presence of a radioactive substance on a surface in quantities in excess of 0.4 Bq/cm^{22} for beta and gamma emitters and low toxicity alpha emitters, or 0.04 Bq/cm^{22}2 for all other alpha emitters.	No similar defined term in Part 71.
215		Non-fixed contamination shall contamination that can be removed from a surface during routine conditions of transport.	No similar defined term in Part 71.
216		Fixed contamination shall mean contamination other than non-fixed contamination.	No similar defined term in Part 71.
217	71.4	Conveyance shall mean for transport by road or rail: any vehicle; for transport by water: any vessel, or any hold, compartment, or defined deck area of a vessel; and for transport by air: any aircraft.	71.4 definition also includes "large freight container" for road and rail transport.
218		Criticality safety index (CSI) assigned to a package, overpack or freight container containing fissile material shall mean a number which is used to provide control over the accumulation of packages, overpacks or freight containers containing fissile material.	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
219		Defined deck area shall mean the area, of the weather deck of a vessel, or of a vehicle deck of a roll-on/roll-off ship or a ferry, which is allocated for the stowage of radioactive material.	No similar defined term in Part 71.
220		Design shall mean the description of special form radioactive material, low dispersible radioactive material, package or packaging which enable such an item to be fully identified. The description may include specifications, engineering drawings, reports demonstrating compliance with regulatory requirements, and other relevant documentation.	No similar defined term in Part 71. See 49 CFR 173.401.
221	71.4	Exclusive use shall mean the sole use, by a single consignor, of a conveyance or of a large freight container....	Part 71 is more prescriptive in its definition, adding radiation protection provisions and requiring specific written instructions. See 49 CFR 173.403 also.
222	71.4	Fissile material shall mean U-233, U-235, Pu-239, and Pu-241.	Essentially the same but Part 71 includes Pu-238.
223		Defines large freight container and small freight container.	No similar defined term in Part 71. See 49 CFR 173.403.
224		Defines intermediate bulk container as a portable packaging that has a capacity of not more than 3 m^{33}, is designed for mechanical	No similar defined term in Part 71. See 49 CFR 171.8 for a similar definition applicable to all hazardous materials IBCs.

		handling, is resistant to the stresses produced in handling and transport, is designed to conform to the standards in the chapter on Recommendations on Intermediate Bulk Containers (IBC's) of the United Nations Recommendations on the Transport of Dangerous Goods.	
225		Low dispersible radioactive material shall mean either a solid radioactive material or a solid radioactive material in a sealed capsule, that has limited dispensability and is not in powder form.	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
226	71.4	Low specific activity material shall mean radioactive material which by its nature has a limited specific activity, or radioactive material for which limits of estimated average specific activity apply. External shielding materials surrounding the LSA material shall not be considered in determining the estimated average specific activity.	Both Part 71 and 49 CFR 173.403 have a similar definition, but there are differences with TS-R-1. The US regulations define LSA-I to include mill tailings, debris, etc. not exceeding 10E-6 A2/g (TS-R-1 does not include this). Part 71 and Title 49 exclude fissile material from LSA-I for unlimited A2 materials. Part 71 and Title 49 do not contain the LSA-I provision for materials up to 30 times the exempt activity concentrations. LSA-II is essentially the same. TS-R-1 excludes powders from LSA-III.
227	71.4	Low toxicity alpha emitters are: natural uranium; depleted uranium; natural thorium; uranium-235 or uranium-238; thorium-232; thorium-228 and thorium-230 when contained in ores or physical and chemical concentrates; or alpha emitters with a half-life of less than 10 days.	Essentially the same. Part 71 adds "tailings".
228	71.4	Maximum normal operating pressure shall mean the maximum pressure above atmospheric pressure at mean sea-level that would develop in the containment system in a period of one year under the conditions of temperature and solar radiation corresponding to environmental conditions in the absence of venting, external cooling by an ancillary system, or operational controls during transport.	Essentially the same.
229		Overpack shall mean an enclosure such as a box or bag which is used by a single consignor to facilitate as a handling unit a consignment of one or more packages for convenience of handling, stowage and carriage.	No similar defined term in Part 71. See 49 CFR 171.8.
230	71.4	Lists each package type and states that packages containing fissile material or UF6 are subject to additional requirements.	The basic definition is the same. Part 71 definition does not include excepted packages, industrial packages, Type A or Type C packages. Part 71 includes additional information on the difference between Type B(U) and Type B(M) and does not mention the additional requirements for packages containing fissile material or UF6.
231	71.4	Packaging shall mean the assembly of components necessary to enclose the radioactive	Basically the same. Part 71 and 49 CFR 173.403 include the provision that the vehicle, tie-down

		contents completely. It may consist of one or more receptacles, absorbent materials, spacing structure, radiation shielding and service equipment for filling, emptying, venting and pressure relief; devices for cooling; absorbing mechanical shocks, handling and tie-down, and thermal insulation; and service devices integral to the package. The packaging may be a box, drum or similar receptacle, or may also be a freight container, tank or intermediate bulk container.	system and auxiliary equipment may be designated as part of the packaging.
232	71.101	Quality assurance shall mean a systematic program of controls and inspections applied by any organization or body involved in the transport of radioactive material which is aimed at providing adequate confidence that the standard of safety prescribed in this Regulation is achieved in practice.	Similar definition is found in 71.101(a).
233		Radiation level shall mean the corresponding dose rate expressed in millisieverts per hour.	No similar defined term in Part 71. See 49 CFR 173.403.
234		Radiation Protection Programme shall mean systematic arrangements which are aimed at providing adequate consideration of radiation protection measures.	No similar defined term.
235		Radioactive contents shall mean the radioactive material together with any contaminated or activated solids, liquids and gases within the packaging.	No similar defined term in Part 71. See 49 CFR 173.403.
236	71.10 (a)	Radioactive material shall mean any material containing radionuclides where both the activity concentration and the total activity in the consignment exceed the values specified in para 401-406.	No similiar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking. However, 71.10 (a) identifies material which is exempt from Part 71, and 49 CFR 173.403 defines radioactive material in an identical way (0.002 microcurie per gram). This definition of radioactive material and the 0.002 microcurie per gram value is SUBSTANTIALLY different from TS-R-1, which is radionuclide-specific.
237		Shipment shall mean the specific movement of a consignment from origin to destination.	No similar defined term.
238		Special arrangement shall mean those provisions, approved by the competent authority, under which consignments which do not satisfy all the applicable requirements of these Regulations may be transported.	No similar defined term. However, a specific exemption (71.8) is similar to a special arrangement.
239	71.4	Special form radioactive material shall mean either an indispersible solid radioactive material or a sealed capsule containing radioactive material.	Definitions are similar, no technical differences. Part 71 includes requirements in the definition while TS-R-1 separates the requirements into another section. Part 71 provides grandfathering for designs in the definition while TS-R-1

			separates these into another section.
240	71.4	Specific activity of a radionuclide shall mean the activity per unit mass of that nuclide. The specific activity of a material shall mean the activity per unit mass or volume of the material in which the radionuclides are essentially uniformly distributed.	Definitions are similar. TS-R-1 also includes "activity per unit volume" for materials.
241	71.4	Surface contaminated object shall mean a solid object which is not itself radioactive but which has radioactive material distributed on its surfaces. SCO shall be in one of two group: SCO-I or SCO-II	Essentially the same.
242		Tank shall mean a tank container, a portable tank, a road tank vehicle, a rail tank wagon or a receptacle with a capacity of not less than 450 litres to contain liquids, powders, granules, slurries or solids which are loaded as gas or liquid and subsequently solidified, and of not less than 1000 litres to contain gases. A tank container shall be capable of being carried on land or on sea and of being loaded and discharged without the need of removal of its structural equipment, shall possess stabilizing members and tie-down attachments external to the shell, and shall be capable of being lifted when full.	No similar defined term.
243	71.4	The transport index assigned to a package, overpack or freight container, or to unpackaged LSA-I or SCO-I, shall mean a number which is used to provide control over radiation exposure.	This is a revised definitiont in TS-R-1 and is being considered in the Part 71 rulemaking. Definitions are technically substantially different. Part 71 (and 49 CFR 173.403) include criticality safety controls in the transport index. TS-R-1 uses a separate criticality safety index.
244		Unirradiated thorium shall mean thorium containing not more than 10E-7 g of U-233 per gram of thorium-232.	No similar defined term in Part 71. See 49 CFR 173.403.
245		Unirradiated uranium is defined as uranium containing not more than 2 x 10E+3 Bq of plutonium per gram of U-235, not more than 9 X 10E+6 Bq of fission products per gram of U-235 and not more than 5 x 10E-3 g of U-236 per gram of U-235.	No similar defined term in Part 71. See 49 CFR 173.403 which is consistent with SS No. 6 but differs from TS-R-1.
246	71.4	Natural uranium shall mean chemically separated uranium containing the naturally occurring distribution of uranium isotopes (approximately 99.28% U-238, and 0.72% U-235 by mass). Depleted uranium shall mean uranium containing a lesser mass percentage of U-235 than in natural uranium. Enriched uranium shall mean uranium containing a greater mass percentage of U-235 than 0.72%. In all cases, a very small mass percentage of U-234 is present.	Essentially the same.

247		Defines vehicle as a road vehicle (including an articulated vehicle, I.e. a tractor and semi-trailer combination) or railroad car or railway wagon. Each trailer shall be considered as a separate vehicle.	No similar defined term.
248		Defined vessel as any seagoing vessel or inland waterway craft used for carrying cargo.	No similar defined term in Part 71. See 49 CFR 171.8 which differs significantly from TS-R-1.
301		**SECTION III - GENERAL PROVISIONS** RADIATION PROTECTION (301-307) Requires a Radiation Protection Program for the transport of radioactive material. The nature and extent of the measures to be employed in the program shall be related to the magnitude and likelihood of radiation exposures. Incorporates the requirements of paras. 302-303 and 305-309. Requires documentation to be available for inspection.	No similar requirement in Part 71.
302		Specifies that doses must be below relevant dose limits. Requires consideration of ""interface between transport and other activities"".	No similar requirement in Part 71.
303		Need for continuous training concerning radiation hazards involved and worker's responsibility. Need for workers to consider affect of their actions on other persons.	No similar requirement in Part 71. See 49 CFR 172.700.
304		Specifies that the competent authority shall arrange for periodic assessments of the radiation doses to persons due to the transport of radioactive material, to ensure that the system of protection and safety complies with the Basic Safety Standards [2].	No similar requirement.
305		States for occupation exposures arising from transport activities, where it is assessed that the effective dose: (a) is most unlikely to exceed 1 mSv in a year, neither special work patterns nor detailed monitoring nor dose assessment programs nor individual record keeping shall be required: (b) is likely to be between 1 and 6 mSv in a year, a dose assessment program via work place monitoring or individual monitoring shall be conducted; (c) is likely to exceed 6 mSv in a year, individual monitoring shall be conducted. When individual monitoring or work place monitoring is conducted, appropriate records shall be kept.	No similar requirement.
306	71.131	Requires sufficient segregation between radioactive materials and workers/members of	No similar requirement in Part 71. Title 49 Parts 174, 175, 176, and 177 do, however, require

		the public. The values for dose shall be used for the purpose of calculating segregation distances or radiation levels: (a) for workers in regularly, occupied working areas, a dose of 5 mSv in a year; (b) for members of the public, in areas where the public has regular access, a dose of 1 mSv in a year to the critical group.	segregation during transit.
307		Segregation between radioactive materials and undeveloped photographic film. The basis for determining segregation distances for this purpose shall be that the radiation exposure of undeveloped photographic film due to the transport of radioactive material be limited to 0.1 mSv per consignment of such film.	No similar requirement in Part 71. Title 49 Parts 175, 176, 177, and 178 do, however, require segregation during transit.
308		EMERGENCY RESPONSE (308-309) Specifies that in the event of accidents or incidents during the transport of radioactive material, emergency provisions, as established by relevant national and/or international organizations, shall be observed to protect persons, property, and the environment.	No similar requirement in Part 71. Title 49 addresses emergency actions and responsibilities in several Parts.
309		Emergency procedures shall take into account the formation of other dangerous substances, that may result from the reaction between the contents of a consignment and the environment in the event of an accident.	No similar requirement.
310	71.37 71 Subpart H	QUALITY ASSURANCE (310) Para. 310 - Quality Assurance QA Programmes shall be established and implemented for the design, manufacture, testing, documentation, use, maintenance, and inspection of all special form radioactive material, low dispersible radioactive material and packages and for transport and in-transit storage operations to ensure compliance with the relevant provisions of these Regulations. The manufacturer, consignor, or user shall be prepared to provide facilities for competent authority inspection during manufacture and use and to demonstrate to any cognizant competent authority that:	Part 71 is much more detailed and specific than TS-R-1, but does not cover areas such as in-transit storage and low dispersible material. Basic approaches are similar.
310 (a)	71.37 71 Subpart H	the manufacturing methods and materials used are in accordance with the approved design specifications; and	

310 (b)	71.37 71 Subpart H	all packages are periodically inspected and, as necessary, repaired and maintained in good condition so that they continue to comply with all relevant requirements and specifications, event after repeated use. Where competent authority approval is required, such approval shall take into account and be contingent upon the adequacy of the quality assurance program.	
311	71.0 (f) 71.93	COMPLIANCE ASSURANCE Competent authority is responsible for assuring compliance with these Regulations. Means to discharge this responsibility include the establishment and execution of a program for monitoring the design, manufacture, testing, inspection, and maintenance of packaging, special form radioactive material and low dispersible radioactive material, and the preparation, documentation, handling, and stowage of packages by consignors and carriers, to provide evident that the provisions of these Regulations are being met in practice.	Part 71 provisions address the NCR's right to inspect, record retention, etc. TS-R-1 addresses the Competent Authority's responsibility to have a program which ensures that the regulations are being met in practice.
312	71.8	SPECIAL ARRANGEMENT States that consignments for which conformity with the other provisions of these Regulations is impracticable shall not be transported except under special arrangement. The competent authority may approve special arrangement transport operations for single or a planned series of multiple consignments. The overall level of safety in transport shall be at least 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.	A Special Arrangement is similar to a specific exemption (71.8) in that it may be granted by the cognizant authority. TS-R-1 requires "equivalent safety" while Part 71 requires the NRC to determine that it is "authorized by law and will not endanger life or property nor the common defense and security".
401	71, Appendix A	**SECTION IV - ACTIVITY LIMITS AND MATERIAL RESTRICTIONS** BASIC RADIONUCLIDE VALUES (401) References Table I (Basic Radionuclide Values) with A1 and A2 values for radionuclides. Table I provides A1 and A2 values, activity concentration for exempt materials, and activity limits for exempt consignments for radionuclides.	Part 71, Appendix A, does not contain values for exempt activity concentrations nor activity limits for exempt consignments of radionuclides. This is being considered in the Part 71 rulemaking.
401 (a)	71, Appendix A	A(sub1) and A(sub2) in TBq	The A-values which were changed between SS No. 6 and its revision to TS-R-1 are not reflected in Part 71. Part 71 and 49 CFR 173.435 are consistent with SS No. 6.

401 (b)	71, Appendix A	activity concentration for exempt material in Bq/g	No similar requirement. This is being considered in the Part 71 rulemaking.
401 (c)	71, Appendix A	activity limits for exempt consignment in Bq	No similar requirement. This is being considered in the Part 71 rulemaking.
402	71, Appendix A II	DETERMINATION OF BASIC RADIONUCLIDE VALUES (402-406) A1 and A2 determination requires competent authority approval if not in table, but no approval if in table. The determination of basic radionuclides values which are not listed in Table 1 shall require competent authority approval or, for international transport, multilateral approval. Where the chemical form of each radionuclide is known, it is permissible to use the A2 value related to its solubility class as recommended by the International Commission on Radiological Protection, if the chemical forms under both normal and accident conditions of transport are taken into consideration. Alternatively, the radionuclide values in Table II may be used without obtaining competent authority approval.	Similar in that both require approval of unlisted radionuclides. TS-R-1 contains more specifics such as allowing chemical form to be considered.
403	71, Appendix A III	In the calculations of A(sub1) and A(sub2) for a radionuclide not in Table I, a single radioactive decay chain in which the radionuclides are present in their naturally occurring proportions, and in which no daughter nuclide has a half-life either longer than 10 days or longer than that of the parent nuclide, shall be considered as a single radionuclide; and the activity to be taken into account and the A(sub1) or A(sub2) value to be applied shall be those corresponding to the parent nuclide of that chain. In the case of radioactive decay chains in which any daughter nuclide has a half-life either longer than 10 days or greater than that of the parent nuclide, the parent and such daughter nuclides shall be considered as mixtures of different nuclides.	Essentially the same
404	71, Appendix A IV	Provides equations that gives conditions that must be met for mixtures of radionuclides.	Essentially the same.
405	71, Appendix A V	States that when the identity of each radionuclide is know but the individual activities of some of the radionuclides are not know, then the radionuclides may be grouped and the lowest radionuclide value for the radionuclides in each group may be used in applying the formulas in paras. 404 and 414. Groups may be based on the total alpha activity	Essentially the same.

		and the total beta/gamma activity when these are known, using the lowest radionuclide values for the alpha emitters or beta/gamma emitters, respectively.	
406	71, Appendix A II	Specifies that the values in Table II should be used for individual radionuclides or for mixtures of radionuclides for which relevant data are not available.	The Tables contain different values (Part 71 is consistent with SS No. 6 while the Table in TS-R-1 has revised values). This is being considered in the Part 71 rulemaking.
407		CONTENTS LIMITS FOR PACKAGES (407-419) Specifies that the quantity of radioactive material in a package should not exceed the relevant limits specified in para 408-419.	No similar general requirement.
408		**Excepted packages** (408-410) An excepted package for radioactive material other than articles manufactured of natural uranium, depleted uranium or natural thorium should not contain activities greater than the listed provisions.	Part 71 does not apply to excepted packages of non-fissile materials (71.10 (b)). See 49 CFR 173.421 - .426.
408 (a)		where radioactive material is enclosed in or is included as a component part of an instrument or other manufactured article, such as a clock or electronic apparatus, the limits specified in columns 2 and 3 of Table III for each individual item and each package, and	Included by reference in 71.5(a) to Title 49.
408 (b)		where the radioactive material is not so enclosed in or is not included as a component of an instrument or other manufactured article, the package limits specified in column 4 of Table III.	Included by reference in 71.5(a) to Title 49.
409		Specifies that for articles manufactured of natural uranium, depleted uranium or natural thorium, an excepted package may contain any quantity of such material provided than the outer surface of the uranium or thorium is enclosed in an inactive sheath made of metal or some other substantial material.	Part 71 does not apply to excepted packages of non-fissile materials (71.10 (b)). See 49 CFR 173.426.
410	71.0 (b)	For transport by post, the total activity in each excepted package should not exceed one tenth of the relevant limit specified in Table III.	Part 71 does not apply to excepted packages of non-fissile materials (71.10 (b)). See U.S. Postal Service regulation (39 CFR 111 and Publication No. 6). The values of TS-R-1 Table III are consistent with 49 CFR 173.425.
411	71.10 (b)(2)	**Industrial packages Type 1, Type 2 and Type 3** (411-412) The radioactive contents in a single package of LSA material or in a single package of SCO shall be so restricted that the radiation level specified in para. 521 shall not be exceeded, and the activity in a single package shall also be so restricted that the activity limits for a	Consistent with 71.10 (b)(2), however, the conveyance limits are not in Part 71, they are in 49 CFR 173.427.

		conveyance specified in para. 525 shall not be exceeded.	
412		Provides activity limit for air transport of packages containing non-combustible solid LSA-II or LSA-III material.	No similar requirement.
413	71.10 (b)(1)	**Type A packages** (413-414) Provides requirements for Type A packages. Type A packages shall not contain activities greater than A1 or A2.	Similar in effect (71.10 exempts packages containing less than Type A quantities of non-fissile or fissile excepted material). See also 49 CFR 173.413.
413 (a)	71.10 (b)(1)	for special form radioactive material - A(sub1)	Similar in effect (71.10 exempts packages containing less than Type A quantities of non-fissile or fissile excepted material). See also 49 CFR 173.413.
413 (b)	71.10 (b)(1)	for all other radioactive material - A (sub2)	Similar in effect (71.10 exempts packages containing less than Type A quantities of non-fissile or fissile excepted material). See also 49 CFR 173.413.
414	71, Appendix A	Provides an equation for determining the maximum radioactive contents for a Type A package that contains a mixture of radionuclides.	See Part 71 Appendix A, IV and 49 CFR 173.431(d) which are similar. TS-R-1 also states that the total amount of both normal and special form material together in a single package must have an A-value ratio of less than 1.
415	71.12, 71.14, and 71.16	**Type B(U) and Type B(M) packages** (415-416) Provides contents limits and restrictions for Type B(U) and Type B(M) packages.	Similar to 71.12 (c)(2), 71.14 (c)(2) and 71.16 (d)(2).
415 (a)	71.12, 71.14, and 71.16	activities greater than those authorized for the package design	Similar to 71.12 (c)(2), 71.14 (c)(2) and 71.16 (d)(2).
415 (b)	71.12, 71.14, and 71.16	radionuclides different from those authorized for the package design, or	Similar to 71.12 (c)(2), 71.14 (c)(2) and 71.16 (d)(2).
415 (c)	71.12, 71.14, and 71.16	contents in a form, or a physical or chemical state different from those authorized for the package design, as specified in their certificates of approval.	Similar to 71.12 (c)(2), 71.14 (c)(2) and 71.16 (d)(2).
416		Provides requirements for Type B(U) and Type B(M) packages transported by air.	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking. These restrictions limit Type B package use for air shipments, effectively imposing Type C package requirements which are not in the US regulations.
416 (a)		for low dispersible radioactive material - as authorized for the package design as specified in the certificate of approval	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.

416 (b)		for special form radioactive material - 3000 A(sub1) or 100,000 A(sub2), whichever is the lower, or	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
416 (c)		for all other radioactive material - 3000 A(sub2)	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
417		**Type C packages** (417) Provides requirements for contents of Type C packages.	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
417 (a)		activities greater than those authorized for the package design,	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
417 (b)		radionuclides different from those authorized for the package design, or	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
417 (c)		contents in a form, or physical or chemical state different from those authorized for the package design,	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
418	71.12, 71.14 to 71.24	**Packages containing fissile material** (418) Provides requirements for the contents of packages containing fissile material.	Similar to 71.12 (c)(2), 71.14 (c)(2), 71.16 (d)(2) which are worded more generally than TS-R-1, and 71.18 - 71.24.
418 (a)		a mass of fissile material different from that authorized for the package design,	Similar to 71.12 (c)(2), 71.14 (c)(2), 71.16 (d)(2) which are worded more generally than TS-R-1, and 71.18 - 71.24.
418 (b)		any radionuclide or fissile material different from those authorized for the package design, or	Similar to 71.12 (c)(2), 71.14 (c)(2), 71.16 (d)(2) which are worded more generally than TS-R-1, and 71.18 - 71.24.
418 (c)		contents in a form or physical or chemical state, or in a spatial arrangement, different from those authorized for the package design.	Similar to 71.12 (c)(2), 71.14 (c)(2), 71.16 (d)(2) which are worded more generally than TS-R-1, and 71.18 - 71.24.
419		**Packages containing uranium hexafluoride** (419) Provides requirements for the contents of packages containing UF6.	No similar requirements. See 49 CFR 173.420.
501	71.85	**SECTION V - REQUIREMENTS AND CONTROLS FOR TRANSPORT** (501) REQUIREMENTS BEFORE THE FIRST SHIPMENT Sets requirements that must be fulfilled before the first shipment of any package.	Similar to 71.85(a) and (b) and 49 CFR 173.474.
501 (a)	71.85 (b)	If the design pressure of the containment system exceeds 35 kPa (gauge), it shall be ensured that the containment system of each package conforms to the approved design requirements relating to the capability of that system to maintain its integrity under that pressure.	Similar to TS-R-1, but with a specific test criteria.

501 (b)	71.85 (a)	For each Type B(U), Type B(M) and Type C package and for each package containing fissile material, it shall be ensured that the effectiveness of its shielding and containment and, where necessary, the heat transfer characteristics and the effectiveness of the confinement system, are within the limits applicable to or specified for the approved design.	Similar in intent to TS-R-1, does not cover Type C packages.
501 (c)	71.85 (a)	For packages containing fissile material, where, in order to comply with the requirements of para. 671, neutron poisons are specifically included as components of the package, checks shall be performed to confirm the presence and distribution of those neutron poisons.	Similar in intent to TS-R-1, does not cover Type C packages.
502	71.87	REQUIREMENTS BEFORE EACH SHIPMENT (502) Sets requirements that must be fulfilled prior to each shipment of any package.	Similar to 71.87 and 49 CFR 173.475.
502 (a)	71.87	For any package it shall be ensured that all the requirements specified in the relevant provisions of these Regulations have been satisfied.	Similar to TS-R-1 and 49 CFR 173.475.
502 (b)	71.87 (h)	It shall be ensured that lifting attachments which do not meet the requirements of para. 607 have been removed or otherwise rendered incapable of being used for lifting the package, in accordance with para. 608.	Similar to TS-R-1.
502 (c)	71.87	For each Type B(U), Type B(M) and Type C package and for each package containing fissile material, it shall be ensured that all the requirements specified in the approval certificates have been satisfied.	71.87, 71.12 (c)(2), 71.14 (c)(2) and 71.16 (d)(2) address compliance with the certificate, but does not include Type C packages.
502 (d)		Each Type B(U), Type B(M) and Type C package shall be held until equilibrium conditions have been approached closely enough to demonstrate compliance with the requirements for temperature and pressure unless an exemption from these requirements has received unilateral approval.	No similar provision. 49 CFR 173.420(a)(5) addresses equilibrium pressure in UF6 packages.
502 (e)	71.87 (c)	For each Type B and Type C package, inspections or tests are required to ensure that all closures, valves and other openings of the containment system are properly closed and sealed.	71.87(c) is similar, but does not include Type C packages.
502 (f)		For special form material, all requirements in the special form approval certificate and the relevant provisions of the regulations must be met.	No similar specific requirement applicable to special form material.
502 (g)	71.55 (c)	For packages containing fissile material, measurements of isotopic composition (if burnup credit is allowed) and tests of the	71.55(c) addresses special design features to prevent water in-leakage. No similar requirement to measure isotopic composition, but 71.83 deals

		closure of the package (if special features are used to avoid in-leakage of water) shall be performed.	with this indirectly.
5 0 2 (h)		For low dispersible material, the requirements in the approval certificate and the relevant provisions of the regulations must be met.	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
503		TRANSPORT OF OTHER GOOD (503-506) Requires that a package shall not contain any other items except such articles and documents that are necessary for the use of radioactive material. The requirements shall not preclude the transport of low specific activity material or surface contaminated objects with other items. The transport of such articles and documents in a package, or of low specific activity material or surface contaminated objects with other items may be permitted provided that there is no interaction between them and the packaging or its radioactive contents that would reduce the safety of the package.	No similar provision. 71.12 (c) requires packages to be used in compliance with their certificates (which specify allowable contents).
504		Requires that tanks and intermediate bulk containers used for the transport of radioactive material shall not be used for the storage or transport of other goods unless decontaminated below the level of 0.4 Bq/cm(super2) for beta and gamma emitters and low toxicity alpha emitters and 0.04 Bq/cm(super2) for all other alpha emitters.	No similar requirement, however, this topic is within DOT's area of responsibility.
505		Specifies that the transport of other goods with consignments being transported under exclusive use shall be permitted provided the arrangements are controlled only by the consignor and it is not prohibited by other regulations.	No similar requirement, however, this topic is within DOT's area of responsibility.
506		States that consignments shall be segregated from other dangerous goods during transport in compliance with the relevant transport regulations for dangerous goods of each of the countries through or into which the materials will be transported, and, where applicable, with the regulations of the cognizant transport organizations, as well as these Regulations.	No similar requirement, however, this topic is within DOT's area of responsibility.
507		OTHER DANGEROUS PROPERTIES OF CONTENTS (507) Sets the requirements of other dangerous properties of the package contents, such as explosiveness, flammability, pyrophoricity, chemical toxicity and corrosiveness that should be taken into account in the packing, labeling, marking, placarding, storage and transport in order to be in compliance with the transport	No similar requirement, however, this topic is within DOT's area of responsibility.

		regulations for dangerous goods of each of the countries through or into which the materials will be transported, and, where applicable, with the regulations of the cognizant transport organizations, as well as these Regulations.	
508	71.87 (i)	REQUIREMENTS AND CONTROLS FOR CONTAMINATION AND FOR LEAKING PACKAGES (508-514) The non-fixed contamination on the external surfaces of any package shall be kept as low as practicable and, under routine conditions of transport, shall not exceed set limits:	Invokes 49 CFR 173.433. The DOT limits and TS-R-1 are the same, but are stated differently.
508 (a)	71.87 (i)	4 Bq/cm2 for beta and gamma emitters and low toxicity alpha emitters, and	Invokes 49 CFR 173.433. The DOT limits and TS-R-1 are the same, but are stated differently.
5 0 8 (b)	71.87 (i)	0.4 Bq/cm2 for all other alpha emitters. These limits are applicable when averaged over any area of 300 cm2 of any part of the surface.	Invokes 49 CFR 173.433. The DOT limits and TS-R-1 are the same, but are stated differently.
509		States that the level of non-fixed contamination on the external and internal surfaces of overpacks, freight containers, tanks and intermediate bulk containers should not exceed the limits specified in para 508, except as provided in para. 514.	Title 49 addresses packages and conveyances.
510		If a package is suspected or is damaged or leaking, access to the package shall be restricted and a qualified person shall assess the extent of contamination and the resultant radiation level of the package. The scope of the assessment shall include the package, the conveyance, the adjacent loading and unloading areas, and, if necessary, all other material which has been carried in the conveyance. When necessary, additional steps for the protection of persons, property and the environment, in accordance with provisions established by the competent authority, shall be taken to overcome and minimize the consequences of such leakage or damage.	No similar requirement. Title 49 addresses this in the modal Parts 174-177.
511		Packages which are damaged or leaking radioactive contents in excess of allowable limits for normal conditions of transport may be removed to an acceptable interim location under supervision, but shall not be forwarded until repaired or reconditioned and decontaminated.	No similar provision. Title 49 addresses this with regard to radioactive material, e.g., 176.710 and hazardous materials in general.
512		A conveyance and equipment used regularly for the transport of radioactive material shall be periodically check to determine the level of contamination. The frequency of such checks shall be related to the likelihood of contamination and the extent to which radioactive material is transported.	No similar requirement. 71.87 (i) invokes 49 CFR173.443 which applies to exclusive use vehicles which transport packages that are allowed to have higher contamination levels.

513		Any contaminated conveyance, equipment or part thereof must be decontaminated if it is contaminated above the limit set in para. 508 or which shows a radiation level in excess of 5 μµSv/h. Radiation level must be in excess of 5 μµSv/h at the surface.	No similar provision. 49 CFR Parts 174, 175, 196 and 177 address conveyance contamination provisions.
514		An overpack, freight container, tank, intermediate bulk container or conveyance dedicated to the transport of radioactive material under exclusive use shall be excepted from the requirements of para 509 and 513 solely with regard to its internal surfaces and only for as long as it remains under that specific exclusive use.	No similar provision in Part 71. 49 CFR 173.443 contains a similar provision.
515		REQUIREMENTS AND CONTROLS FOR TRANSPORT OF EXCEPTED PACKAGES (515-519) Sets requirements and controls for transport of excepted packages.	Part 71exempts excepted packages containing non-fissile or fissile-excepted contents (71.10 (b)(1)).
515 (a)		Requirements specified in paras. 507, 508, 511, 516, 534-536, 549 (c), 554 and, as applicable 517-520.	Part 71exempts excepted packages containing non-fissile or fissile-excepted contents (71.10 (b)(1)). 49 CFR 173.421 is similar.
515 (b)		The requirements for excepted packages specified in para. 620.	No similar provision in Part 71. 49 CFR 173.421 is similar.
515 (c)	71.10 (b)(1)	If the excepted package contains fissile material, one of the fissile exceptions provided by para. 672 shall apply and the requirement of para. 634 shall be met.	Part 71exempts excepted packages containing non-fissile or fissile-excepted contents (71.10 (b)(1)). 49 CFR 173.421 limits excepted packages to less than 15 g U-235.
515 (d)		The requirements in paras. 579 and 580 if transported by post	Part 71exempts excepted packages containing non-fissile or fissile-excepted contents (71.10 (b)(1)). See 39 CFR 124 for postal requirements.
516		States that the radiation level at any point on the external surface of an excepted package shall not exceed 5 μµSv/h.	Part 71exempts excepted packages containing non-fissile or fissile-excepted contents (71.10 (b)(1)). 49 CFR 173.421 (a) and 173.424 contain a similar provision.
517		Sets requirements for radioactive material transport in an excepted package. States that radioactive material which is enclosed in or is included as a component part of an instrument or other manufactured article, with activity not exceeding the item and package limits specified in columns 2 and 3 of Table III (Activity Limits for Excepted Packages), may be transported in an excepted package provided that certain conditions are met.	Part 71exempts excepted packages containing non-fissile or fissile-excepted contents (71.10 (b)(1)). 49 CFR 173.424 contains similar provisions.
517 (a)		The radiation level at 10 cm from any point on the external surface of any unpackaged instrument or article is not greater than 0.1 mSv/h.	Part 71exempts excepted packages containing non-fissile or fissile-excepted contents (71.10 (b)(1)). 49 CFR 173.424 contains a similar provision.
517 (b)		Each instrument or article (except radioluminescent timepieces or devices) bears	No similar requirement, however, this topic is within DOT's area of responsibility.

		the marking "RADIOACTIVE".	
517 (c)		The active material is completely enclosed by non-active components (a device performing the sole function of containing radioactive material shall not be considered to be an instrument or manufactured article).	No similar requirement, however, this topic is within DOT's area of responsibility.
518		Radioactive material in forms other than as specified in para. 517, with an activity not exceeding the limit specified in column 4 of Table III, may be transported in an excepted package, provided that certain conditions are met.	Part 71exempts excepted packages containing non-fissile or fissile-excepted contents (71.10 (b)(1)). 49 CFR 173.421 contains similar provisions.
518 (a)		The package retains its radioactive contents under routine conditions of transport.	Part 71exempts excepted packages containing non-fissile or fissile-excepted contents (71.10 (b)(1)). 49 CFR 173.421 contains a similar provision.
5 1 8 (b)		The package bears the marking "RADIOACTIVE" on an internal surface in such a manner that a warning of the presence of radioactive material is visible on opening the package.	Part 71exempts excepted packages containing non-fissile or fissile-excepted contents (71.10 (b)(1)). 49 CFR 173.421 contains a similar provision.
519		A manufactured article in which the sole radioactive material is unirradiated natural uranium, unirradiated depleted uranium or unirradiated natural thorium may be transported as an excepted package provided that the outer surface of the uranium or thorium is enclosed in an inactive sheath made of metal or some other substantial material.	No similar provision in Part 71. 49 CFR 173.426 contains a similar provision.
520		**Additional requirements and controls for transport of empty packagings** (520) Additional requirements and controls for transport of empty packagings. Empty packages which had previously contained radioactive material may be transported as an excepted package provided that they meet certain conditions.	Part 71exempts excepted packages containing non-fissile or fissile-excepted contents (71.10 (b)(1)). 49 CFR 173.428 contains similar provisions. Exmpty certified packagings are also addressed in the appropriate package safety analysis report in the chapter on operating procedures.
520 (a)		It is in a well maintained condition and securely closed.	Part 71exempts excepted packages containing non-fissile or fissile-excepted contents (71.10 (b)(1)). 49 CFR 173.428 contains a similar requirement.
5 2 0 (b)		The outer surface of any uranium or thorium in its structure is covered with an inactive sheath made of metal or some other substantial material.	Part 71exempts excepted packages containing non-fissile or fissile-excepted contents (71.10 (b)(1)). No similar requirement in Title 49.
520 (c)		The level of internal non-fixed contamination does not exceed one hundred times the levels specified in para. 508.	Part 71exempts excepted packages containing non-fissile or fissile-excepted contents (71.10 (b)(1)). 49 CFR 173.428 contains this requirement.
5 2 0 (d)		Any labels which may have been displayed on it in conformity with para. 541 are no longer	Part 71exempts excepted packages containing non-fissile or fissile-excepted contents (71.10

		visible.	(b)(1)). 49 CFR 173.428 contains a similar requirement.
521	71.10 (b)(2)	REQUIREMENTS AND CONTROLS FOR TRANSPORT OF LSA MATERIAL AND SCO IN INDUSTRIAL PACKAGES OR UNPACKAGED (521-525) States that the quantity of LSA material of SCO in a single Industrial package Type 1, Industrial package Type 2, Industrial package Type 3, or object or collection of objects shall be so restricted that the external radiation level at 3 m from the unshielded material or object or collection of objects does not exceed 10 mSv/h.	71.10(b)(2) and 49 CFR 173.427(a)(1) are essentially the same as this requirement.
522		LSA material and SCO which is or contains fissile material shall meet the applicable requirements of para 568, 569 and 671.	Part 71 limits LSA and SCO, per se, to non-fissile and fissile excepted materials.
523		LSA material and SCO in groups LSA-I and SCO-I may be transported unpackaged under specified conditions.	71.10(b)(2) exempts LSA and 49 CFR 173.427(c) addresses LSA-I and SCO-I transported in bulk packagings, but the provisions are very different than TS-R-1.
523 (a)		All unpackaged material other than ores containing only naturally occurring radionuclides shall be transported in such a manner that under routine conditions of transport there will be no escape of radioactive contents from the conveyance nor will there be any loss of shielding.	71.10(b)(2) exempts LSA and 49 CFR 173.427(c) addresses LSA-I and SCO-I transported in bulk packagings, but the provisions are very different than TS-R-1.
5 2 3 (b)		Each conveyance shall be under exclusive use, except when only transporting SCO-I on which the contamination on the accessible and the inaccessible surfaces is not greater than ten times the applicable level specified in para. 214.	71.10(b)(2) exempts LSA and 49 CFR 173.427(c) addresses LSA-I and SCO-I transported in bulk packagings, but the provisions are very different than TS-R-1.
523 (c)		For SCO-I where it is suspected that non-fixed contamination exists on inaccessible surfaces in excess of the values specified in para. 241(a)(I), measures shall be taken to ensure that the radioactive material is not released into the conveyance.	71.10(b)(2) exempts LSA and 49 CFR 173.427(c) addresses LSA-I and SCO-I transported in bulk packagings, but the provisions are very different than TS-R-1.
524		LSA material and SCO, except as otherwise specified in para. 523, shall be packaged in accordance with Table IV (Industrial Package Requirements for LSA Material and SCO)	71.10(b)(2) exempts LSA. 49 CFR 173.427(f), Table 8, is identical this TS-R-1 provision.
525		The total activity in a single hold or compartment of an inland water craft, or in another conveyance, for carriage of LSA material or SCO in Type IP-I, Type IP-2, Type IP-3 or unpackaged, shall not exceed the limits shown in Table V.	71.10(b)(2) exempts LSA. 49 CFR 173.427(f), Table 9, is identical this TS-R-1 provision, except that the limits for inland water craft are not included.

526	71.4	Specifies that the TI for a package, overpack, or freight container, or for unpackaged LSA-I or SCO-I, shall be the number derived in accordance with the procedure in paras. 526 - 527.	The definition of the TI is different between Part 71/Title 49 and TS-R-1 (see TS-R-1 para. 243 above). The application of the TI is different as well. TS-R-1 applies the TI concept to unpackaged LSA and SCO and freight containers.
526 (a)	71.4	Determine the maximum radiation level in units of mSv/h at a distance of 1 m from the external surfaces of the package, overpack, freight container, or unpackaged LSA-I and SCO-I. The value determined is multiplied by 100 and the resulting number is the transport index. For uranium and thorium ores and their concentrates, the maximum radiation level at any point 1 m from the external surface of the load may be takes as: .4 mSv/h for ores and physical concentrates of uranium and thorium; 0.3 mSv/h for chemical concentrates of thorium; 0.02 mSv/h for chemical concentrates of uranium, other than uranium hexafluoride.	For non-fissile and fissile excepted packages Part 71 and 49 CFR 173.403 are the same as TS-R-1. The TS-R-1 provisions for ores, concentrates, etc. are not reflected in Part 71 and Title 49.
5 2 6 (b)		DETERMINATION OF TRANSPORT INDEX (TI) (526-527) For tanks, freight containers and unpackaged LSA-I and SCO-I, the value determined in step (a) shall be multiplied by the appropriate factor from Table VI (Multiplication Factors for Large Dimension Loads).	No similar requirement, however, this topic is within DOT's area of responsibility.
526 (c)	71.4	The value obtained in (a) and (b) shall be rounded up to the first decimal place (e.g. 1.13 becomes l.2) except that a value of 0.05 or less may be considered as zero.	71.4 and 49 CFR 173.403 are similar to the TS-R-1 requirement to round up the TI and 49 CFR 172.403 (footnote 2) is essentially the same as the TS-R-1 provision to consider a value of 0.05 or less to be zero.
527		The transport index for each overpack, freight container or conveyance shall be determined as either the sum of the TIs of all the packages contained, or by direct measurement of radiation level, except in the case of non-rigid overpacks for which the transport index shall be determined only as the sum of the TIs of all the packages.	Part 71 does not address this. 49 CFR 173.448(g) has similar requirements.
528		DETERMINATION OF CRITICALITY SAFETY INDEX (CSI) (528-529) The CSI for packages containing fissile material shall be obtained by dividing the number 50 by the smaller of the two values of N derived in para 681 and 682. The value of the CSI may be zero, provided that an unlimited number of packages is subcritical.	Part 71 and Title 49 do not have the Criticality Safety Index (CSI) provisions. However, the way in which the Transport Index is used for criticality control purposes is similar to the TS-R-1 CSI provisions.

529		The CSI for each consignment shall be determined as the sum of the CSIs of all the packages contained in that consignment.	Part 71 and Title 49 do not have the Criticality Safety Index (CSI) provisions. However, the limits which placed on accumulation of TI's are similar to TS-R-1.
530	71.47 (a) 71.59	LIMITS ON TRANSPORT INDEX, CRITICALITY SAFETY INDEX AND RADIATION LEVELS FOR PACKAGES AND OVERPACKS (530-532) The transport index of any package or overpack shall not exceed 10, nor shall the CSI of any package or overpack exceed 50 except for consignments under exclusive use.	71.47 (a) and (b) and 49 CFR 173.441 (a) and (b) are consistent with the TI limit. Part 71 and Title 49 do not have the Criticality Safety Index (CSI) provisions, but the limits on the accumulation of TI's for fissile packages are similar to TS-R-1.
531	71.47 (a)	Sets the maximum radiation level on external surfaces of packages or overpacks except for those: transported under exclusive use by rail and road under the conditions specified in subpara. 572(a); or, under exclusive use and special arrangement by vessel or by air under the conditions specified in paras. 574 or 578. The maximum radiation level at any point on any external surface of a package or overpack shall not exceed 2 mSv/h.	71.47 and 49 CFR (173.441 and Parts 174-177) are essentially the same as TS-R-1.
532	71.47 (b) (1)	The maximum radiation level at any point on any external surface of a package under exclusive use shall not exceed 10 mSv/h.	71.47 and 49 CFR 173.441 are essentially the same as TS-R-1.
533	71.5 (a)	CATEGORIES (533) Sets requirements for assigning packages and overpacks to either category I-WHITE, II-YELLOW, or III-YELLOW in accordance with the conditions in Table VII (Categories of Packages and Overpacks).	Part 71 does not address labeling, except by reference to Title 49 in 71.5(a). 49 CFR 172.403 is essentially the same as TS-R-1.
533 (a)	71.5 (a)	Both the transport index and the surface radiation level conditions shall be taken into account in determining which is the appropriate category. Where the TI satisfies the condition for one category, but the surface radiation level satisfies the condition for a different category, the package or overpack shall be assigned to the higher category. For this purpose, category I-WHITE shall be regarded as the lowest category.	Part 71 does not address labeling, except by reference to Title 49 in 71.5(a). 49 CFR 172.403 is essentially the same as TS-R-1.
533 (b)	71.5 (a)	The TI shall be determined following the procedures specified in paras. 526 and 527.	See paras. 526 and 527 above.
533 (c)	71.47	If the surface radiation level is greater than 2 mSv/h, the package or overpack shall be transported under exclusive use and under the provisions of paras. 572(a), 547 or 578.	71.47 and 49 CFR (173.441 and Parts 174-177) are essentially the same as TS-R-1.

5 3 3 (d)		A package transported under a special arrangement shall be assigned to category III-YELLOW.	Part 71 and title 49 do not use the concept of "special arrangement". The provisions for exemptions are the closest analogy.
533 (e)		An overpack which contains packages transported under special arrangement shall be assigned to category II-YELLOW.	Part 71 and title 49 do not use the concept of "special arrangement". The provisions for exemptions are the closest analogy.
534	71.5 (a)	MARKING, LABELLING AND PLACARDING (534-547) **Marking** (534-540) Each package shall be legibly and durably marked on the outside of the packaging with an identification of either the consignor or consignee, or both.	Part 71 does not address this marking, except by reference to Title 49 in 71.5 (a). 49 CFR 172.302 (d) is essentially the same as TS-R-1 (with some domestic exceptions).
535	71.5 (a)	Provides that the United Nations number preceded by the letters "UN", and the proper shipping name shall be legibly and durably marked on the outside of the packages that are not excepted packages. In the case of excepted packages, other than those accepted for international movement by post, only the UN number, preceded by the letters "UN", shall be required. For packages accepted for international movement by post, the requirement of para. 580 shall apply. Table VIII - Excerpts from List of United Nations Numbers, Proper Shipping Names and Descriptions, Subsidiary Risks and Their Relationship to the Schedules.	Part 71 does not address marking, except by reference to Title 49 in 71.5 (a). 49 CFR 172.302 addresses marking: UN number and proper shipping name - see 49 CFR 302 (a) [173.421 (a) provides the exception for excepted packages]; UN number on excepted packages - Part 71 and Title 49 do not contain this requirement; packages by post - see US Postal Publication No. 6 which is consistent with the IAEA requirement; Table VIII - Title 49 is not consistent with TS-R-1 which has extensively revised the proper shipping names and UN numbers.
536	71.85 (c)	Each package of gross mass exceeding 50 kg shall have its permissible gross mass legibly and durably marked on the outside of the packaging.	71.85 addresses all packages subject to Part 71, regardless of mass.. 49 CFR 172.310(a) is essentially the same as TS-R-1.
537 (a)		Each package which conforms to an Industrial package Type 1, and Industrial package Type 2, or an Industrial package Type 3 design shall be legibly and durably marked on the outside of the packaging with ““TYPE IP-1””, ““TYPE IP-2””, or ““TYPE IP	No similar requirement, however, this topic is within DOT's area of responsibility.
5 3 7 (b)	71.85	Each package which conforms to a Type A package design shall be legibly and durably marked on the outside the packaging with "TYPE A".	71.85(c) would apply to Type AF (fissile) packages. 49 CFR 172.310(b) is essentially the same as TS-R-1.
537 (c)	71.5 (a)	Each package which conforms to an Industrial package Type 2, an Industrial package Type 3 or a Type A package design shall be legibly and durably marked on the outside of the packaging with the international vehicle registration code of the country of origin of design and the name of the manufacturers, or other identification of the packaging specified by the competent	Part 71 does not address this marking, except by reference to Title 49 in 71.5(a) and indirectly by 71.85(c) for fissile package designs (since the vehicle registration code is included in the package identification number). 49 CFR 172.310(d) is consistent with TS-R-1 (except for IP-type packages). There is no requirement for marking the name of the manufacturer.

		authority.	
538	71.5 (a), 71.85(c)	Sets marking requirements for packages which conform to an approved design under paras, 805-814 or 816-817.	Part 71 addresses marking with a combination of reference to Title 49 in 71.5(a) and requirements in 71.85(c). 49 CFR 172.310 is partially consistent with TS-R-1: UF6 packages and Type C package - not consistent; B(U), B(M) and fissile packages - consistent;
538 (a)	71.5 (a), 71.85(c)	Each package shall be legibly and durably marked on the outside of the packaging with the identification mark allocated to that design by the competent authority.	71.85 requires marking the identification number assigned by NRC and 71.5(c) references Title 49. 49 CFR 172.310(d), 173.471 (b), 173.472(c) and 174.473(b) are consistent with TS-R-1.
538 (b)	71.13 (a) (b); 71.85	Each package shall be legibly and durably marked on the outside of the packaging with a serial number to uniquely identify each packaging which conforms to that design.	Part 71 is essentially the same as TS-R-1.
538 (c)	71.5 (a)	Each package shall be legibly and durably marked on the outside of the packaging with "TYPE B(U)" or "TYPE B(M) in the case of a Type B(U) or Type B(M) package design.	Part 71 does not address this requirement for marking the package type, except by reference to Title 49 in 71.5(a). 49 CFR 172.302(b) requires "TYPE B", but not "B(U)" or "B(M)".
538 (d)		Each package shall be legibly and durably marked on the outside of the packaging with "TYPE C" in the case of a Type C package design.	Part 71 and 49 CFR do not contain requirements for Type C packages.
539	71.5 (a)	Each package which conforms to a Type B(U), Type B(M) or Type C package design shall have the outside of the outermost receptacle which is resistant to the effects of fire and water plainly marked by embossing, stamping or other means resistant to the effects of fire and water with the trefoil symbol shown in Fig. I.	Part 71 does not address this marking, except by reference to Title 49 in 71.5(a). 49 CFR 172.310(c) is consistent with TS-R-1 (except for Type C packages) but resistance to the effects of fire and water are not stated.
540	71.5 (a)	LSA-1 or SCO-1 material is contained in receptacles or wrapping materials and is transported under exclusive use as permitted by para. 523, the outer surface of these receptacles or wrapping materials may bear the marking 'RADIOACTIVE LSA-' or 'RADIOACTIVE SCO-I' as appropriate.	Part 71 does not address this marking, except by reference to Title 49 in 71.5(a). 49 CFR 173.427(a)(6) is similar to TS-R-1 but is limited to domestic transport.
541	71.5 (a)	**Labelling** (541-542) Sets labeling requirements for packages, overpacks, and freight containers. Each package, overpack and freight container shall bear the labels which conform to the models in Fig. 2, Fig. 3, or Fig. 4, except as allowed under the alternative provisions of para. 546 for large freight containers and tanks. Each package, overpack, and freight container containing fissile material, other than fissile material excepted under the provisions of para. 672 shall bear labels which conform to the model in Fig. 5. Any labels which do not relate to the contents shall be removed or covered. For	Part 71 does not address labeling except by reference to Title 49 in 71.5(a). Title 49 is similar to TS-R-1: packages and overpacks - consistent except for "fissile" label requirements which are not in Title 49; freight containers - not consistent with TS-R-1; removal of irrelevant labels - 49 CFR 172.401 is similar;

		radioactive material having other dangerous properties see para. 507.	
542	71.5 (a)	Indicates requirements for fixing labels. The labels conforming to the models in Fig. 2, Fig. 3 and Fig 4 shall be affixed to two opposite sides of the outside of a package or overpack or on the outside of all four sides of a freight container or tank. The labels conforming the model in Fig. 5 shall be affixed adjacent to the labels conforming to the models in Fig 2, Fig. 3, and Fig. 4. The labels should not cover the markings specified in paras. 534-539.	Part 71 does not address labeling except by reference to Title 49 in 71.5(a). Title 49 is similar to TS-R-1: labeling of packages and overpacks are identical; Title 49 does not require labeling freight containers nor does it provide for the "fissile" label.
543	71.5 (a)	**Labelling for radioactive contents** (543) Indicates requirements for labeling radioactive contents. Each label conforming to the models in Fig. 2, Fig. 3, and Fig. 4 should be completed with the following information.	Part 71 does not address labeling except by reference to Title 49 in 71.5(a). 49 CFR 172.403(g) is similar to TS-R-1.
543 (a)	71.5 (a)	Contents - except for LSA-I material, the names(s) of the radionuclide(s) as taken from Table I, using the symbols prescribed therein. For mixtures of radionuclides, the most restrictive nuclides must be listed to the extent the space on the line permits. The group of LSA or SCO shall be shown following the name(s) of the radionuclide(s). The terms "LSA-II", "LSA-III", "SCO-I", and "SCO-II" shall be used for this purpose. For LSA-I material, the term "LSA-I" is all that is necessary; the name of the radionuclide is not necessary.	Part 71 does not address labeling except by reference to Title 49 in 71.5(a). 49 CFR 172.403(g) is similar but does not permit using "LSA" and "SCO" entries generically for the contents.
5 4 3 (b)	71.5 (a)	Activity: The maximum activity of the radioactive contents during transport expressed in units of Bq with the appropriate SI prefix (see Annex II). For fissile material, the mass of fissile material in units of grams (g), or multiples thereof, may be used in place of activity.	Part 71 does not address labeling except by reference to Title 49 in 71.5(a). 49 CFR 172.403(g) is similar but allows use of customary units (Curies) and has other requirements for Pu-238, -239, and -241.
543 (c)	71.5 (a)	For overpacks and freight containers the "contents" and "activity" entries on the label shall bear the information required in subparas. 543(a) and 543(b), respectively, totaled together for the entire contents of the overpack or freight container except that on labels for overpacks or freight containers containing different radionuclides, such entries may read "See Transport Documents".	Part 71 does not address labeling except by reference to Title 49 in 71.5(a). 49 CFR 173.448(g) is similar for overpacks.
544		**Labelling for criticality safety** (544-545) Sets requirements for criticality safety labels. Each label conforming to the model in Fig. 5 shall be completed with the CSI as stated in the certificate of approval for special arrangement	No similar requirement, however, this topic is within DOT's area of responsibility.

		of the certificate of approval for the package design issued by the competent authority.	
545		The CSI on the label shall bear the information required in para. 544 totaled together for the fissile contents of the overpack or freight container.	No similar requirement, however, this topic is within DOT's area of responsibility.
546	71.5 (a)	**Placarding** (546-547) Sets requirements for placarding. Large freight containers carrying packages other than excepted packages, and tanks shall bear four placards, which conform with the model in Fig. 6. The placards shall be affixed in a vertical orientation to each side wall and each end wall of the large freight container or tank. Any placards which do not relate to the contents shall be removed. Instead of using both labels and placards, it is permitted as an alternative to use enlarged labels only, as shown in Fig. 2, Fig. 3, Fig. 4 and Fig. 5 with dimensions of the minimum size shown in Fig. 6.	Part 71 does not address placarding except by reference to Title 49 in 71.5(a). 49 CFR 172.504 addresses placarding but only requires placarding freight containers when a YELLOW-III package or an exclusive use LSA/SCO shipment under the provisions of 173.427(a) is being transported. Placement of placards is similar. Title 49 does not permit use of enlarged labels.
547	71.5 (a)	States that where the consignment in the freight container or tank is unpackaged LSA-I or SCO-I or where an exclusive use consignment in a freight container is packaged radioactive material with a single United Nations number, the appropriate United Nations number for the consignment (see Table VIII) shall also be displaced in black digits not less than 65 mm high, either: in the lower half of the placard shown in Fig. 6, preceded by the letters "UN" and against the white background, or on the placard shown in Fig. 7.	Part 71 does not address placarding except by reference to Title 49 in 71.5(a). 49 CFR 172.504 addresses placarding and 172.332 addresses display of the UN identification number. Title 49 does not require display of the UN identification number for LSA/SCO shipments nor for packaged material in most cases.
548	71.5 (a)	CONSIGNOR'S RESPONSIBILITIES (548-561) Compliance with the requirements of paras. 520 (d) and 534-547 for marking, labelling and placarding shall be the responsibility of the consignor.	71.5(a) and 49 CFR 173.1(b) are similar to TS-R-1.
549	71.5 (a)	**Particulars of consignment** (549) Must include in the transport documents with each consignment, the following:	Part 71 does not address shipment documentation except by reference to Title 49 in 71.5(a). 49 CFR 172.200 (a) is similar to TS-R-1.
549 (a)	71.5 (a)	The proper shipping name, as specified in Table VIII.	Part 71 does not address shipment documentation except by reference to Title 49 in 71.5(a). 49 CFR 172.202(a) is identical, but it refers to the hazardous materials table in Title 49 which contains different proper shipping names than TS-R-1.
5 4 9	71.5 (a)	The United Nation Class number "7".	Part 71 does not address shipment documentation

(b)			except by reference to Title 49 in 71.5(a). 49 CFR 172.202(a) is identical, but it refers to the hazardous materials table in Title 49 which contains different proper shipping names than TS-R-1.
549 (c)	71.5 (a)	The United Nations number assigned to the material as specified in Table VIII, preceded by the letters "UN"	Part 71 does not address shipment documentation except by reference to Title 49 in 71.5(a). 49 CFR 172.202(a) is identical, but it refers to the hazardous materials table in Title 49 which contains different proper shipping names than TS-R-1.
5 4 9 (d)	71.5 (a)	The name or symbol of each radionuclide or, for mixtures of radionuclides, an appropriate general description or a list of the most restrictive nuclides.	Part 71 does not address shipment documentation except by reference to Title 49 in 71.5 (a). 49 CFR 172.203 (d) is similar to TS-R-1.
549 (e)	71.5 (a)	A description of the physical and chemical form of the material, or a notation that the material is special form radioactive material or low dispersible radioactive material. A generic chemical description is acceptable for chemical form	Part 71 does not address shipment documentation except by reference to Title 49 in 71.5(a). 49 CFR 172.203(d) is similar but does not contain provisions for low dispersible material.
549 (f)	71.5 (a)	The maximum activity of the radioactive contents during transport expressed in units of Bq with an appropriate SI prefix (see Annex II). For fissile material, the mass of fissile material in units of grams, or appropriate multiples may be used in place of activity.	Part 71 does not address shipment documentation except by reference to Title 49 in 71.5(a). 49 CFR 172.203(d) is similar but contains special requirements for Pu-238, -239, and -241 and customary units.
5 4 9 (g)	71.5 (a)	The category of the package, I.e. I-WHITE, II-YELLOW, III-YELLOW	Part 71 does not address shipment documentation except by reference to Title 49 in 71.5(a). 49 CFR 172.203(d) is similar to TS-R-1.
5 4 9 (h)	71.5 (a)	The transport index (categories II-YELLOW and III-YELLOW only)	Part 71 does not address shipment documentation except by reference to Title 49 in 71.5(a). 49 CFR 172.203(d) is similar to TS-R-1.
549 (i)	71.5 (a)	For consignments including fissile material other than consignments excepted under para. 672, the criticality safety index	No similar requirement, however, this topic is within DOT's area of responsibility.
549 (j)	71.5 (a)	The identification mark for each competent authority approval certificate	Part 71 does not address shipment documentation except by reference to Title 49 in 71.5 (a). 49 CFR 172.203 (d), 173.471, 173.472, and 173.473 are similar to TS-R-1.
5 4 9 (k)	71.5 (a)	For consignments of packages in an overpack or freight container, a detailed statement of the contents of each package within the overpack or freight container and, where appropriate, of each overpack or freight container in the consignment. If packages are to be removed from the overpack or freight container at a point of intermediate unloading, appropriate transport documents shall be made available	Part 71 does not address shipment documentation except by reference to Title 49 in 71.5 (a). 49 CFR 172.203 (d) is similar in that it requires information on each package.

549 (l)	71.5 (a)	Where a consignment is required to be shipped under exclusive use, the statement "EXCLUSIVE USE SHIPMENT"	Part 71 does not address shipment documentation except by reference to Title 49 in 71.5 (a). 49 CFR 172.203 (d) is similar to TS-R-1.
549 (m)		For LSA-II, LSA-III, SCO-I, and SCO-II, the total activity of the consignment as a multiple of A2	No similar requirement, however, this topic is within DOT's area of responsibility.
550	71.5 (a)	**Consignor's declaration** (550-553) The consignor shall include in the transport documents a declaration in the following terms or in terms having an equivalent meaning: "I hereby declare that the contents of this consignment are fully and accurately described above by the proper shipping name and are classified, packed, marked and labeled, and are in all respects in proper condition for transport by (insert mode(s) of transport involved) according to the applicable international and national governmental regulations."	Part 71 does not address shipment documentation except by reference to Title 49 in 71.5 (a). 49 CFR 172.204 is similar to TS-R-1.
551		If the intent of the declaration is already a condition of transport within a particular international convention, the consignor need not provide such a declaration for that part of the transport covered by the convention.	No similar provision, however, this topic is within DOT's area of responsibility.
552		The declaration shall be signed and dated by the consignor. Facsimile signatures are acceptable where applicable laws and regulations recognize the legal validity of facsimile signatures.	Part 71 does not address shipment documentation except by reference to Title 49 in 71.5(a). 49 CFR 172.204 is similar to TS-R-1.
553	71.5 (a)	The declaration shall be made on the same transport document which contains the particulars of consignment listed in para 549.	Part 71 does not address shipment documentation except by reference to Title 49 in 71.5(a). 49 CFR 172.204 is similar to TS-R-1.
554	71.5 (a)	**Removal or covering of labels** (554) Removal or covering of labels **-** When an empty packaging is transported as an excepted package under the provisions of para. 520, the previously displayed labels shall not be visible	Part 71 does not address shipment documentation except by reference to Title 49 in 71.5 (a). 49 CFR 172.204 is similar to TS-R-1.
555		**Information for Carriers** (555-556) The consignor shall provide in the transport documents a statement regarding actions that are required to be taken by the carrier. The statement shall be in the languages deemed necessary by the carrier or the authorities concerned, and shall include at the following points:	No similar requirement, however, this topic is within DOT's area of responsibility.
555 (a)		Supplementary requirements for loading, stowage, carriage, handling and unloading of the package, overpack or freight container including any special stowage provisions for the	No similar requirement, however, this topic is within DOT's area of responsibility.

		safe dissipation of heat (see para. 565), or a statement that no such requirements are necessary.	
5 5 5 (b)	71.5 (a)	Restrictions on the mode of transport or conveyance and any necessary routing instructions.	Part 71 does not address shipment documentation except by reference to Title 49 in 71.5 (a). 49 CFR 172.203 (d) requires entering the words "Highway Route Controlled Quantity" when appropriate.
555 (c)	71.5 (a)	Emergency arrangement appropriate to the consignment.	Part 71 does not address shipment documentation except by reference to Title 49 in 71.5 (a). 49 CFR 172.604 addresses the information which must appear on the shipping documentation such as an emergency contact number.
556		The applicable competent authority certificates need not necessarily accompany the consignment. The consignor shall make them available to the carrier(s) before loading and unloading.	No similar provision in Part 71.49 CFR 173.473 (a) has a similar requirement for foreign made packages.
557	71.5 (a)	**Notification of competent authorities** (557-560) States that before the first shipment of any package requiring competent authority approval, the consignor shall ensure that copies of each applicable competent authority certificate applying to the package design have been submitted to the competent authority of each country through or into which the consignment is to be transported. The consignor is not required to await an acknowledgement for the competent authority, nor is the competent authority required to make such acknowledgement of receipt of the certificate.	Part 71 does not address shipment documentation except by reference to Title 49 in 71.5 (a). 49 CFR 173.471 (d) and 173.472 (e) are similar to TS-R-1.
558		Sets requirements that for each of the given shipments, notification must be made to the proper authorities of each country through or into which the consignment is to be transported. The notification shall be in the hands of each competent authority prior to the commencement of the shipment, and preferably at least 7 days in advance.	No similar requirement, however, this topic is within DOT's area of responsibility.
558 (a)		Type C packages containing radioactive material with an activity greater than 3000 A(sub1) or 3000 A(sub2), as appropriate, or 1000 TBq, whichever is the lower	No similar requirement, however, this topic is within DOT's area of responsibility.
5 5 8 (b)		Type B(U) packages containing radioactive material with an activity greater than 3000 A(sub1) or 3000 A(sub2), as appropriate, or 1000 TBq, whichever is the lower	No similar requirement, however, this topic is within DOT's area of responsibility.
558 (c)		Type B(M) packages	No similar requirement, however, this topic is within DOT's area of responsibility.

5 5 8 (d)		Shipment under special arrangement	No similar requirement, however, this topic is within DOT's area of responsibility.
559		The consignment notification shall include:	No similar requirement, however, this topic is within DOT's area of responsibility.
559 (a)		Sufficient information to enable the identification of the package or packages including all applicable certificate numbers and identification marks.	No similar requirement, however, this topic is within DOT's area of responsibility.
5 5 9 (b)		Information on the date of shipment, the expected date of arrival and proposed routing.	No similar requirement, however, this topic is within DOT's area of responsibility.
559 (c)		The names of the radioactive materials or nuclides.	No similar requirement, however, this topic is within DOT's area of responsibility.
5 5 9 (d)		Descriptions of the physical and chemical forms of the radioactive material, or whether it is special form radioactive material or low dispersible radioactive material.	No similar requirement, however, this topic is within DOT's area of responsibility.
559 (e)		The maximum activity of the radioactive contents during transport expressed in units of becquerels (Bq) with an appropriate SI prefer (See Annex II). For fissile material, the mass of fissile material in units of grams (g), or multiples thereof, may be used in place of activity.	No similar requirement, however, this topic is within DOT's area of responsibility.
560		The consignor is not required to sent a separate notification if the required information has been included in the application for shipment approval; see para. 822.	No similar provision, however, this topic is within DOT's area of responsibility.
561	71.12 (c)	**Possession of Certificates and Instructions** (561) The consignor shall have in his or her possession a copy of each certificate required under Section VIII of these Regulations and a copy of the instructions with regard to the proper closing of the package and other preparations for shipment before making any shipment under the terms of the certificates.	71.12 (c) is similar to TS-R-1; TS-R-1 refers to "……proper closing …….and other preparations for shipment……" while 71.12 (c) refers to "……use and maintenance ……and to the actions to be taken before shipment."
562		TRANSPORT AND STORAGE IN TRANSIT (562-580) **Segregation During Transport and Storage in Transit** (562-563) Packages, overpacks and freight containers containing radioactive material shall be segregated during transport and during storage in transit:	Part 71 does not have a similar requirement. 49 CFR 173.447 addresses this.

562 (a)		from places occupied by persons and from undeveloped photographic film, for radiation exposure control purposes, in accordance with paras. 306 and 307, and	Part 71 does not have a similar requirement. 49 CFR Parts 173, 174, 175, 176, and 177 address segregation requirements.
5 6 2 (b)		from other dangerous goods in accordance with para. 506.	Part 71 does not have a similar requirement. 49 CFR Parts 173, 174, 175, 176, and 177 address segregation requirements.
563		Category II-YELLOW or III-YELLOW packages or overpacks shall not be carried in compartments occupied by passengers, except those exclusively reserved for couriers specially authorized to accompany such packages or overpacks.	Part 71 does not have a similar requirement. 49 CFR 173.448(c),174.700, 175.701, 176.708 and 177.843 address this, and 173.448 and 176.708 include provisions for couriers.
564	71.47 (b) (1) (ii)	**Stowage during transport and storage in transit** (564-567) Consignments shall be securely stowed during transport and storage in transit.	71.47(b)(1)(ii) and 49 CFR 173.441addresse securing high radiation level packages. 49 CFR Parts 174 - 177 address carrier requirements, including securing loads.
565		Provided that its average surface heat flux does not exceed 15 W/m2 and that the immediately surrounding cargo is not in sacks or bags, a package or overpack may be carried or stored among packaged general cargo without any special stowage provisions except as may be specifically required by the competent authority in an applicable approval certificate.	Part 71 does not have a similar requirement. 49 CFR 173.448(b) has a similar requirement.
566		States that loading of freight containers and accumulation of packages, overpacks, and freight containers shall be controlled as follows:	Included by reference in 71.5(a) to Title 49.
566 (a)		The total number of packages, overpacks and freight containers aboard a single conveyance shall be so limited that the total sum of the transport indexes aboard the conveyance does not exceed the values shown in Table IX (Limits for Freight Containers and Conveyances Not Under Exclusive Use), except under the condition of exclusive use. For consignments of LSA-I material there shall be no limit on the sum of the TIs.	Part 71 does not have a similar requirement. 49 CFR Parts 174, 175, 176 and 177 address limits on accumulation of transport indexes per conveyance.
5 6 6 (b)		Where a consignment is transported under exclusive use, there shall be no limit on the sum of the TIs aboard a single conveyance.	Part 71 does not have a similar requirement. 49 CFR Parts 174, 175, 176 and 177 address limits on accumulation of transport indexes per conveyance.
566 (c)	71.47 (b) (2) and (3)	The radiation level under routine conditions of transport shall not exceed 2 mSv/h at any point on, and 0.1 mSv/h at 2 m from, the external surface of the conveyance.	71.47(b)(2) and (3) and 49 CFR 173.441(b)(2) and (3) are essentially the same as TS-R-1.
5 6 6 (d)		The total sum of the criticality safety indexes in a freight container and aboard a conveyance shall not exceed the values shown in Table X (CSI Limits for Freight Containers and	No similar requirement since Part 71 and 49 CFR do not include the criticality index requirements. This is being considered in the DOT rulemaking.

		Conveyance Containing Fissile Material).	
567	71.47 (b)	States that any package or overpack having either a transport index greater than 10, or any consignment having a criticality safety index greater than 50, shall be transported only under exclusive use.	71.47(b) and 49 CFR 173.441(b) are essentially the same as this requirement with regard to the transport index requirement. There is no similar requirement regarding the limit on the criticality safety index since Part 71 and 49 CFR do not include any criticality safety index requirements.
568		**Segregation of packages containing fissile material during transport and storage in transi**t (568-569) The number of packages, overpacks, and freight containers containing fissile material stored in transit in any one storage area shall be so limited that the total sum of the criticality safety indexes in any group of such packages, overpacks or freight containers does not exceed 50. Groups of such packages, overpacks and freight containers shall be stored so as to maintain a spacing of at least 6 m from other groups of such packages, overpacks, or freight containers.	No similar requirement since Part 71 and Title 49 do not include the criticality safety index. However, see 173.447 and 173.459 which address limiting accumulations of transport indexes. 49 CFR Parts 174, 175, 176 and 177 address limits on accumulation of transport indexes.
569		Where the total sum of the criticality safety indexes on board a conveyance or in a freight containers exceeds 50, as permitted in Table X (CSI Limits for Freight Containers and Conveyances Containing Fissile Material), storage shall be such as to maintain a spacing of at least 6 m from other groups of packages, overpacks, or freight containers containing fissile material or other conveyances carrying radioactive material.	No similar requirement since Part 71 and Title 49 do not include the criticality safety index. However, see 173.447 and 173.459 which address limiting accumulations of transport indexes. 49 CFR Parts 174, 175, 176 and 177 address limits on accumulation of transport indexes.
570	71.5 (a)	**Additional requirements relating to transport by rail and by road** (570-573) Rail and road vehicles carrying packages, overpacks or freight containers labeled with any of the labels shown in Fig. 2, Fig. 3, Fig. 4 or Fig. 5, or carrying consignments under exclusive use, shall display the placard shown in Fig. 6 on each of the two external lateral walls in the case of rail vehicle; the two external lateral walls and the external real wall in the case of a road vehicle. In the case of vehicles which have insufficient area to allow the fixing of larger placards, the dimensions of the placard as described in Fig. 6 may be reduced to 100 mm.	Reference to 49 CFR invokes the DOT placarding requirements in 172.504. However, these requirements are SUBSTANTIALLY different than TS-R-1. Title 49 only requires placarding when transporting a Yellow-III labeled package or when transporting an exclusive use shipment of LSA/SCO in accordance with 49 CFR 173.427(a). TS-R-1 requires placarding when transporting any package which is required to bear a "Radioactive" label.
571		Requires that in certain instances, a placard displaying the appropriate UN number must be displayed. Where the consignment in or on the vehicle in	No similar requirement, however, this topic is within DOT's area of responsibility.

		unpackaged LSA-I material or SCO-I or where an exclusive use consignment is packaged radioactive material with a single UN number, the appropriate UN number shall also be displayed, in black digits not less than 65 mm high, either:	
571 (a)		In the lower half of the placard shown in Fig. 6, against the white background, or	No similar requirement, however, this topic is within DOT's area of responsibility.
5 7 1 (b)		On the placard shown in Fig. 7.	No similar requirement, however, this topic is within DOT's area of responsibility.
572	71.47	For consignments under exclusive use, the radiation level shall not exceed:	71.47 and 49 CFR 173.441 requirements are essentially the same as TS-R-1.
572 (a)	71.47	10 mSv/h at any point on the external surface of any package or overpack, and may only exceed 2 mSv/h provided that the vehicle is equipped with an enclosure which prevents the access of unauthorized persons to the interior of the enclosure, and provisions are made to secure the package or overpack so that its position within the vehicle remains fixed during routine conditions of transport, and there is no loading or unloading during the shipment.	71.47 and 49 CFR 173.441 requirements are essentially the same as TS-R-1.
5 7 2 (b)	71.47	2 mSv/h at any point on the outer surfaces of the vehicle	71.47 and 49 CFR 173.441 requirements are essentially the same as TS-R-1.
572 (c)	71.47	0.1 mSv/h at any point 2 m from the vertical planes represented by the outer lateral surfaces of the vehicle, or, if the load is transported in an open vehicle, at any point 2 m from the vertical planes projected from the outer edges of the vehicle.	71.47 and 49 CFR 173.441 requirements are essentially the same as TS-R-1.
573	71.5 (a)	Set requirements for road vehicles. States that in case of road vehicles, no persons other than the driver and assistants shall be permitted in vehicles carrying packages, overpacks or freight containers bearing category II-YELLOW or III-YELLOW labels.	Part 71 addresses this by reference to Title 49. 49 CFR 173.448(c) is similar and Parts 174, 175, 176 and 177 also address some aspects of this requirement.
574	71.5 (a)	**Additional requirements relating to transport by vessels** (547-575) Details additional requirements relating to transport by vessels. Packages or overpacks having a surface radiation level greater than 2 mSv/h, unless being carried in or on a vehicle under exclusive use in accordance with Table IX, footnote (a), shall not be transported by vessel except under special arrangement.	Part 71 addresses this by reference to Title 49. 49 CFR 176.704(f), Table III, footnote (e) includes a requirement that high radiation level packages must not be removed from their exclusive use vehicle. By implication, if the high radiation package does not remain in the exclusive use vehicle, an exemption (special arrangement equivalent) would be required.
575		States that the transport of consignments by means of a special use vessel which is dedicated to the purpose of carrying radioactive materials, shall be excepted from the requirements	No similar requirement in Part 71. 49 CFR 176.708(d) includes requirements which are consistent, but not as comprehensive as the TS-R-1 requirements.

		specified in para. 566 provided that the following conditions are met:	
575 (a)		A radiation protection programme for the shipment shall be approved by the competent authority of the flag state of the vessel and, when requested, by the competent authority at each port of call;	No similar requirement, however, this topic is within DOT's area of responsibility.
575 (b)	71.5 (a)	Stowage arrangements shall be predetermined for the whole voyage including any consignments to be loaded at ports of call en route; and	Part 71 addresses this by reference to Title 49. 49 CFR 176.708(d) includes requirements which are consistent, but not as comprehensive as the TS-R-1 requirements.
575 (c)		The loading, carriage and unloading of the consignments shall be supervised by persons qualified in the transport of radioactive material.	No similar requirement, however, this topic is within DOT's area of responsibility.
576	71.5 (a)	**Additional requirements relating to transport by air** (576-578) Type B(M) packages and consignments under exclusive use shall not be transported on passenger aircraft.	Part 71 partially addresses this by reference to Title 49. 49 CFR 175.700(d) prohibits Type B(M) packages on passenger aircraft. No similar prohibition on exclusive use consignments.
577	71.43 (h)	Vented Type B(M) packages, packages which require external cooling by an ancillary cooling system, packages subject to operational controls during transport, and packages containing liquid pyrophoric materials shall not be transported by air.	Part 71 addresses this by reference to Title 49 and 49 CFR 175.703(d) is essentially the same as TS-R-1. Part 71 prohibits packages designed to allow continuous venting during transport (71.43(h)) regardless of mode of transport.
578	71.5 (a)	Packages or overpacks having a surface radiation level greater than 2 mSv/h shall not be transported by air except by special arrangement.	Part 71 addresses this by reference to Title 49 and 49 CFR 175.703(d) is similar to TS-R-1 but requires "approval" in lieu of special arrangement.
579	71.0 (b), footnote 1	**Additional requirements relating to transport by post** (579-580) Indicates that requirements for transport by post is subject to the requirements of the national postal authorities.	Reference is made to the USPS requirements which are similar to TS-R-1.
580	71.0 (b), footnote 1	States that a consignment that conforms with the requirements of para. 515, and in which the activity of the radioactive contents does not exceed one tenth of the limits prescribed in Table III, may be accepted for international movement by post, subject in particular to the following additional requirements as prescribed by the Acts of the Universal Postal Union:	Reference is made to the USPS requirements which are similar to TS-R-1.
580 (a)	71.0 (b) footnote 1	it shall be deposed with the postal service only by consignors authorized by the national authority;	Reference is made to the USPS requirements which are similar to TS-R-1.
580 (b)	71.0 (b) footnote 1	it shall be dispatched by the quickest route, normally by air;	Reference is made to the USPS requirements which are similar to TS-R-1.
580 (c)	71.0 (b) footnote	it shall be plainly and durably marked on the outside with the words "RADIOACTIVE	Reference is made to the USPS requirements which are similar to TS-R-1.

	1	MATERIAL - QUANTITIES PERMITTED FOR MOVEMENT BY POST"; these words shall be crossed out if the packaging is returned empty;	
580 (d)	71.0 (b) footnote 1	it shall carry on the outside the name and address of the consignor with the request that the consignment be returned in the case of non-delivery; and	Reference is made to the USPS requirements which are similar to TS-R-1.
580 (e)	71.0 (b) footnote 1	the name and address of the consignor and the contents of the consignment shall be indicated on the internal packaging.	Reference is made to the USPS requirements which are similar to TS-R-1.
581		CUSTOM OPERATIONS (581) Para. 581 - Customs Operations States that customs operations involving the inspection of the radioactive contents of a package shall be carried out only in a place where adequate means of controlling radiation exposure are provided and in the presence of qualified persons. Any package opened on customs instructions shall, before being forwarded to the consignee, be restored to its original condition.	No similar requirement.
582		UNDELIVERABLE CONSIGNMENTS (582) Para 582 - Undeliverable Consignments Specifies that when a consignment is undeliverable, the consignment shall be placed in a safe location and the appropriate competent authority shall be informed as soon as possible and a request made for instruction on further action.	No similar requirement, however, this topic is within DOT's area of responsibility.
601	71.77 (b) (4)	**SECTION VI - REQUIREMENTS FOR RADIOACTIVE MATERIALS AND FOR PACKAGINGS AND PACKAGES** REQUIREMENTS FOR RADIOACTIVE MATERIALS (601-605) **Requirements for LSA-III material** (601) LSA-III material shall be a solid of such a nature that if the entire contents of a package were subject to the test specified in para. 703 the activity in the water would not exceed 0.1 A2.	71.77 (b) (4) and 49 CFR 173.468 are essential the same as TS-R-1.
602	71.4	**Requirements for special form radioactive material** (602-604) Special form radioactive material shall have at least one dimension not less than 5 mm.	71.4 and 49 CFR 173.403 contain this requirement in their definition of "special form radioactive material".

603	71.75	Sets requirements for special form radioactive material when subjected to the tests specified in paras. 704-711. Should meet the following requirements below:	71.75 and 49 CFR 173.469 are similar to TS-R-1.
603 (a)	71.75 (a) (2)	It would not break or shatter under the impact, percussion and bending tests in paras. 705, 706, 707, and 709(a) as applicable;	See also 49 CFR 173.469 (a) (2).
6 0 3 (b)	71.75 (a) (3)	It would not melt or disperse in the heat test in para. 708 or para 709(b) as applicable; and	See also 49 CFR 173.469 (a) (3).
603 (c)	71.75 (a) (4) and (5)	The activity in the water from the leaching tests specified in paras. 710 and 711 would not exceed 2 kBq; or alternatively for sealed sources, the leakage rate for the volumetric leakage assessment test specified in the International Organizations for Standardization document ISO 9978: Radiation Protection - Sealed Radioactive Sources - Leakage Test Methods" [8], would not exceed the applicable acceptance threshold acceptable to the competent authority.	See also 49 CFR 173.469 (a) (4).
604	71.4	States that when a sealed capsule constitutes part of the special form radioactive material, the capsule shall be so manufactured that it can be opened only by destroying it.	71.4 and 49 CFR 173.403 contain this requirement in their definition of "special form radioactive material".
605		**Requirements for low dispersible radioactive material** (605) States that low dispersible radioactive material shall be such that the total amount of this radioactive material in a package shall meet the following requirements:	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
605 (a)		The radiation level at 3 m from the unshielded radioactive material does not exceed 10 mSv/h;	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
6 0 5 (b)		If subjected to the tests specified in paras. 736 and 737, the airborne release in gaseous and particulate forms of up to 100 um aerodynamic equivalent diameter would not exceed 100 A (sub2). A separate specimen may be used for each test; and	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
605 (c)		If subject to the test specified in para. 703 the activity in the water would not exceed 100 A (sub2). In the application of this test, the damaging effects of the tests specified in (b) above shall be taken into accounts.	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
606		GENERAL REQUIREMENTS FOR ALL PACKAGINGS AND PACKAGES (606-616) Specifies that the package shall be so designed in relation to its mass, volume and shape that it	No similar requirement in Part 71. 49 CFR 173.410 (a) contains this requirement.

		can be easily and safely transported. In addition, the package shall be so designed that it can be properly secured in or on the conveyance during transport.	
607	71.45 (a)	Specifies that the design shall be such that any lifting attachment on the package willl not fall when used in the intended manner and that, if failure of the attachments should occur, the ability of the package to meet other requirements of these Regulations would not be impaired. The design shall take account of appropriate safety factors to cover snatch lifting.	71.45 (a) and 49 CFR 173.410 (b) are similar to TS-R-1. Part 71 and Title 49 impose a safety factor of 3. 71.45 (b) has similar requirements for tie-down devices and structural parts of a package.
608	71.45	Specifies that the package shall be so designed in relation to its mass, volume and shape that it can be easily and safely transported. In addition, the package shall be so designed that it can be properly secured in or on the conveyance during transport.	71.45 and 49 CFR 173.410 (b) are similar to TS-R-1.
609		As far as practicable, the packaging shall be so designed and finished that the external surfaces are free from protruding features and can be easily decontaminated.	No similar requirement in Part 71. 49 CFR 173.410 (c) is essentially the same as TS-R-1.
610		As far as practicable, the outer layer of the package shall be so designed as to prevent the collection and the retention of water.	No similar requirement in Part 71. 49 CFR 173.410 (d) is similar to TS-R-1.
611		States that any features added to the package at the time of transport which are not part of the package shall not reduce its safety.	No similar requirement in Part 71. 49 CFR 173.410 (e) is similar to TS-R-1.
612	71.71 (c) (5)	States that the package shall be capable of withstanding the effect of any acceleration, vibration or vibration resonance which may arise under routine conditions of transport without any deterioration in the effectiveness of the closing devices on the various receptacles or in the integrity of the package as a whole. In particular, nuts, bolts, and other securing devices shall be so designed as to prevent them from becoming lose or being released unintentionally, even after repeated use.	71.71 (c) (5) and 49 CFR 173.410 (f) contain similar requirements to TS-R-1.
613	71.43 (d)	States that the materials of the packaging and any components or structures shall be physically and chemically compatible with each other and with the radioactive contents. Account shall be taken of their behavior under irradiation.	71.43(d) and 49 CFR 173.410(g) are both similar to TS-R-1.
614	71.43 (e)	All valves through which the radioactive contents could otherwise escape shall be protected against unauthorized operation.	71.43(e) also includes a requirement for an enclosure. 49 CFR 173.410(h) is similar to TS-R-1.
615		States that the design of the package shall take into account ambient temperatures and pressures that are likely to be encountered in	49 CFR 173.24 is similar, but neither Part 71 nor Title 49 contain the "routine conditions of transport" since this is a new term in TS-R-1.

		routine conditions of transport.	
616		For radioactive material having other dangerous properties the package design shall take into account those properties; see paras. 109 and 507.	No similar requirement in Part 71. 49 CFR 173.2a and 173.423 address limited quantity radioactive materials which meet the definition of another hazard class. These requirements are more limited in scope than TS-R-1.
617	71.43 (g)	ADDITIONAL REQUIREMENTS FOR PACKAGES TRANSPORTED BY AIR (617-619) Sets requirements for packages transported by air. For packages to be transported by air, the temperature of the accessible surfaces shall not exceed 50°°C at an ambient temperature of 38°°C with no account taken for insulation.	71.43(g), while not specific to air transport, is consistent with 49 CFR 173.410(i) and TS-R-1.
618		Packages to be transported by air shall be so designed that, if they were exposed to ambient temperatures ranging from -40°°C to + 55°°C, the integrity of containment would not be impaired.	Part 71 does not have a similar provision except for plutonium transport by air (71.64 (b) ((1) (ii)). 49 CFR 174.410 (i) is essentially the same as TS-R-1.
619		Packages containing radioactive material transported by air shall have a containment system able to withstand without leakage a reduction in ambient pressure to 5 kPa.	49 CFR 173.410(i) addresses liquid contents and uses an internal pressure test which is roughly equivalent for a rigid package (95 kPa).
620	71.5 (a)	REQUIREMENTS FOR EXCEPTED PACKAGES (620) States that an excepted package shall be designed to meet the requirements specified in paras. 606-616 and in addition, the requirements of paras. 617-619 if carried by air.	Part 71 does not address excepted packages. 49 CFR has their requirements in various sections which are roughly equivalent with TS-R-1.
621	71.5 (a)	REQUIREMENTS FOR INDUSTRIAL PACKAGES (621-628) **Requirements for industrial package Type 1 (Type IP-1)** (621) Industrial package Type I (Type IP-1) shall be designed to meet the requirements specified in paras. 606-616 and 634, and, in addition, the requirements of paras. 617-619 if carried by air.	Part 71 does not address industrial packages. 49 CFR 173.411 has requirements which are similar to TS-R-1.
622	71.5 (a)	**Requirements for Industrial package Type 2 (Type IP-2)** (622) Sets requirements for industrial package Type 2 (Type IP-2) as specified in para. 621 and, in addition, if it were subject to the tests specified in paras. 722 and 723, it would prevent:	Part 71 does not address industrial packages. 49 CFR 173.411 has requirements which are similar to TS-R-1.
622 (a)	71.5 (a)	loss of dispersal of the radioactive contents	Part 71 does not address industrial packages. 49 CFR 173.411 has requirements which are similar to TS-R-1.

6 2 2 (b)	71.5 (a)	loss of shielding integrity which would result in more than a 20% increase in the radiation level at any external surface of the package.	Part 71 does not address industrial packages. 49 CFR 173.411 has requirements which are similar to TS-R-1 (20% increase in TS-R-1, "no significant increase in Title 49).
623	71.5 (a)	**Requirements for industrial packages Type 3 (Type IP-3)** (623) Sets requirements for industrial package Type 3 (Type IP-3) as specified in para. 621 and, in addition, the requirements specified in paras. 634-647.	Part 71 does not address industrial packages. 47 CFR 173.411 has requirements which are similar to TS-R-1.
624		**Alternative Requirements for Industrial Packages Types 2 and 3 (Type IP-2) and Type IP-3)** (624-628) States that packages may be used as Industrial package Type 2 (Type IP-2) provided that:	No similar provision, however, this topic is within DOT's area of responsibility.
624 (a)		They satisfy the requirements for Type IP-I specified in para. 621;	No similar requirement, however, this topic is within DOT's area of responsibility.
6 2 4 (b)		They are designed to conform to the standards prescribed in the chapter on General Recommendations on Packing of the United Nations Recommendations on the Transport of Dangerous Goods [7], or other requirements at least equivalent to those standards; and	No similar requirement, however, this topic is within DOT's area of responsibility.
624 (c)		When subject to the tests required for UN Packing Group I or II, they would prevent: loss or dispersal of the radioactive contents; and loss of shielding integrity which would result in more than a 20% increase in the radiation level at any external surface of the package.	No similar requirement, however, this topic is within DOT's area of responsibility.
625	71.5 (a)	States that tank containers may also be used as Industrial package Types 2 or 3 (Type IP-2) or (Type IP-3), as long as certain provisions are met.	Part 71 does not address industrial packages. 49 CFR 173.411 has requirements which are more restrictive than TS-R-1.
625 (a)	71.5 (a)	They satisfy the requirements for Type IP-I specified in para. 621;	Part 71 does not address industrial packages. 49 CFR 173.411 has requirements which are more restrictive than TS-R-1, specifying that only IMO101 or 102 portable tanks can be used which meet the requirements of IP-2 or IP-3.
6 2 5 (b)	71.5 (a)	They are designed to conform to the standards prescribed in the chapter on Recommendations on Multimodal Tank Transport of the UN Recommendations on the Transport of Dangerous Goods [7], or other requirements at least equivalent to those standards, and are capable of withstanding a test pressure of 265 kPa; and	Part 71 does not address industrial packages. 49 CFR 173.411 is similar, but limited to IMO 101 or 102 portable tanks.
625 (c)	71.5 (a)	They are designed so that any additional shielding which is provided shall be capable of withstanding the static and dynamic stresses resulting from handling and routine conditions	Part 71 does not address industrial packages. 49 CFR 173.411 is similar, but limited to IMO 101 or 102 portable tanks.

		of transport and of preventing a loss of shielding integrity which would result in more than a 20% increase in the radiation level at any external surface of the tank containers.	
626		Tanks, other than tank containers, may also be used as Industrial package Types 2 or 3 (Type IP-2) or (Type IP-3) for transporting LSA-I and LSA-II liquids and gases as prescribed in Table IV, provided that they conform to standards at least equivalent to those prescribed in para. 625.	No similar provision, however, this topic is within DOT's area of responsibility.
627	71.5 (a)	Sets provision by which freight containers may also be used as industrial package Types 2 or 3. Provisions are:	Part 71 does not address industrial packages. 49 CFR 173.411 has requirements which are more restrictive than TS-R-1.
627 (a)		that the radioactive contents are restricted to solid materials;	No similar requirement, however, this topic is within DOT's area of responsibility.
627 (b)	71.5 (a)	that they satisfy the requirements for Type IP-I specified in para. 621; and	Part 71 does not address industrial packages. 49 CFR 173.411 has requirements which are more restrictive than TS-R-1, requiring that the freight container meet the requirements for IP-2 or IP-3.
627 (c)	71.5 (a)	that they are designed to conform to the standards prescribed in the International Organization for Standardization document ISO 1496/1: "Series 1 Freight Containers - Specifications and Testing - Part 1: General Cargo Containers" [9] excluding dimensions and ratings. They shall be designed such that if subjected to the tests prescribed in that document and the accelerations occurring during routing conditions of transport they would prevent: loss or dispersal of the radioactive contents and loss of shielding integrity which would result in more than a 20% increase in the radiation level at any external surface of the freight containers	Part 71 does not address industrial packages. 49 CFR 173.411 has requirements that are similar.
628		Provisions by which metal intermediate bulk containers may be used as Industrial package Type 2 or 3 (Type IP-2) or (Type IP-3). Provisions are:	No similar provision, however, this topic is within DOT's area of responsibility.
628 (a)		that they satisfy the requirements for Type IP-I specified in para. 621; and	No similar requirement, however, this topic is within DOT's area of responsibility.
628 (b)		that they are designed to conform to the standards, prescribed in the chapter on Recommendation on Intermediate Bulk Containers (IBC's) of the United Nations Recommendations on the transport of Dangerous Goods [7], for Packing Group I or II, and if they were subjected to the tests prescribed in that document, but with the drop test conducted in the most damaging orientation, they would prevent: loss or dispersal of the radioactive contents and loss of	No similar requirement, however, this topic is within DOT's area of responsibility.

		shielding integrity which would result in more than a 20% increase in the radiation level at any external surface of the intermediate bulk container.	
629	71.5 (a)	REQUIREMENTS FOR PACKAGES CONTAINING URANIUM HEXAFLUORIDE (629-632) Sets requirements for packages containing uranium hexafluoride (UF6). Uranium hexafluoride shall be packaged and transported in accordance with the provision ISO 7195 [10], and the requirements of paras. 630-631. The package shall also meet the requirements prescribed elsewhere in these Regulation which pertain to the radioactive and fissile properties of the material.	Part 71 does not address UF6 directly, although it addresses fissile package design standards which are applicable to fissile UF6. 49 CFR 173.420 contains specific requirements for UF6.
630		States that each package designed to contain 0.1 kg or more of UF6 shall be designed so that it would meet the following requirements.	Part 71 does not address UF6 directly, although it addresses fissile package design standards which are applicable to fissile UF6. 49 CFR 173.420 contains specific requirements for UF6. Part 71 does not address UF6 directly.
630 (a)	71.5 (a)	Withstand without leakage and without unacceptable stress, as specified in ISO 7195 [10], the structural test as specified in para. 718 [an internal pressure test];	Part 71 does not address UF6 directly. 49 CFR 173.420 requires compliance with ANSI N14.1 which is similar.
6 3 0 (b)		Withstand without loss or dispersal of the uranium hexafluoride the test specified in para. 722 [the Type A drop test]; and	No similar requirement, however, this topic is within DOT's area of responsibility.
630 (c)		Withstand without rupture of the containment system the test specified in para. 728 [the Type B fire test].	No similar requirement, however, this topic is within DOT's area of responsibility.
631		Packages designed to contain 0.1 kg or more of UF6 shall not be provided with pressure relief devices.	No similar requirement, however, this topic is within DOT's area of responsibility.
632		Subject to the approval of the competent authority, packages designed to contain 0.1 kg or move of UF6 may be transported if:	No similar requirement.
632 (a)		the packages are designed to requirements other than those given in ISO 7195 [10] and paras. 630-631 but, notwithstanding, the requirements of paras. 630-631 are met as far as practicable;	No similar requirement.
6 3 2 (b)		the packages are designed to withstand without leakage and without unacceptable stress a test pressure less and 2.8 MPa as specified in para. 718; or	No similar requirement.
632 (c)		for packages designed to contain 9000 kg or more of uranium hexafluoride, the packages do not meet the requirement of para. 630(c).	No similar requirement.
633	71.5 (a)	REQUIREMENTS FOR TYPE A PACKAGES (633-649)	Part 71 does not address non-fissile and fissile excepted Type A packages directly but does address fissile package designs which may be

		Type A packages shall be designed to meet the requirements specified in paras. 606-616 and, in addition, the requirements of paras. 617-619 if carried by air, and of paras. 634-649.	Type A. 49 CFR 173.412 is similar to TS-R-1.
634	71.43 (a)	The smallest overall external dimension of the package shall not be less than 10 cm.	71.43(a) and 49 CFR 173.412 (b) are essentially the same as TS-R-1.
635	71.43 (b)	The outside of the package shall incorporate a feature such as a seal, which is not readily breakable and which, while intact, will be evident that it has not been opened.	71.43(b) and 49 CFR 173.412 (a) are essentially the same as TS-R-1, however, 49 CFR 173.412 (a) allows sealing of the cargo compartment of a closed transport vehicle in lieu of individual package seals.
636	71.45 (b) (3)	Any tie-down attachments on the package shall be so designed that, under normal and accident conditions of transport, the forces in those attachments shall not impair the ability of the package to meet the requirements of these Regulations.	71.45 (b) (3) and 49 CFR 173.412 (i) are essentially the same as TS-R-1.
637	71.5 (a), 71.71 (b)	The design of the package shall take into account temperatures ranging from -40°°C to +70°°C for the components of the packaging. Attention shall be given to freezing temperatures for liquids and to the potential degradation of packaging materials within the given temperature range.	71.71 (b) contains requirements for "initial conditions" as part of the normal conditions of transport, and these are similar, but differ numerically from TS-R-1. 49 CFR 173.412 (c) is similar to TS-R-1.
638	71.31 (c)	The design and manufacturing techniques shall be in accordance with national or international standards, or other requirements, acceptable to the competent authority.	National or international standards may be used but are not required by 71.31 (c). Compliance with the requirements in Part 71 and Title 49 have been deemed acceptable to the competent authority (by implication).
639	71.43 (c)	The design shall include a containment system securely closed by a positive fastening device which cannot be opened unintentionally or by a pressure which may arise within the package.	71.43 (c) and 49 CFR 173.412 (d) are essentially the same as TS-R-1.
640		Special form radioactive material may be considered as a component of the containment system.	No similar provision in Part 71. 49 CFR 173.412(d) is essentially the same as TS-R-1.
641	71.43 (c)	If the containment system forms a separate unit of the package, it shall be capable of being securely closed by a positive fastening device which is independent of any other part of the packaging.	71.43 (c) is similar to TS-R-1. 49 CFR 173.412(d) is essentially the same as TS-R-1.
642	71.43 (d)	The design of any component of the containment system shall take into account, where applicable, the radiolytic decomposition of liquids and other vulnerable materials and the generation of gas by chemical reaction and radiolysis.	71.43 (d) and 49 CFR 173.412 (e) are similar to TS-R-1.
643	71.71 (c) (3)	The containment system shall retain its radioactive contents under a reduction of ambient pressure to 60 kPa.	71.71 (c) (3) and 49 CFR 173.412(f) are more restrictive (reduced pressure to 25 kPa).
644	71.43 (e)	All valves, other than pressure relief valves, shall be provided with an enclosure to retain	71.43 (e) and 49 CFR 173.412 (g) are essentially the same as TS-R-1.

		any leakage from the valve.	
645	71.5 (a)	A radiation shield which encloses a component of the package specified as a part of the containment system shall be so designed as to prevent the unintentional release of that component from the shield. Where the radiation shield and such component within it form a separate unit, the radiation shield shall be capable or being securely closed by a positive fastening device which is independent of any other packaging structure.	49 CFR 173.412 (h) is similar to TS-R-1.
646	71.43 (f)	A package shall be so designed that if it were subjected to the tests specified in paras. 719-724 [Type A package tests], it would prevent:	71.43 (f) and 49 CFR 173.412 (j) are essentially the same as TS-R-1.
646 (a)	71.43 (f)	Loss or dispersal of the radioactive content; and	71.43(f) and 49 CFR 173.412(j) are identical to TS-R-1.
646 (b)	71.43 (f)	Loss of shielding integrity which would result in more than a 20% increase in the radiation level at any external surface of the package.	Both 71.43(f) and 49 CFR 173.412(j) limit radiation level increases to "no significant increase".
647	71.5 (a)	The design of a package intended for liquid radioactive material shall make provision for ullage to accommodate variations in the temperature of the contents, dynamic effects and filling dynamics.	49 CFR 173.412 (k) is essentially the same as TS-R-1.
648	71.5 (a)	Sets additional requirements for Type A packages designed to contain liquids.	49 CFR 173.412 (k) is essentially the same as TS-R-1.
648 (a)	71.5 (a)	Be adequate to meet the conditions specified in para. 646 if the package is subjected to the tests specified in para. 725; and	49 CFR 173.412 (k) is essentially the same as TS-R-1.
648 (b)	71.5 (a)	Either be provided with sufficient absorbent material to absorb twice the volume of the liquid contents. Such absorbent material must be suitably positioned so as to contact the liquid in the event of leakage; or be provided with a containment system composed of primary inner and secondary outer containment components designed to ensure retention of the liquid contents, within the secondary outer containment components, even if the primary inner components leak.	49 CFR 173.412 (k) is essentially the same as TS-R-1.
649	71.5 (a)	A package designed for gases shall prevent loss or dispersal of the radioactive contents if the package were subjected to the tests specified in para. 725. A Type A package designed for tritium gas or for noble gases shall be excepted from this requirement.	49 CFR 173.412 (k) is essentially the same as TS-R-1.
650	71.51	REQUIREMENTS FOR TYPE B (U) PACKAGES (650-664) Type B(U) packages shall be designed to meet the requirements specified in paras. 606-616,	49 CFR 173.413 refers to 10 CFR Part 71 for these requirements (which is similar to TS-R-1).

		the requirements of paras. 617-619 if carried by air, and of paras. 634-647, except as specified in para. 646(a), and in addition, the requirements specified in paras. 651-664.	
651	71.71 (c)	A package shall be so designed that, under the ambient conditions specified in paras. 653 and 654, heat generated within the package by the radioactive contents shall not, under normal conditions of transport, as demonstrated by the tests in paras. 719-724, adversely affect the package in such a way that it would fail to meet the applicable requirements for containment and shielding if left unattended for period of one week. Particular attention shall be paid to the effects of heat, which may:	71.71 (c) requires identical ambient heat conditions as TS-R-1 for the normal conditions of transport.
651 (a)	71.43 (d)	Alter the arrangement, the geometrical form or the physical state of the radioactive contents or, if the radioactive material is enclosed in a can or receptacle (for example, clad fuel elements), cause the can, receptacle or radioactive material to deform or melt; or	71.43 (d) establishes a broad requirement which is similar in application to the TS-R-1 requirement.
651 (b)	71.43 (d)	Lessen the efficiency of the packaging through differential thermal expansion or cracking or melting of the radiation shielding material; or	71.43 (d) establishes a broad requirement which is similar in application to the TS-R-1 requirement.
651 (c)	71.43 (d)	In combination with moisture, accelerate corrosion.	71.43 (d) establishes a broad requirement which is similar in application to the TS-R-1 requirement.
652	71.43 (g)	A package shall be so designed that the temperature of the accessible surfaces of a package shall not exceed 50°°C, unless the package is transported under exclusive use.	Similar to TS-R-1 with 85 degree limit on exclusive use shipments, but Part 71 does not tie 71.71(c) to a maximum allowable surface temperature.
653	71.71 (c)	The ambient temperature shall be assumed to be 38°°C.	71.71 (c) uses the same ambient temperature as TS-R-1.
654	71.71 (c)	The solar insulation conditions shall be assumed to be as specified in Table XI.	Essentially the same as TS-R-1.
655	71.73 (a) 71.73 (c)(4)	Requires that for a package which includes thermal protection in order to satisfy the 30 minute thermal test, the protection on the exterior of the package shall not be rendered ineffective by ripping, cutting, skidding, abrasion, or rough handling.	Part 71 requires essentially the same test sequencing prior to thermal testing, but does not include consideration of "ripping, cutting," etc.
656	71.51	Sets requirements indicating that a package shall be so designed that if it were subjected to certain tests (a and b below) it could meet given requirements.	Similar to TS-R-1.
656 (a)	71.51	The tests specified in paras. 719-724, it would restrict the loss of radioactive contents to not more than 10(super)—6 A(sub)2 per hour; and	Similar to TS-R-1, same release limit.
656 (b)	71.51	The tests specified in paras. 726, 727 (b), 728 and 729 and the tests in paras:	Similar to TS-R-1, same acceptance criteria. However, TS-R-1 specifies that only the crush

		(i) 727(c), when the package has a mass not greater than 500 kg, an overall density not greater than 1000 kg/m³³ based on the external dimensions, and the radioactive contents greater than 1000 A(sub)2 for not as special form radioactive material, or (ii) 727(a) for all other packages, it would: (i) retain sufficient shielding to ensure that the radiation level at 1 m from the surface of the package would not exceed 10 mSv/h with the maximum radioactive contents and (ii) restrict the accumulated loss of radioactive contents in a period of one week to not more than 20 A(sub)2 for krypton-85 and not more than A(sub)2 for all other radionuclides.	test need be performed on those packages subject to it. Part 71 requires that the free drop test and crush tests be performed on such packages.
657	71.61	A package for radioactive contents with activity greater than 10(super)5 A(sub)2 shall be so designed that if it were subject to the enhanced water immersion test specified in para. 730, there would be no rupture of the containment system.	71.61 applies to only irradiated nuclear fuel exceeding 10(super)6 curies, TS-R-1 applies to all radionuclides exceeding10(super)5 A(sub)2. 71.61 requires "without collapse, buckling, or inleakage of water" while TS-R-1 requires "no rupture of the containment system".
658	71.51 (c)	Compliance with the permitted activity release limits shall depend neither upon filters nor upon a mechanical cooling system.	Equivalent with TS-R-1.
659	71.4	A package shall not include a pressure relief system from the containment system which would allow the release of radioactive material to the environment under the conditions of the tests specified in paras. 719-724 and 726-729.	Essentially the same as TS-R-1.
660	71.51	A package shall be so designed that if were at the maximum normal operating pressure and it were subjected to the tests specified in para 719-724 and 726-729 the level of strain in the containment system would not adversely affect the package.	71.51 (a) (1) requires "no substantial reduction in the effectiveness of the packaging following the normal conditions of transport and 71.51 (a) (2) is similarly applied (in practice) following the hypothetical accident conditons.
661	71.4	A package shall not have a maximum normal operating pressure in excess of a gauge pressure of 700 kPa.	Essentially the same as TS-R-1.
662	71.43 (g)	The maximum temperature of any surface readily accessible during transport of a package shall not exceed 85ººC in the absence of insolation under the ambient conditions specified in para. 653 (except as required in para. 617 for a package transported by air). The package shall be carried under exclusive use if this maximum temperature exceeds 50ººC. Account may be taken of barriers or screens intended to give protection to persons without the need for the barriers or screens being subject to any test.	Similar to TS-R-1.
663		A package containing low dispersible radioactive material shall be so designed that any features added to the low dispersible	No similar requirement (LDM is introduced by TS-R-1).

		radioactive material that are not part of it, or any internal components of the packaging shall not adversely affect the performance of the low dispersible radioactive material.	
664	7 1 . 7 1 (c)	A package shall be designed for an ambient temperature range from —40ººC to +38ººC.	Essentially the same as TS-R-1.
665	7 1 . 4 , 71.41.(c)	REQUIREMENTS FOR TYPE B(M) PACKAGES (665-666) Type B(M) packages shall meet the requirements for Type B(U) packages specified in para. 650, except that for packages to be transported solely within a specified country or solely between specified countries, conditions other than those given in paras. 637, 653, 654 and 657-664 above may be assumed with the approval of the competent authorities of these countries. Notwithstanding, the requirements for Type B(U) packages specified in paras. 657-664 shall be met as far as practicable.	Type B(M) is defined in 71.4 in a more limited way than in TS-R-1. The provisions of 71.41(c) are similar to para. 665 in that it allows other conditions to be used in the design and analysis of a package, provided that approval is obtained.
666		Intermittent venting of Type B(M) packages may be permitted during transport, provided that the operational controls for venting are acceptable to the relevant competent authorities.	No similar provision.
667		REQUIREMENTS FOR TYPE C PACKAGES (667-670) Type C packages shall be designed to meet the requirements specified in para 606-619 and others paras.	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
668		A package shall be capable of meeting the assessment criteria prescribed for tests in paras. 656(b) and 660 after burial in an environment defined by a thermal conductivity of 0.33 W/(m··K) and a temperature of 38ººC in the steady state. Initial conditions for the assessment shall assume that any thermal insulation of the package remains intact, the package is at the maximum normal operating pressure and the ambient temperature is 38ººC.	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
669		A package shall be so designed that, if it were at the maximum normal operating pressure and subjected to: If subjected to (a) and (b) below where mixtures of different radionuclides are present the provisions of paras. 404-406 shall apply except that for krypton-85 an effective A(sub2)(i) value equal to 10 A(sub)2 may be used. For case (a), the assessment shall take into account the external contamination limits of para. 508.	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.

669 (a)		the tests specified in paras. 719-724, it would restrict the loss of radioactive contents to not more than 10 (sup—6) A(sub2) per hours; and	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
6 6 9 (b)		the test sequences in para. 734, it would meet the following requirements: retain sufficient shielding to ensure that the radiation level at 1 m from the surface of the package would not exceed 10 mSv/h with the maximum radioactive contents which the package is designed to contain; and restrict the accumulated loss of radioactive contents in a period of 1 week to not more than 10 A(sub2) for krypton-85 and not more than A(sub2) for all other radionuclides.	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
670		A package shall be so designed that there will be no rupture of the containment system following performance of the enhanced water immersion test specified in para. 730.	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
671	71.55	REQUIREMENTS FOR PACKAGES CONTAINING FISSILE MATERIAL (671-682) Fissile material shall be transported so as to:	Similar to TS-R-1.
671 (a)	71.55	maintain subcriticality during normal and accident conditions of transport; in particular, the following contingencies shall be considered: water leaking into or out of packages; the loss of efficiency of built-in neutron absorbers or moderators; rearrangement of the contents either within the package or as a result of loss from the package; reduction of spaces within or between packages; packages becoming immersed in water or buried in snow; and temperature changes; and	Similar to TS-R-1.
6 7 1 (b)	71.55	meet the requirements: of para. 634 for fissile material contained in packages; prescribed elsewhere in these Regulations which pertain to the radioactive properties of the material; and specified in paras. 673-682, unless excepted by para. 672.	Similar to TS-R-1.
672	71.53	**Exceptions from the requirements for packages containing fissile material** (672) Fissile material meeting of the of the provision of this para is excepted from the requirement to be transported in packages that comply with paras. 673-682 as well as the other requirements of these Regulations that apply to fissile material. Only one type of exception is allowed per consignment.	The provisions of 71.53 are similar to TS-R-1. The fissile exceptions in 49 CFR 173.453 are different than both of those.
672 (a)	71.53 (a)	A mass limit per consignment such that: Refer to equation and Table XII - Consignment	Essentially the same as TS-R-1.

		Mass Limits for Exception from the Requirements for Packages Containing Fissile Material.	
6 7 2 (b)	71.53 (b)	Uranium enriched in uranium-235 to a maximum of 1% by mass, and with a total plutonium and uranium-233 content not exceeding 1% of the mass of uranium-235, provided that the fissile material is distributed essentially homogeneously through the material. In addition, if uranium-235 is present in metallic, oxide, or carbide forms, it shall form a lattice arrangement.	Similar to TS-R-1.
672 (c)	71.53 (c)	Liquid solutions of uranyl nitrate enriched in uranium-235 to a maximum of 2% by mass, with a total plutonium and uranium-233 content not exceeding 0.002% of the mass of uranium, and with a minimum nitrogen to uranium atomic ration (N/U) of 2.	Similar to TS-R-1.
6 7 2 (d)	71.53 (d)	Packages containing, individually, a total plutonium mass not more than 1 kg, of which not more than 20% by mass may consist of plutonium-239, plutonium-241 or any combination of those radionuclides.	Similar to TS-R-1.
673	71.83	**Content specification for assessments of packages containing fissile material** (673-674) Contains requirements for fissile material for which the chemical or physical form, isotopic composition, mass or concentration, moderation ratio or density, or geographic configuration is not known.	Similar to TS-R-1.
674	71.83	For irradiated nuclear fuel the assessment of paras. 677-682 shall be based on an isotopic composition demonstrated to provide (a) the maximum neutron multiplication during the irradiation history, or (b) a conservative estimate of the neutron multiplication for the package assessments. After irradiation but prior to shipment, a measurement shall be performed to confirm the conservatism of the isotopic composition.	71.83 requires that for unknown properties of fissile material, credible values shall be used that cause maximum neutron mutliplication. There are no provisions similar to 674 (b) nor a requirement for measurement prior to shipment.
675	71.55 (d) (4)	**Geometry and temperature requirements** (675-676) The packaging, after being subject to the tests specified in paras. 719-724, must prevent the entry of a 10 cm cube.	Essentially the same as TS-R-1.
676	71.71 (c)	A package for fissile material shall be designed for an ambient temp. range of -40 degrees C to +38 degrees.	Part 71 applies this requirement to all packages.
677	71.55 (b) and (c)	**Assessment of an individual package in isolation** (677-680)	Similar to TS-R-1.

		For a package in isolation, it shall be assumed that water can leak into or out of all void spaces of the package, including those within the containment system. However, if the design incorporates, special features to prevent such leakage of water into or out of certain void spaces, even as a result of error, absence of leakage may be assumed in respect of those void spaces. Special features shall include the following:.	
677 (a)	71.55 (c)	Multiple high standard water barriers, each of which would remain watertight if the package were subject to the tests prescribed in para. 682(b), a high degree of quality control in the manufacture, maintenance and repair of packagings and tests to demonstrate the closure of each package before each shipment; of	Similar to TS-R-1.
6 7 7 (b)	71.55 (c)	For packages containing UF6 only: packages where there is no physical contact between the valve and any other component of the packaging other that at its original point of attachment and where, in addition the valves remain leaktight; and 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.	Part 71 does not contain specific requirements for UF6, but 71.55(c) has been applied with respect to moderator exclusion.
678	71.55 (e)	It shall be assumed that the confinement system shall be closely reflected by at least 20 cm of water or such greater reflection as may additionally be provided by the surrounding material of the packaging. However, when it can be demonstrated that the confinement system remains within the packaging following the tests prescribed in para. 682(b), close reflection of the package by at least 20 cm of water may be assumed in para 679(c).	71.55(e) has a similar requirement requiring full reflection on all sides, but applies it under the hypothetical accident condition tests (damaged condition). TS-R-1 applies this to both normal and accident conditions.
679	71.55	The package shall be subcritical under the conditions of paras. 677 and 678 with the package conditions that result in the maximum neutron multiplication consistent with: (a) routine conditions of transport (incident free); (b) the tests specified in para. 681(b); (c), the test specified in para. 682(b)	Part 71 has requirements for both the normal and hypothetical accident conditions of transport which are similar to TS-R-1.
680		For packages to be transported by air: (1) the package shall be subcritical under conditions consistent with the tests prescribed in para. 734 assuming reflection by at least 20 cm of water but no water inleakage; and (b) allowance shall not be made for special features of para. 677 unless, following the tests specified in para. 734 and, subsequently, para. 733, leakage of water into or out of the void spaces is prevented.	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
681	71.59 (a)	**Assessment of package arrays under normal**	Similar to TS-R-1.

		conditions of transport (681) A number ““N”” shall be derived, such that five time ““N”” shall be subcritical for the arrangement and package conditions that provide the maximum neutron multiplication consistent with the following:	
681 (a)	71.59	There shall not be anything between the packages, and the package arrangement shall be reflected on all sides by at least 20 cm of water, and	71.59 is similar to TS-R-1 except for the words requiring "full reflection" on all sides.
681 (b)	71.59	The state of the packages shall be their assessed or demonstrated condition if they had been subjected to the test specified in paras. 719-724.	Similar by inference. Since the normal condition tests must not result in a reduction of the performance of the package.
682	71.59 (a)	**Assessment of package arrays under accident conditions of transport** A number "N" shall be derived, such that two times "N" shall be subcritical for the arrangement and package conditions that provide the maximum neutron multiplication consistent with the following:	Similar to TS-R-1.
682 (a)	71.55 (e), 71.59 (a)	Hydrogenous moderation between packages, and the package arrangement reflected on all sides by at least 20 cm of water; and	Similar to TS-R-1.
682 (b)	71.55 (e), 71.59 (a), 71.73	The tests specified in paras. 719-724 followed by whichever of the following if the more limiting: the tests specified in para. 727(b) and, either para. 727(c) for packages having a mass not greater than 500 kg and an overall density not greater than 1000 kg/m³³ based on the external dimensions, or para. 727(a) for all other packages; followed by the test specified in para. 728 and completed by the tests specified in paras. 731-733; or the test specified in para. 729; and	Similar to TS-R-1.
682 (c)	71.59 (a)(2)	Where any part of the fissile material escapes from the containment system following the tests specified in para. 682(b), it shall be assumed that fissile material escapes from each package in the array and all of the fissile material shall be arranged in the configuration and moderation that results in the maximum neutron multiplication with close reflection by at least 20 cm of water.	71.59(a)(2) is similar in that damaged packages must be assessed with optimum interspersed hydrogenous moderation.
701	71.41 (a)	**SECTION VII - TEST PROCEDURES** DEMONSTRATION OF COMPLIANCE (701-702) Demonstration of compliance with the performance standards required in Section VI	71.41(a) and 49 CFR 173.461 both address this topic.

		shall be accomplished by any of the methods listed below of a combination thereof.	
701 (a)	71.41 (a)	Performance of tests with specimens representing LSA-III material, or special form radioactive material, or low dispersible radioactive material, or with prototypes or samples of the packagings, where the contents of the specimen or the packaging for the tests shall stimulate as closely as practicable the expected range of radioactive contents and the specimen or packaging to be tested shall be prepared as presented for transport	71.41(a) is a similar requirement to TS-R-1, but more limited in scope (does not cover LSA III or low dispersible material). 49 CFR 173.461(a)(1) is a similar requirements to TS-R-1.
701 (b)	71.41 (a)	Reference to previous satisfactory demonstrations of a sufficiently similar nature.	71.41(a) allows "another method of demonstration acceptable to the Commission", which is a similar requirement to TS-R-1. 49 CFR 173.461(a)(2) is essentially the same as TS-R-1.
701 (c)	71.41 (a)	Performance of tests with models of appropriate scale incorporate those features which are significant with respect to the item under investigation when engineering experience has shown results of such tests to be suitable for design purposes. When a scale model is used, the need for adjusting certain test parameters, such as penetrator diameter or compressive load, shall be taken into account.	71.41(a) is a similar requirement to TS-R-1 and 49 CFR 173.461(a)(3) is essentially the same as TS-R-1.
701 (d)	71.41 (a)	Calculation, or reasoned argument, when the calculation procedures and parameters are generally agreed to be reliable or conservative.	71.41(a) is similar to TS-R-1 (since calculations are routinely used to demonstrate compliance) and 49 CFR 173.461(a)(4) is essentially the same as TS-R-1.
702	71.41 (a)	Appropriate methods of assessment shall be used to ensure that the requirements of this sections have been fulfilled in compliance with the performance and acceptance standards prescribed in Section VI.	71.41(a) and 49 CFR 173.461(a) are consistent with TS-R-1 in that an assessment must be performed to demonstrate compliance.
703	71.77	TEST FOR LSA-III MATERIAL (703) A solid material sample representing the entire contents of the package shall be immersed for 7 days in water at ambient temperature. The volume of water to be used in the test shall be sufficient to ensure that at the end of the 7 day test period the free volume of he unabsorbed and unreacted water remaining shall be at least 10% of the volume of the solid test sample itself. The water shall have an initial pH of 6-8 and a maximum conductivity of 1 mS/m at 20ººC. The total activity of the free volume of water shall be measured following the 7 day immersion of the test sample.	71.77 and 49 CFR 173.468 are essentially the same as TS-R-1.
704	71.75 (a)	TESTS FOR SPECIAL FORM RADIOACTIVE MATERIAL (704-711)	71.75(a) and 49 CFR 173.469 contain similar requirements to TS-R-1.

		General (704) Specimens that comprise or simulate special form radioactive material shall be subjected to the impact test, the percussion test, the bending test, and the heat test specified in paras. 705-709. A different specimen may be used for each of the tests. Following each test, a leaching assessment or volumetric leakage test shall be performed on the specimen by a method no less sensitive than the methods given in para. 710 for indispersible solid material or para. 711 for encapsulated material.	
705	71.75 (b) (1)	**Test methods** (705-709) Impact test: The specimen shall drop onto the target from a height of 9 m. The target shall be as defined in para. 717.	71.75 (b) (1) is a similar requirement to TS-R-1 but also require that the specimen strike the target "in the orientation expected to result in maximum damage." 49 CFR 173.469 (b) (1) is essentially the same as TS-R-1.
706	71.75 (b) (2)	Percussion test: The specimen shall be placed on a sheet of lead which is supported by a smooth solid surface and struck by the flat face of a mild steel bar so as to cause an impact equivalent to that resulting from a free drop of 1.4 kg through 1 m. The lower part of the bar shall be 25 mm in diameter with the edges rounded off to a radium of (3.0 ±± 0.3) mm. The lead, of hardness number 3.5 to 4.5 on the Vickers scale and not more than 25 mm thick, shall cover an area greater than that covered by the specimen. A fresh surface of lead shall be used for each impact. The bar shall strike the specimen so as to cause maximum damage.	71.75 (b) (2) and 49 CFR 173.469 (b) (2) contain similar requirements to TS-R-1. The only difference is that the lead target in TS-R-1 must be "not more than 25 mm thick" and in Part 71 and Title 49 the lead target must be "25 mm or greater".
707	71.75 (b) (3)	Bending test: The test shall apply only to long, slender sources with both a minimum length of 10 cm and a length to minimum width ratio of not less than 10. The specimen shall be rigidly clamped in a horizontal position so that one half of its length protrudes from the face of the clamp. The orientation of the specimen shall be such that the specimen will suffer maximum damage when its free end is struck by the flat face of a steel bar. The bar shall strike the specimen so as to cause an impact equivalent to that resulting from a free vertical drop of 1.4 kg through 1 m. The lower part of the bar shall be 25 mm in diameter with the edges rounded off to a radius of (3.0 ±± 0.3)mm.	71.75 (b) (3) and 49 CFR 173.469(b) (3) are essentially the same as TS-R-1.
708	71.75 (b) (4)	Heat test: The specimen shall be heated in air to a temperature of 800^{oo}C and held at that temperature for a period 10 minutes and shall then be allowed to cool.	71.75 and 49 CFR 173.469 (b) (4) are essentially the same as TS-R-1.

709	71.75 (d)	Specimens that comprise or simulate radioactive material enclosed in a sealed capsule may be excepted from:	71.75 (d) and 49 CFR 173.469 (d) contain similar requirements to TS-R-1.
709 (a)	71.75 (d) (1)	The test prescribed in paras. 705 and 706 provided the mass of the special form radioactive material is less than 200 g and they are alternatively subjected to the Class 4 impact test prescribed in the International Organization for Standardization document ISO 2919: "Sealed Radioactive Sources - Classification" [11], and	71.75 (d) (1) and 49 CFR 173.469 (d) (1) contain similar requirements to TS-R-1, but do not contain the limitation that the mass of the special form must be less than 200 g.
7 0 9 (b)	71.75 (d) (2)	The test prescribed in para. 708 provided they are alternatively subjected to the Class 6 temperatures test specified in ISO 2919: "Sealed Radioactive Sources - Classification" [11].	71.75 (d) (2) and 49 CFR 173.469 (d) (2) are essentially the same as TS-R-1.
710	71.75 (c)	**Leaching and volumetric leakage assessment methods** (710-711) For specimens which comprise or simulate indispersible solid material, a leaching assessment shall be performed as follows:	71.75(c) and 49 CFR 173.469(c) contain similar requirements to TS-R-1.
710 (a)	71.75 (c) (1) (i)	The specimen shall be immersed for 7 days in water at ambient temperature. The volume of water to be used in the test shall be sufficient to ensure that at the end of the 7 day test period the free volume of the unabsorbed and unreacted water remaining shall be at least 10% of the volume of the solid test sample itself. The water shall have an initial pH of 6-8 and a maximum conductivity of 1 mS/m at 20ººC.	71.75(c)(1)(i) and 49 CFR 173.469(c)(1)(i) contain similar requirements to TS-R-1. They do not require that the volume of free water at the end of the test be at least 10% of the volume of the test sample.
7 1 0 (b)	71.75 (c) (1) (ii)	The water with specimen shall then be heated to a temperature of (50 ±± 5)°°C and maintained at this temperature for 4 hours.	71.75(c)(1)(ii) and 49 CFR 173.469(c)(1)(ii) are essentially the same as TS-R-1.
710 (c)	71.75 (c) (1) (iii)	The activity of the water shall then be determined.	71.75(c)(1)(iii) and 49 CFR 173.469(c)(1)(iii) are essentially the same as TS-R-1.
7 1 0 (d)	71.75 (c) (1) (iv)	The specimen shall then be kept for a least 7 days in still air at not less than 30°°C and relative humidity not less than 90%.	71.75(c)(1)(iv) and 49 CFR 173.469(c)(1)(iv) are essentially the same as TS-R-1.
710 (e)	71.75 (c) (1) (v)	The specimen shall then be immersed in water of the same specification as in (a) above and the water with the specimen heated to (50 ±± 5)°°C and maintained at this temperature for 4 hours.	71.75(c)(1)(v) and 49 CFR 173.469(c)(1)(v) are essentially the same as TS-R-1.
710 (f)	71.75 (c) (1) (vi)	The activity of the water shall then be determined.	71.75(c)(1)(vi) and 49 CFR 173.469(c)(1)(vi) are essentially the same as TS-R-1, with the addition of the acceptance criteria for the activity of the water (which TS-R-1 places in a different para.)..
711	71.75 (c)	For specimens which comprise of simulate	71.75(c)(2) and 49 CFR 173.469(c)(2) contain

	(2)	radioactive material enclosed in a sealed capsule, either a leaching assessment or a volumetric leakage assessment shall be performed as follows:	similar requirements to TS-R-1.
711 (a)	71.75 (c)(2)	Identifies the steps of the leaching assessment.	71.75(c)(2) and 49 CFR 173.469(c)(2) contain similar requirements to TS-R-1. Part 71 and Title 49 do not contain the TS-R-1 requirement that the 7 day storage period be in air with relative humidity of 90% or greater.
711 (b)	71.75 (a)(4)	The alternative volumetric leakage assessment shall comprise any of these tests prescribed in the ISO 9978 [8], which are acceptable to the competent authority.	71.75(a)(4) and 49 CFR 173.469(a)(4)(i) contain similar requirements to TS-R-1. The cited ISO standards are different (the TS-R-1 citation is more recent) and the acceptance criteria are specified in different unit.
712		TESTS FOR LOW DISPERSIBLE RADIOACTIVE MATERIAL (712) A specimen that comprises or simulates low dispersible radioactive material shall be subjected to the enhanced thermal test specified in para. 736 and the impact test specified in para. 737. A different specimen may be used for each of the tests. Following each test, the specimen shall be subjected to the leach test specified in para. 703. After each test it shall be determined if the applicable requirements of para. 605 have been met.	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
713		TESTS FOR PACKAGES (713-737) **Preparation of a specimen for testing** (713-715) All specimens shall be inspected before testing in order to identify and record faults or damage including the following:	No similar requirement in Part 71. 49 CFR 173.462 is essentially the same as TS-R-1.
713 (a)		divergence from the design;	No similar requirement in Part 71. 49 CFR 173.462 is essentially the same as TS-R-1.
713 (b)		defects in manufacture;	No similar requirement in Part 71. 49 CFR 173.462 refers to "defects in construction".
713 (c)		corrosion or other deterioration; and	No similar requirement in Part 71. 49 CFR 173.462 is identical to TS-R-1.
713 (d)		distortion of features.	No similar requirement in Part 71. 49 CFR 173.462 is identical to TS-R-1.
714	71.33 (a)(4)	The containment system of the package shall be clearly specified.	71.33(a)(4) and 49 CFR 173.462(c) are essentially the same as TS-R-1.
715	71.33	The external features of the specimen shall be clearly identified so that reference may be made simply and clearly to any part of such specimen.	71.33 is similar to TS-R-1 and 49 CFR 173.462(d) is essentially the same as TS-R-1.
716	71.35	**Testing the integrity of the containment**	71.35 requires that an application for package

		system and shielding and assessing critical safety (716) After each of the applicable tests specified in paras. 718-737:	approval include demonstration that all applicable requirements are met.
716 (a)	71.35	Faults and damage shall be identified and recorded:	71.35 requires that an application for package approval include demonstration that all applicable requirements are met.
7 1 6 (b)	71.35	It shall be determined whether the integrity of the containment system and shielding has been retained to the extent required in Section VI for the package under test; and	71.35 requires that an application for package approval include demonstration that all applicable requirements are met.
716 (c)	71.35	For packages containing fissile material, it shall be determined whether the assumptions and conditions used in the assessments required by paras. 671-682 for one or more packages are valid.	71.35 requires that an application for package approval include demonstration that all applicable requirements are met.
717	71.73 (c) (1)	**Target for drop tests** (717) The target for the drop tests specified in paras. 705, 722, 725(a), 727, 735 and 737 shall be a flat, horizontal surface of such a character that any increase in its resistance to displacement or deformation upon impact by the specimen would not significantly increase the damage to the specimen.	71.73 (c) (1) is similar to TS-R-1 and 49 CFR 173.465 (c) (5) is essential the same as TS-R-1.
718		**Test for packagings designed to contain uranium hexafluoride** (718) Provides requirements for testing and approval of packagings designed to contain UF6. Specimens that comprise or simulate packagings designed to contain 0.1 kg or more of UF6 shall be tested hydraulically at an internal pressure of at least 1.4 MPa but, when the test pressure is less than 2.8 MPa, the design shall require multilateral approval. For retesting packagings, any other equivalent nondestructive testing may be applied subject to multilateral approval.	No similar requirements in Part 71. 49 CFR 173.420 requires compliance with ANSI N14.1 which specifies internal test pressures for uranium hexafluoride packages which are similar to TS-R-1.
719	71.71	**Test for demonstrating ability to withstand normal condition of transport** (719-724) The tests are: the water spray test, the free drop test, the stacking test, and the penetration test. Specimens of the package shall be subjected to the free drop test, the stacking test and the penetration test, preceded in each case by the water spray test. One specimen may be used for all the tests, provided that the requirements of para. 720 are fulfilled.	71.71 contains similar requirements to TS-R-1 and 49 CFR 173.465 is essentially the same as TS-R-1.
720	71.71 (c) (7)	The time interval between the conclusion of the water spray test and the succeeding test shall be	71.71 (c) (7) requires a period of time between 1.5 and 2.5 hours between the water spray and

		such that the water has soaked in to the maximum extent, without appreciable drying of the exterior of the specimen. In the absence of any evidence to the contrary, this interval shall be taken to be two hours if the water spray is applied from four direction simultaneously. No time interval shall elapse if the water spray is applied from each of the four directions consecutively.	drop tests. 49 CFR 173.465 (b) is essentially the same as TS-R-1.
721	71.71 (c) (6)	Water spray test: The specimen shall be subjected to a water spray test that simulates exposure to rainfall of approximately 5 cm per hour for at least one hour.	71.71 (c) (6) and 49 CFR 173.465 (b) are essentially the same as TS-R-1.
722	71.71 (c) (7)	Free drop test: The specimen shall drop onto the target so as to suffer maximum damage in respect of the safety features to be used.	71.71 (c) (7) and 49 CFR 173.465 (c) are essentially the same as TS-R-1.
722 (a)	71.71 (c) (7)	The height of drop measured from the lowest point of the specimen to the upper surface of the target shall be not less than the distance specified in Table XIII - Free Drop Distance for Testing Packages to Normal Conditions of Transport.	71.71 (c) (7) and 49 CFR 173.465 (c) are essentially the same as TS-R-1, except that for packages with a mass of exactly 15,000 kg, Part 71 and Title 49 require a drop of 0.6 m and TS-R-1 requires a drop of 0.3 m.
722 (b)	71.71 (c) (8)	For rectangular fiberboard or wood packages not exceeding a mass of 50 kg, a separate specimen shall be subjected to a free drop onto each corner from a height of 0.3.	71.71 (c) (8) and 49 CFR 173.465 (c) (3) are essentially the same as TS-R-1.
722 (c)	71.71 (c) (8)	For cylindrical fiberboard packages not exceeding a mass of 100 kg, a separate specimen shall be subjected to a free drop onto each of the quarters of each rim from a height of 0.3.	71.71 (c) (8) extends the test requirement to wooden cylindrical packages and fissile materials packages. 49 CFR 173.465 (c) (3) is essentially the same as TS-R-1, but 49 CFR 173.465 (c) (2) imposes additional drop tests for fissile materials packages.
723	71.71 (c) (9)	Stacking test: Unless the shape of the packaging effectively prevents stacking, the specimen shall be subjected, for a period 24 h, to a compressive load equal to the greater of the following:	71.71 (c) (9) is similar to TS-R-1 but limits the test to packages weighing 5,000 kg or less. 49 CFR 173.465 (d) is similar to TS-R-1. Neither Part 71 nor Title 49 contains the TS-R-1 provision excepting packages whose shape effectively prevents stacking.
723 (a)	71.71 (c) (9)	the equivalent of 5 times the mass of the actual package;	71.71 (c) (9) and 49 CFR 173.465 (d) (1) (i) are essentially the same as TS-R-1.
723 (b)	71.71 (c) (9)	the equivalent of 13 kPa multiplied by the vertically projected areas of the package. The load shall be applied uniformly to two opposite sides of the specimen, on of which shall be the base on which the package would typically test.	71.71 (c) (9) and 49 CFR 173.465 (d) (1) (ii) and (d) (2) are essentially the same as TS-R-1.
724		Penetration test: The specimen shall be placed on a rigid, flat, horizontal surface which will not move significantly while the test is being carried out.	No similar requirement in Part 71. 49 CFR 173.465 (e) is essentially the same as TS-R-1.
724 (a)	71.71 (c) (10)	A bar of 3.2 cm in diameter with a hemispherical end and a mass of 6 kg shall be dropped and directed to fall, with its	71.71 (c) (10) is similar to TS-R-1, but does not contain the requirements relating to striking the containment system nor concerning bar

		longitudinal axis vertical, onto the center of the weakest part of the specimen, so that, if it penetrates sufficiently far, it will hit the containment system. The bar shall not be significantly deformed by the test performance.	deformation. 49 CFR 173.465 (e) (1) is essentially the same as TS-R-1.
7 2 4 (b)	71.71 (c) (10)	The height of drop of the bar measured from its lower end to the intended point of impact on the upper surface of the specimen shall be 1 m.	71.71 (c) (10) and.49 CFR 173.465 (e) (2) are essentially the same as TS-R-1.
725		**Additional tests for type A packages designed for liquids and gases** (725) A specimen or separate specimens shall be subjected to each of the following tests unless it can be demonstrated that one test is more severe for the specimen in question than the other, in which case one specimen shall be subjected to the more severe test.	Part 71 does not address liquids in Type A packages. 49 CFR 173.466 is similar, but it requires that both tests be taken into account.
725 (a)	71.5 (a)	Free drop test: The specimen shall drop onto the target so as to suffer the maximum damage in respect of containment. The height of the drop measured from the lowest part of the specimen to the upper surface of the target shall be 9 m. The target shall be as defined in para. 717.	Part 71 does not address liquids in Type A packages. 49 CFR 173.466 is essentially the same as TS-R-1.
7 2 5 (b)	71.5 (a)	Penetration test: The specimen shall be subjected to the test specified in para. 724 except that the height of drop shall be increased to 1.7 m from the 1 m specified in para. 724(b).	Part 71 does not address liquids in Type A packages. 49 CFR 173.466 is essentially the same as TS-R-1.
726	71.73 (a)	**Tests for demonstrating ability to withstand accident conditions of transport** (726-729) The specimen shall be subject to the cumulative effects of the tests specified in para. 727 and para. 728, in that order. Following these tests, either this specimen or a separate specimen shall be subjected to the effect(s) of the water immersion test(s) as specified in para. 729 and, if applicable, para 730.	71.73 (a) is similar to TS-R-1, however, the deep water immersion test in (para. 730) is limited in Part 71 to just irradiated nuclear fuel packages (see TS-R-1 para. 657 above).
727	71.73 (c) (1), (2) and (3)	Mechanical Test: The mechanical test consists of three difference drop tests. Each specimen shall be subjected to the applicable drops as specified in para. 656 or para. 682. The order in which the specimen is subjected to the drops shall be such that, on the completion of the mechanical test, the specimen shall have suffered such damage as will lead to the maximum damage in the thermal test which follows.	71.73 (c) (1), (2) and (3) are similar to TS-R-1. Part 71 requires that the free drop test be performed before the crush and puncture test, rather than in the most damaging order.
727 (a)	71.73 (c) (1)	For drop I, the specimen shall drop onto the target so as to suffer the maximum damage, and the height of the drop measured from the lowest point of the specimen to the upper surface of the	Essentially the same as TS-R-1.

		target shall be 9 m. The target shall be as defined in para. 717.	
7 2 7 (b)	71.73 (c) (3)	For drop II, the specimen shall drop so as to suffer the maximum damage onto a bar rigidly mounted perpendicularly on the target. The height of the drop measured from the intended point of impact of the specimen to the upper surface of the shall be 1 m. The bar shall be of solid mild steel of circular section, (15.0 ±± 0.5) cm in diameter and 20 cm long unless a longer bar would cause greater damage. The upper end of the bar shall be flat and horizontal with its edges rounded off to a radius of not more than 6 mm. The target on which the bar is mounted shall be as described in para. 717.	Essentially the same as TS-R-1.
727 (c)	71.73 (c) (2)	For drop III, the specimen shall be subjected to a dynamic crush test by positioning, the specimen on the target so as to suffer maximum damage by the drop of a 500 kg mass from 9 m onto the specimen. The mass shall consist of a solid mild steel plate 1 m and shall fall in a horizontal attitude. The height of the drop shall be measured from the underside of the plate to the highest point of the specimen. The target on which the specimen rests shall be as defined in para. 717.	Essentially the same test as TS-R-1, however, in Part 71 it is applied in addition to drop I (free drop).
728	71.73 (c) (4)	Thermal test: The specimen shall be in thermal equilibrium under conditions of an ambient temperature of 38ºº C, subject to the solar insolation conditions specified in Table IX and subject to the design maximum rate of internal heat generation within the package from the radioactive contents. Alternatively, any of these parameters are allowed to have different values prior to and during the test, providing due account is taken of them in the subsequent assessment of package response. The thermal test shall consist of (a), (b) below. During and following the test the specimen shall not be artificially cooled and any combustion of materials of the specimen shall be permitted to proceed naturally.	71.73 (c) (4) contains similar requirements to TS-R-1.
728 (a)	71.73 (c) (4)	Exposure of a specimen for a period 30 minutes to a thermal environment which provides a heat flux at least equivalent to that of a hydrocarbon fuel/air fire in sufficiently quiescent ambient conditions to give a minimum average flame emissivity coefficient of 0.9 and an average temperature of at least 800°°C, fully engulfing the specimen, with a surface absorptivity coefficient of 0.8 or that value which the package may be demonstrated to possess if exposed to the fire specified, followed by:	71.73(c)(4) contains similar requirements to TS-R-1.

7 2 8 (b)	71.73 (c)(4)	Exposure of the specimen to an ambient temperature of 38°°C, subject to the solar insolation conditions specified in Table XI and subject to the design maximum rate of internal heat generation within the package by the radioactive contents for a sufficient period to ensure that temperatures in the specimen are everywhere decreasing and/or are approaching initial steady state conditions.	71.73(c)(4) contains similar requirements to TS-R-1.
729	71.73 (c)(6)	Water immersion test: The specimen shall be immersed under a head of water of a least 15 m for a period of not less than eight hours in the attitude which will lead to maximum damage. For demonstration purposes, an external gauge pressure of at least 150 kPa shall be considered to meet these conditions.	71.73(c)(6) contains similar requirements to TS-R-1, but does not include the 8 hour test duration.
730	71.61	**Enhanced water immersion test for Type B(U) and Type B(M) packages containing more than 10(sup)5 A(sub)2 and Type C packages** (730) Enhanced water immersion test: The specimen shall be immersed under a head of water of at least 200 m for a period of not less than one hour. For demonstration purposes, an external gauge pressure of at least 2 Map shall be considered to meet these conditions.	71.61 contains similar requirements to TS-R-1, but TS-R-1 applies the test to all packages, not just irradiated nuclear fuel packages, above a certain activity threshold.
731	71.73 (c)(5)	**Water leakage test for packages containing fissile material** (731-733) Packages for which water in-leakage or out-leakage to the extent which results in greatest reactivity has been assumed for purposes of assessment under paras. 677-682 shall be excepted from the test.	71.73(c)(5) contains similar requirements to TS-R-1, but does not impose the test where water out-leakage has not been assumed.
732	71.73	Before the specimen is subjected to the water leakage test specified below, it shall be subjected to the tests in para. 727(b), and either para. 727(a) or (c) as required by para. 682, and the test specified in para. 728.	71.73 contains similar requirements to TS-R-1 regarding the order of tests.
733	71.73 (c)(5)	The specimen shall be immersed under a head of water of a least 0.9 m for a period of not less than eight hours and in the attitude for which maximum leakage is expected.	71.73(c)(5) contains similar requirements to TS-R-1, but does not include the 8 hour test duration.
734		**Test for Type C packages** (734-737) Specimens shall be subjected to the effects of each of the following test (a and b below) sequences in the orders specified: Separate specimens are allowed to be used for each of the sequences (a) and (b).	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.

734 (a)		the test specified in para 727(a), 727(c), 735 and 736; and	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
7 3 4 (b)		the test specified in para. 737. Separate specimens are allowed to be used for each of the sequences (a) and (b).	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
735		Puncture/tearing test: The specimen shall be subjected to the damaging effects of a solid probe made of mild steel. The orientation of the probe to the surface of the specimen shall be such as to cause maximum damage at the conclusion of the test sequence specified in para. 734(a).	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
735 (a)		The specimen, representing a package having a mass less than 250 kg, shall be placed on a target and subjected to a probe having a mass of 250 kg falling from a height of 3 m above the intended impact point. For this test the probe shall be a 20 cm diameter cylindrical bar with the striking end forming a frustum of a right circular cone with the following dimensions: 30 cm height and 2.5 cm diameter at the top. The target on which the specimen is placed shall be as specified in para. 717.	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
7 3 5 (b)		For packages having a mass of 250 kg or more, the base of the probe shall be placed on a target and the specimen dropped onto the probe. The height of the drop, measured from the point of impact with the specimen to the upper surface of the probe shall be 3 m. For this test the probe shall have the same properties and dimensions as specified in (a) above, except that the length and mass of the probe shall be such as to incur maximum damage to the specimen. The target on which the base of the probe is placed shall be as specified in para. 717.	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
736		Enhanced thermal test: The conditions for this test shall be as specified in para 728, except that the exposure to the thermal environment shall be for a period of 60 minutes.	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
737		Impact test: The specimen shall be subject to an impact on a target at a velocity of not less than 90 m/s, at such an orientation as to suffer maximum damage. The target shall be as defined in para. 717.	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
801		**Section VIII - APPROVAL AND ADMINISTRATIVE REQUIREMENTS** GENERAL (801-802) For packages designs where it is not required that a competent authority issue an approval	No similar requirement in Part 71 as this is within DOT's area of responsibility. 49 CFR 173.411(c) requires this for IP-2 and IP-3 packages and 173.415(a) requires this for Type A packages.

		certificate the consignor shall, on request, make available for inspection by the relevant competent authority, documentary evidence of the compliance of the package design with all the applicable requirements.	
802		Competent authority approval shall be required for:	Both Part 71 and Title 49 have similar requirements.
802 (a)	71.12	designs for: (i)special form radioactive material; (ii) low dispersible radioactive material; (iii) packages containing 0.1 kg or more of UF_6; (iv) all packages containing fissile material unless excepted by para. 672; (v) Type B(U) packages and Type B(M) packages; (vi) Type C packages	71.12 provides a general license which is dependent on the licensee using a package for which a license or certificate of compliance or other approval has been issued. This applies to all fissile material and Type B and packages (consistent with TS-R-1 802(a)(iv) and (a)(v)). Title 49 has requirements that are similar to (i), (iv) and (v). There are no similar requirements for (ii), (iii) and (vi).
8 0 2 (b)	71.8	special arrangements	71.8 (specific exemptions) has similar requirements to TS-R-1 802 (b). 49 CFR Part 107, Subpart B has similar requirements for exemptions.
802 (c)		certain shipments	No similar requirement, however, this topic is within DOT's area of responsibility.
8 0 2 (d)		radiation protection program for special use vessels; and	No similar requirement, however, this topic is within DOT's area of responsibility.
802 (e)		calculation of radionuclide values that are not listed in Table I.	Part 71,Appendix A, II and 49 CFR 173.433(b) are essentially the same as TS-R-1.
803		APPROVAL OF SPECIAL FORM RADIOACTIVE MATERIAL AND LOW DISPERSIBLE RADIOACIVE MATERIAL (803-804) The design for special form radioactive material shall require unilateral approval. The design for low dispersible radioactive material shall require multilateral approval. In both cases, an application shall include:	No similar requirement in Part 71. 49 CFR 173.476 contains a similar requirement for special form material. No similar requirement exists for low dispersible material.
803 (a)		a detailed description of the radioactive material, or, if a capsule, the contents; particular reference shall be made to both physical and chemical states;	No similar requirement in Part 71. 49 CFR 173.476 contains a similar requirement for special form material. No similar requirement exists for low dispersible material.
8 0 3 (b)		a detailed statement of the design of any capsule to be used;	No similar requirement in Part 71. 49 CFR 173.476 contains a similar requirement for special form material. No similar requirement exists for low dispersible material.
803 (c)		a statement of the tests which have been done and their results, or evidence based on calculative methods to show that the radioactive material is capable of meeting the performance	No similar requirement in Part 71. 49 CFR 173.476 contains a similar requirement for special form material. No similar requirement exists for low dispersible material.

		standards, or other evidence that the special form radioactive material or low dispersible radioactive material meets the applicable requirements of these Regulations.	
8 0 3 (d)		a specification of the applicable quality assurance program as required in para. 310; and	No similar requirement in Part 71. 49 CFR 173.476 contains a similar requirement for special form material. No similar requirement exists for low dispersible material.
803 (e)		any proposed pre-shipment actions for use in the consignment of special form radioactive material or low dispersible radioactive material	No similar requirement.
804		The competent authority shall establish an approval certificate stating that the approved design meets the requirements for special form radioactive material or low dispersible radioactive material and shall attribute to that design an identification mark.	No similar requirement in Part 71. 49 CFR 173.476 contains a similar requirement for special form material. No similar requirement exists for low dispersible material.
805		APPROVAL OF PACKAGE DESIGNS (805-814) **Approval of package designs to contain uranium hexafluoride** (805) Provides requirements for the approval of packages containing 0.1 kg or more of UF6.	Part 71 does not contain package approval requirements for uranium hexafluoride packages. 49 CFR 173.420 contains packaging and transport requirements, but does not contain requirements for package design approval.
805 (a)		After 31 December 2000, each design that meets the requirements of para. 632 shall require multilateral approval. After 31 December 2003, each design that meets the requirements of paras. 623-631 shall require unilateral approval by the competent authority of the country of origin of the design;	Part 71 does not contain package approval requirements for uranium hexafluoride packages. 49 CFR 173.420 contains packaging and transport requirements, but does not contain requirements for package design approval.
8 0 5 (b)		The application for approval shall include all information necessary to satisfy the competent authority that the design meets the requirements of para. 629, and a specification of the applicable quality assurance program as required in para. 310;	Part 71 does not contain package approval requirements for uranium hexafluoride packages. 49 CFR 173.420 contains packaging and transport requirements, but does not contain requirements for package design approval.
805 (c)		The competent authority shall establish an approval certificate stating that the approved design meets the requirements of para. 629 and shall attribute to that design an identification mark.	Part 71 does not contain package approval requirements for uranium hexafluoride packages. 49 CFR 173.420 contains packaging and transport requirements, but does not contain requirements for package design approval.
806	71.12	**Approval of Type B(U) and Type C package designs** (806-808) Each Type B(U) and Type C package design requires unilateral approval, except that:	71.12 is similar to TS-R-1 for Type B(U) packages. 49 CFR 173.471 and 173.472 are similar for Type B packages. No similar requirement for Type C packages.

806 (a)	71.12	a package design for fissile material, which is also subject to paras 812-814, shall require multilateral approval; and	71.12 is similar to TS-R-1 for fissile packages. 49 CFR 173.471 and 173.472 are similar for fissile packages. However, both Part 71 and Title 49 only require approval by NRC and DOT, they do not impose multilateral approval (approval by other countries) requirements.
806 (b)		a Type B(U) package design for low dispersible radioactive material shall require multilateral approval.	No similar requirement; this is a new requirement in TS-R-1 and is being considered in the Part 71 rulemaking.
807	71.31, 71.33	Provides the requirements for the information which must be contained in an application for approval for Type B(U) and Type C packages.	71.31and 71.33, and 49 CFR 173.471, 173.472 and 173.473 contain similar requirements for Type B packages. No similar requirements exist for Type C packages.
807 (a)	71.31, 71.33	a detailed description of the proposed radioactive contents with reference to their physical and chemical states and the nature of the radiation emitted;	71.31and 71.33, and 49 CFR 173.471, 173.472 and 173.473 contain similar requirements for Type B packages. No similar requirements exit for Type C packages.
807 (b)	71.31, 71.33	a detailed statement of the design, including complete engineering drawings and schedules of materials and methods of manufacture;	71.31and 71.33, and 49 CFR 173.471, 173.472 and 173.473 contain similar requirements for Type B packages. No similar requirements exist for Type C packages.
807 (c)	71.31, 71.33	a statement of the tests which have been done and their results, or evidence based on calculative methods or other evidence that the design is adequate to meet the applicable requirements;	71.31and 71.33, and 49 CFR 173.471, 173.472 and 173.473 contain similar requirements for Type B packages. No similar requirements exist for Type C packages.
807 (d)		the proposed operating and maintenance instructions for the use of the packaging;	No similar requirement.
807 (e)	71.33	if the package is designed to have a maximum normal operating pressure in excess of 100 kPa gauge, a specification of the materials of manufacture of the containment system, the samples to be taken, and the tests to be made;	71.33 requires detailed descriptions of the materials of construction for all packages.
807 (f)	71.35 (c)	where the proposed radioactive contents are irradiated fuel, the applicant shall state and justify any assumption in the safety analysis relating to the characteristics of the fuel and describe any preshipment measurement required by para. 674(b);	71.35(c) contains a similar requirement.
807 (g)	71.33	any special stowage provisions necessary to ensure the safe dissipation of heat from the package considering the various modes of transport to be used and type of conveyance or freight container;	71.33 contains a similar requirement.
807 (h)	71.33	a reproducible illustration, not larger than 21 cm by 30 cm, showing the make-up of the package; and`	71.33 requires a description of the package. 49 CFR 173.471and 173.472 contain similar requirements for Type B packages. No similar requirements for Type C packages.

807 (i)	71.37 (a) 71.37 (b)	a specification of the applicable quality assurance program as required in para. 310.	71.37 contains a similar requirement.
808		The competent authority shall establish an approval certificate stating that the design meets requirements for Type B(U) or Type C packages and shall attribute an identification mark to the design.	No similar requirement in Part 71, although this is done in practice. 49 CFR 173.471and 173.472 contain similar requirements for Type B packages. No similar requirements for Type C packages.
809	71.12	**Approval of Type B(M) package design** (809-811) Each Type B(M) package, including those for fissile material and low dispersible radioactive material, shall require multilateral approval.	71.12 requires approval of Type B packages, including Type B(M). 49 CFR 173.471 is similar. 71.16 and 49 CFR 173.473 also require that foreign approved packages receive US approval. There are no requirements related to obtaining multilateral approval for US origin design Type B(M) packages.
810	71.31 71.33	An application for approval of a Type B(M) package design shall include, in addition to the information required in para. 807 for Type B(U) packages:	All package approval applications must meet 71.31 and 71.33.
810 (a)		a list of the requirements specified in paras. 637, 653, 654, and 657-664 with which the package does not conform;	No similar requirement. Non-conforming designs are only allowed under exemption (71.8).
8 1 0 (b)		any proposed supplementary operational controls to be applied during transport not regularly provided for in these Regulations, but which are necessary to ensure the safety of the package or to compensate for the deficiencies listed in (a) above;	No similar requirement. Non-conforming designs are only allowed under exemption (71.8).
810 (c)		a statement relative to any restrictions on the mode of transport and to any special loading, carriage, unloading or handling procedures; and	No similar requirement. Non-conforming designs are only allowed under exemption (71.8).
8 1 0 (d)		the range of ambient conditions (temperature, solar radiation) which are expected to be encountered during transport and which have been taken into account in the design.	No similar requirement. Non-conforming designs are only allowed under exemption (71.8).
811		The competent authority shall establish an approval certificate stating that the approved design meets the applicable requirements for Type B(M) packages and shall attribute to that design an identification mark.	No similar requirement in Part 71, although this is done in practice. 49 CFR 173.471and 173.472 contain similar requirements for Type B packages.
812	71.12	**Approval of package designs to contain fissile material** (812-814) Each package design for fissile material which is not excepted according to para. 672 from the requirements that apply specifically to packages containing fissile material shall require multilateral approval.	71.12 requires approval of fissile material package designs. 49 CFR 173.471 is similar. 71.16 and 49 CFR 173.473 also require that foreign approved packages receive US approval. There are no requirements related to obtaining multilateral approval for US origin design fissile material packages.
813	7 1 . 3 3 , 71.35, 71.37	An application for approval shall include all information necessary to satisfy the competent authority that the design meets the requirements	71.33, 71.35 and 71.37 contain similar requirements. 49 CFR 173.471 contains similar requirements.

		of para. 671, and a specification of the applicable quality assurance program as required in para. 310.	
814		The competent authority shall establish an approval certificate stating that the approved design meets the requirements of para. 671 and shall attribute to that design an identification mark.	No similar requirement in Part 71, although this is done in practice. 49 CFR 173.471and 173.472 contain similar requirements for Type B packages.
815		TRANSITIONAL ARRANGEMENTS (815-818) **Packages not requiring competent authority approval of design under the 1985 and 1985 (As Amended 1990) Editions of These Regulation** Excepted packages, Industrial packages Type IP-1, Type IP-2, and Type IP-3 and Type A packages that did not require approval of design by the competent authority and which meet the requirements of the 1985 or 1985 (As Amended 1990) Editions of these Regulations may continue to be used subject to the mandatory program of QA and the activity limits of TS-R-1. Any packaging modified, unless to improve safety, or manufactured after 31 December 2003, shall meet this Edition of the Regulations in full. Packages prepared for transport not later than 31 December 2003 under the 1985 or 1985 (As Amended 1990) Editions of these Regulations may continue in transport. Packages prepared for transport after this date shall meet this Edition of the Regulations in full.	No similar provision, however, this topic is within DOT's area of responsibility. 49 CFR 173.415(a) contains a similar requirement for packages complying with earlier versions of Title 49.
816	71.13 (b)	**Packages approved under the 1973, 1973 (As Amended), 1985 and 1985 (As Amended 199) Editions of these Regulations (816-817)** Packagings manufactured to a design approved under the provision of the 1973 or 1973 (As Amended) Editions of the IAEA regulations may continue to be used, subject to: multilateral approval of package design; a mandatory program of quality assurance; the activity limits and material restrictions of TS-R-1; and, for a package containing fissile material and transported by air, the requirements of para. 680. No new manufacture of such packagings is allowed to begin. Changes in the package design or nature or quantity of the contents which, as determined by the competent authority, would significantly affect safety shall require that this Edition of the Regulations by met in full. A serial number shall be assigned	71.13(b) contains a similar provisions. 71.13(b) is applicable to packages approved under the 1973 (As Amended) regulations. 71.13(a) contains provisions for packages approved under the 1967 regulations and TS-R-1 does not contain this provision.

		and marked on the outside of each packaging.	
817		Packagings manufactured to a package design approved by the competent authority under the provisions of the 1985 or 1985 (As Amended 1990) Editions of these Regulations may continue to be used until 31 December 2003, subject to: the mandatory program of quality assurance; the activity limits and material restrictions of Section IV,; and, for a package containing fissile material and transported by air, the requirement of para. 680. After this date use may continue subject, additionally, to multilateral approval of package design, Changes in the design of the packaging or in the nature or quantity of the authorized radioactive contents which, as determined by the competent authority, would significantly affect safety shall require that this Edition of the Regulations be met in full. All packagings for which manufacture begins after 30 December 2006 shall meet this Edition of the Regulations in full.	No similar provision; this is a new provision in TS-R-1 and is being considered in the Part 71 rulemaking.
818		**Special form radioactive material approved under the 1973, 1973 (As Amended), 1985 and 1985 (As Amended 1990) Editions of these Regulations** (818) Special form radioactive material approved under the 1973, 1973 (As Amended), 1985 and 1985 (As Amended 1990) Editions of these Regulations may continue to used when in compliance with the mandatory program of QA in accordance with the applicable requirements of para. 310. All special form radioactive material manufactured after 31 December 2003 shall meet this Edition of the Regulations in full.	No similar provision.
819		NOTIFICATION AND REGISTRATION OF SERIAL NUMBERS (819) The competent authority should maintain a register of serial numbers.	No similar requirement.
820		APPROVAL OF SHIPMENTS (820-823) Multilateral approval shall be required:	No similar requirement.
820 (a)		the shipment of Type B(M) packages not conforming with the requirements of para. 637 or designed to allow controlled intermittent venting;	No similar requirement.
820 (b)		the shipment of Type B(M) packages containing radioactive material with an activity greater than 3000 A (sub)1 or 3000 A (sub) 2, as	No similar requirement.

		appropriate, or 1000 TBq, whichever is the lower;	
820 (c)		the shipment of packages containing fissile materials if the sum of the criticality safety indexes of the packages exceeds 50; and	No similar requirement.
8 2 0 (d)		radiation protection programs for shipments by special use vessels according to para. 575 (a).	No similar requirement.
821		A competent authority may authorize transport into or through its country without shipment approval, by a specific provision in its design approval (see para. 827).	No similar requirement.
822		An application for shipment approval shall include:	No similar requirement.
822 (a)		the period of time, related to the shipment, for which the approval is sought;	No similar requirement.
8 2 2 (b)		the actual radioactive contents, the expected modes of transport, the type of conveyance, and the probable or proposed route; and	No similar requirement.
822 (c)		the details of how the precautions and administrative or operational controls, referred to in the package design approval certificates issued under paras. 808, 811, and 814, are to be put into effect.	No similar requirement.
823		Upon approval of the shipment, the competent authority shall issue an approval certificate.	No similar requirement.
824		APPROVAL OF SHIPMENTS UNDER SPECIAL ARRANGEMENT (824-865) Each consignment transported internationally under special arrangement shall require multilateral approval.	If the special arrangement uses a foreign-made package, 49 CFR 173.473 requires Competent Authority 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:	No similar requirement. 71.8 provides for specific exemptions which are similar to special arrangements, but 71.8 does not include an "equivalent safety" provision. 49 CFR 107 Subpart B is similar to TS-R-1.
825 (a)		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	No similar requirement in Part 71. 49 CFR 107 Subpart B is similar to TS-R-1.
8 2 5 (b)		A statement of any special precautions or special administrative or operational controls which are to be employed during transport to	No similar requirement in Part 71. 49 CFR 107 Subpart B is similar to TS-R-1.

		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.	No similar requirement in Part 71. 49 CFR 107 Subpart B is similar to TS-R-1.
827		COMPETENT AUTHORITY APPROVAL CERTIFICATES (827-829) Five types of approval certificates may be issued: special form radioactive material, low dispersible radioactive material, special arrangement, shipment, and package design. The package design and shipment approval certificates may be combined into a single certificate.	No similar statement.
828		**Competent authority identification marks** (828-829) Each approval certificate issued by a competent authority shall be assigned an identification mark. The mark shall be of the following generalized type: VRI/Number/Type Code Each approval certificate issued by a competent authority shall be assigned an identification mark. The mark shall be of the following generalized type:	No similar requirement, although this is done in practice.
828 (a)		Except as provided in para. 829(b), VRI represents the international vehicle registration identification code of the country issuing the certificate	No similar requirement, although this is done in practice.
8 2 8 (b)		The number shall be assigned by the competent authority, and shall be unique and specific with regard to the particular design or shipment. The shipment approval identification marks shall be clearly related to the design approval identification mark.	No similar requirement, although this is done in practice.
828 (c)		The following type codes shall be used in the order listed: AF Type A package design for fissile material B(U) Type B(U) package design [B(U)F if for fissile material] B(M) Type B(M) package design [B(M)F if for fissile material] C Type C package design [CF if for fissile material] IF Industrial package design for fissile material	No similar requirement, although this is done in practice. Type C, IF, LD, H(U) or H(M) package designs are not reflected in Part 71 nor Title 49.

		S Special form radioactive material LD Low dispersible radioactive material T Shipment X Special arrangement. For non-fissile or fissile excepted UF6, where none of the above codes apply: H(U) Unilateral approval H(M) Multilateral approval	
8 2 8 (d)		For package design and special form radioactive material approval certificates, other than those issued under the provisions of paras. 816-818, and for low dispersible radioactive material approval certificates, the symbols "—96" shall be added to the type code.	No similar requirement.
829		Type codes shall be applied as follows:	No similar requirement, although this is done in practice.
829 (a)		Each certificate and each package shall bear the appropriate identification mark, comprising the symbols prescribed in para. 828(a), (b), (c) and (d) above, except that, for packages, only the applicable design type codes including, if applicable, the symbols '—96', shall appear following the second stroke, that is, the 'T', or 'X' shall not appear in the identification marking on the package. Where the design approval and shipment approval are combined, the applicable type codes do not need to be repeated. Examples of these are given in TS-R-1. A/132/B(M)F-96: A Type B(M) package design approved for fissile material, requiring multilateral approval, for which the competent authority of Austria has assigned the design number 132 (to be marked on both the package and on the package design approval certificate).	No similar requirement, although this is done in practice. However, the "-96" provision is not contained in Part 71 nor Title 49.
8 2 9 (b)		Where multilateral approval is effected by validation according to para. 834, only the identification mark issued by the country of origin of the design or shipment shall be used. Where multilateral approval is effected by issue of certificates by successive countries, each certificate shall bear the appropriate identification mark and the package whose design was so approved shall bear all appropriate identification marks. For example: A/132/B(M)F-96 and CH/28/B(M)F-96 would be the identification mark of a package which was originally approved by Austria and was subsequently approved, by separate certificate, by Switzerland. Additional identification marks would be tabulated in a similar manner on the package.	No similar requirement in Part 71. 49 CFR 173.473 requires that the US issued identification mark be displayed on the shipping paper and the package.

829 (c)		The revision of a certificate shall be indicated by a parenthetical expression following the identification mark on the certificate. For example, A/132/B(M)F-96(Rev.2) would indicate revision 2 of the Austrian package design approval certificate; or A/132/B(M)F-96(Rev.0) would indicate the original issuance of the Austrian package design approval certificate. For original issuance, the parenthetical entry is optional and other words such as 'original issuance' may also be used in place of 'Rev0'. Certificate revision numbers may only be issued by the country issuing the original approval certificate.	No similar requirement, although a revision number is used in practice.
8 2 9 (d)		Additional symbols (as may be necessitated by national requirements) may be added in brackets to the end of the identification mark; for example, A/13/B(M)F-96(SP503).	No similar provision.
829 (e)		It is not necessary to alter the identification mark on the packaging each time that a revision to the design certificate is made. Such re-marking shall be required only in those cases where the revision to the package design certificate involves a change in the letter type codes for the package design following the second stroke.	No similar requirement, although this is done in practice.
830		CONTENTS OF APPROVAL CERTIFICATES (830-833) **Special form radioactive material and low dispersible radioactive material approval certificates** (830) Each approval certificate issued by a competent authority for special form radioactive material shall include the following information:	No similar requirement, although this is done in practice by DOT for special form. No similar requirement exists for low dispersible material.
830 (a)		Type of certificate	No similar requirement, although this is done in practice by DOT.
8 3 0 (b)		The competent authority identification mark	No similar requirement, although this is done in practice by DOT.
830 (c)		The issue date and expiration date.	No similar requirement, although this is done in practice by DOT.
8 3 0 (d)		List of applicable nation and international regulations, including the edition of the IAEA Regulations for the Safe Transport of Radioactive Material under which the special form radioactive material or low dispersible radioactive material is approved	No similar requirement, although this is done in practice by DOT for special form material.
830 (e)		The identification of the special form radioactive material or low dispersible radioactive or low dispersible radioactive material	No similar requirement, although this is done in practice by DOT for special form material.

830 (f)		A description of the special form radioactive material or low dispersible radioactive material.	No similar requirement, although this is done in practice by DOT for special form material.
8 3 0 (g)		Design specifications for the special form radioactive material or low dispersible radioactive material which may include references to drawings.	No similar requirement, although this is done in practice by DOT for special form material.
8 3 0 (h)		A specification of the radioactive contents which includes the activities involved and which may include the physical and chemical form.	No similar requirement, although this is done in practice by DOT for special form material.
830 (i)		A specification of the applicable quality assurance program as required in para. 310	No similar requirement, although this is done in practice by DOT for "-85" approvals.
830 (j)		Reference to information provided by the applicant relating to specific actions to be taken prior to shipment	No similar requirement, although this may be done in practice by DOT.
8 3 0 (k)		If deemed appropriate by the competent authority, reference to the identity of the applicant.	No similar requirement, although this is done in practice by DOT for special form material.
830 (l)		Signature and identification of the certifying official.	No similar requirement, although this is done in practice by DOT for special form material.
831		**Special arrangement approval certificates** (831) Each approval certificate issued by a competent authority for a special arrangement shall include the following information:	No similar requirement, although this is done in practice.
831 (a)		Type of certificate	No similar requirement, although this is done in practice.
8 3 1 (b)		The competent authority identification mark.	No similar requirement, although this is done in practice.
831 (c)		The issue date and an expiration date.	No similar requirement, although this is done in practice.
8 3 1 (d)		Mode(s) of transport.	No similar requirement, although this is done in practice.
831 (e)		Any restrictions on the modes of transport, type of conveyance, freight container, and any necessary routing instructions.	No similar requirement, although this is done in practice.
831 (f)		List of applicable national and international regulations, including the edition of the IAEA Regulations for the Safe Transport of Radioactive Material under which the special arrangement is approved.	No similar requirement, although this is done in practice.
8 3 1 (g)		The following statement: "This certificates does not relieve the consignor from compliance with any requirement of the government of any country through or into	No similar requirement, although this is done in practice.

		which the package will be transported."	
8 3 1 (h)		References to certificates for alternative radioactive contents, other competent authority validation, or additional technical data or information, as deemed appropriate by the competent authority.	No similar requirement, although this is done in practice.
831 (i)		Description of the packaging by a reference to the drawings or a specification of the design. If deemed appropriate by the competent authority, a reproducible illustration, not larger than 21 cm by 30 cm, showing the make-up of the package should also be provided, accompanied by a brief description of the packaging, including materials of manufacture, gross mass, general outside dimension and appearance.	No similar requirement, although this is done in practice.
831 (j)		A specification of the authorized radioactive contents, including any restrictions on the radioactive contents which might not be obvious from the nature of the packaging. This shall include the physical and chemical forms, the activities involved (including those of the various isotopes, if appropriate), amounts in grams (for fissile material), and whether special form radioactive material or low dispersible radioactive material, if applicable.	No similar requirement, although this is done in practice.
8 3 1 (k)		Additionally, for packages of fissile material: a detailed description of the authorized radioactive contents; the value of the criticality safety index; reference to the documentation that demonstrates the criticality safety of the contents; any special features, on the basis of which the absence of water from certain void spaces has been assumed in the criticality assessment; any allowance (based on para. 674(b)) for a change in neutron multiplication assumed in the criticality assessment as a result of actual irradiation experience; and the ambient temperature range for which the special arrangement has been approved.	No similar requirement, although this is done in practice. There are no provision for the criticality safety index (the transport index for criticality control purposes is used).
831 (l)		A detailed listing of any supplementary operational controls required for preparation, loading, carriage, unloading and handling of the consignment, including any special stowage provisions for the safe dissipation of heat.	No similar requirement, although this is done in practice.
8 3 1 (m)		If deemed appropriate by the competent authority, reasons for the special arrangement	No similar requirement, although this is done in practice.
8 3 1 (n)		Description of the compensatory measures to be applied as a result of the shipment being under special arrangement.	No similar requirement, although this is done in practice.
8 3 1 (o)		Reference to information provided by the applicant relating to the use of the packaging or specific actions to be taken prior to the	No similar requirement, although this is done in practice.

		shipment.	
8 3 1 (p)		A statement regarding the ambient conditions assumed for purposes of design if these are not in accordance with those specified in paras. 653, 654, and 664, as applicable.	No similar requirement, although this may be done in practice.
8 3 1 (q)		Any emergency arrangement deemed necessary by the competent authority.	No similar requirement, although this may be done in practice.
831 (r)		A specification of the applicable quality assurance program as required in para. 310.	No similar requirement, although this may be done in practice.
831 (s)		If deemed appropriate by the competent authority, reference to the identity of the applicant and to the identity of the carrier.	No similar requirement, although this is done in practice.
831 (t)		Signature and identification of the certifying official.	No similar requirement, although this is done in practice.
832		**Shipment approval certificates** (832) Each approval certificate for a shipment issued by a competent authority shall include the following information:	No similar requirement, although this is done in practice by DOT.
832 (a)		Type of certificate	No similar requirement, although this is done in practice by DOT.
8 3 2 (b)		The competent authority identification mark(s).	No similar requirement, although this is done in practice by DOT.
832 (c)		The issue date and an expiry date.	No similar requirement, although this is done in practice by DOT.
8 3 2 (d)		List of applicable national and international regulations, including the edition of the IAEA Regulations for the Safe Transport of Radioactive Material under which the shipment is approved.	No similar requirement, although this is done in practice by DOT.
832 (e)		Any restrictions on the modes of transport, type of conveyance, freight container, and any necessary routing instructions.	No similar requirement, although this is done in practice by DOT.
832 (f)		The following statement: "This certificates does not relieve the consignor from compliance with any requirement of the government of any country through or into which the package will be transported."	No similar requirement, although this is done in practice by DOT.
8 3 2 (g)		A detailed listing of any supplementary operational controls required for preparation, loading, carriage, unloading, and handling of the consignment, including any special stowage provisions for the safe dissipation of heat or maintenance of criticality safety.	No similar requirement, although this is done in practice by DOT.
8 3 2 (h)		Reference to information provided by the applicant relating to specific actions to be taken prior to shipment.	No similar requirement, although this is done in practice by DOT.
832 (i)		Reference to the applicable design approval certificate(s).	No similar requirement, although this is done in practice by DOT.
832 (j)		A specification of the actual radioactive contents, including any restrictions on the	No similar requirement, although this is done in practice by DOT.

		radioactive contents which might not be obvious from the nature of the packaging. This shall include the physical and chemical forms, the total activities involved (including those of the various isotopes, if appropriate), amounts in grams (for fissile material), and whether special form radioactive material or low dispersible radioactive material, if applicable.	
8 3 2 (k)		Any emergency arrangements deemed necessary by the competent authority.	No similar requirement, although this is done in practice by DOT.
832 (l)		A specification of the applicable quality assurance program as required in para. 310.	No similar requirement, although this may be done in practice by DOT.
8 3 2 (m)		If deemed appropriate by the competent authority, reference to the identity of the applicant.	No similar requirement, although this is done in practice by DOT.
8 3 2 (n)		Signature and identification of the certifying official.	No similar requirement, although this is done in practice by DOT.
833		**Package design approval certificates** (833) Each approval certificate of the design of a package issued by a competent authority shall include the following information:	No similar requirement, although this is done in practice.
833 (a)		Type of certificate.	No similar requirement, although this is done in practice.
8 3 3 (b)		The competent authority identification mark.	No similar requirement, although this is done in practice.
833 (c)		The issue date and expiry date.	No similar requirement, although this is done in practice.
8 3 3 (d)		Any restriction on the modes of transport, if appropriate.	No similar requirement, although this is done in practice.
833 (e)		List of applicable national and international regulations, including the edition of the IAEA Regulations for the Safe Transport of Radioactive Material under the design is approved.	No similar requirement, although this is done in practice.
833 (f)		The following statement: "This certificates does not relieve the consignor from compliance with any requirement of the government of any country through or into which the package will be transported."	No similar requirement, although this is done in practice by DOT.
8 3 3 (g)		References to certificates for alternative radioactive contents, other competent authority validation, or additional technical data or information, as deemed appropriate by the competent authority.	No similar requirement, although this is done in practice.
8 3 3 (h)		A statement authorizing shipment where shipment approval is required under para. 820, if deemed appropriate.	No similar requirement, although this is done in practice by DOT.

833 (i)		Identification of the packaging.	No similar requirement, although this is done in practice.
833 (j)		Description of the packaging by a reference to the drawings or specification of the design. If deemed appropriate by the competent authority, a reproducible illustration, not larger than 21 cm by 30 cm, showing the make-up of the package should also be provided, accompanied by a brief description of the packaging, including materials of manufacture, gross mass, general outside dimensions and appearance.	No similar requirement, although this is done in practice.
8 3 3 (k)		Specification of the design by reference to the drawings.	No similar requirement, although this is done in practice.
833 (l)		A specifications of the authorized radioactive content, including any restrictions on the radioactive contents which might not be obvious from the nature of the packaging. This shall include the physical and chemical forms the activities involved (including those of the various isotopes, if appropriate), amounts in grams (for fissile material), and whether special form radioactive material or low dispersible radioactive material, if applicable.	No similar requirement, although this is done in practice (except for low dispersible material).
8 3 3 (m)		Additionally, for packages of fissile material: a detailed description of the authorized radioactive contents; the value of the criticality safety index; reference to the documentation that demonstrates the criticality safety of the contents; any special features, on the basis of which the absence of water from any certain void spaces has been assumed in the criticality assessment; any allowance (based on para. 674(b)) for a change in neutron multiplication assumed in the criticality assessment as a result of actual irradiation experience; and the ambient temperature range for which the package design has been approved.	No similar requirement, although this is done in practice, as deemed appropriate (except for the criticality safety index, since the transport index for criticality control purposes is used).
8 3 3 (n)		For Type B(M) packages, a statement specifying those prescriptions of paras. 637, 653, 654, and 657-664 with which the package does not conform and any amplifying information which may be useful to other competent authorities.	No similar requirement, although this is done in practice.
8 3 3 (o)		A detailed listing of any supplementary operational controls required for preparation, loading, carriage, unloading, and handling of the consignment, including any special stowage provisions for the safe dissipation of heat.	No similar requirement, although this is done in practice.
8 3 3 (p)		Reference to information provided by the applicant relating the use of the packaging or specific actions to be taken prior to shipment.	No similar requirement, although this is done in practice.

8 3 3 (q)		A statement regarding the ambient conditions assumed for purpose of design if these are not in accordance with those specified in paras. 653, 654, and 664, as applicable.	No similar requirement, although this may done in practice.
833 (r)		A specification of the applicable QA program as required in para. 310.	No similar requirement, although this is done in practice.
833 (s)		Any emergency arrangements deemed necessary by the competent authority.	No similar requirement, although this is done in practice by DOT.
833 (t)		If deemed appropriate by the competent authority, reference to the identity of the applicant.	No similar requirement, although this is done in practice.
8 3 3 (u)		Signature and identification of the certifying official.	No similar requirement, although this is done in practice.
834		VALIDATION OF CERTIFICATES Multilateral approval may be by validation of the original certificate issued by the competent authority of the country of origin of the design or shipment. Such validation may take the form of an endorsement on the original certificate or the issuance of a separate endorsement, annex, supplement, etc., by the competent authority of the country through or into which the shipment is made.	No similar requirement, although this is done in practice by DOT.
	71.0 (a)		This statement of purpose and scope is narrower than the broader scope of TS-R-1.
	71.0 (d)		TS-R-1 does not contain similar requirements pertaining to licensing. Some of the requirements such as applications for package approval and operating controls and procedures are similar to TS-R-1.
	71.1 (a) (b)		TS-R-1 does not address record retention requirements
	71.2		TS-R-1 does not contain similar requirements.
	71.3		TS-R-1 does not contain similar requirements.
	71.5 (b)		TS-R-1 does not contain similar requirements.
	71.6 (a) (b)		TS-R-1 does not contain similar text on information collection.
	71.7 (a) (b)		TS-R-1 does not contain similar requirements concerning completeness and accuracy of information.
	71.9		TS-R-1 does not contain similar text on information collection.
	71.10 (b) (3)		TS-R-1 does not contain similar exceptions for special form americium and plutonium.
	71.10 (c)		TS-R-1 does not contain similar requirements.
	71.11 (a) (b) (c) (d)		TS-R-1 does not contain similar requirements pertaining to deliberate misconduct.

	71.14 (a) (b) (c)		TS-R-1 does not contain similar provisions as it does not include "specification containers".
	71.16 (a) (b) (c)		TS-R-1 does not contain similar requirements.
	71.18 (a) (b) (c) (d) (e)		TS-R-1 does not contain provisions for fissile material, limited quantity per package.
	71.20 (a) (b) (c)		TS-R-1 does not contain provisions for fissile material, limited moderator per package.
	71.22 (a) (b) (c) (d) (e) (f)		TS-R-1 does not contain provisions for fissile material, limited quantity, controlled shipments.
	71.24 (a) (b) (c)		TS-R-1 does not contain provisions for fissile material, limited moderator, controlled shipments.
	71.35 (a) (b)		TS-R-1 paras. 807 and 808 are somewhat similar, but are more specific for Type B packages and less specific for fissile material packages.
	71.38 (a) (b) (c)		TS-R-1 does not contain requirements addressing renewal of certificates and approvals.
	71.39		TS-R-1 does not contain similar requirements.
	71.41 (b) (c)		TS-R-1 does not contain similar requirements. TS-R-1 has requirements similar to 71.41(c), but these are tied to Type B(M) package designs.
	71.57		Reserved
	71.63 (a) (b)		TS-R-1 does not contain special requirements for plutonium shipments.
	71.64 (a) (b)		TS-R-1 does not contain special requirements for plutonium air shipments.
	71.65		TS-R-1 does not contain similar requirements
	71.74 (a) (b) (c)		TS-R-1 does not contain similar requirements for plutonium packages, although the Type C package performance requirements are similar (but less severe).
	71.81		TS-R-1 does not contain a similar general statement of requirement.
	71.88 (a) (b) (c)		TS-R-1 does not contain similar requirements for the air transport of plutonium.
	71.89		TS-R-1 does not contain similar requirements.
	71.91 (a) (b) (c)		TS-R-1 does not contain similar requirements
	71.93 (a)		TS-R-1 does not contain similar requirements.

	71.93 (b)		TS-R-1 does not contain similar requirements.
	71.93 (c)		TS-R-1 does not contain similar requirements.
	71.95		TS-R-1 does not contain similar requirements.
	71.97 (a) (b) (c) (d) (e) (f)		TS-R-1 does not contain similar requirements for advance notification of shipments of irradiated nuclear fuel and nuclear waste.
	71.99 (a) (b)		TS-R-1 does not contain similar requirements.
	71.100 (a) (b)		TS-R-1 does not address criminal penalties.
	71.101 (a) (b) (c) (d) (e) (f) (g)		TS-R-1 does not contain similar detailed requirements, although para. 310 requires a QA program based on standards acceptable to the Competent Authority.
	71.103 (a) (b) (c) (d) (e) (f)		TS-R-1 does not contain similar detailed requirements, although para. 310 requires a QA program based on standards acceptable to the Competent Authority.
	71.105 (a) (b) (c) (d)		TS-R-1 does not contain similar detailed requirements, although para. 310 requires a QA program based on standards acceptable to the Competent Authority.
	71.107 (a) (b) (c)		TS-R-1 does not contain similar detailed requirements, although para. 310 requires a QA program based on standards acceptable to the Competent Authority.
	71.109		TS-R-1 does not contain similar detailed requirements, although para. 310 requires a QA program based on standards acceptable to the Competent Authority.
	71.111		TS-R-1 does not contain similar detailed requirements, although para. 310 requires a QA program based on standards acceptable to the Competent Authority.
	71.113		TS-R-1 does not contain similar detailed requirements, although para. 310 requires a QA program based on standards acceptable to the Competent Authority.
	71.115 (a) (b) (c)		TS-R-1 does not contain similar detailed requirements, although para. 310 requires a QA program based on standards acceptable to the Competent Authority.
	71.117		TS-R-1 does not contain similar detailed requirements, although para. 310 requires a QA program based on standards acceptable to the Competent Authority.
	71.119		TS-R-1 does not contain similar detailed requirements, although para. 310 requires a QA

			program based on standards acceptable to the Competent Authority.
	71.121		TS-R-1 does not contain similar detailed requirements, although para. 310 requires a QA program based on standards acceptable to the Competent Authority
	71.123		TS-R-1 does not contain similar detailed requirements, although para. 310 requires a QA program based on standards acceptable to the Competent Authority.
	71.125		TS-R-1 does not contain similar detailed requirements, although para. 310 requires a QA program based on standards acceptable to the Competent Authority.
	71.127		TS-R-1 does not contain similar detailed requirements, although para. 310 requires a QA program based on standards acceptable to the Competent Authority.
	71.129 (a) (b)		TS-R-1 does not contain similar detailed requirements, although para. 310 requires a QA program based on standards acceptable to the Competent Authority.
	71.131		TS-R-1 does not contain similar detailed requirements, although para. 310 requires a QA program based on standards acceptable to the Competent Authority.
	71.133		TS-R-1 does not contain similar detailed requirements, although para. 310 requires a QA program based on standards acceptable to the Competent Authority.
	71.135		TS-R-1 does not contain similar requirements regarding quality assurance records. However, para. 310 does require that a manufacturer, consignor or user be able to demonstrate compliance with the QA program, which necessarily would include records.
	71.137		TS-R-1 does not contain similar detailed requirements, although para. 310 requires a QA program based on standards acceptable to the Competent Authority.

[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 71

RIN: 3150 - AG-71

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

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

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

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

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

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

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

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

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

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I. Background

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

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

Past NRC-IAEA Compatibility Revisions.

Recognizing that its international regulations for the safe transportation of radioactive material should be revised from time to time to reflect knowledge gained in scientific and technical advances and accumulated experience, IAEA invited Member States (the U.S. is a Member State) to submit comments and suggest changes to the regulations in 1969. As a result of this initiative, the IAEA issued revised regulations in 1973 (Regulations for the Safe Transport of Radioactive Material, 1973 Edition, Safety Series No. 6). The IAEA also decided to periodically review its transportation regulations, at intervals of about 10 years, to ensure that the regulations are kept current. In 1979, a review of IAEA's transportation regulations was initiated that resulted in the publication of revised regulations in 1985 (Regulations for the Safe Transport of Radioactive Material, 1985 Edition, Safety Series No. 6).

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

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

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

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

Scope of 10 CFR Part 71 Rulemaking.

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

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

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

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

II. Summary of Public Comments

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

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

August 10, 2000 Meeting.

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

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

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

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

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

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

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

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

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

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

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

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

September 20, 2000 Meeting,

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

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

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

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

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

September 26, 2000 Meeting.

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

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

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

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

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

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

Comments received on the website and by mail.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

III. Discussion

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

A. TS-R-1 Compatibility Issues

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

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

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

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

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

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

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

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

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

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

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

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

Affected Sections. None (not adopted).

Issue 2. Radionuclide Exemption Values

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Affected Sections. 71.10, 71.88, Appendix A.

Issue 3. Revision of A_1 and A_2

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

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

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

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

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

Additional comments were received concerning the A_1 and A_2 values for molybdenum-99 and californium-252. Currently, in Part 71, the A_1 and A_2 values for these radionuclides are: molybdenum-99: A_1: 0.6 TBq (16.2 Ci); A_2: 0.5 TBq (13.5 Ci), and californium-252: A_1: 0.1TBq (2.7 Ci); A_2: 1.0x10^{-3} TBq (2.7 E-2 Ci). Further, Appendix A, Table A-1, the A_2 value for molybdenum-99 has a footnote that indicates for domestic use, the A_2 value is 0.74 TBq (20 Ci). The values from TS-R-1 for these radionuclides are: molybdenum-99: A_1: 1 TBq (27 Ci); A_2: 0.6 TBq (16.2 Ci), and californium-252: A_1: 5.0x10^{-2} TBq (1.35 Ci); A_2: 3.0x10^{-3} TBq (0.08 Ci). Pharmaceutical industry representatives indicated that a change to the new (lower) A_2 value for molybdenum-99 (16.2 Ci vs 20 Ci) would result in a significant increase in the number of packagings shipped, and in occupational doses. DOT is proposing to retain the current exception for molybdenum-99 for domestic commerce, and NRC also believes the current exception for this radionuclide should be retained.

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

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

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

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

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

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

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

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

Affected Sections. Appendix A**.**

Issue 4. Uranium Hexafluoride Package Requirements

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

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

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

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

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

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

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

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

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

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

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

Affected Sections. 71.55.

Issue 5. Introduction of the Criticality Safety Index Requirements

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

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

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

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

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

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

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

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

Issue 6. Type C Packages and Low Dispersible Material

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

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

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

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

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

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

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

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

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

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

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

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

Affected Sections. None (not adopted).

Issue 7. Deep Immersion Test

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

In its September 28, 1995 (60 FR 50264), rulemaking for Part 71 compatibility with the 1985 edition of Safety Series No. 6, the NRC addressed the new Safety Series No. 6 requirement for spent fuel packages by adding § 71.61, "Special requirements for irradiated nuclear fuel shipments." Currently, § 71.61 is more conservative than Safety Series No. 6 with respect to irradiated fuel package design requirements. It requires that a package for irradiated nuclear fuel with activity greater than 37 PBq (10^6 Ci) must be designed so that its undamaged containment system can withstand an external water pressure of 2 MPa (290 psi) for a period of not less than one hour *without collapse, buckling, or inleakage of water*. The conservatism lies in the test criteria of no collapse, buckling, or inleakage as compared to the *"no rupture"* criteria found in Safety Series No. 6 and TS-R-1. The draft advisory document for TS-R-1 (TS-G-1.1, paragraphs 657.1 to 657.7) recognizes that leakage into the package and subsequent leakage from the package is possible while still meeting the IAEA requirement.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Affected Sections. 71.41, 71.51, 71.61.

Issue 8. Grandfathering Previously Approved Packages

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Affected Sections. 71.13.

Issue 9. Changes to Various Definitions

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

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

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

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

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

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

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

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

Affected Sections. 71.4.

Issue 10. Crush Test for Fissile Material Package Design

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

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

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

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

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

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

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

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

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

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

<u>**NRC Proposed Position.**</u> The NRC proposes to adopt the requirement for a crush test for fissile material packages, and eliminate the 1000 A_2 criterion. However, because there is

no new information that addresses concerns from the previous rulemaking regarding the difference in test requirements between Part 71 and Safety Series No. 6, the NRC proposes not to change the testing sequence nor to change the drop and crush test requirements in this revision.

Affected Sections. 71.73.

Issue 11. Fissile Material Package Design for Transport by Aircraft

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

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

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

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

Type A fissile package by air must:

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

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

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

Type B fissile package by air must:

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

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

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

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

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

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

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

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

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

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

Affected Sections. 71.55.

B. NRC-Initiated Issues

Issue 12. Special Package Authorizations

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

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

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

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

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

PGE acknowledged that the TRVP could not meet Type B regulations and applied for a Type B package CoC for the TRVP itself, either under § 71.41(c), "Demonstration of compliance," or § 71.8, "Specific exemptions." Section 71.41(c) provides that "Environmental and test conditions different from those specified in §§ 71.71 and 71.73 may be approved by the Commission if the controls proposed to be exercised by the shipper are demonstrated to be adequate to provide equivalent safety of the shipment." Section 71.41(c) has been used to accommodate minor deviations in test environments (e.g., initial temperatures), and was not intended to be used to establish new test conditions for Type B packages. The use of this provision in the Trojan case would essentially have resulted in establishing new (and less rigorous) Type B test conditions that the Trojan vessel could meet. A CoC for a Type B package could then have been issued for Trojan, but the level of performance reflected in that Certificate would have been significantly different from that in other Type B Certificates. NRC decided against using § 71.41(c), and to use the § 71.8 exemption provision - the only other option available.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Affected Sections. 71.41.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Issue 14. Adoption of ASME Code

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

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

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

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

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

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

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

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

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

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

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

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

Affected Sections.   None (not adopted).

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Issue 16. Fissile Material Exemptions and General License Provisions

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Table 16-1

Summary of Recommended Changes in NUREG/CR-5342

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

The petitioner further stated that:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Affected Sections. 71.63.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Affected Sections. None (not adopted).

Issue 19. Modifications of Event Reporting Requirements

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

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

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

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

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

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

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

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

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

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

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

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

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

Affected Sections. 71.95.

IV. Section-By-Section Analysis

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

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

Subpart A—General Provisions

10 CFR 71.0 Purpose and scope.

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

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

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

10 CFR 71.1 Communications and reports.

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

10 CFR 71.2 Interpretations.

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

10 CFR 71.3 Requirement for license.

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

10 CFR 71.4 Definitions.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

10 CFR 71.5 Transportation of licensed material.

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

10 CFR 71.6 Information collection requirements: OMB approval.

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

10 CFR 71.7 Completeness and accuracy of information.

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

10 CFR 71.8 Deliberate misconduct.

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

§ 10 CFR 71.9 Employee protection.

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

10 CFR 71.10 Public Inspection of Application.

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

10 CFR 71.11 (Reserved)

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

Subpart B - Exemptions

10 CFR 71.12 Specific exemptions.

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

10 CFR 71.13 Exemption of physicians.

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

10 CFR 71.14 Exemption for low-level materials.

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

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

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

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

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

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

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

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

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

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

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

10 CFR 71.16 (Reserved)

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

Subpart C—General Licenses

§ 71.17 General license: NRC-approved package.

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

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

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

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

10 CFR 71.19 Previously approved package.

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

10 CFR 71.20 General license: DOT specification container.

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

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

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

10 CFR 71.22 General license: Fissile material.

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

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

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

Paragraph (b) would remain unchanged.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

10 CFR 71.24 (Reserved)

10 CFR 71.25 (Reserved)

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

Subpart D—Application for Package Approval

10 CFR 71.41 Demonstration of Compliance.

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

10 CFR 71.51 Additional Requirements for Type B Packages.

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

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

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

10 CFR 71.55 General requirements for fissile material packages.

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

10 CFR 71.59 Standards for arrays of fissile material packages.

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

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

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

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

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

10 CFR 71.63 Special requirement for plutonium shipments.

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

Paragraph (b) would be removed.

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

10 CFR 71.73 Hypothetical accident conditions.

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

10 CFR 71.88 Air transport of plutonium.

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

Subpart G—Operating Controls and Procedures

10 CFR 71.91 Records.

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

10 CFR 71.93 Inspection and tests.

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

10 CFR 71.95 Reports.

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

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

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

10 CFR 71.100 Criminal penalties.

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

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

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

Subpart H—Quality Assurance

10 CFR 71.101 Quality assurance requirements.

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

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

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

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

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

10 CFR 71.103 Quality assurance organization.

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

10 CFR 71.105 Quality assurance program.

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

10 CFR 71.107 Package design control.

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

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

10 CFR 71.109 Procurement document control.

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

10 CFR 71.111 Instructions, procedures, and drawings.

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

10 CFR 71.113 Document control.

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

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

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

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

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

10 CFR 71.119 Control of special processes.

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

10 CFR 71.121 Internal inspection.

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

10 CFR 71.123 Test control.

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

10 CFR 71.125 Control of measuring and test equipment.

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

10 CFR 71.127 Handling, storage, and shipping control.

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

10 CFR 71.129 Inspection, test, and operating status.

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

Paragraph (b) would remain unchanged.

10 CFR 71.131 Nonconforming materials, parts, or components.

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

10 CFR 71.133 Corrective Action.

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

10 CFR 71.135 Quality assurance records.

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

10 CFR 71.137 Audits.

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

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

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

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

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

10 CFR 71.153 Contents of application.

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

10 CFR 71.155 Package description.

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

10 CFR 71.157 Package evaluation.

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

10 CFR 71.159 Quality assurance.

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

10 CFR 71.161 Requirement for additional information.

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

10 CFR 71.163 Issuance of an NRC certificate of compliance.

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

10 CFR 71.165 Conditions for package reapproval.

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

10 CFR 71.167 Application to amend a certificate of compliance.

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

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

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

10 CFR 71.171 Inspections and tests.

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

10 CFR 71.173 Recordkeeping and reports.

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

10 CFR 71.175 Changes.

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

10 CFR 71.177 Safety analysis report updating.

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

Appendix A to Part 71 — Determination of A_1 and A_2

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

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

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

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

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

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

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

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

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

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

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

V. Criminal Penalties

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

The following is a list of substantive rule sections being revised or added in this rulemaking: §§ 71.1, 71.3, 71.5, 71.8, 71.9, 71.12, 71.13, 71.14, 71.15, 71.17, 71.18, 71.19, 71.20, 71.21, 71.22, 71.23, 71.61, 71.63, 71.88, 71.91, 71.93, 71.95, 71.101, 71.103, 71.105, 71.107, 71.109, 71.111, 71.113, 71.115, 71.117, 71.119, 71.121, 71.123, 71.125, 71.127, 71.129, 71.131, 71.133, 71.135, 71.137, 71.171, 71.173, 71.175, and 71.177

VI. Issues of Compatibility for Agreement States

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

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

Part 71 - PACKAGING AND TRANSPORTATION OF RADIOACTIVE MATERIAL

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

VII. Plain Language

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

VIII. Voluntary Consensus Standards

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

IX. Environmental Assessment: Finding of No Significant Environmental Impact

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

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

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

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

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

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

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

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

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

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

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

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

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

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

The Commission's “Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes,'' NUREG-0170[18], dated December 1977, is NRC's generic EIS, covering all types of radioactive material transportation by all modes (road, rail, air, and water). From the Commission's latest survey of radioactive material shipments and their characteristics, “Transport of Radioactive Material in the United States,'' SAND 84-7174, April 1985, the NRC concluded that current radioactive material shipments are not so different from those evaluated in NUREG-0170 as to invalidate the results or conclusions of that EIS. Environmental assessment of the impacts associated with this rulemaking are evaluated in “Environmental Assessment of Major Revision to Packaging and Transportation of Radioactive Material Regulations (10 CFR Part 71),'' dated February 2000.

NUREG-0170 established the nonaccident related radiation exposures associated with transportation of radioactive material in the United States as 98 person-Sv (9800 person-rem) which, based on the conservative linear radiation dose hypothesis, resulted in a maximum of 1.7 genetic effects and 1.2 latent cancer effects per year. More than half this impact resulted from shipment of medical-use radioactive materials. Accident related impacts were established

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[18] Copies of NUREG-0170 may be purchased from the Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082. Copies are also available from the National Technical Information Service, 5285 Port Royal Road, Springfield, VA 22161. A copy is also available for inspection and copying for a fee in the NRC Public Document Room, 11555 Rockville Pike, Room 0-1F21, Rockville, MD.

at a maximum of one genetic effect and one latent cancer fatality for 200 years of transporting radioactive materials. The principal nonradiological impacts were found to be two injuries per year, and less than one accidental death per 4 years. In contrast, nonaccident related radiation exposures and accident related impacts associated with this rulemaking would not change from the impact of the current Part 71 requirements (i.e., no increase or decrease). Nonradiological traffic injuries and nonradiological traffic deaths would not change. These impacts are judged to be insignificant compared with the baseline impacts established in NUREG-0170.

The environmental assessment and finding of no significant impact on which this determination is based are available, for inspection, at the NRC Public Document Room, 11555 Rockville Pike, Room 0-1F21, Rockville, MD. The environmental assessment is also available on the NRC rulemaking website.

X. Paperwork Reduction Act Statement

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

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

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

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

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

XI. Public Protection Notification

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

XII. Regulatory Analysis

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

XIII. Regulatory Flexibility Act Certification

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

XIV. Backfit Analysis

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

List of Subjects in 10 CFR Part 71

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

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; and 5 U.S.C. 553, the Commission is proposing to revise 10 CFR Part 71 as follows:

PART 71 -- PACKAGING AND TRANSPORTATION OF RADIOACTIVE MATERIAL

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

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

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

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

Sec.

71.0 Purpose.

71.1 Communications and records.

71.2 Interpretations.

71.3 Requirement for license.

71.4 Definitions.

71.5 Transportation of licensed material.

71.6 Information collection requirements: OMB approval.

71.7 Completeness and accuracy of information.

71.8 Deliberate misconduct.

71.9 Employee protection.

71.10 Public inspection of application.

71.11 [Reserved]

Subpart A - General Provisions

§ 71.0 Purpose and scope.

(a) This part establishes --

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

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

(b) The packaging and transport of licensed material are also subject to other parts of this chapter (e.g., 10 CFR parts 20, 21, 30, 40, 70, and 73) and to the regulations of other agencies (e.g., the U.S. Department of Transportation (DOT) and the U.S. Postal Service[19]) having jurisdiction over means of transport. The requirements of this part are in addition to, and not in substitution for, other requirements.

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

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

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

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

package to be used satisfies the package approval standards contained in subpart E of this part, as related to the tests of subpart F of this part.

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

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

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

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

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

§ 71.1 Communications and records.

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

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

§71.2 Interpretations.

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

§71.3 Requirement for license.

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

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

(b) Transport licensed material.

§71.4 Definitions.

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

A_1 means the maximum activity of special form radioactive material permitted in a Type A package. This value is either listed in Appendix A of this part, Table A-1, or may be derived in accordance with the procedures prescribed in Appendix A of this part.

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

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

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

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

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

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

Conveyance means:

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

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

(3) For transport by aircraft any aircraft.

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

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

DOT means the U.S. Department of Transportation.

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

Fissile material means the radionuclides uranium-233, uranium-235, plutonium-239, and plutonium-241, or any combination of these radionuclides. Fissile material means the fissile nuclides themselves, not material containing fissile nuclides. Unirradiated natural uranium and depleted uranium and natural uranium or depleted uranium, that has been irradiated in thermal reactors only, are not included in this definition. Certain exclusions from fissile material controls are provided in § 71.15.

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

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

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

(1) *LSA - I*.

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

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

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

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

(2) *LSA - II.*

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

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

(3) *LSA - III*. Solids (e.g., consolidated wastes, activated materials) that satisfy the requirements of § 71.77, in which:

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

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

(iii) The average specific activity of the solid does not exceed 2 x10^{-3} A_2/g.

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

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

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

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

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

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

(1) *Fissile material* package or Type *AF package, Type BF package, Type B(U)F package,* or *Type B(M)F package* means a fissile material packaging together with its fissile material contents.

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

(3) *Type B package* means a Type B packaging together with its radioactive contents. On approval, a Type B package design is designated by NRC as B(U) unless the package has a maximum normal operating pressure of more than 700 kPa (100 lbs/in^2) gauge or a pressure relief device that would allow the release of radioactive material to the environment under the tests specified in § 71.73 (hypothetical accident conditions), in which case it will receive a designation B(M). B(U) refers to the need for unilateral approval of international shipments; B(M) refers to the need for multilateral approval of international shipments. There is no distinction made in how packages with these designations may be used in domestic transportation. To determine their distinction for international transportation, see DOT regulations in 49 CFR Part 173. A Type B package approved before September 6, 1983, was designated only as Type B. Limitations on its use are specified in § 71.19.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(1) SCO - I: A solid object on which:

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

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

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

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

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

(ii) The fixed contamination on the accessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 8x10^5 Bq/cm^2 (20 microcuries/cm^2) for beta and gamma and low toxicity alpha emitters, or 8x10^4 Bq/cm^2 (2 microcuries/cm^2) for all other alpha emitters; and

(iii) The nonfixed contamination plus the fixed contamination on the inaccessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 8x10^5 Bq/cm^2 (20 microcuries/cm^2) for beta and gamma and low toxicity alpha emitters, or 8x10^4 Bq/cm^2 (2 microcuries/cm^2) for all other alpha emitters.

Transport index (TI) means the dimensionless number (rounded up to the next tenth) placed on the label of a package, to designate the degree of control to be exercised by the carrier during transportation. The transport index is the number determined by multiplying the maximum radiation level in millisievert (mSv) per hour at 1 meter (3.3 ft) from the external surface of the package by 100 (equivalent to the maximum radiation level in millirem per hour at 1 meter (3.3 ft)).

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

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

Uranium -- natural, depleted, enriched

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

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

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

§71.5 Transportation of licensed material.

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

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

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

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

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

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

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

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

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

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

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

(ii) Air -- 49 CFR part 175.

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

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

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

§ 71.6 Information collection requirements: OMB approval.

(a) The Nuclear Regulatory Commission has submitted the information collection requirements contained in this part to the Office of Management and Budget (OMB) for approval as required by the Paperwork Reduction Act (44 U.S.C. 3501 et seq.). The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number. OMB has approved the information collection requirements contained in this part under control number 3150-0008.

(b) The approved information collection requirements contained in this part appear in §§ 71.5, 71.7, 71.8, 71.12, 71.13, 71.17, 71.18, 71.19, 71.20, 71.31, 71.33, 71.35, 71.37, 71.38, 71.39, 71.41, 71.47, 71.85, 71.87, 71.89, 71.90, 71.91, 71.93, 71.95, 71.97, 71.101, 71 103, 71.105, 71.107, 71.109, 71.111, 71.113, 71.115, 71.117, 71.119, 71.121, 71.123, 71.125, 71.127, 71.129, 71.131, 71.133, 71.135, 71.137, 71.151, 71.153, 71.155, 71.157, 71.159, 71.161, 71.165, 71.167, 71.171, 71.173, 71.175, 71.177, and Appendix A.

§ 71.7 Completeness and accuracy of information.

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

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

§ 71.8 Deliberate misconduct.

(a) This section applies to any--

(1) Licensee;

(2) Certificate holder;

(3) Quality assurance program approval holder;

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

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

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

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

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

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

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

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

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

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

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

§ 71.9 Employee protection.

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

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

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

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

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

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

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

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

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

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

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

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

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

(3) Other enforcement action.

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

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

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

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

§ 71.10 Public inspection of application.

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

§ 71.11 [Reserved.]

Subpart B - Exemptions

§ 71.12 Specific exemptions.

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

§ 71.13 Exemption of physicians.

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

§ 71.14 Exemption for low-level materials.

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

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

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

(b) A licensee is exempt from all requirements of this part, other than §§ 71.5 and 71.88, with respect to shipment or carriage of the following packages, provided the packages do not contain any fissile material, or the material is exempt from classification as fissile material under § 71.15:

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

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

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

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

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

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

§ 71.15 Exemption from classification as fissile material.

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

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

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

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

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

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

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

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

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

§ 71.16 [Reserved]

Sec.

Subpart C - General Licenses

§ 71.17 General license: NRC-approved package.

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

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

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

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

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

(3) Submits in writing to the NRC, before the licensee's first use of the package, the licensee's name and license number and the package identification number specified in the package approval. A licensee shall submit this information in accordance with § 71.1.

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

(e) For a Type B or fissile material package, the design of which was approved by NRC before April 1, 1996, the general license is subject to the additional restrictions of § 71.19.

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

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

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

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

(1) Has a copy of the CoC, or other approval, of the Type B(DP) package; a copy of the updated final safety analysis report for the package; and the drawings and other documents referenced in the CoC, or other approval, relating to the use and maintenance of the packaging and to the actions to be taken before shipment;

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

(3) Submits in writing to the NRC, before the licensee's first use of the package, the licensee's name and license number and the package identification number specified in the package approval. A licensee shall submit this information in accordance with § 71.1.

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

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

§71.19 Previously approved package.

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

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

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

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

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

(b) A Type B(U) package, a Type B(M) package, or a fissile material package, previously approved by the NRC but without the designation "-85" in the identification number of the NRC Certificate of Compliance, may be used under the general license of § 71.17 with the following additional conditions:

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

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

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

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

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

(2) After December 31, 2003, a package used for a shipment to a location outside the United States is subject to multilateral approval as defined in DOT regulations at 49 CFR 173.403.

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

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

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

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

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

§ 71.20 General license: DOT specification container.

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

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

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

(1) Has a copy of the specification; and

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

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

§ 71.21 General License: Use of foreign approved package.

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

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

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

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

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

(2) Complies with the terms and conditions of the certificate and revalidation, and with the applicable requirements of subparts A, G, and H of this part. With respect to the quality assurance provisions of subpart H of this part, the licensee is exempt from design, construction, and fabrication considerations.

§ 71.22 General license: Fissile material.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(iv) substances having a moderating effectiveness (i.e., an average hydrogen density greater than H_2O) [e.g., certain hydrocarbon oils or plastics] are present in any form, except as polyethylene used for packing or wrapping.

TABLE 71-1. MASS LIMITS FOR GENERAL LICENSE PACKAGES CONTAINING MIXED QUANTITIES OF FISSILE MATERIAL OR URANIUM-235 OF UNKNOWN ENRICHMENT PER § 71.22(e)

Fissile material	Fissile material mass mixed with moderating substances having an average hydrogen density less than or equal to H_2O. (grams)	Fissile material mass mixed with moderating substances having an average hydrogen density greater than H_2O.[a] (grams)
^{235}U (*X*)..	60	38
^{233}U (*Y*)..	43	27
^{239}Pu or ^{241}Pu (Z)...	37	24

[a] When mixtures of moderating substances are present, the lower mass limits shall be used if more than 15 percent of the moderating substance has an average hydrogen density greater than H_2O.

TABLE 71-2 — MASS LIMITS FOR GENERAL LICENSE PACKAGES CONTAINING URANIUM-235 OF KNOWN ENRICHMENT PER § 71.22(e)

Uranium enrichment in weight percent of ^{235}U not exceeding	Fissile material mass of ^{235}U (X). (grams)
24	60
20	63
15	67
11	72
10	76
9.5	78
9	81
8.5	82
8	85
7.5	88
7	90
6.5	93
6	97
5.5	102
5	108
4.5	114
4	120
3.5	132
3	150
2.5	180
2	246
1.5	408
1.35	480
1	1,020
0.92	1.800

§ 71.23 General license: Plutonium-beryllium special form material.

(a) A general license is issued to any licensee of the Commission to transport fissile material in the form of plutonium-beryllium (Pu-Be) special form sealed sources, or to deliver Pu-Be sealed sources to a carrier for transport, if the material is shipped in accordance with this section. This material need not be contained in a package which meets the standards of subparts E and F of this part; however, the material must be contained in a Type A package. The Type A package must also meet the DOT requirements of 49 CFR 173.417(a).

(b) The general license applies only to a license who has a quality assurance program approved by the Commission as satisfying the provisions of subpart H of this part.

(c)The general license applies only when a package's contents:

(1) Contain less than a Type A quantity of material; and

(2) Contain less than 1000 g of plutonium, provided that: plutonium-239, plutonium-241, or any combination of these radionuclides, constitutes less than 240 g of the total quantity of plutonium in the package.

(d) The general license applies only to packages labeled with a CSI which:

(1) has been determined in accordance with paragraph (e) of this section;

(2) has a value less than or equal to 100.0; and

(3) for a shipment of multiple fissile material packages, contained in a single conveyance, the sum of the CSI must meet the requirements of § 71.59(c).

(e)(1) The value for the CSI must be greater than or equal to the number calculated by the following equation:

$$\mathrm{CSI} = 10\left|\frac{\text{grams of } {}^{239}\mathrm{Pu} + \text{grams of } {}^{241}\mathrm{Pu}}{24}\right|\text{; and}$$

(2) The calculated CSI must be rounded up to the first decimal place.

§ 71.24 [Reserved]

§ 71.25 [Reserved]

3. In § 71.41, paragraph (a) is revised and a new paragraph (d) is added to read as follows:

§ 71.41 Demonstration of compliance.

(a) The effects on a package of the tests specified in § 71.71 (“Normal conditions of transport”), and the tests specified in § 71.73 (“Hypothetical accident conditions”), and § 71.61 (“Special requirements for Type B packages containing more than 10^5 A_2 ”), must be evaluated by subjecting a specimen or scale model to a specific test, or by another method of demonstration acceptable to the Commission, as appropriate for the particular feature being considered.

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(d) Packages for which compliance with the other provisions of these regulations is impracticable shall not be transported except under special package authorization. Provided the applicant demonstrates that compliance with the other provisions of the regulations is impracticable and that the requisite standards of safety established by these regulations have been demonstrated through means alternative to the other provisions, a special package authorization may be approved for one-time shipments. The applicant shall demonstrate that the overall level of safety in transport for these shipments is at least equivalent to that which would be provided if all the applicable requirements had been met.

4. In § 71.51, the introductory text of paragraph (a) is revised, and a new paragraph (d) is added to read as follows:

§ 71.51 Additional requirements for Type B packages.

(a) A Type B package, in addition to satisfying the requirements of §§ 71.41 through 71.47, must be designed, constructed, and prepared for shipment so that under the tests specified in:

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(d) For packages which contain radioactive contents with activity greater than 10^5 A_2, the requirements of § 71.61 must be met. Except for Type B(DP) packages, a package must meet the requirements of § 71.61, if the package contents contain radioactive material with an activity greater than 10^5 A_2.

5. Section 71.53 is removed and reserved.

§ 71.53 [Reserved]

6. In § 71.55, paragraph (b) is revised, and new paragraphs (f) and (g) are added to read as follows:

§ 71.55 General requirements for fissile material packages.

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(b) Except as provided in paragraph (c) or (g) of this section, a package used for the shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system, or liquid contents were to leak out of the containment system so that, under the following conditions, maximum reactivity of the fissile material would be attained:

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(f) For fissile material package designs to be transported by air:

(1) The package must be designed and constructed, and its contents limited so that it would be subcritical, assuming reflection by 20 cm (7.9 in) of water but no water inleakage, when subjected to sequential application of:

(i) The free drop test in §71.73(c)(1);

(ii) The crush test in §71.73(c)(2);

(iii) A puncture test, for packages of 250 kg or more, consisting of a free drop of the specimen through a distance of 3 m (120 in) in a position for which maximum damage is expected at the conclusion of the test sequence, onto the upper end of a solid, vertical, cylindrical, mild steel probe mounted on an essentially unyielding, horizontal surface. The probe must be 20 cm (7.9 in) in diameter, with the striking end forming the frustum of a right circular cone with the dimensions of 30 cm height, 2.5 cm top diameter, and a top edge rounded to a radius of not more than 6 mm (0.25 in). For packages less than 250 kg, the puncture test must be the same, except that a 250 kg probe must be dropped onto the specimen which must be placed on the surface; and

(iv) The thermal test in §71.73(c)(4), except that the duration of the test must be 60 minutes.

(2) The package must be designed and constructed, and its contents limited so that it would be subcritical, assuming reflection by 20 cm (7.9 in) of water but no water inleakage, when subjected to an impact on an unyielding surface at a velocity of 90 m/s normal to the surface, at such orientation so as to result in maximum damage. A separate, undamaged specimen can be used for this evaluation.

(3) Allowance may not be made for the special design features in paragraph (c) of this section, unless water leakage into or out of void spaces is prevented following application of the

tests in paragraphs (f)(1) and (f)(2) of this section, and subsequent application of the immersion test in § 71.73(c)(5).

(g) Packages containing uranium hexafluoride only are excepted from the requirements of paragraph (b) of this section provided that:

(1) Following the tests specified in § 71.73 (“Hypothetical accident conditions”), there is no physical contact between the valve body and any other component of the packaging, other than at its original point of attachment, and the valve remains leak tight;

(2) There is an adequate quality control in the manufacture, maintenance and repair of packagings;

(3) Each package is tested to demonstrate closure before each shipment; and

(4) The uranium is enriched to not more than 5 weight percent uranium-235.

7. In § 71.59, paragraphs (b) and (c) are revised to read as follows:

§ 71.59 Standards for arrays of fissile material packages.

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(b) The CSI must be determined by —

(1) Dividing the number 50 by the value of "N" derived using the procedures specified in paragraph (a) of this section. The value of the CSI may be zero provided that an unlimited number of packages is subcritical, such that the value of "N" is effectively equal to infinity under the procedures specified in paragraph (a) of this section. Any CSI greater than zero must be rounded up to the first decimal place; or

(2) The number calculated by the equations in either §§ 71.22 or 71.23, for packages shipped under the general license provisions of §§ 71.22 or 71.23.

(c) Where a fissile material package is assigned a CSI value —

(1) Less than or equal to 10.0, that package may be shipped by any carrier, and that carrier must provide adequate criticality control by limiting the sum of the CSIs to less than 50.0 in a nonexclusive use vehicle, and to less than 100.0 in an exclusive use vehicle.

(2) Greater than 10.0, that package must be shipped by exclusive use vehicle or other shipper controlled system specified by DOT for fissile material packages. The shipper must provide adequate criticality control by limiting the sum of the CSIs to less than 100.0 in an exclusive use vehicle.

8. Section 71.61 is revised to read as follows:

§ 71.61 Special requirements for Type B packages containing more than 10^5 A_2.

A Type B package containing more than 10^5 A_2 must be so designed that its undamaged containment system can withstand an external water pressure of 2 MPa (290 psi) for a period of not less than one hour without collapse, buckling, or inleakage of water.

9. Section 71.63 is revised to read as follows:

§ 71.63 Special requirement for plutonium shipments.

Shipments containing plutonium must be made with the contents in solid form, if the contents contain greater than 0.74 TBq (20 Ci) of plutonium.

10. In § 71.73, paragraph (c)(2) is revised to read as follows:

§ 71.73 Hypothetical accident conditions.

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(c) ★ ★ ★

(2) *Crush*. Subjection of the specimen to a dynamic crush test by positioning the specimen on a flat, essentially unyielding horizontal surface so as to suffer maximum damage by the drop of a 500-kg (1100-lb) mass from 9 m (30 ft) onto the specimen. The mass must consist of a solid mild steel plate 1 m (40 in) by 1 m and must fall in a horizontal attitude. The crush test is required only when the specimen has a mass not greater than 500 kg (1100 lbs), an overall density not greater than 1000 kg/m^3 (62.4 lbs/ft^3) based on external dimension, and radioactive contents greater than 1000 A_2 not as special form radioactive material. For packages containing fissile material, the radioactive contents greater than 1000 A_2 criterion does not apply.

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11. In § 71.88, paragraph (a)(2) is revised to read as follows:

§ 71.88 Air transport of plutonium

(a) ★ ★ ★

(2) The plutonium is contained in a material in which the specific activity is less than or equal to the activity concentration values for plutonium specified in Appendix A, Table A-2 of this part, and in which the radioactivity is essentially uniformly distributed; or

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12. In § 71.91, paragraphs (b) and (c) are revised, and a new paragraph (d) is added to read as follows:

§ 71.91 Records.

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(b) Each certificate holder shall maintain, for a period of 3 years after the life of the packaging to which they apply, records identifying the packaging by model number, serial number, and date of manufacture.

(c) The licensee, certificate holder, and an applicant for a CoC, shall make available to the Commission for inspection, upon reasonable notice, all records required by this part. Records are only valid if stamped, initialed, or signed and dated by authorized personnel or otherwise authenticated.

(d) The licensee, certificate holder , and an applicant for a CoC, shall maintain sufficient written records to furnish evidence of the quality of packaging. The records to be maintained include results of the determinations required by § 71.85; design, fabrication, and assembly records, results of reviews, inspections, tests, and audits; results of monitoring work performance and materials analyses; and results of maintenance, modification and repair activities. Inspection, test, and audit records must identify the inspector or data recorder, the type of observation, the results, the acceptability and the action taken in connection with any deficiencies noted. These records must be retained for 3 years after the life of the packaging to which they apply.

13. Section 71.93 is revised to read as follows:

§ 71.93 Inspection and tests.

(a) The licensee, certificate holder, and applicant for a CoC shall permit the Commission, at all reasonable times, to inspect the licensed material, packaging, premises, and facilities in which the licensed material or packaging is used, provided, constructed, fabricated, tested, stored, or shipped.

(b) The licensee, certificate holder, and applicant for a CoC shall perform, and permit the Commission to perform, any tests the Commission deems necessary or appropriate for the administration of the regulations in this chapter.

(c) The certificate holder and applicant for a CoC shall notify the NRC, in accordance with § 71.1, 45 days in advance of starting fabrication of the first packaging under a CoC. This paragraph applies to any packaging used for the shipment of licensed material which has either—

(1) A decay heat load in excess of 5 kW; or

(2) A maximum normal operating pressure in excess of 103 kPa (15 lbf/in^2) gauge.

14. Section 71.95 is revised to read as follows:

§ 71.95 Reports.

(a) The licensee, after requesting the certificate holder's input, shall submit a written report to the Commission of—

(1) Instances in which there is a significant reduction in the effectiveness of any NRC-approved Type B or Type A(F) packaging during use; or

(2) Details of any defects with safety significance in any NRC-approved Type B or fissile material packaging, after first use.

(b) The licensee shall submit a written report to the Commission of instances in which the conditions in the certificate of compliance were not followed during a shipment.

(c) *Written report*. Each licensee shall submit, in accordance with § 71.1, a written report required by paragraphs (a) or (b) of this section within 60 days of the event or discovery of the event. The licensee shall also provide a copy of each report submitted to the NRC to the applicable certificate holder. Written reports prepared pursuant to other regulations may be submitted to fulfill this requirement if the reports contain all the necessary information, and the appropriate distribution is made. These written reports must include the following:

(1) A brief abstract describing the major occurrences during the event, including all component or system failures that contributed to the event and significant corrective action taken or planned to prevent recurrence.

(2) A clear, specific, narrative description of the event that occurred so that knowledgeable readers conversant with the requirements of Part 71, but not familiar with the design of the packaging, can understand the complete event. The narrative description must include the following specific information as appropriate for the particular event.

(i) Status of components, or systems that were inoperable at the start of the event and that contributed to the event;

(ii) Dates and approximate times of occurrences;

(iii) The cause of each component or system failure or personnel error, if known;

(iv) The failure mode, mechanism, and effect of each failed component, if known;

(v) A list of systems or secondary functions that were also affected for failures of components with multiple functions;

(vi) The method of discovery of each component or system failure or procedural error;

(vii) For each human performance related root cause, a discussion of the cause(s) and circumstances;

(viii) The manufacturer and model number (or other identification) of each component that failed during the event; and

(ix) For events occurring during use of a packaging, the quantities and chemical and physical form(s) of the package contents.

(3) An assessment of the safety consequences and implications of the event. This assessment must include the availability of other systems or components that could have performed the same function as the components and systems that failed during the event.

(4) A description of any corrective actions planned as a result of the event, including the means employed to repair any defects, and actions taken to reduce the probability of similar events occurring in the future.

(5) Reference to any previous similar events involving the same packaging that are known to the licensee or certificate holder.

(6) The name and telephone number of a person within the licensee's organization who is knowledgeable about the event and can provide additional information.

(7) The extent of exposure of individuals to radiation or to radioactive materials without identification of individuals by name.

(d) *Report legibility*. The reports submitted by licensees and/or certificate holders under this section must be of sufficient quality to permit reproduction and micrographic processing.

15. In § 71.100, paragraph (b) is revised to read as follows:

§ 71.100 Criminal penalties.

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(b) The regulations in part 71 that are not issued under §§ 161b, 161i, or 161o for the purposes of § 223 are as follows: §§ 71.0, 71.2, 71.4, 71.6, 71.7, 71.10, 71.31, 71.33, 71.35, 71.37, 71.38, 71.39, 71.40, 71.41, 71.43, 71.45, 71.47, 71.51, 71.55, 71.59, 71.65, 71.71, 71.73, 71.74, 71.75, 71.77, 71.99, 71.100, and 71.151 through 71.169.

16. Subpart H to Part 71 is revised to read as follows:

Sec.

Subpart H—Quality Assurance

§ 71.101 Quality assurance requirements.

(a) *Purpose*. This subpart describes quality assurance requirements applying to design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of components of packaging that are important to safety. As used in this subpart, "quality assurance" comprises all those planned and systematic actions necessary to provide adequate confidence that a system or component will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to control of the physical characteristics and quality of the material or component to predetermined requirements. The licensee, certificate holder, and applicant for a CoC are responsible for the quality assurance requirements as they apply to design, fabrication, testing, and modification of packaging. Each licensee is responsible for the quality assurance provision which applies to its use of a packaging for the shipment of licensed material subject to this subpart.

(b) *Establishment of program*. Each licensee, certificate holder, and applicant for a CoC shall establish, maintain, and execute a quality assurance program satisfying each of the applicable criteria of §§ 71.101 through 71.137 and satisfying any specific provisions that are applicable to the licensee's activities including procurement of packaging. The licensee, certificate holder, and applicant for a CoC shall execute the applicable criteria in a graded approach to an extent that is commensurate with the quality assurance requirement's importance to safety.

(c) *Approval of program*. (1) Before the use of any package for the shipment of licensed material subject to this subpart, each licensee shall obtain Commission approval of its quality assurance program. Each licensee shall, in accordance with § 71.1, file a description of its quality assurance program, including a discussion of which requirements of this subpart are applicable and how they will be satisfied.

(2) Before the fabrication, testing, or modification of any package for the shipment of licensed material subject to this subpart, each licensee, certificate holder, or applicant for a CoC shall obtain Commission approval of its quality assurance program. Each certificate holder or applicant for a CoC shall, in accordance with § 71.1, file a description of its quality assurance program, including a discussion of which requirements of this subpart are applicable and how they will be satisfied.

(d) *Existing package designs*. The provisions of this paragraph deal with packages that have been approved for use in accordance with this part before January 1, 1979, and which have been designed in accordance with the provisions of this part in effect at the time of application for package approval. Those packages will be accepted as having been designed in accordance with a quality assurance program that satisfies the provisions of paragraph (b) of this section.

(e) *Existing packages*. The provisions of this paragraph deal with packages that have been approved for use in accordance with this part before January 1, 1979, have been at least partially fabricated before that date, and for which the fabrication is in accordance with the provisions of this part in effect at the time of application for approval of package design. These packages will be accepted as having been fabricated and assembled in accordance with a quality assurance program that satisfies the provisions of paragraph (b) of this section.

(f) *Previously approved programs*. A Commission-approved quality assurance program that satisfies the applicable criteria of subpart H of this part, Appendix B of part 50 of this chapter, or subpart G of part 72 of this chapter, and that is established, maintained, and executed regarding transport packages, will be accepted as satisfying the requirements of paragraph (b) of this section. Before first use, the licensee, certificate holder, and applicant for a CoC shall notify the NRC, in accordance with § 71.1, of its intent to apply its previously approved subpart H, Appendix B, or subpart G quality assurance program to transportation activities. The licensee, certificate holder, and applicant for a CoC shall identify the program by date of submittal to the Commission, Docket Number, and date of Commission approval.

(g) *Radiography containers*. A program for transport container inspection and maintenance limited to radiographic exposure devices, source changers, or packages transporting these devices and meeting the requirements of § 34.31(b) of this chapter or equivalent Agreement State requirement, is deemed to satisfy the requirements of §§ 71.17(b) and 71.101(b).

§ 71.103 Quality assurance organization.

(a) The licensee,[20] certificate holder, and applicant for a CoC shall be responsible for the establishment and execution of the quality assurance program. The licensee, certificate holder, and applicant for a CoC may delegate to others, such as contractors, agents, or consultants, the work of establishing and executing the quality assurance program, or any part of the quality assurance program, but shall retain responsibility for the program. The licensee, certificate holder, and applicant for a CoC shall clearly establish and delineate, in writing, the authority and

[20] While the term "licensee" is used in these criteria, the requirements are applicable to whatever design, fabrication, assembly, and testing of the package is accomplished with respect to a package before the time a package approval is issued.

duties of persons and organizations performing activities affecting the functions of structures, systems, and components that are important to safety. These activities include performing the functions associated with attaining quality objectives and the quality assurance functions.

(b) The quality assurance functions are—

(1) Assuring that an appropriate quality assurance program is established and effectively executed; and

(2) Verifying, by procedures such as checking, auditing, and inspection, that activities affecting the functions that are important to safety have been correctly performed.

(c) The persons and organizations performing quality assurance functions must have sufficient authority and organizational freedom to—

(1) Identify quality problems;

(2) Initiate, recommend, or provide solutions; and

(3) Verify implementation of solutions.

(d) The persons and organizations performing quality assurance functions shall report to a management level that assures that the required authority and organizational freedom, including sufficient independence from cost and schedule, when opposed to safety considerations, are provided.

(e) Because of the many variables involved, such as the number of personnel, the type of activity being performed, and the location or locations where activities are performed, the organizational structure for executing the quality assurance program may take various forms, provided that the persons and organizations assigned the quality assurance functions have the required authority and organizational freedom.

(f) Irrespective of the organizational structure, the individual(s) assigned the responsibility for assuring effective execution of any portion of the quality assurance program, at any location where activities subject to this section are being performed, must have direct access to the levels of management necessary to perform this function.

§ 71.105 Quality assurance program.

(a) The licensee, certificate holder, and applicant for a CoC shall establish, at the earliest practicable time consistent with the schedule for accomplishing the activities, a quality assurance program that complies with the requirements of §§ 71.101 through 71.137. The licensee, certificate holder, and applicant for a CoC shall document the quality assurance program by written procedures or instructions and shall carry out the program in accordance with those procedures throughout the period during which the packaging is used. The licensee, certificate holder, and applicant for a CoC shall identify the material and components to be covered by the quality assurance program, the major organizations participating in the program, and the designated functions of these organizations.

(b) The licensee, certificate holder, and applicant for a CoC, through its quality assurance program, shall provide control over activities affecting the quality of the identified materials and components to an extent consistent with their importance to safety, and as necessary to assure conformance to the approved design of each individual package used for the shipment of radioactive material. The licensee, certificate holder, and applicant for a CoC shall assure that activities affecting quality are accomplished under suitably controlled conditions. Controlled conditions include the use of appropriate equipment; suitable environmental conditions for accomplishing the activity, such as adequate cleanliness; and assurance that all prerequisites for the given activity have been satisfied. The licensee, certificate holder, and applicant for a CoC shall take into account the need for special controls, processes, test equipment, tools, and skills to attain the required quality, and the need for verification of quality by inspection and test.

(c) The licensee, certificate holder, and applicant for a CoC shall base the requirements and procedures of its quality assurance program on the following considerations concerning the complexity and proposed use of the package and its components:

(1) The impact of malfunction or failure of the item to safety;

(2) The design and fabrication complexity or uniqueness of the item;

(3) The need for special controls and surveillance over processes and equipment;

(4) The degree to which functional compliance can be demonstrated by inspection or test; and

(5) The quality history and degree of standardization of the item.

(d) The licensee, certificate holder, and applicant for a CoC shall provide for indoctrination and training of personnel performing activities affecting quality, as necessary to assure that suitable proficiency is achieved and maintained. The licensee, certificate holder, and applicant for a CoC shall review the status and adequacy of the quality assurance program at established intervals. Management of other organizations participating in the quality assurance program shall review regularly the status and adequacy of that part of the quality assurance program they are executing.

§ 71.107 Package design control.

(a) The licensee, certificate holder, and applicant for a CoC shall establish measures to assure that applicable regulatory requirements and the package design, as specified in the license or CoC for those materials and components to which this section applies, are correctly translated into specifications, drawings, procedures, and instructions. These measures must include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from standards are controlled. Measures must be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the functions of the materials, parts, and components of the packaging that are important to safety.

(b) The licensee, certificate holder, and applicant for a CoC shall establish measures for the identification and control of design interfaces and for coordination among participating design organizations. These measures must include the establishment of written procedures, among participating design organizations, for the review, approval, release, distribution, and revision of documents involving design interfaces. The design control measures must provide for verifying or checking the adequacy of design, by methods such as design reviews, alternate or simplified calculational methods, or by a suitable testing program. For the verifying or checking process, the licensee shall designate individuals or groups other than those who were responsible for the original design, but who may be from the same organization. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, the licensee, certificate holder, and applicant for a CoC shall include suitable qualification testing of a prototype or sample unit under the most adverse design conditions. The licensee, certificate holder, and applicant for a CoC shall apply design control measures to the following:

(1) Criticality physics, radiation shielding, stress, thermal, hydraulic, and accident analyses;

(2) Compatibility of materials;

(3) Accessibility for inservice inspection, maintenance, and repair;

(4) Features to facilitate decontamination; and

(5) Delineation of acceptance criteria for inspections and tests.

(c) The licensee, certificate holder, and applicant for a CoC shall subject design changes, including field changes, to design control measures commensurate with those applied to the original design. Changes in the conditions specified in the CoC require NRC prior approval.

§ 71.109 Procurement document control.

The licensee, certificate holder, and applicant for a CoC shall establish measures to assure that adequate quality is required in the documents for procurement of material, equipment, and services, whether purchased by the licensee, certificate holder, and applicant for a CoC or by its contractors or subcontractors. To the extent necessary, the licensee, certificate holder, and applicant for a CoC shall require contractors or subcontractors to provide a quality assurance program consistent with the applicable provisions of this part.

§ 71.111 Instructions, procedures, and drawings.

The licensee, certificate holder, and applicant for a CoC shall prescribe activities affecting quality by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall require that these instructions, procedures, and drawings be followed. The instructions, procedures, and drawings must include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

§ 71.113 Document control.

The licensee, certificate holder, and applicant for a CoC shall establish measures to control the issuance of documents such as instructions, procedures, and drawings, including changes, which prescribe all activities affecting quality. These measures must assure that documents, including changes, are reviewed for adequacy, approved for release by authorized personnel, and distributed and used at the location where the prescribed activity is performed. These measures must assure that changes to documents are reviewed and approved.

§ 71.115 Control of purchased material, equipment, and services.

(a) The licensee, certificate holder, and applicant for a CoC shall establish measures to assure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents. These measures must include provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by the contractor or subcontractor, inspection at the contractor or subcontractor source, and examination of products on delivery.

(b) The licensee, certificate holder, and applicant for a CoC shall have available documentary evidence that material and equipment conform to the procurement specifications before installation or use of the material and equipment. The licensee, certificate holder, and applicant for a CoC shall retain, or have available, this documentary evidence for the life of the package to which it applies. The licensee, certificate holder, and applicant for a CoC shall assure that the evidence is sufficient to identify the specific requirements met by the purchased material and equipment.

(c) The licensee, certificate holder, and applicant for a CoC shall assess the effectiveness of the control of quality by contractors and subcontractors at intervals consistent with the importance, complexity, and quantity of the product or services.

§ 71.117 Identification and control of materials, parts, and components.

The licensee, certificate holder, and applicant for a CoC shall establish measures for the identification and control of materials, parts, and components. These measures must assure that identification of the item is maintained by heat number, part number, or other appropriate means, either on the item or on records traceable to the item, as required throughout fabrication, installation, and use of the item. These identification and control measures must be designed to prevent the use of incorrect or defective materials, parts, and components.

§ 71.119 Control of special processes.

The licensee, certificate holder, and applicant for a CoC shall establish measures to assure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

§ 71.121 Internal inspection.

The licensee, certificate holder, and applicant for a CoC shall establish and execute a program for inspection of activities affecting quality by or for the organization performing the activity, to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. The inspection must be performed by individuals other than those who performed the activity being inspected. Examination, measurements, or tests of material or products processed must be performed for each work operation where necessary to assure quality. If direct inspection of processed material or products is not carried out, indirect control by monitoring processing methods, equipment, and personnel must be provided. Both inspection and process monitoring must be provided when quality control is inadequate without both. If mandatory inspection hold points, which require witnessing or inspecting by the licensee's designated representative and beyond which work should not proceed without the consent of its designated representative, are required, the specific hold points must be indicated in appropriate documents.

§ 71.123 Test control.

The licensee, certificate holder, and applicant for a CoC shall establish a test program to assure that all testing required to demonstrate that the packaging components will perform satisfactorily in service is identified and performed in accordance with written test procedures that incorporate the requirements of this part and the requirements and acceptance limits contained in the package approval. The test procedures must include provisions for assuring that all prerequisites for the given test are met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions. The licensee, certificate holder, and applicant for a CoC shall document and evaluate the test results to assure that test requirements have been satisfied.

§ 71.125 Control of measuring and test equipment.

The licensee, certificate holder, and applicant for a CoC shall establish measures to assure that tools, gauges, instruments, and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated, and adjusted at specified times to maintain accuracy within necessary limits.

§ 71.127 Handling, storage, and shipping control.

The licensee, certificate holder, and applicant for a CoC shall establish measures to control, in accordance with instructions, the handling, storage, shipping, cleaning, and preservation of materials and equipment to be used in packaging to prevent damage or deterioration. When necessary for particular products, special protective environments, such as inert gas atmosphere, and specific moisture content and temperature levels must be specified and provided.

§ 71.129 Inspection, test, and operating status.

(a) The licensee, certificate holder, and applicant for a CoC shall establish measures to indicate, by the use of markings such as stamps, tags, labels, routing cards, or other suitable means, the status of inspections and tests performed upon individual items of the packaging. These measures must provide for the identification of items that have satisfactorily passed required inspections and tests, where necessary to preclude inadvertent bypassing of the inspections and tests.

(b) The licensee shall establish measures to identify the operating status of components of the packaging, such as tagging valves and switches, to prevent inadvertent operation.

§ 71.131 Nonconforming materials, parts, or components.

The licensee, certificate holder, and applicant for a CoC shall establish measures to control materials, parts, or components that do not conform to the licensee's requirements to prevent their inadvertent use or installation. These measures must include, as appropriate, procedures for identification, documentation, segregation, disposition, and notification to affected organizations. Nonconforming items must be reviewed and accepted, rejected, repaired, or reworked in accordance with documented procedures.

§ 71.133 Corrective action.

The licensee, certificate holder, and applicant for a CoC shall establish measures to assure that conditions adverse to quality, such as deficiencies, deviations, defective material and equipment, and nonconformances, are promptly identified and corrected. In the case of a significant condition adverse to quality, the measures must assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken must be documented and reported to appropriate levels of management.

§ 71.135 Quality assurance records.

The licensee, certificate holder, and applicant for a CoC shall maintain sufficient written records to describe the activities affecting quality. The records must include the instructions, procedures, and drawings required by § 71.111 to prescribe quality assurance activities and must include closely related specifications such as required qualifications of personnel, procedures, and equipment. The records must include the instructions or procedures which establish a records retention program that is consistent with applicable regulations and designates factors such as duration, location, and assigned responsibility. The licensee, certificate holder, and applicant for a CoC shall retain these records for 3 years beyond the date when the licensee, certificate holder, and applicant for a CoC last engages in the activity for which the quality assurance program was developed. If any portion of the written procedures or instructions is superseded, the licensee, certificate holder, and applicant for a CoC shall retain the superseded material for 3 years after it is superseded.

§ 71.137 Audits.

The licensee, certificate holder, and applicant for a CoC shall carry out a comprehensive system of planned and periodic audits, to verify compliance with all aspects of the quality assurance program, and to determine the effectiveness of the program. The audits must be performed in accordance with written procedures or checklists by appropriately trained personnel not having direct responsibilities in the areas being audited. Audited results must be documented and reviewed by management having responsibility in the area audited. Follow-up action, including reaudit of deficient areas, must be taken where indicated.

17. A new subpart I is added to Part 71 to read as follows:

Sec.

71.151 Procedures for applying for a Type B(DP) package approval.

71.153 Contents of application.

71.155 Package description.

71.157 Package evaluation.

71.159 Quality assurance.

71.161 Requirement for additional information.

71.163 Issuance of an NRC certificate of compliance.

71.165 Conditions for package reapproval.

71.167 Application to amend a certificate of compliance.

Subpart I - Type B(DP) Package Approval

§ 71.151 Procedures for applying for a Type B(DP) package approval.

(a) Spent fuel storage casks that have been issued a Certificate of Compliance (CoC) under subpart L of part 72 of this chapter may also be approved under this subpart as a Type B(DP) package for the transportation of spent fuel. A copy of the part 72 CoC issued for the cask, and any drawings and other documents referenced in the part 72 CoC, must be included with the application.

(b) An application for approval of a Type B(DP) package design must contain the information required by § 71.153 and be submitted in accordance with § 71.1.

(c) *Public inspection.* An application for the approval of a Type B(DP) package, or amendment of a Type B(DP) package, may be made available for public inspection under § 71.10.

(d) *Fees.* Fees for reviews and evaluations related to issuance of a Type B(DP) CoC and inspections related to package fabrication are those shown in § 170.31 of this chapter.

§ 71.153 Contents of application.

(a) An application for an approval of a Type B(DP) package under this subpart must include, for each proposed Type B(DP) packaging design, the following information:

(1) A package description as required by § 71.155;

(2) A package evaluation as required by § 71.157; and

(3) A quality assurance program description, as required by § 71.159, or a reference to a previously approved quality assurance program.

(b) A safety analysis report describing —

(1) The proposed Type B(DP) package design;

(2) How the package would be used to transport spent fuel safely;

(3) An analysis of potential accidents, package response to these potential accidents, and any consequences to the public; and

(4) How the package is suitable for the transportation of spent fuel for a period of at least 20 years.

(c) Except as provided in § 71.19, an application for modification of a Type B(DP) package design, whether for modification of the packaging or the authorized contents, must include sufficient information to demonstrate that the proposed design satisfies the Type B(DP) package standards in effect at the time the application is filed.

(d) The applicant shall identify any established codes and standards proposed for use in package design, fabrication, assembly, testing, maintenance, and use. In the absence of any codes and standards, the applicant shall describe and justify the basis and rationale used to formulate the package quality assurance program.

§ 71.155 Package description.

The application must include a description of the proposed Type B(DP) package in sufficient detail to identify the Type B(DP) package accurately and provide a sufficient basis for evaluation of the Type B(DP) package. The description must include—

(a) With respect to the packaging—

(1) Gross weight;

(2) Model number;

(3) Identification of the containment system;

(4) Specific materials of construction, weights, dimensions, and fabrication methods of—

(i) Receptacles;

(ii) Materials specifically used as nonfissile neutron absorbers or moderators;

(iii) Internal and external structures supporting or protecting receptacles;

(iv) Valves, sampling ports, lifting devices, and tie-down devices; and

(v) Structural and mechanical means for the transfer and dissipation of heat; and

(5) Identification and volumes of any receptacles containing coolant.

(b) With respect to the contents of the package—

(1) Identification and maximum radioactivity of radioactive constituents;

(2) Identification and maximum quantities of fissile constituents;

(3) Chemical and physical form;

(4) Extent of reflection, the amount and identity of nonfissile materials used as neutron absorbers or moderators, and the atomic ratio of moderator to fissile constituents;

(5) Maximum normal operating pressure;

(6) Maximum weight;

(7) Maximum amount of decay heat; and

(8) Identification and volumes of any coolants.

§ 71.157 Package evaluation.

The application submitted under § 71.151 must include the following:

(a) A demonstration that the Type B(DP) package satisfies the standards specified in subparts E and F of this part. The application need not address the requirements of §§ 71.61, 71.64, 71.74, 71.75, and 71.77;

(b) The number "N" for the Type B(DP) package as determined in accordance with § 71.59; and

(c) Any proposed special controls and precautions for transport, loading, unloading, and handling and any proposed special controls in case of an accident or delay.

§ 71.159 Quality assurance.

(a) The applicant shall describe the quality assurance program (see subpart H of this part) for the design, fabrication, assembly, testing, maintenance, repair, modification, and use of the proposed Type B(DP) package.

(b) The applicant shall identify any specific provisions of the quality assurance program that are applicable to the particular Type B(DP) package design under consideration, including a description of any leak testing.

§ 71.161 Requirement for additional information.

The Commission may at any time require additional information in order to enable it to determine whether a license, CoC, or other approval should be granted, renewed, denied, modified, suspended, or revoked.

§ 71.163 Issuance of an NRC certificate of compliance.

The NRC will issue a CoC for a Type B(DP) package on a finding that the requirements in §§ 71.151 through 71.159 are met. The term of a Type B(DP) CoC is to up to 20 years.

§ 71.165 Conditions for package reapproval.

(a) Except as provided in paragraph (b) of this section, each CoC for a Type B(DP) package or Quality Assurance Program Approval expires at the end of the day, in the month and year stated in the approval.

(b) *Timely renewal.* If a person holding a CoC for a Type B(DP) package or Quality Assurance Program Approval issued under this part has filed a proper application requesting renewal of either the CoC or the Quality Assurance Program Approval, then the CoC or Quality Assurance Program Approval is not considered to have expired until the Commission has taken final action on the application. The application must be submitted to the Commission not less than 2 years before the expiration of the CoC or the Quality Assurance Program Approval.

(c) In applying for renewal of an existing CoC for a Type B(DP) package or Quality Assurance Program Approval, an applicant may be required to submit a consolidated application that incorporates all changes to its program — that are incorporated by reference in the existing approval or certificate — into as few referenceable documents as reasonably achievable.

(d) Applications for renewal of an existing CoC for a Type B(DP) package or Quality Assurance Program Approval must be submitted to the Commission in accordance with § 71.1.

§ 71.167 Application to amend a certificate of compliance.

A certificate holder desiring to amend its CoC for a Type B(DP) package — including a change to the terms, conditions, or specifications of the CoC — shall submit an application for amendment with the Commission, in accordance with § 71.1. The application must fully describe the changes desired and the reasons for these changes. The application should follow, as far as applicable, the form prescribed for an original application in § 71.151.

§ 71.169 Issuance of an amendment to a certificate of compliance.

In determining whether an amendment to a CoC for a Type B(DP) package will be issued to the applicant, the Commission will be guided by the considerations that govern the issuance of an initial CoC.

§ 71.171 Inspections and tests.

(a) The certificate holder and applicant for a CoC for a Type B(DP) package shall permit, and make provisions for, the NRC to inspect the premises and facilities where a Type B(DP) package is designed, fabricated, and tested.

(b) The certificate holder and applicant for a CoC for a Type B(DP) package shall make available to the NRC for inspection, upon reasonable notice, records kept by them pertaining to the design, fabrication, and testing of a Type B(DP) package.

(c) The certificate holder and applicant for a CoC for a Type B(DP) package shall perform, and make provisions that permit the NRC to perform tests that the Commission deems necessary or appropriate for the administration of the regulations in this part.

§ 71.173 Recordkeeping and reports.

(a) Each certificate holder or applicant shall maintain any records and produce any reports that may be required by the conditions of the CoC or by the rules, regulations, and orders of the NRC in effectuating the purposes of the Act.

(b) Records that are required by the regulations in this part or by conditions of the CoC must be maintained for the period specified by the appropriate regulation or the CoC conditions. If a retention period is not specified, the records must be maintained until the NRC terminates the CoC.

(c) Any record maintained under this part may be either the original or a reproduced copy by any state-of-the-art method provided that any reproduced copy is duly authenticated by authorized personnel and is capable of producing a clear and legible copy after storage for the period specified by NRC regulations.

(d) Each certificate holder shall maintain a record of each Type B(DP) package it has manufactured. The record must contain the following information:

(1) The package identification number;

(2) The package serial number;

(3) The date fabrication of the package was commenced; and

(4) The date fabrication of the package was completed.

§ 71.175 Changes.

(a) Definitions for the purposes of this section:

(1) *Change* means a modification or addition to, or removal from, a Type B(DP) package design or procedures that affect a design function, method of performing or controlling the function, or an evaluation that demonstrates that intended functions will be accomplished.

(2) *Departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses* means:

(i) Changing any of the elements of the method described in the FSAR (as updated) unless the results of the analysis are conservative or essentially the same; or

(ii) Changing from a method described in the FSAR to another method unless that method has been approved by NRC for the intended application.

(3) *A Type B(DP) package design as described in the Final Safety Analysis Report (FSAR) (as updated)* means:

(i) The structures, systems, and components (SSC) that are described in the FSAR (as updated),

(ii) The design and performance requirements for such SSCs described in the FSAR (as updated), and

(iii) The evaluations or methods of evaluation included in the FSAR (as updated) for such SSCs which demonstrate that their intended function(s) will be accomplished.

(4) *Final Safety Analysis Report (as updated)* means the Safety Analysis Report for a Type B(DP) package design as submitted, amended, and updated in accordance with § 71.177.

(5) *Procedures as described in the FSAR (as updated)* means those procedures that contain information described in the safety analysis report such as how SSCs are operated and controlled (including assumed operator actions and response times).

(b) This section applies to each holder of a CoC for Type B(DP) package issued under this subpart.

(c)(1) A certificate holder may make changes to a Type B(DP) package design, as described in the FSAR (as updated), and make changes in the procedures, as described in the FSAR (as updated), without obtaining a CoC amendment under § 71.167 if:

(i) A change in the terms, conditions, or specifications incorporated in the CoC is not required; and

(ii) The change does not meet any of the criteria in paragraph (c)(2) of this section.

(2) A certificate holder shall obtain a CoC amendment under § 71.167, before implementing a proposed change if the change would:

(i) Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR (as updated);

(ii) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a system, structure, or component (SSC) important to safety previously evaluated in the FSAR (as updated);

(iii) Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR (as updated);

(iv) Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR (as updated);

(v) Create a possibility for an accident of a different type than any previously evaluated in the FSAR (as updated);

(vi) Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR (as updated);

(vii) Result in a design basis limit for a fission product barrier as described in the FSAR (as updated) being exceeded or altered; or

(viii) Result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses.

(3) In implementing this paragraph, the FSAR (as updated) is considered to include FSAR changes resulting from evaluations performed under this section and analyses performed under § 71.161, since the last update of the FSAR as required by § 71.177.

(4) The provisions in this section do not apply to changes to procedures when the applicable regulations of this part establish more specific criteria for accomplishing such changes.

(d)(1) The certificate holder shall maintain records of changes to a Type B(DP) package and of changes in procedures made under paragraph (c) of this section. These records must include a written evaluation that provides the bases for the determination that the change does not require a CoC amendment under paragraph (c)(2) of this section.

(2) The certificate holder shall submit, as specified in § 71.1, a report containing a brief description of any changes, including a summary of the evaluation of each. A report must be submitted at intervals not to exceed 24 months.

(3) The records of changes in a Type B(DP) package design must be maintained until:

(i) The Commission terminates the CoC issued under this part; or

(ii) The package is permanently removed from service.

(4) The records of changes in procedures must be maintained for a period of 5 years.

(5) The holder of a Type B(DP) package design CoC, who permanently ceases operation, shall provide the records of changes to the new certificate holder or to the Commission, in accordance with § 71.1, as appropriate.

(6) A certificate holder shall provide a copy of the record for any changes to a Type B(DP) package design to any licensee using the package design within 60 days of implementing the change.

§ 71.177 Safety analysis report updating.

(a) Each certificate holder for a Type B(DP) package approved under this subpart shall update periodically, as provided in paragraph (b) of this section, the final safety analysis report

(FSAR) to assure that the information included in the report contains the latest information developed.

(1) Each certificate holder shall submit an original FSAR to the Commission, in accordance with § 71.1, within 90 days after the Type B(DP) package design has been approved under § 71.163.

(2) The original FSAR must be based on the safety analysis report submitted with the application and reflect any changes and applicant commitments developed during the Type B(DP) package design review process. The original FSAR must be updated to reflect any changes to requirements contained in the issued CoC.

(b) Each update must contain all the changes necessary to reflect information and analyses submitted to the Commission by the certificate holder or prepared by the certificate holder pursuant to Commission requirements since the submission of the original FSAR or, as appropriate, the last update to the FSAR under this section. The update must include the effects[21] of:

(1) All changes made in the dual-purpose spent fuel transportation package procedures as described in the FSAR;

(2) All safety analyses and evaluations performed by the certificate holder either in support of approved CoC amendments, or in support of conclusions that changes did not require a CoC amendment in accordance with § 71.175; and

(3) All analyses of new safety issues performed by or on behalf of the certificate holder at Commission request. The information shall be appropriately located within the updated FSAR.

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[21] Effects of changes includes appropriate revisions of descriptions in the FSAR so that the FSAR (as updated) is complete and accurate.

(c)(1) The update of the FSAR must be filed in accordance with § 71.1, on a replacement-page basis;

(2) The update must include a list that identifies the current pages of the FSAR following page replacement;

(3) Each replacement page must include both a change indicator for the area changed, e.g., a bold line vertically drawn in the margin adjacent to the portion actually changed, and a page change identification (date of change or change number or both);

(4) The update must include:

(i) A certification by a duly authorized officer of the certificate holder that either the information accurately presents changes made since the previous submittal, or that no such changes were made; and

(ii) An identification of changes made by the certificate holder under the provisions of § 71.175, but not previously submitted to the Commission;

(5) The update must reflect all changes implemented up to a maximum of 6 months before the date of filing;

(6) Updates must be filed every 24 months from the date of issuance of the CoC;

(7) Updates must be filed within 90 days of issuance from the date of an amendment to the CoC; and

(8) The certificate holder shall provide a copy of the updated FSAR to each licensee who is using its Type B(DP) package design.

(d) The updated FSAR must be retained by the certificate holder until the Commission terminates the certificate.

(e) A certificate holder who permanently ceases operation shall provide the updated FSAR to the new certificate holder or to the Commission, in accordance with § 71.1, as appropriate.

18. Appendix A to Part 71 is revised to read as follows:

APPENDIX A TO PART 71 - DETERMINATION OF A_1 AND A_2

I. Values of A_1 and A_2 for individual radionuclides, which are the bases for many activity limits elsewhere in these regulations, are given in Table A-1. The curie (Ci) values specified are obtained by converting from the Terabecquerel (TBq) figure. The curie values are expressed to three significant figures to assure that the difference in the TBq and Ci quantities is one tenth of one percent or less. Where values of A_1 and A_2 are unlimited, it is for radiation control purposes only. For nuclear criticality safety, some materials are subject to controls placed on fissile material.

II.(a) For individual radionuclides whose identities are known, but which are not listed in Table A-1, the A_1 and A_2 values contained in Table A-3 may be used. Otherwise, the licensee shall obtain prior Commission approval of the A_1 and A_2 values for radionuclides not listed in Table A-1, before shipping the material.

(b) For individual radionuclides whose identities are known, but which are not listed in Table A-2, the exempt material activity concentration and exempt consignment activity values contained in Table A-3 may be used. Otherwise, the licensee shall obtain prior Commission approval of the exempt material activity concentration and exempt consignment activity values, for radionuclides not listed in Table A-2, before shipping the material.

(c) The licensee shall submit requests for prior approval, described under paragraphs II(a) and II(b) of this Appendix, to the Commission, in accordance with § 71.1 of this part.

III. In the calculations of A_1 and A_2 for a radionuclide not in Table A-1, a single radioactive decay chain, in which radionuclides are present in their naturally occurring proportions, and in which no daughter radionuclide has a half-life either longer than 10 days, or longer than that of the parent radionuclide, shall be considered as a single radionuclide, and the activity to be taken into account, and the A_1 or A_2 value to be applied shall be those corresponding to the parent radionuclide of that chain. In the case of radioactive decay chains in which any daughter radionuclide has a half-life either longer that 10 days, or greater than that of the parent radionuclide, the parent and those daughter radionuclides shall be considered as mixtures of different radionuclides.

IV. For mixtures of radionuclides whose identities and respective activities are know**n**, the following conditions apply:

(a) For special form radioactive material, the maximum quantity transported in a Type A package is as follows:

$$ {}_{i} \ \frac{B(i)}{A_1(i)} \leq 1 $$

Where B(i) is the activity of radionuclide I**,** and A_1(i) is the A_1 value for radionuclide I.

(b) For normal form radioactive material, the maximum quantity transported in a Type A package is as follows:

$$ {}_{i} \ \frac{B(i)}{A_2(i)} \leq 1 $$

Where B(i) is the activity of radionuclide I**,** and A_2(i) is the A_2 value for radionuclide I.

(c) Alternatively, the A_1 value for mixtures of special form material may be determined as follows:

$$A_1 \text{ for mixture } = \frac{1}{\sum_{i} \frac{f(i)}{A_1(i)}}$$

Where f(i) is the fraction of activity for radionuclide I in the mixture**,** and $A_1(i)$ is the appropriate A_1 value for radionuclide I.

(d) Alternatively, the A_2 value for mixtures of normal form material may be determined as follows:

$$A_2 \text{ for mixture } = \frac{1}{\sum_{i} \frac{f(i)}{A_2(i)}}$$

Where f(i) is the fraction of activity for radionuclide I in the mixture**,** and $A_2(i)$ is the appropriate A_2 value for radionuclide I.

(e) The exempt activity concentration for mixtures of nuclides may be determined as follows:

$$\text{Exempt activity concentration for mixture } = \frac{1}{\sum_{i} \frac{f(i)}{[A](i)}}$$

Where f(i) is the fraction of activity concentration of radionuclide I in the mixture, and [A] is the activity concentration for exempt material containing radionuclide I.

(f) The activity limit for an exempt consignment for mixtures of radionuclides may be determined as follows:

$$\text{Exempt consignment activity limit for mixture} = \frac{1}{{}_{i}\,\dfrac{f(i)}{A(i)}}$$

Where f(i) is the fraction of activity of radionuclide I in the mixture, and A is the activity limit for exempt consignments for radionuclide I.

V. When the identity of each radionuclide is known, but the individual activities of some of the radionuclides are not known, the radionuclides may be grouped and the lowest A_1 or A_2 value, as appropriate, for the radionuclides in each group may be used in applying the formulas in paragraph IV. Groups may be based on the total alpha activity and the total beta/gamma activity when these are known, using the lowest A_1 or A_2 values for the alpha emitters and beta/gamma emitters.

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES							
Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Ac-225 (a)	Actinium (89)	$8.0X10^{-1}$	$2.2X10^{1}$	$6.0X10^{-3}$	$1.6X10^{-1}$	$2.1X10^{3}$	$5.8X10^{4}$
Ac-227 (a)		$9.0X10^{-1}$	$2.4X10^{1}$	$9.0X10^{-5}$	$2.4X10^{-3}$	2.7	$7.2X10^{1}$
Ac-228		$6.0X10^{-1}$	$1.6X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$8.4X10^{4}$	$2.2X10^{6}$
Ag-105	Silver (47)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$1.1X10^{3}$	$3.0X10^{4}$
Ag-108m (a)		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$9.7X10^{-1}$	$2.6X10^{1}$
Ag-110m (a)		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.8X10^{2}$	$4.7X10^{3}$
Ag-111		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$5.8X10^{3}$	$1.6X10^{5}$
Al-26	Aluminum (13)	$1.0X10^{-1}$	2.7	$1.0X10^{-1}$	2.7	$7.0X10^{-4}$	$1.9X10^{-2}$
Am-241	Americium (95)	$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$1.3X10^{-1}$	3.4
Am-242m (a)		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$3.6X10^{-1}$	$1.0X10^{1}$
Am-243 (a)		5.0	$1.4X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$7.4X10^{-3}$	$2.0X10^{-1}$
Ar-37	Argon (18)	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$3.7X10^{3}$	$9.9X10^{4}$
Ar-39		$2.0X10^{1}$	$5.4X10^{2}$	$4.0X10^{1}$	$1.1X10^{3}$	1.3	$3.4X10^{1}$
Ar-41		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.5X10^{6}$	$4.2X10^{7}$
As-72	Arsenic (33)	$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$6.2X10^{4}$	$1.7X10^{6}$
As-73		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$8.2X10^{2}$	$2.2X10^{4}$
As-74		1.0	$2.7X10^{1}$	$9.0X10^{-1}$	$2.4X10^{1}$	$3.7X10^{3}$	$9.9X10^{4}$
As-76		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$5.8X10^{4}$	$1.6X10^{6}$
As-77		$2.0X10^{1}$	$5.4X10^{2}$	$7.0X10^{-1}$	$1.9X10^{1}$	$3.9X10^{4}$	$1.0X10^{6}$
At-211 (a)	Astatine (85)	$2.0X10^{1}$	$5.4X10^{2}$	$5.0X10^{-1}$	$1.4X10^{1}$	$7.6X10^{4}$	$2.1X10^{6}$
Au-193	Gold (79)	7.0	$1.9X10^{2}$	2.0	$5.4X10^{1}$	$3.4X10^{4}$	$9.2X10^{5}$
Au-194		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$1.5X10^{4}$	$4.1X10^{5}$

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Au-195	Gold (79)	$1.0X10^{1}$	$2.7X10^{2}$	6.0	$1.6X10^{2}$	$1.4X10^{2}$	$3.7X10^{3}$
Au-198		1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$9.0X10^{3}$	$2.4X10^{5}$
Au-199		$1.0X10^{1}$	$2.7X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$7.7X10^{3}$	$2.1X10^{5}$
Ba-131 (a)	Barium (56)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$3.1X10^{3}$	$8.4X10^{4}$
Ba-133		3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	9.4	$2.6X10^{2}$
Ba-133m		$2.0X10^{1}$	$5.4X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.2X10^{4}$	$6.1X10^{5}$
Ba-140 (a)		$5.0X10^{-1}$	$1.4X10^{1}$	$3.0X10^{-1}$	8.1	$2.7X10^{3}$	$7.3X10^{4}$
Be-7	Beryllium (4)	$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{1}$	$5.4X10^{2}$	$1.3X10^{4}$	$3.5X10^{5}$
Be-10		$4.0X10^{1}$	$1.1X10^{3}$	$6.0X10^{-1}$	$1.6X10^{1}$	$8.3X10^{-4}$	$2.2X10^{-2}$
Bi-205	Bismuth (83)	$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$1.5X10^{-3}$	$4.2X10^{4}$
Bi-206		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$3.8X10^{3}$	$1.0X10^{5}$
Bi-207		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	1.9	$5.2X10^{1}$
Bi-210		1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$4.6X10^{3}$	$1.2X10^{5}$
Bi-210m (a)		$6.0X10^{-1}$	$1.6X10^{1}$	$2.0X10^{-2}$	$5.4X10^{-1}$	$2.1X10^{-5}$	$5.7X10^{-4}$
Bi-212 (a)		$7.0X10^{-1}$	$1.9X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$5.4X10^{5}$	$1.5X10^{7}$
Bk-247	Berkelium (97)	8.0	$2.2X10^{2}$	$8.0X10^{-4}$	$2.2X10^{-2}$	$3.8X10^{-2}$	1.0
Bk-249 (a)		$4.0X10^{1}$	$1.1X10^{3}$	$3.0X10^{-1}$	8.1	$6.1X10^{1}$	$1.6X10^{3}$
Br-76	Bromine (35)	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$9.4X10^{4}$	$2.5X10^{6}$
Br-77		3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$2.6X10^{4}$	$7.1X10^{5}$
Br-82		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{4}$	$1.1X10^{6}$
C-11	Carbon (6)	1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.1X10^{7}$	$8.4X10^{8}$
C-14		$4.0X10^{1}$	$1.1X10^{3}$	3.0	$8.1X10^{1}$	$1.6X10^{-1}$	4.5

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Ca-41	Calcium (20)	Unlimited	Unlimited	Unlimited	Unlimited	$3.1X10^{-3}$	$8.5X10^{-2}$
Ca-45		$4.0X10^{1}$	$1.1X10^{3}$	1.0	$2.7X10^{1}$	$6.6X10^{2}$	$1.8X10^{4}$
Ca-47 (a)		3.0	$8.1X10^{1}$	$3.0X10^{-1}$	8.1	$2.3X10^{4}$	$6.1X10^{5}$
Cd-109	Cadmium (48)	$3.0X10^{1}$	$8.1X10^{2}$	2.0	$5.4X10^{1}$	$9.6X10^{1}$	$2.6X10^{3}$
Cd-113m		$4.0X10^{1}$	$1.1X10^{3}$	$5.0X10^{-1}$	$1.4X10^{1}$	8.3	$2.2X10^{2}$
Cd-115 (a)		3.0	$8.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.9X10^{4}$	$5.1X10^{5}$
Cd-115m		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$9.4X10^{2}$	$2.5X10^{4}$
Ce-139	Cerium (58)	7.0	$1.9X10^{2}$	2.0	$5.4X10^{1}$	$2.5X10^{2}$	$6.8X10^{3}$
Ce-141		$2.0X10^{1}$	$5.4X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.1X10^{3}$	$2.8X10^{4}$
Ce-143		$9.0X10^{-1}$	$2.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.5X10^{4}$	$6.6X10^{5}$
Ce-144 (a)		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$1.2X10^{2}$	$3.2X10^{3}$
Cf-248	Californium (98)	$4.0X10^{1}$	$1.1X10^{3}$	$6.0X10^{-3}$	$1.6X10^{-1}$	$5.8X10^{1}$	$1.6X10^{3}$
Cf-249		3.0	$8.1X10^{1}$	$8.0X10^{-4}$	$2.2X10^{-2}$	$1.5X10^{-1}$	4.1
Cf-250		$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{-3}$	$5.4X10^{-2}$	4.0	$1.1X10^{2}$
Cf-251		7.0	$1.9X10^{2}$	$7.0X10^{-4}$	$1.9X10^{-2}$	$5.9X10^{-2}$	1.6
Cf-252 (h)		$1.0X10^{-1}$	2.7	$1.0X10^{-3}$	$2.7X10^{-2}$	$2.0X10^{1}$	$5.4X10^{2}$
Cf-253 (a)		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{-2}$	1.1	$1.1X10^{3}$	$2.9X10^{4}$
Cf-254		$1.0X10^{-3}$	$2.7X10^{-2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$3.1X10^{2}$	$8.5X10^{3}$
Cl-36	Chlorine (17)	$1.0X10^{1}$	$2.7X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.2X10^{-3}$	$3.3X10^{-2}$
Cl-38		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$4.9X10^{6}$	$1.3X10^{8}$
Cm-240	Curium (96)	$4.0X10^{1}$	$1.1X10^{3}$	$2.0X10^{-2}$	$5.4X10^{-1}$	$7.5X10^{2}$	$2.0X10^{4}$
Cm-241		2.0	$5.4X10^{1}$	1.0	$2.7X10^{1}$	$6.1X10^{2}$	$1.7X10^{4}$

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES							
Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Cm-242	Curium (96)	$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-2}$	$2.7X10^{-1}$	$1.2X10^{2}$	$3.3X10^{3}$
Cm-243		9.0	$2.4X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$1.9X10^{-3}$	$5.2X10^{1}$
Cm-244		$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{-3}$	$5.4X10^{-2}$	3.0	$8.1X10^{1}$
Cm-245		9.0	$2.4X10^{2}$	$9.0X10^{-4}$	$2.4X10^{-2}$	$6.4X10^{-3}$	$1.7X10^{-1}$
Cm-246		9.0	$2.4X10^{2}$	$9.0X10^{-4}$	$2.4X10^{-2}$	$1.1X10^{-2}$	$3.1X10^{-1}$
Cm-247 (a)		3.0	$8.1X10^{1}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$3.4X10^{-6}$	$9.3X10^{-5}$
Cm-248		$2.0X10^{-2}$	$5.4X10^{-1}$	$3.0X10^{-4}$	$8.1X10^{-3}$	$1.6X10^{-5}$	$4.2X10^{-3}$
Co-55	Cobalt (27)	$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$1.1X10^{5}$	$3.1X10^{6}$
Co-56		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.1X10^{3}$	$3.0X10^{4}$
Co-57		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$3.1X10^{2}$	$8.4X10^{3}$
Co-58		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$1.2X10^{3}$	$3.2X10^{4}$
Co-58m		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$2.2X10^{5}$	$5.9X10^{6}$
Co-60		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$4.2X10^{1}$	$1.1X10^{3}$
Cr-51	Chromium (24)	$3.0X10^{1}$	$8.1X10^{2}$	$3.0X10^{1}$	$8.1X10^{2}$	$3.4X10^{3}$	$9.2X10^{4}$
Cs-129	Cesium (55)	4.0	$1.1X10^{2}$	4.0	$1.1X10^{2}$	$2.8X10^{4}$	$7.6X10^{5}$
Cs-131		$3.0X10^{1}$	$8.1X10^{2}$	$3.0X10^{1}$	$8.1X10^{2}$	$3.8X10^{3}$	$1.0X10^{5}$
Cs-132		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$5.7X10^{3}$	$1.5X10^{5}$
Cs-134		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$4.8X10^{1}$	$1.3X10^{3}$
Cs-134m		$4.0X10^{1}$	$1.1X10^{3}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.0X10^{5}$	$8.0X10^{6}$
Cs-135		$4.0X10^{1}$	$1.1X10^{3}$	1.0	$2.7X10^{1}$	$4.3X10^{-5}$	$1.2X10^{-3}$
Cs-136		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$2.7X10^{3}$	$7.3X10^{4}$
Cs-137 (a)		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	3.2	$8.7X10^{1}$

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Cu-64	Copper (29)	6.0	$1.6X10^{2}$	1.0	$2.7X10^{1}$	$1.4X10^{5}$	$3.9X10^{6}$
Cu-67		$1.0X10^{1}$	$2.7X10^{2}$	$7.0X10^{-1}$	$1.9X10^{1}$	$2.8X10^{4}$	$7.6X10^{5}$
Dy-159	Dysprosium (66)	$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{1}$	$5.4X10^{2}$	$2.1X10^{2}$	$5.7X10^{3}$
Dy-165		$9.0X10^{-1}$	$2.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.0X10^{5}$	$8.2X10^{6}$
Dy-166 (a)		$9.0X10^{-1}$	$2.4X10^{1}$	$3.0X10^{-1}$	8.1	$8.6X10^{3}$	$2.3X10^{5}$
Er-169	Erbium (68)	$4.0X10^{1}$	$1.1X10^{3}$	1.0	$2.7X10^{1}$	$3.1X10^{3}$	$8.3X10^{4}$
Er-171		$8.0X10^{-1}$	$2.2X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$9.0X10^{4}$	$2.4X10^{6}$
Eu-147	Europium (63)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$1.4X10^{3}$	$3.7X10^{4}$
Eu-148		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$6.0X10^{2}$	$1.6X10^{4}$
Eu-149		$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{1}$	$5.4X10^{2}$	$3.5X10^{2}$	$9.4X10^{3}$
Eu-150 (short lived)		2.0	$5.4X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$6.1X10^{4}$	$1.6X10^{6}$
Eu-150 (long lived)		2.0	$5.4X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$6.1X10^{4}$	$1.6X10^{6}$
Eu-152		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	6.5	$1.8X10^{2}$
Eu-152m		$8.0X10^{-1}$	$2.2X10^{1}$	$8.0X10^{-1}$	$2.2X10^{1}$	$8.2X10^{4}$	$2.2X10^{6}$
Eu-154		$9.0X10^{-1}$	$2.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	9.8	$2.6X10^{2}$
Eu-155		$2.0X10^{1}$	$5.4X10^{2}$	3.0	$8.1X10^{1}$	$1.8X10^{1}$	$4.9X10^{2}$
Eu-156		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$2.0X10^{3}$	$5.5X10^{4}$
F-18	Fluorine (9)	1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.5X10^{6}$	$9.5X10^{7}$
Fe-52 (a)	Iron (26)	$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$2.7X10^{5}$	$7.3X10^{6}$
Fe-55		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$8.8X10^{1}$	$2.4X10^{3}$
Fe-59		$9.0X10^{-1}$	$2.4X10^{1}$	$9.0X10^{-1}$	$2.4X10^{1}$	$1.8X10^{3}$	$5.0X10^{4}$
Fe-60 (a)		$4.0X10^{1}$	$1.1X10^{3}$	$2.0X10^{-1}$	5.4	$7.4X10^{-4}$	$2.0X10^{-2}$

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Ga-67	Gallium (31)	7.0	$1.9X10^{2}$	3.0	$8.1X10^{1}$	$2.2X10^{4}$	$6.0X10^{5}$
Ga-68		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$1.5X10^{6}$	$4.1X10^{7}$
Ga-72		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.1X10^{5}$	$3.1X10^{6}$
Gd-146 (a)	Gadolinium (64)	$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$6.9X10^{2}$	$1.9X10^{4}$
Gd-148		$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{-3}$	$5.4X10^{-2}$	1.2	$3.2X10^{1}$
Gd-153		$1.0X10^{1}$	$2.7X10^{2}$	9.0	$2.4X10^{2}$	$1.3X10^{2}$	$3.5X10^{3}$
Gd-159		3.0	$8.1X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.9X10^{4}$	$1.1X10^{6}$
Ge-68 (a)	Germanium (32)	$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$2.6X10^{2}$	$7.1X10^{3}$
Ge-71		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$5.8X10^{3}$	$1.6X10^{5}$
Ge-77		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.3X10^{5}$	$3.6X10^{6}$
Hf-172 (a)	Hafnium (72)	$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$4.1X10^{1}$	$1.1X10^{3}$
Hf-175		3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$3.9X10^{2}$	$1.1X10^{4}$
Hf-181		2.0	$5.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$6.3X10^{2}$	$1.7X10^{4}$
Hf-182		Unlimited	Unlimited	Unlimited	Unlimited	$8.1X10^{-6}$	$2.2X10^{-4}$
Hg-194 (a)	Mercury (80)	1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$1.3X10^{-1}$	3.5
Hg-195m (a)		3.0	$8.1X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$1.5X10^{4}$	$4.0X10^{5}$
Hg-197		$2.0X10^{1}$	$5.4X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$9.2X10^{3}$	$2.5X10^{5}$
Hg-197m		$1.0X10^{1}$	$2.7X10^{2}$	$4.0X10^{-1}$	$1.1X10^{1}$	$2.5X10^{4}$	$6.7X10^{5}$
Hg-203		5.0	$1.4X10^{2}$	1.0	$2.7X10^{1}$	$5.1X10^{2}$	$1.4X10^{4}$
Ho-166	Holmium (67)	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$2.6X10^{4}$	$7.0X10^{5}$
Ho-166m		$6.0X10^{-1}$	$1.6X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$6.6X10^{-2}$	1.8

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES							
Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
I-123	Iodine (53)	6.0	$1.6X10^{2}$	3.0	$8.1X10^{1}$	$7.1X10^{4}$	$1.9X10^{6}$
I-124		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$9.3X10^{3}$	$2.5X10^{5}$
I-125		$2.0X10^{1}$	$5.4X10^{2}$	3.0	$8.1X10^{1}$	$6.4X10^{2}$	$1.7X10^{4}$
I-126		2.0	$5.4X10^{1}$	1.0	$2.7X10^{1}$	$2.9X10^{3}$	$8.0X10^{4}$
I-129		Unlimited	Unlimited	Unlimited	Unlimited	$6.5X10^{-6}$	$1.8X10^{-4}$
I-131		3.0	$8.1X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$4.6X10^{3}$	$1.2X10^{5}$
I-132		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$3.8X10^{5}$	$1.0X10^{7}$
I-133		$7.0X10^{-1}$	$1.9X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$4.2X10^{4}$	$1.1X10^{6}$
I-134		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$9.9X10^{5}$	$2.7X10^{7}$
I-135 (a)		$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.3X10^{5}$	$3.5X10^{6}$
In-111	Indium (49)	3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$1.5X10^{4}$	$4.2X10^{5}$
In-113m		4.0	$1.1X10^{2}$	2.0	$5.4X10^{1}$	$6.2X10^{5}$	$1.7X10^{7}$
In-114m (a)		$1.0X10^{1}$	$2.7X10^{2}$	$5.0X10^{-1}$	$1.4X10^{1}$	$8.6X10^{2}$	$2.3X10^{4}$
In-115m		7.0	$1.9X10^{2}$	1.0	$2.7X10^{1}$	$2.2X10^{5}$	$6.1X10^{6}$
Ir-189 (a)	Iridium (77)	$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$1.9X10^{3}$	$5.2X10^{4}$
Ir-190		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$2.3X10^{3}$	$6.2X10^{4}$
Ir-192		1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.4X10^{2}$	$9.2X10^{3}$
Ir-194		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$3.1X10^{4}$	$8.4X10^{5}$
K-40	Potassium (19)	$9.0X10^{-1}$	$2.4X10^{1}$	$9.0X10^{-1}$	$2.4X10^{1}$	$2.4X10^{-7}$	$6.4X10^{-6}$
K-42		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$2.2X10^{5}$	$6.0X10^{6}$
K-43		$7.0X10^{-1}$	$1.9X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.2X10^{5}$	$3.3X10^{6}$

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Kr-81	Krypton (36)	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$7.8X10^{-4}$	$2.1X10^{-2}$
Kr-85		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$1.5X10^{1}$	$3.9X10^{2}$
Kr-85m		8.0	$2.2X10^{2}$	3.0	$8.1X10^{1}$	$3.0X10^{5}$	$8.2X10^{6}$
Kr-87		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$1.0X10^{6}$	$2.8X10^{7}$
La-137	Lanthanum (57)	$3.0X10^{1}$	$8.1X10^{2}$	6.0	$1.6X10^{2}$	$1.6X10^{-3}$	$4.4X10^{-2}$
La-140		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$2.1X10^{4}$	$5.6X10^{5}$
Lu-172	Lutetium (71)	$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$4.2X10^{3}$	$1.1X10^{5}$
Lu-173		8.0	$2.2X10^{2}$	8.0	$2.2X10^{2}$	$5.6X10^{1}$	$1.5X10^{3}$
Lu-174		9.0	$2.4X10^{2}$	9.0	$2.4X10^{2}$	$2.3X10^{1}$	$6.2X10^{2}$
Lu-174m		$2.0X10^{1}$	$5.4X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$2.0X10^{2}$	$5.3X10^{3}$
Lu-177		$3.0X10^{1}$	$8.1X10^{2}$	$7.0X10^{-1}$	$1.9X10^{1}$	$4.1X10^{3}$	$1.1X10^{5}$
Mg-28 (a)	Magnesium (12)	$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$2.0X10^{5}$	$5.4X10^{6}$
Mn-52	Manganese (25)	$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.6X10^{4}$	$4.4X10^{5}$
Mn-53		Unlimited	Unlimited	Unlimited	Unlimited	$6.8X10^{-5}$	$1.8X10^{-3}$
Mn-54		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$2.9X10^{2}$	$7.7X10^{3}$
Mn-56		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$8.0X10^{5}$	$2.2X10^{7}$
Mo-93	Molybdenum (42)	$4.0X10^{1}$	$1.1X10^{3}$	$2.0X10^{1}$	$5.4X10^{2}$	$4.1X10^{-2}$	1.1
Mo-99 (a) (h)		1.0	$2.7X10^{1}$	$7.4X10^{-1}$	$2.0X10^{1}$	$1.8X10^{4}$	$4.8X10^{5}$
N-13	Nitrogen (7)	$9.0X10^{-1}$	$2.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$5.4X10^{7}$	$1.5X10^{9}$
Na-22	Sodium (11)	$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$2.3X10^{2}$	$6.3X10^{3}$
Na-24		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$3.2X10^{5}$	$8.7X10^{6}$

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES							
Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Nb-93m	Niobium (41)	$4.0X10^{1}$	$1.1X10^{3}$	$3.0X10^{1}$	$8.1X10^{2}$	8.8	$2.4X10^{2}$
Nb-94		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$6.9X10^{-3}$	$1.9X10^{-1}$
Nb-95		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$1.5X10^{3}$	$3.9X10^{4}$
Nb-97		$9.0X10^{-1}$	$2.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$9.9X10^{5}$	$2.7X10^{7}$
Nd-147	Neodymium (60)	6.0	$1.6X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.0X10^{3}$	$8.1X10^{4}$
Nd-149		$6.0X10^{-1}$	$1.6X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$4.5X10^{5}$	$1.2X10^{7}$
Ni-59	Nickel (28)	Unlimited	Unlimited	Unlimited	Unlimited	$3.0X10^{-3}$	$8.0X10^{-2}$
Ni-63		$4.0X10^{1}$	$1.1X10^{3}$	$3.0X10^{1}$	$8.1X10^{2}$	2.1	$5.7X10^{1}$
Ni-65		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$7.1X10^{5}$	$1.9X10^{7}$
Np-235	Neptunium (93)	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$5.2X10^{1}$	$1.4X10^{3}$
Np-236 (short-lived)		$2.0X10^{1}$	$5.4X10^{2}$	2.0	$5.4X10^{1}$	$4.7X10^{-4}$	$1.3X10^{-2}$
Np-236 (long-lived)		$2.0X10^{1}$	$5.4X10^{2}$	2.0	$5.4X10^{1}$	$4.7X10^{-4}$	$1.3X10^{-2}$
Np-237		$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{-3}$	$5.4X10^{-2}$	$2.6X10^{-5}$	$7.1X10^{-4}$
Np-239		7.0	$1.9X10^{2}$	$4.0X10^{-1}$	$1.1X10^{1}$	$8.6X10^{3}$	$2.3X10^{5}$
Os-185	Osmium (76)	1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$2.8X10^{2}$	$7.5X10^{3}$
Os-191		$1.0X10^{1}$	$2.7X10^{2}$	2.0	$5.4X10^{1}$	$1.6X10^{3}$	$4.4X10^{4}$
Os-191m		$4.0X10^{1}$	$1.1X10^{3}$	$3.0X10^{1}$	$8.1X10^{2}$	$4.6X10^{4}$	$1.3X10^{6}$
Os-193		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.0X10^{4}$	$5.3X10^{5}$
Os-194 (a)		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.1X10^{1}$	$3.1X10^{2}$
P-32	Phosphorus (15)	$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$1.1X10^{4}$	$2.9X10^{5}$
P-33		$4.0X10^{1}$	$1.1X10^{3}$	1.0	$2.7X10^{1}$	$5.8X10^{3}$	$1.6X10^{5}$

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES							
Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Pa-230 (a)	Protactinium (91)	2.0	$5.4X10^{1}$	$7.0X10^{-2}$	1.9	$1.2X10^{3}$	$3.3X10^{4}$
Pa-231		4.0	$1.1X10^{2}$	$4.0X10^{-4}$	$1.1X10^{-2}$	$1.7X10^{-3}$	$4.7X10^{-2}$
Pa-233		5.0	$1.4X10^{2}$	$7.0X10^{-1}$	$1.9X10^{1}$	$7.7X10^{2}$	$2.1X10^{4}$
Pb-201	Lead (82)	1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$6.2X10^{4}$	$1.7X10^{6}$
Pb-202		$4.0X10^{1}$	$1.1X10^{3}$	$2.0X10^{1}$	$5.4X10^{2}$	$1.2X10^{-4}$	$3.4X10^{-3}$
Pb-203		4.0	$1.1X10^{2}$	3.0	$8.1X10^{1}$	$1.1X10^{4}$	$3.0X10^{5}$
Pb-205		Unlimited	Unlimited	Unlimited	Unlimited	$4.5X10^{-6}$	$1.2X10^{-4}$
Pb-210 (a)		1.0	$2.7X10^{1}$	$5.0X10^{-2}$	1.4	2.8	$7.6X10^{1}$
Pb-212 (a)		$7.0X10^{-1}$	$1.9X10^{1}$	$2.0X10^{-1}$	5.4	$5.1X10^{4}$	$1.4X10^{6}$
Pd-103 (a)	Palladium (46)	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$2.8X10^{3}$	$7.5X10^{4}$
Pd-107		Unlimited	Unlimited	Unlimited	Unlimited	$1.9X10^{-5}$	$5.1X10^{-4}$
Pd-109		2.0	$5.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$7.9X10^{4}$	$2.1X10^{6}$
Pm-143	Promethium (61)	3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$1.3X10^{2}$	$3.4X10^{3}$
Pm-144		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$9.2X10^{1}$	$2.5X10^{3}$
Pm-145		$3.0X10^{1}$	$8.1X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	5.2	$1.4X10^{2}$
Pm-147		$4.0X10^{1}$	$1.1X10^{3}$	2.0	$5.4X10^{1}$	$3.4X10^{1}$	$9.3X10^{2}$
Pm-148m (a)		$8.0X10^{-1}$	$2.2X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$7.9X10^{2}$	$2.1X10^{4}$
Pm-149		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.5X10^{4}$	$4.0X10^{5}$
Pm-151		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.7X10^{4}$	$7.3X10^{5}$
Po-210	Polonium (84)	$4.0X10^{1}$	$1.1X10^{3}$	$2.0X10^{-2}$	$5.4X10^{-1}$	$1.7X10^{2}$	$4.5X10^{3}$
Pr-142	Praseodymium (59)	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$4.3X10^{4}$	$1.2X10^{6}$
Pr-143		3.0	$8.1X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.5X10^{3}$	$6.7X10^{4}$

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES							
Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Pt-188 (a)	Platinum (78)	1.0	$2.7X10^{1}$	$8.0X10^{-1}$	$2.2X10^{1}$	$2.5X10^{3}$	$6.8X10^{4}$
Pt-191		4.0	$1.1X10^{2}$	3.0	$8.1X10^{1}$	$8.7X10^{3}$	$2.4X10^{5}$
Pt-193		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	1.4	$3.7X10^{1}$
Pt-193m		$4.0X10^{1}$	$1.1X10^{3}$	$5.0X10^{-1}$	$1.4X10^{1}$	$5.8X10^{3}$	$1.6X10^{5}$
Pt-195m		$1.0X10^{1}$	$2.7X10^{2}$	$5.0X10^{-1}$	$1.4X10^{1}$	$6.2X10^{3}$	$1.7X10^{5}$
Pt-197		$2.0X10^{1}$	$5.4X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.2X10^{4}$	$8.7X10^{5}$
Pt-197m		$1.0X10^{1}$	$2.7X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.7X10^{5}$	$1.0X10^{7}$
Pu-236	Plutonium (94)	$3.0X10^{1}$	$8.1X10^{2}$	$3.0X10^{-3}$	$8.1X10^{-2}$	$2.0X10^{1}$	$5.3X10^{2}$
Pu-237		$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{1}$	$5.4X10^{2}$	$4.5X10^{2}$	$1.2X10^{4}$
Pu-238		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$6.3X10^{-1}$	$1.7X10^{1}$
Pu-239		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$2.3X10^{-3}$	$6.2X10^{-2}$
Pu-240		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$8.4X10^{-3}$	$2.3X10^{-1}$
Pu-241 (a)		$4.0X10^{1}$	$1.1X10^{3}$	$6.0X10^{-2}$	1.6	3.8	$1.0X10^{2}$
Pu-242		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$1.5X10^{-4}$	$3.9X10^{-3}$
Pu-244 (a)		$4.0X10^{-1}$	$1.1X10^{1}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$6.7X10^{-7}$	$1.8X10^{-5}$
Ra-223 (a)	Radium (88)	$4.0X10^{-1}$	$1.1X10^{1}$	$7.0X10^{-3}$	$1.9X10^{-1}$	$1.9X10^{3}$	$5.1X10^{4}$
Ra-224 (a)		$4.0X10^{-1}$	$1.1X10^{1}$	$2.0X10^{-2}$	$5.4X10^{-1}$	$5.9X10^{3}$	$1.6X10^{5}$
Ra-225 (a)		$2.0X10^{-1}$	5.4	$4.0X10^{-3}$	$1.1X10^{-1}$	$1.5X10^{3}$	$3.9X10^{4}$
Ra-226 (a)		$2.0X10^{-1}$	5.4	$3.0X10^{-3}$	$8.1X10^{-2}$	$3.7X10^{-2}$	1.0
Ra-228 (a)		$6.0X10^{-1}$	$1.6X10^{1}$	$2.0X10^{-2}$	$5.4X10^{-1}$	$1.0X10^{1}$	$2.7X10^{2}$

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Rb-81	Rubidium (37)	2.0	$5.4X10^{1}$	$8.0X10^{-1}$	$2.2X10^{1}$	$3.1X10^{5}$	$8.4X10^{6}$
Rb-83 (a)		2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$6.8X10^{2}$	$1.8X10^{4}$
Rb-84		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$1.8X10^{3}$	$4.7X10^{4}$
Rb-86		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$3.0X10^{3}$	$8.1X10^{4}$
Rb-87		Unlimited	Unlimited	Unlimited	Unlimited	$3.2X10^{-9}$	$8.6X10^{-8}$
Rb(nat)		Unlimited	Unlimited	Unlimited	Unlimited	$6.7X10^{6}$	$1.8X10^{8}$
Re-184	Rhenium (75)	1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$6.9X10^{2}$	$1.9X10^{4}$
Re-184m		3.0	$8.1X10^{1}$	1.0	$2.7X10^{1}$	$1.6X10^{2}$	$4.3X10^{3}$
Re-186		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$6.9X10^{3}$	$1.9X10^{5}$
Re-187		Unlimited	Unlimited	Unlimited	Unlimited	$1.4X10^{-9}$	$3.8X10^{-8}$
Re-188		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$3.6X10^{4}$	$9.8X10^{5}$
Re-189 (a)		3.0	$8.1X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.5X10^{4}$	$6.8X10^{5}$
Re(nat)		Unlimited	Unlimited	Unlimited	Unlimited	0.0	$2.4X10^{-8}$
Rh-99	Rhodium (45)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$3.0X10^{3}$	$8.2X10^{4}$
Rh-101		4.0	$1.1X10^{2}$	3.0	$8.1X10^{1}$	$4.1X10^{1}$	$1.1X10^{3}$
Rh-102		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$4.5X10^{1}$	$1.2X10^{3}$
Rh-102m		2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$2.3X10^{2}$	$6.2X10^{3}$
Rh-103m		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$1.2X10^{6}$	$3.3X10^{7}$
Rh-105		$1.0X10^{1}$	$2.7X10^{2}$	$8.0X10^{-1}$	$2.2X10^{1}$	$3.1X10^{4}$	$8.4X10^{5}$
Rn-222 (a)	Radon (86)	$3.0X10^{-1}$	8.1	$4.0X10^{-3}$	$1.1X10^{-1}$	$5.7X10^{3}$	$1.5X10^{5}$

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES							
Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Ru-97	Ruthenium (44)	5.0	$1.4X10^{2}$	5.0	$1.4X10^{2}$	$1.7X10^{4}$	$4.6X10^{5}$
Ru-103 (a)		2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$1.2X10^{3}$	$3.2X10^{4}$
Ru-105		1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.5X10^{5}$	$6.7X10^{6}$
Ru-106 (a)		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$1.2X10^{2}$	$3.3X10^{3}$
S-35	Sulphur (16)	$4.0X10^{1}$	$1.1X10^{3}$	3.0	$8.1X10^{1}$	$1.6X10^{3}$	$4.3X10^{4}$
Sb-122	Antimony (51)	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.5X10^{4}$	$4.0X10^{5}$
Sb-124		$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$6.5X10^{2}$	$1.7X10^{4}$
Sb-125		2.0	$5.4X10^{1}$	1.0	$2.7X10^{1}$	$3.9X10^{1}$	$1.0X10^{3}$
Sb-126		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$3.1X10^{3}$	$8.4X10^{4}$
Sc-44	Scandium (21)	$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$6.7X10^{5}$	$1.8X10^{7}$
Sc-46		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$1.3X10^{3}$	$3.4X10^{4}$
Sc-47		$1.0X10^{1}$	$2.7X10^{2}$	$7.0X10^{-1}$	$1.9X10^{1}$	$3.1X10^{4}$	$8.3X10^{5}$
Sc-48		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$5.5X10^{4}$	$1.5X10^{6}$
Se-75	Selenium (34)	3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$5.4X10^{2}$	$1.5X10^{4}$
Se-79		$4.0X10^{1}$	$1.1X10^{3}$	2.0	$5.4X10^{1}$	$2.6X10^{-3}$	$7.0X10^{-2}$
Si-31	Silicon (14)	$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.4X10^{6}$	$3.9X10^{7}$
Si-32		$4.0X10^{1}$	$1.1X10^{3}$	$5.0X10^{-1}$	$1.4X10^{1}$	3.9	$1.1X10^{2}$
Sm-145	Samarium (62)	$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$9.8X10^{1}$	$2.6X10^{3}$
Sm-147		Unlimited	Unlimited	Unlimited	Unlimited	$8.5X10^{-1}$	$2.3X10^{-8}$
Sm-151		$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{1}$	$2.7X10^{2}$	$9.7X10^{-1}$	$2.6X10^{1}$
Sm-153		9.0	$2.4X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.6X10^{4}$	$4.4X10^{5}$

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Sn-113 (a)	Tin (50)	4.0	$1.1X10^{2}$	2.0	$5.4X10^{1}$	$3.7X10^{2}$	$1.0X10^{4}$
Sn-117m		7.0	$1.9X10^{2}$	$4.0X10^{-1}$	$1.1X10^{1}$	$3.0X10^{3}$	$8.2X10^{4}$
Sn-119m		$4.0X10^{1}$	$1.1X10^{3}$	$3.0X10^{1}$	$8.1X10^{2}$	$1.4X10^{2}$	$3.7X10^{3}$
Sn-121m (a)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-1}$	$2.4X10^{1}$	2.0	$5.4X10^{1}$
Sn-123		$8.0X10^{-1}$	$2.2X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.0X10^{2}$	$8.2X10^{3}$
Sn-125		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{3}$	$1.1X10^{5}$
Sn-126 (a)		$6.0X10^{-1}$	$1.6X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.0X10^{-3}$	$2.8X10^{-2}$
Sr-82 (a)	Strontium (38)	$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$2.3X10^{3}$	$6.2X10^{4}$
Sr-85		2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$8.8X10^{2}$	$2.4X10^{4}$
Sr-85m		5.0	$1.4X10^{2}$	5.0	$1.4X10^{2}$	$1.2X10^{6}$	$3.3X10^{7}$
Sr-87m		3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$4.8X10^{5}$	$1.3X10^{7}$
Sr-89		$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.1X10^{3}$	$2.9X10^{4}$
Sr-90 (a)		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	5.1	$1.4X10^{2}$
Sr-91 (a)		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.3X10^{5}$	$3.6X10^{6}$
Sr-92 (a)		1.0	$2.7X10^{1}$	$3.0X10^{-1}$	8.1	$4.7X10^{5}$	$1.3X10^{7}$
T(H-3)	Tritium (1)	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$3.6X10^{2}$	$9.7X10^{3}$
Ta-178 (long-lived)	Tantalum (73)	1.0	$2.7X10^{1}$	$8.0X10^{-1}$	$2.2X10^{1}$	$4.2X10^{6}$	$1.1X10^{8}$
Ta-179		$3.0X10^{1}$	$8.1X10^{2}$	$3.0X10^{1}$	$8.1X10^{2}$	$4.1X10^{1}$	$1.1X10^{3}$
Ta-182		$9.0X10^{-1}$	$2.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$2.3X10^{2}$	$6.2X10^{3}$
Tb-157	Terbium (65)	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$5.6X10^{-1}$	$1.5X10^{1}$
Tb-158		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$5.6X10^{-1}$	$1.5X10^{1}$
Tb-160		1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$4.2X10^{2}$	$1.1X10^{4}$

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Tc-95m (a)	Technetium (43)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$8.3X10^{2}$	$2.2X10^{4}$
Tc-96		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.2X10^{4}$	$3.2X10^{5}$
Tc-96m (a)		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.4X10^{6}$	$3.8X10^{7}$
Tc-97		Unlimited	Unlimited	Unlimited	Unlimited	$5.2X10^{-5}$	$1.4X10^{-3}$
Tc-97m		$4.0X10^{1}$	$1.1X10^{3}$	1.0	$2.7X10^{1}$	$5.6X10^{2}$	$1.5X10^{4}$
Tc-98		$8.0X10^{-1}$	$2.2X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$3.2X10^{-5}$	$8.7X10^{-4}$
Tc-99		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-1}$	$2.4X10^{1}$	$6.3X10^{-4}$	$1.7X10^{-2}$
Tc-99m		$1.0X10^{1}$	$2.7X10^{2}$	4.0	$1.1X10^{2}$	$1.9X10^{5}$	$5.3X10^{6}$
Te-121	Tellurium (52)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$2.4X10^{3}$	$6.4X10^{4}$
Te-121m		5.0	$1.4X10^{2}$	3.0	$8.1X10^{1}$	$2.6X10^{2}$	$7.0X10^{3}$
Te-123m		8.0	$2.2X10^{2}$	1.0	$2.7X10^{1}$	$3.3X10^{2}$	$8.9X10^{3}$
Te-125m		$2.0X10^{1}$	$5.4X10^{2}$	$9.0X10^{-1}$	$2.4X10^{1}$	$6.7X10^{2}$	$1.8X10^{4}$
Te-127		$2.0X10^{1}$	$5.4X10^{2}$	$7.0X10^{-1}$	$1.9X10^{1}$	$9.8X10^{4}$	$2.6X10^{6}$
Te-127m (a)		$2.0X10^{1}$	$5.4X10^{2}$	$5.0X10^{-1}$	$1.4X10^{1}$	$3.5X10^{2}$	$9.4X10^{3}$
Te-129		$7.0X10^{-1}$	$1.9X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$7.7X10^{5}$	$2.1X10^{7}$
Te-129m (a)		$8.0X10^{-1}$	$2.2X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.1X10^{3}$	$3.0X10^{4}$
Te-131m (a)		$7.0X10^{-1}$	$1.9X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$3.0X10^{4}$	$8.0X10^{5}$
Te-132 (a)		$5.0X10^{-1}$	$1.4X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.1X10^{4}$	$8.0X10^{5}$
Th-227	Thorium (90)	$1.0X10^{1}$	$2.7X10^{2}$	$5.0X10^{-3}$	$1.4X10^{-1}$	$1.1X10^{3}$	$3.1X10^{4}$
Th-228 (a)		$5.0X10^{-1}$	$1.4X10^{1}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$3.0X10^{1}$	$8.2X10^{2}$
Th-229		5.0	$1.4X10^{2}$	$5.0X10^{-4}$	$1.4X10^{-2}$	$7.9X10^{-3}$	$2.1X10^{-1}$
Th-230		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$7.6X10^{-4}$	$2.1X10^{-2}$

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Th-231	Thorium (90)	$4.0X10^{1}$	$1.1X10^{3}$	$2.0X10^{-2}$	$5.4X10^{-1}$	$2.0X10^{4}$	$5.3X10^{5}$
Th-232		Unlimited	Unlimited	Unlimited	Unlimited	$4.0X10^{-9}$	$1.1X10^{-7}$
Th-234 (a)		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$8.6X10^{2}$	$2.3X10^{4}$
Th(nat)		Unlimited	Unlimited	Unlimited	Unlimited	$8.1X10^{-9}$	$2.2X10^{-7}$
Ti-44 (a)	Titanium (22)	$5.0X10^{-1}$	$1.4X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	6.4	$1.7X10^{2}$
Tl-200	Thallium (81)	$9.0X10^{-1}$	$2.4X10^{1}$	$9.0X10^{-1}$	$2.4X10^{1}$	$2.2X10^{4}$	$6.0X10^{5}$
Tl-201		$1.0X10^{1}$	$2.7X10^{2}$	4.0	$1.1X10^{2}$	$7.9X10^{3}$	$2.1X10^{5}$
Tl-202		2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$2.0X10^{3}$	$5.3X10^{4}$
Tl-204		$1.0X10^{1}$	$2.7X10^{2}$	$7.0X10^{-1}$	$1.9X10^{1}$	$1.7X10^{1}$	$4.6X10^{2}$
Tm-167	Thulium (69)	7.0	$1.9X10^{2}$	$8.0X10^{-1}$	$2.2X10^{1}$	$3.1X10^{3}$	$8.5X10^{4}$
Tm-170		3.0	$8.1X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.2X10^{2}$	$6.0X10^{3}$
Tm-171		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$
U-230 (fast lung absorption) (a)(d)	Uranium (92)	$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-1}$	2.7	$1.0X10^{3}$	$2.7X10^{4}$
U-230 (medium lung absorption) (a)(e)		$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-1}$	2.7	$1.0X10^{3}$	$2.7X10^{4}$
U-230 (slow lung absorption) (a)(f)		$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-1}$	2.7	$1.0X10^{3}$	$2.7X10^{4}$
U-232 (fast lung absorption) (d)		$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-2}$	$2.7X10^{-1}$	$8.3X10^{-1}$	$2.2X10^{1}$
U-232 (medium lung absorption) (e)		$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-2}$	$2.7X10^{-1}$	$8.3X10^{-1}$	$2.2X10^{1}$
U-232 (slow lung absorption) (f)		$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-2}$	$2.7X10^{-1}$	$8.3X10^{-1}$	$2.2X10^{1}$

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES							
Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
U-233 (fast lung absorption) (d)	Uranium (92)	$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-2}$	2.4	$3.6X10^{-4}$	$9.7X10^{-3}$
U-233 (medium lung absorption) (e)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-2}$	2.4	$3.6X10^{-4}$	$9.7X10^{-3}$
U-233 (slow lung absorption) (f)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-2}$	2.4	$3.6X10^{-4}$	$9.7X10^{-3}$
U-234 (fast lung absorption) (d)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-2}$	2.4	$2.3X10^{-4}$	$6.2X10^{-3}$
U-234 (medium lung absorption) (e)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-2}$	2.4	$2.3X10^{-4}$	$6.2X10^{-3}$
U-234 (slow lung absorption) (f)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-2}$	2.4	$2.3X10^{-4}$	$6.2X10^{-3}$
U-235 (all lung absorption types) (a),(d),(e),(f)		Unlimited	Unlimited	Unlimited	Unlimited	$8.0X10^{-8}$	$2.2X10^{-6}$
U-236 (fast lung absorption) (d)		Unlimited	Unlimited	Unlimited	Unlimited	$2.4X10^{-6}$	$6.5X10^{-5}$
U-236 (medium lung absorption) (e)		Unlimited	Unlimited	Unlimited	Unlimited	$2.4X10^{-6}$	$6.5X10^{-5}$
U-236 (slow lung absorption) (f)		Unlimited	Unlimited	Unlimited	Unlimited	$2.4X10^{-6}$	$6.5X10^{-5}$
U-238 (all lung absorption types) (d),(e),(f)		Unlimited	Unlimited	Unlimited	Unlimited	$1.2X10^{-8}$	$3.4X10^{-7}$
U (nat)		Unlimited	Unlimited	Unlimited	Unlimited	$2.6X10^{-8}$	$7.1X10^{-7}$
U (enriched to 20% or less)(g)		Unlimited	Unlimited	Unlimited	Unlimited	N/A	N/A
U (dep)		Unlimited	Unlimited	Unlimited	Unlimited	0.0	(See Table A-3)
V-48	Vanadium (23)	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$6.3X10^{3}$	$1.7X10^{5}$
V-49		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$3.0X10^{2}$	$8.1X10^{3}$

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES							
Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
W-178 (a)	Tungsten (74)	9.0	$2.4X10^{2}$	5.0	$1.4X10^{2}$	$1.3X10^{3}$	$3.4X10^{4}$
W-181		$3.0X10^{1}$	$8.1X10^{2}$	$3.0X10^{1}$	$8.1X10^{2}$	$2.2X10^{2}$	$6.0X10^{3}$
W-185		$4.0X10^{1}$	$1.1X10^{3}$	$8.0X10^{-1}$	$2.2X10^{1}$	$3.5X10^{2}$	$9.4X10^{3}$
W-187		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.6X10^{4}$	$7.0X10^{5}$
W-188 (a)		$4.0X10^{-1}$	$1.1X10^{1}$	$3.0X10^{-1}$	8.1	$3.7X10^{2}$	$1.0X10^{4}$
Xe-122 (a)	Xenon (54)	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$4.8X10^{4}$	$1.3X10^{6}$
Xe-123		2.0	$5.4X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$4.4X10^{5}$	$1.2X10^{7}$
Xe-127		4.0	$1.1X10^{2}$	2.0	$5.4X10^{1}$	$1.0X10^{3}$	$2.8X10^{4}$
Xe-131m		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$3.1X10^{3}$	$8.4X10^{4}$
Xe-133		$2.0X10^{1}$	$5.4X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$6.9X10^{3}$	$1.9X10^{5}$
Xe-135		3.0	$8.1X10^{1}$	2.0	$5.4X10^{1}$	$9.5X10^{4}$	$2.6X10^{6}$
Y-87 (a)	Yttrium (39)	1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$1.7X10^{4}$	$4.5X10^{5}$
Y-88		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$5.2X10^{2}$	$1.4X10^{4}$
Y-90		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$2.0X10^{4}$	$5.4X10^{5}$
Y-91		$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$9.1X10^{2}$	$2.5X10^{4}$
Y-91m		2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$1.5X10^{6}$	$4.2X10^{7}$
Y-92		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$3.6X10^{5}$	$9.6X10^{6}$
Y-93		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.2X10^{5}$	$3.3X10^{6}$
Yb-169	Ytterbium (79)	4.0	$1.1X10^{2}$	1.0	$2.7X10^{1}$	$8.9X10^{2}$	$2.4X10^{4}$
Yb-175		$3.0X10^{1}$	$8.1X10^{2}$	$9.0X10^{-1}$	$2.4X10^{1}$	$6.6X10^{3}$	$1.8X10^{5}$

TABLE A - 1: A_1 AND A_2 VALUES FOR RADIONUCLIDES							
Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Zn-65	Zinc (30)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$3.0X10^{2}$	$8.2X10^{3}$
Zn-69		3.0	$8.1X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.8X10^{6}$	$4.9X10^{7}$
Zn-69m (a)		3.0	$8.1X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.2X10^{5}$	$3.3X10^{6}$
Zr-88	Zirconium (40)	3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$6.6X10^{2}$	$1.8X10^{4}$
Zr-93		Unlimited	Unlimited	Unlimited	Unlimited	$9.3X10^{-5}$	$2.5X10^{-3}$
Zr-95 (a)		2.0	$5.4X10^{1}$	$8.0X10^{-1}$	$2.2X10^{1}$	$7.9X10^{2}$	$2.1X10^{4}$
Zr-97 (a)		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$7.1X10^{4}$	$1.9X10^{6}$

NOTES
(a) A1 and/or A2 values include contributions from daughter nuclides with half-lives less than 10 days
(b) Parent nuclides and their progeny included in secular equilibrium are listed in the following:

Sr-90	Y-90
Zr-93	Nb-93m
Zr-97	Nb-97
Ru-106	Rh-106
Cs-137	Ba-137m
Ce-134	La-134
Ce-144	Pr-144
Ba-140	La-140
Bi-212	Tl-208 (0.36), Po-212 (0.64)
Pb-210	Bi-210, Po-210
Pb-212	Bi-212, Tl-208 (0.36), Po-212 (0.64)
Rn-220	Po-216
Rn-222	Po-218, Pb-214, Bi-214, Po-214
Ra-223	Rn-219, Po-215, Pb-211, Bi-211, Tl-207
Ra-224	Rn-220, Po-216, Pb-212, Bi-212, Tl-208 (0.36), Po-212 (0.64)
Ra-226	Rn-222, Po-218, Pb-214, Bi-214, Po-214, Pb-210, Bi-210, Po-210
Ra-228	Ac-228
Th-226	Ra-222, Rn-218, Po-214
Th-228	Ra-224, Rn-220, Po-216, Pb212, Bi-212, Tl208 (0.36), Po-212 (0.64)
Th-229	Ra-225, Ac-225, Fr-221, At-217, Bi-213, Po-213, Pb-209
Th-nat	Ra-228, Ac-228, Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208 (0.36), Po-212 (0.64)
Th-234	Pa-234m
U-230	Th-226, Ra-222, Rn-218, Po-214
U-232	Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208 (0.36), Po-212 (0.64)
U-235	Th-231

U-238	Th-234, Pa-234m
U-nat	Th-234, Pa-234m, U-234, Th-230, Ra-226, Rn-222, Po-218, Pb-214, Bi-214, Po-214,
U-240	Np-240m
Np-237	Pa-233
Am-242m	Am-242
Am-243	Np-239

(c) The quantity may be determined from a measurement of the rate of decay or a measurement of the radiation level at a prescribed distance from the source.

(d) These values apply only to compounds of uranium that take the chemical form of UF6, UO2F2 and UO2(NO3)2 in both normal and accident conditions of transport.

(e) These values apply only to compounds of uranium that take the chemical form of UO3, UF4, UCl4 and hexavalent compounds in both normal and accident conditions of transport.

(f) These values apply to all compounds of uranium other than those specified in (d) and (e) above.

(g) These values apply to unirradiated uranium only.

(h) These values apply to domestic transport only. For international transport use the values in the table below.

A_1 AND A_2 VALUES FOR RADIONUCLIDES FOR INTERNATIONAL SHIPMENTS

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Cf-252	Californium (98)	5.0×10^{-2}	1.4	3.0×10^{-3}	8.1×10^{-2}	2.0×10^{1}	5.4×10^{2}
Mo-99 (a)	Molybdenum (42)	1.0	2.7×10^{1}	6.0×10^{-1}	1.6×10^{1}	1.8×10^{4}	4.8×10^{5}

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Ac-225 (a)	Actinium (89)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Ac-227 (a)		$1.0X10^{-1}$	$2.7X10^{-12}$	$1.0X10^{3}$	$2.7X10^{-8}$
Ac-228		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ag-105	Silver (47)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ag-108m (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ag-110m (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ag-111		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Al-26	Aluminum (13)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Am-241	Americium (95)	1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Am-242m (a)		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Am-243 (a)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Ar-37	Argon (18)	$1.0X10^{6}$	$2.7X10^{-5}$	$1.0X10^{8}$	$2.7X10^{-3}$
Ar-39		$1.0X10^{7}$	$2.7X10^{-4}$	$1.0X10^{4}$	$2.7X10^{-7}$
Ar-41		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{9}$	$2.7X10^{-2}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
As-72	Arsenic (33)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
As-73		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
As-74		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
As-76		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
As-77		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
At-211 (a)	Astatine (85)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Au-193	Gold (79)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Au-194		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Au-195		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Au-198		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Au-199		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ba-131 (a)	Barium (56)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ba-133		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ba-133m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ba-140 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Be-7	Beryllium (4)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Be-10		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{6}$	$2.7X10^{-5}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Bi-205	Bismuth (83)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Bi-206		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Bi-207		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Bi-210		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Bi-210m (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Bi-212 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Bk-247	Berkelium (97)	1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Bk-249 (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Br-76	Bromine (35)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Br-77		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Br-82		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
C-11	Carbon (6)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
C-14		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Ca-41	Calcium (20)	$1.0X10^{5}$	$2.7X10^{-6}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ca-45		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ca-47 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cd-109	Cadmium (48)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cd-113m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cd-115 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cd-115m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ce-139	Cerium (58)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ce-141		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ce-143		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ce-144 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cf-248	Californium (98)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cf-249		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Cf-250		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cf-251		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Cf-252		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cf-253 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Cf-254		1.0	2.7×10^{-11}	1.0×10^{3}	2.7×10^{-8}
Cl-36	Chlorine (17)	1.0×10^{4}	2.7×10^{-7}	1.0×10^{6}	2.7×10^{-5}
Cl-38		1.0×10^{1}	2.7×10^{-10}	1.0×10^{5}	2.7×10^{-6}
Cm-240	Curium (96)	1.0×10^{2}	2.7×10^{-9}	1.0×10^{5}	2.7×10^{-6}
Cm-241		1.0×10^{2}	2.7×10^{-9}	1.0×10^{6}	2.7×10^{-5}
Cm-242		1.0×10^{2}	2.7×10^{-9}	1.0×10^{5}	2.7×10^{-6}
Cm-243		1.0	2.7×10^{-11}	1.0×10^{4}	2.7×10^{-7}
Cm-244		1.0×10^{1}	2.7×10^{-10}	1.0×10^{4}	2.7×10^{-7}
Cm-245		1.0	2.7×10^{-11}	1.0×10^{3}	2.7×10^{-8}
Cm-246		1.0	2.7×10^{-11}	1.0×10^{3}	2.7×10^{-8}
Cm-247 (a)		1.0	2.7×10^{-11}	1.0×10^{4}	2.7×10^{-7}
Cm-248		1.0	2.7×10^{-11}	1.0×10^{3}	2.7×10^{-8}
Co-55	Cobalt (27)	1.0×10^{1}	2.7×10^{-10}	1.0×10^{6}	2.7×10^{-5}
Co-56		1.0×10^{1}	2.7×10^{-10}	1.0×10^{5}	2.7×10^{-6}
Co-57		1.0×10^{2}	2.7×10^{-9}	1.0×10^{6}	2.7×10^{-5}
Co-58		1.0×10^{1}	2.7×10^{-10}	1.0×10^{6}	2.7×10^{-5}
Co-58m		1.0×10^{4}	2.7×10^{-7}	1.0×10^{7}	2.7×10^{-4}

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Co-60		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cr-51	Chromium (24)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Cs-129	Cesium (55)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cs-131		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cs-132		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cs-134		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cs-134m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cs-135		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Cs-136		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cs-137 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cu-64	Copper (29)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cu-67		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Dy-159	Dysprosium (66)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Dy-165		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Dy-166 (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Er-169	Erbium (68)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Er-171		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Eu-147	Europium (63)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-148		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-149		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Eu-150 (short lived)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-150 (long lived)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-152		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-152 m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-154		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-155		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Eu-156		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
F-18	Fluorine (9)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Fe-52 (a)	Iron (26)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Fe-55		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{6}$	$2.7X10^{-5}$
Fe-59		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Fe-60 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES

Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Ga-67	Gallium (31)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ga-68		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ga-72		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Gd-146 (a)	Gadolinium (64)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Gd-148		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Gd-153		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Gd-159		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ge-68 (a)	Germanium (32)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ge-71		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{8}$	$2.7X10^{-3}$
Ge-77		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Hf-172 (a)	Hafnium (72)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hf-175		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hf-181		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hf-182		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Hg-194 (a)	Mercury (80)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hg-195m (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hg-197		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Hg-197m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hg-203		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ho-166	Holmium (67)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ho-166m		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
I-123	Iodine (53)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
I-124		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
I-125		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
I-126		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
I-129		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
I-131		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
I-132		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
I-133		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
I-134		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
I-135 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
In-111	Indium (49)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
In-113m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
In-114m (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
In-115m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ir-189 (a)	Iridium (77)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ir-190		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ir-192		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Ir-194		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
K-40	Potassium (19)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
K-42		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
K-43		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Kr-81	Krypton (36)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Kr-85		$1.0X10^{5}$	$2.7X10^{-6}$	$1.0X10^{4}$	$2.7X10^{-7}$
Kr-85m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{10}$	$2.7X10^{-1}$
Kr-87		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{9}$	$2.7X10^{-2}$
La-137	Lanthanum (57)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
La-140		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Lu-172	Lutetium (71)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Lu-173		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Lu-174		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Lu-174m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Lu-177		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Mg-28 (a)	Magnesium (12)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Mn-52	Manganese (25)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Mn-53		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{9}$	$2.7X10^{-2}$
Mn-54		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Mn-56		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Mo-93	Molybdenum (42)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{8}$	$2.7X10^{-3}$
Mo-99 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
N-13	Nitrogen (7)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{9}$	$2.7X10^{-2}$
Na-22	Sodium (11)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Na-24		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Nb-93m	Niobium (41)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Nb-94		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Nb-95		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Nb-97		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Nd-147	Neodymium (60)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Nd-149		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ni-59	Nickel (28)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{8}$	$2.7X10^{-3}$
Ni-63		$1.0X10^{5}$	$2.7X10^{-6}$	$1.0X10^{8}$	$2.7X10^{-3}$
Ni-65		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Np-235	Neptunium (93)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Np-236 (short-lived)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Np-236 (long-lived)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Np-237		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Np-239		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Os-185	Osmium (76)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Os-191		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Os-191m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Os-193		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Os-194 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
P-32	Phosphorus (15)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{5}$	$2.7X10^{-6}$
P-33		$1.0X10^{5}$	$2.7X10^{-6}$	$1.0X10^{8}$	$2.7X10^{-3}$
Pa-230 (a)	Protactinium (91)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pa-231		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Pa-233		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pb-201	Lead (82)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pb-202		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pb-203		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pb-205		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pb-210 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Pb-212 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Pd-103 (a)	Palladium (46)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{8}$	$2.7X10^{-3}$
Pd-107		$1.0X10^{5}$	$2.7X10^{-6}$	$1.0X10^{8}$	$2.7X10^{-3}$
Pd-109		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pm-143	Promethium (61)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pm-144		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pm-145		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pm-147		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pm-148m (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pm-149		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pm-151		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Po-210	Polonium (84)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Pr-142	Praseodymium (59)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Pr-143		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{6}$	$2.7X10^{-5}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Pt-188 (a)	Platinum (78)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pt-191		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pt-193		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pt-193m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pt-195m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pt-197		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pt-197m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pu-236	Plutonium (94)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Pu-237		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pu-238		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Pu-239		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Pu-240		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Pu-241 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Pu-242		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Pu-244 (a)		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES

Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Ra-223 (a)	Radium (88)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ra-224 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ra-225 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ra-226 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Ra-228 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Rb-81	Rubidium (37)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Rb-83 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Rb-84		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Rb-86		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Rb-87		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Rb(nat)		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Re-184	Rhenium (75)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Re-184m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Re-186		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Re-187		$1.0X10^{6}$	$2.7X10^{-5}$	$1.0X10^{9}$	$2.7X10^{-2}$
Re-188		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Re-189 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Re(nat)		$1.0X10^{6}$	$2.7X10^{-5}$	$1.0X10^{9}$	$2.7X10^{-2}$
Rh-99	Rhodium (45)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Rh-101		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Rh-102		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Rh-102m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Rh-103m		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{8}$	$2.7X10^{-3}$
Rh-105		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Rn-222 (a)	Radon (86)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{8}$	$2.7X10^{-3}$
Ru-97	Ruthenium (44)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ru-103 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ru-105		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ru-106 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
S-35	Sulphur (16)	$1.0X10^{5}$	$2.7X10^{-6}$	$1.0X10^{8}$	$2.7X10^{-3}$
Sb-122	Antimony (51)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{4}$	$2.7X10^{-7}$
Sb-124		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sb-125		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sb-126		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Sc-44	Scandium (21)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Sc-46		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sc-47		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sc-48		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Se-75	Selenium (34)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Se-79		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Si-31	Silicon (14)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Si-32		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sm-145	Samarium (62)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Sm-147		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Sm-151		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{8}$	$2.7X10^{-3}$
Sm-153		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Sn-113 (a)	Tin (50)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Sn-117m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sn-119m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Sn-121m (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Sn-123		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sn-125		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Sn-126 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Sr-82 (a)	Strontium (38)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Sr-85		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sr-85m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Sr-87m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sr-89		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sr-90 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{4}$	$2.7X10^{-7}$
Sr-91 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Sr-92 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
T(H-3)	Tritium (1)	$1.0X10^{6}$	$2.7X10^{-5}$	$1.0X10^{9}$	$2.7X10^{-2}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Ta-178 (long-lived)	Tantalum (73)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ta-179		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ta-182		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Tb-157	Terbium (65)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Tb-158		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tb-160		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tc-95m (a)	Technetium (43)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tc-96		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tc-96m (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Tc-97		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{8}$	$2.7X10^{-3}$
Tc-97m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Tc-98		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tc-99		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Tc-99m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Te-121	Tellurium (52)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Te-121m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Te-123m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Te-125m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Te-127		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Te-127m (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Te-129		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Te-129m (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Te-131m (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Te-132 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Th-227	Thorium (90)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Th-228 (a)		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Th-229		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Th-230		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Th-231		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Th-232		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Th-234 (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{5}$	$2.7X10^{-6}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES

Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Th (nat)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Ti-44 (a)	Titanium (22)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Tl-200	Thallium (81)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tl-201		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tl-202		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tl-204		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{4}$	$2.7X10^{-7}$
Tm-167	Thulium (69)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tm-170		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tm-171		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{8}$	$2.7X10^{-3}$
U-230 (fast lung absorption) (a)(d)	Uranium (92)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
U-230 (medium lung absorption) (a)(e)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
U-230 (slow lung absorption) (a)(f)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
U-232 (fast lung absorption) (d)	Uranium (92)	1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
U-232 (medium lung absorption) (e)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
U-232 (slow lung absorption) (f)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
U-233 (fast lung absorption) (d)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-233 (medium lung absorption) (e)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-233 (slow lung absorption) (f)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-234 (fast lung absorption) (d)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-234 (medium lung absorption) (e)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-234 (slow lung absorption) (f)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-235 (all lung absorption types) (a),(d),(e),(f)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
U-236 (fast lung absorption) (d)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-236 (medium lung absorption) (e)	Uranium (92)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-236 (slow lung absorption) (f)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-238 (all lung absorption types) (d),(e),(f)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U (nat)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
U (enriched to 20% or less)(g)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
U (dep)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
V-48	Vanadium (23)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
V-49		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
W-178 (a)	Tungsten (74)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
W-181		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
W-185		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
W-187		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
W-188 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Xe-122 (a)	Xenon (54)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{9}$	$2.7X10^{-2}$
Xe-123		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{9}$	$2.7X10^{-2}$
Xe-127		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{5}$	$2.7X10^{-6}$
Xe-131m		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{4}$	$2.7X10^{-7}$
Xe-133		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{4}$	$2.7X10^{-7}$
Xe-135		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{10}$	$2.7X10^{-1}$
Y-87 (a)	Yttrium (39)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Y-88		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Y-90		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{5}$	$2.7X10^{-6}$
Y-91		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Y-91m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Y-92		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Y-93		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Yb-169	Ytterbium (79)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Yb-175		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Zn-65	Zinc (30)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Zn-69		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{6}$	$2.7X10^{-5}$

TABLE A - 2: EXEMPT MATERIAL ACTIVITY CONCENTRATIONS AND EXEMPT CONSIGNMENT ACTIVITY LIMITS FOR RADIONUCLIDES					
Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Zn-69m (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Zr-88	Zirconium (40)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Zr-93		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Zr-95 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Zr-97 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$

NOTES

(a) A1 and/or A2 values include contributions from daughter nuclides with half-lives less than 10 days

(b) Parent nuclides and their progeny included in secular equilibrium are listed in the following:

Sr-90	Y-90
Zr-93	Nb-93m
Zr-97	Nb-97
Ru-106	Rh-106
Cs-137	Ba-137m
Ce-134	La-134
Ce-144	Pr-144
Ba-140	La-140
Bi-212	Tl-208 (0.36), Po-212 (0.64)
Pb-210	Bi-210, Po-210
Pb-212	Bi-212, Tl-208 (0.36), Po-212 (0.64)
Rn-220	Po-216
Rn-222	Po-218, Pb-214, Bi-214, Po-214
Ra-223	Rn-219, Po-215, Pb-211, Bi-211, Tl-207
Ra-224	Rn-220, Po-216, Pb-212, Bi-212, Tl-208 (0.36), Po-212 (0.64)
Ra-226	Rn-222, Po-218, Pb-214, Bi-214, Po-214, Pb-210, Bi-210, Po-210
Ra-228	Ac-228
Th-226	Ra-222, Rn-218, Po-214
Th-228	Ra-224, Rn-220, Po-216, Pb212, Bi-212, Tl208 (0.36), Po-212 (0.64)
Th-229	Ra-225, Ac-225, Fr-221, At-217, Bi-213, Po-213, Pb-209
Th-nat	Ra-228, Ac-228, Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208 (0.36), Po-212 (0.64)
Th-234	Pa-234m
U-230	Th-226, Ra-222, Rn-218, Po-214
U-232	Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208 (0.36), Po-212 (0.64)

U-235	Th-231
U-238	Th-234, Pa-234m
U-nat	Th-234, Pa-234m, U-234, Th-230, Ra-226, Rn-222, Po-218, Pb-214, Bi-214, Po-214,
U-240	Np-240m
Np-237	Pa-233
Am-242m	Am-242
Am-243	Np-239

(c) The quantity may be determined from a measurement of the rate of decay or a measurement of the radiation level at a prescribed distance from the source.

(d) These values apply only to compounds of uranium that take the chemical form of UF6, UO2F2 and UO2(NO3)2 in both normal and accident conditions of transport.

(e) These values apply only to compounds of uranium that take the chemical form of UO3, UF4, UCl4 and hexavalent compounds in both normal and accident conditions of transport.

(f) These values apply to all compounds of uranium other than those specified in (d) and (e) above.

(g) These values apply to unirradiated uranium only.

TABLE A-3: GENERAL VALUES FOR A_1 AND A_2

Contents	A_1		A_2		Activity concentration for exempt material	Activity concentration for exempt material	Activity limits for exempt consignments	Activity limits for exempt consignments
	(TBq)	(Ci)	(TBq)	(Ci)	(Bq/g)	(Ci/g)	(Bq)	(Ci)
Only beta or gamma emitting radionuclides are known to be present	1×10^{-1}	2.7×10^{0}	2×10^{-2}	5.4×10^{-1}	1×10^{1}	2.7×10^{-10}	1×10^{4}	2.7×10^{-7}
Only alpha emitting radionuclides are known to be present	2×10^{-1}	5.4×10^{0}	9×10^{-5}	2.4×10^{-3}	1×10^{-1}	2.7×10^{-12}	1×10^{3}	2.7×10^{-8}
No relevant data are available	1×10^{-3}	2.7×10^{-2}	9×10^{-5}	2.4×10^{-3}	1×10^{-1}	2.7×10^{-12}	1×10^{3}	2.7×10^{-8}

TABLE A-4: ACTIVITY-MASS RELATIONSHIPS FOR URANIUM

Uranium Enrichment[1] wt % U-235 present	Specific Activity	
	TBq/g	Ci/g
0.45	1.8×10^{-8}	5.0×10^{-7}
0.72	2.6×10^{-8}	7.1×10^{-7}
1.0	2.8×10^{-8}	7.6×10^{-7}
1.5	3.7×10^{-8}	1.0×10^{-6}
5.0	1.0×10^{-7}	2.7×10^{-6}
10.0	1.8×10^{-7}	4.8×10^{-6}
20.0	3.7×10^{-7}	1.0×10^{-5}
35.0	7.4×10^{-7}	2.0×10^{-5}
50.0	9.3×10^{-7}	2.5×10^{-5}
90.0	2.2×10^{-6}	2.8×10^{-5}
93.0	2.6×10^{-6}	7.0×10^{-5}
95.0	3.4×10^{-6}	9.1×10^{-5}

[1] The figures for uranium include representative values for the activity of the uranium-234 that is concentrated during the enrichment process.

Dated at Rockville, Maryland, this _________day of _____________, 2001.

For the Nuclear Regulatory Commission.

____

Annette L. Vietti-Cook,
Secretary for the Commission

NUREG/CR-6712

Summary and Categorization of Public Comments on the Major Revision of 10 CFR Part 71

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ABSTRACT

This report presents, in digest form, all comments the Nuclear Regulatory Commission (NRC) received on its issues paper to modify 10 CFR Part 71 requirements pertaining to the packaging and transport of radioactive materials, including fissile materials. NRC first published the issues paper in the Federal Register (65 FR 44360) on July 17, 2000. The NRC proposed 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, and the U.S. Department of Transportation's 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. As part of its enhanced public participatory process, NRC invited written comments on the issues paper, established an interactive web site, and held public meetings during August and September 2000 in Oakland, CA; Atlanta, GA; and Rockville, MD. Extensive and wide-ranging comments were received from almost 100 members of the public and industry at these public meetings and during the 75-day public comment period. (All comments received after the comment period ended were included in both the decision-making process and this digest.) This report synthesizes those comments into a publicly accessible digest form without analyzing or otherwise responding to the comments. The issues paper is included in this report as an Appendix.

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FOREWORD

The NRC is conducting an enhanced public participatory process to evaluate its proposal to harmonize 10 CFR Part 71 with the International Atomic Energy Agency's most recent transportation standards, TS-R-1, as well as with U.S. Department of Transportation regulations, 49 CFR. NRC published an Issues Paper in the Federal Register (65 FR 44360) on July 17, 2000 to seek public input on these alternatives and invite written comments. NRC also held public meetings during August and September 2000 in Oakland, CA; Atlanta, GA; and Rockville, MD. The commentary on the alternatives and fundamental issues solicited from interested parties, who participated in these meetings and submitted comments directly, forms part of the official record that NRC's proposed rulemaking to harmonize 10 CFR Part 71 will address. This report summarizes, and presents in digest form, the comments that were categorized from transcripts of the three public meetings and NRC docketed letters from individuals and organizations. The full text of these comments, as well as additional supporting materials, can be accessed from the docket maintained by NRC and the dedicated web site that was developed both for disseminating information and for obtaining comments on the Issues Paper (http://www.nrc.gov/NMSS/IMNS/transport.html). Comments received with respect to this published report will also be included in the formal docket and be accessible therefrom.

This report includes letters and comments received from July 24, 2000 to December 20, 2000. While the public comment period ended September 30, 2000, letters and comments received after this time were incorporated into both the decision-making process and this digest. The results, approaches, and methods described in this report are provided for information only. Publication of this report does not necessarily constitute NRC approval or agreement with the information contained herein.

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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	Indpendent 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

1.0 INTRODUCTION

1.1 BACKGROUND

The U.S. Nuclear Regulatory Commission (NRC or the Commission) is conducting an enhanced public participatory process to evaluate its proposal to harmonize 10 CFR Part 71 with the International Atomic Energy Agency's (IAEA) most recent transportation standards, TS-R-1, as well as with U.S. Department of Transportation (DOT) regulations, 49 CFR. NRC sought early public input on the major issues associated with this effort in order to confirm the validity of its approach. Towards this end, NRC developed an Issues Paper that presents the key issues associated with conforming NRC regulations with IAEA and DOT regulations. This Issues Paper was published with the goal of developing a public discussion of the issues associated with harmonizing 10 CFR Part 71 with TS-R-1.

The Issues Paper was published in the Federal Register (65 FR 44360) on July 17, 2000. The Federal Register Notice invited public comment on the Issues Paper and, to provide further opportunity for public input, NRC held three facilitated public meetings during August and September 2000. These public meetings included 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.

In the Issues Paper, NRC discussed initiating a proposed rulemaking to: (1) conform its transportation regulations found in 10 CFR Part 71 ("Packaging and Transport of Radioactive Material") with the most recent transportation regulations established by the IAEA in TS-R-1; and (2) address the Commission's goals for risk-informed regulations and eliminating inconsistencies between Part 71 and other parts of 10 CFR.

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, NRC is responsible for controlling the transport of radioactive materials. NRC shares responsibility for radioactive material transport with the 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 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.

As specified by the Commission in SRM-SECY-00-0117 (June 28, 2000), NRC is now proceeding towards developing 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 will be considered during the decision-making process.

1.2 OVERVIEW OF COMMENTS

NRC received comments from almost 100 individuals, citizen and environmental groups, state government agencies, and members of industry on its Issues Paper. Fifty written comments were submitted to NRC's interactive web site, with another 46 comments received during discussions at public meetings. The Issues Paper is included in this report as Appendix A.

The public meetings were all well-attended events with local citizen groups being present as well as industry and environmental group representatives. Attendees included: Federal agencies (e.g., U.S. DOT; U.S. Department of Energy), state and local government agencies (e.g., Attorney General's Office, State of New Mexico; Clark County Department of Comprehensive Planning), educational institutions (e.g., Oregon State University), members of industry (e.g., AEA Technology; Calvert Cliffs Nuclear Power Plant, Inc.; Mallinckrodt Inc.), as well as private citizens and environmental groups (e.g., Action for a Clean Environment, Tri-Valley CARES).

NRC received extensive and wide-ranging comments during each of the three public meetings as well as via the interactive web site.

NRC received general comments on issues related to the proposed rulemaking. These comments included things such as concerns that regulatory materials were either unavailable or not written for a lay audience or that NRC and DOT should develop a coordinated process for managing the harmonization of NRC, DOT, and IAEA regulations.

The majority of comments received addressed the specific issues under consideration in the rulemaking, and requests for input made by NRC in its Issues Paper. For example, NRC was told that many commenters preferred to continue using dual units of measurement. Other commenters were more than willing to provide new or edited definitions or to ask NRC to clarify particular definitions.

This report presents comment summaries in an easily accessible format. The public meeting transcripts and the written public comments are all available on the NRC's interactive web site.

The public comment period extended from July 24, 2000 to September 30, 2000. But NRC decided that including all comments received after that date in both the decision-making process as well as this digest had merit. Therefore, this digest includes comments received from July 24, 2000 through December 20, 2000. A listing of the commenters, and the issues they addressed, is included in Appendix B.

The organization of this report is similar to the Issues Paper. Chapter 2 presents the general issues and questions while Chapter 3 corresponds to the first issue discussed in the Issues Paper, "Changing Part 71 to SI Units Only." Subsequent chapters focus on Issues 2 through 18.

Comment summaries are found in Chapters 2 through 21. Each comment summary includes a unique comment number assigned to each of the commenters who submitted comments to NRC, either during public meetings, in writing, or via the NRC web site. Although an individual or organization may have addressed an issue in several letters, or in a meeting and a comment letter, the summary includes reference to that commenter only once for any given issue.

When there are multiple submissions by one commenter or organization, the first submission's comment number is used as the comment number for this report. For example, Commenter Number 15 attended the

Rockville, MD meeting and then submitted comments to NRC twice (i.e., Commenter Number 69 and 72). In this instance, the report uses Commenter Number 15, and notes that Commenter Numbers 69 and 72 are the same person in Appendix C.

To try to orient the reader further, the comment number's first two digits identify what public meeting the comment is from. The Rockville, Maryland meeting is denoted with an "MD" while the Atlanta, Georgia and Oakland, California meetings are denoted with "AT" and "OA," respectively. Comments submitted to NRC via its interactive web site, or in the mail, are denoted with two zeros preceding the comment number (e.g., 0073).

Readers can identify the commenter numbers applicable to an individual or organization by referencing Appendix B. Alternatively, the reader may identify the individual or organization name applicable to a comment number by referencing Appendix C. Appendix B also identifies the issues addressed by each commenter in subsections of Chapters 2 through 21.

2.0 GENERAL ISSUES

Commenters provided general comments on NRC's proposed rulemaking. Some commenters were supportive of NRC's efforts while others were not so inclined.

Commenters also spoke to issues not directly included in the Issues Paper, such as the process NRC used to disseminate information to the public or how NRC and DOT would coordinate an international harmonization effort.

2.1 SUPPORT NRC'S EFFORTS

Several commenters supported NRC's efforts with the proposed rule and noted particular benefits that could result.

- Appreciate use of enhanced rulemaking process and encouraged us to continue using this process (OA43) (0094)

- Shifting to risk-informed regulation will increase the safety of nuclear power plants by allowing the operators to focus on risk-significant issues (MD18)

- Adopt regulations based on technical merit (0052)

- Continue with safety and performance-based regulatory focus (MD08) (MD17) (MD20)

- Adopt uniform regulations to ensure both the domestic and international safe use and transport of radioactive materials (MD08) (MD17) (MD20) (0051)

- Regulatory consistency promotes compliance with minimal confusion (0051)

- Support efforts to incorporate TS-R-1 into 10 CFR Part 71 because regulations affecting movement of radioactive materials around the world need to be applied and adopted uniformly as demands on transport of radioactive materials grow (AT27) (0079)

Other commenters wanted to ensure that any changes to NRC's regulations, whether in the context of conformity with international regulations, or solely affecting domestic shipments of radioactive materials, would not result in a reduction in transportation safety for the public.

- Safety considerations are important but also support NRC's shift towards performance-based regulation, similar to the way NRC revised 10 CFR Parts 50 and 70 (MD08)

- Support revising requirements in 10 CFR Part 71 if a publicly available technical justification demonstrates that safety margins are not reduced by the revisions (0050) (0073)

- Do not revise 10 CFR Part 71 solely to be compatible with IAEA TS-R-1. A technical basis document, similar to NUREG-1230 used in the revision of the Emergency Core Cooling System, needs to be cited in support of this proposed revision (0054)

- NRC and DOT should not support changes which increase radiation doses to the general public or increase adverse impacts on the environment (MD16) (0095)

- Aspects of the proposed rules would be beneficial but other portions would be overly burdensome without improving public health, potentially even doing harm (MD04)

- No cost-benefit analysis has justified why the change is necessary but the proposed rule can be successfully developed and still improve public health and safety (MD04)
- Public safety and the integrity of the regulatory process should not be compromised as a result of a cost/benefit analysis -- i.e., cost/benefit analysis should not be an overriding criterion in decision-

making because it is based on challengeable assumptions (0096)

- While considering TS-R-1 changes, NRC and DOT should evaluate performance standards from across the world so that international commerce activities are not disrupted (MD08)

2.2 PUBLIC ACCESS TO INFORMATION

Commenters were also concerned with the process NRC uses to make documents and information publicly available that are pertinent to transportation regulations.

- Asked to be placed on NRC distribution list for all correspondence issued related to this rulemaking effort (0050)

- Find alternative publication methods, such as posting documents on the web, materials informing the public of specific proposed changes -- e.g., TS-R-1, pertinent sections from the CFR -- and why they are proposed (AT22) (AT23) (AT25) (AT27) (AT33) (AT35) (AT36) (AT37) (OA43) (0050) (0073)

- Purchase an IAEA web document distribution license, which should not be too expensive (OA43)

- Web site difficult to access and to navigate and find pertinent information (AT23) (AT33) (AT35) (AT36) (AT37) (0050) (0059) (0073)

- Information not readily available prior to public meetings (AT27) (0063) (0095)

- Translate proposed changes and their impacts into language a layperson can understand. One suggestions was to use plain language footnotes (AT23) (AT33) (AT35) (AT36) (AT37) (0050) (0059) (0063) (0073) (0095)

- Entire process is frustrating (AT33) (0095)

- Forced to track down information that NRC should have provided (MD05)

- Identify where people can learn about package routing through their community (AT35) (AT36)

- White paper afforded participants a limited, possibly distorted, view of the proposed changes (MD16)

- Unavailable documents, abridged discussion papers, and limited public meetings must not form the basis for substantive changes in regulations (MD16)

- No substantive information should be suppressed, and no decisions should be made without full public consensus (MD16)

- Supporting documents should not be too expensive for the public to purchase or otherwise access them (AT22) (AT27) (0095)

- Documents expensive and delivery takes too long -- e.g., weeks passed before receiving a copy of TS-R-1 from the contractor listed in the Federal Register (MD06) (MD15)

- NRC and DOT must provide a publicly accessible version of the proposed regulations, and related documents, and make the regulatory process transparent, which is critical if NRC is to develop international standards (MD03) (MD06) (MD15) (MD16) (AT30) (0063)

- NRC and DOT do not have authority to encourage an international reduction in public protection which could preempt more protective, existing national standards (MD06) (MD15) (MD16)

Commenters addressed the public comment period and issues surrounding public meetings.

- Lengthen the comment period and/or otherwise allow for additional public meetings (MD05) (MD15) (AT27) (AT30) (AT40) (OA41) (OA43) (OA44) (0073)

- Provide additional notice of public meetings (AT27) (0063) (0095)

- Hold meetings in locations likely to be affected by any changes in NRC's transportation regulations -- e.g., communities near Yucca Mountain, communities near major transport hub cities (MD15) (AT30) (OA41)

- Coordinate NRC's public meetings for all rulemakings or actions related to transportation (e.g., the Modal Study) so the public can see the interrelationships of various NRC actions (MD15)

- Allow every transport community to have the opportunity to request a formal public hearing (MD16)

- Schedule representative group sessions with Agreement States, affected cities, citizens' groups, and industry representatives, to discuss TS-R-1 (OA44)

- Extend the public comment period by at least 30 days (0073)

- Extend the public comment period by at least 60 days because NRC's white paper is insufficient and does not adequately characterize the proposed changes (MD05)

- Extend the public comment period by at least 180 days (MD06) (MD15)

- Extend the public comment period because, using a mostly trucking scenario, all Yucca Mountain truck shipments will pass through downtown Las Vegas and approximately seven percent of the national shipment miles to Yucca Mountain will occur in Clark County of Nevada (OA43)

- Start the clock on the public comment period once plain language information, including NRC's proposed rule and the basis for its adoption, is publicly available (MD06) (MD15)

- Make the regulatory process as open and democratic as possible (AT22) (AT40)

- To date, inadequate and unrepresentative public participation -- e.g., public meetings scheduled too close to the end of the public comment period, IAEA standards were not established under a cost-benefit regulatory standard, as is Congressionally mandated) for the proposed rule -- which contradicts the Administrative Procedure Act (OA43) (0090)

- To date, inadequate mechanisms exist to encourage public involvement in discussions of modifications to internationally significant policies, and without this, the modifications may lack legitimacy (OA43)

2.3 GENERAL ISSUES

NRC also heard from commenters about general issues related to NRC and the proposed rule.

- NRC and DOT need a coordinated process to jointly study and, after a reconciliation process, address public comments (OA43)

- NRC should limit its focus to areas for which its responsible -- e.g., fissile material, Type B shipments -- and not develop parallel regulations (0078)

Commenters were interested in NRC's proposed standards and their strength of protection.

- NRC should only suggest changing existing standards if said changes improve or otherwise strengthen existing standards (AT22) (AT23) (AT27) (AT34) (AT39) (0090)

- If NRC's regulations are more stringent than IAEA regulations, then NRC regulations should be maintained (AT27) (AT30) (OA41)

- International standards should be considered a regulatory floor, not a ceiling (MD05) (AT22) (AT34) (OA41) (0096)

- NRC should not lower its standards but should work to strengthen international standards (OA41)

- Cost should only be considered if the changes will not decrease public safety (AT27)

- Any change that does not improve public health, safety, and the environment -- e.g., strengthening double-casking requirements -- is not likely to be worth its regulatory costs and should be carefully considered (AT22) (AT34) (0050) (0073)

- Clarify whether the proposed changes discussed in the issues paper would strengthen or weaken public health and safety in the U.S. (0090)

- IAEA should periodically examine its regulations against more stringent ones to ensure the IAEA regulations are as protective of public health as they can be. After such a review, and as necessary, IAEA should revise its regulations (AT30)

- Recent NRC rulemaking initiatives have improved neither public safety nor safety margins, and appear designed only to relieve regulatory burden (0073)

- Current process is being driven by the European nuclear industry, which does not have the safety interests of corridor communities as a first priority (AT30)

Several commenters asked about NRC's plans to regulate Naturally Occurring Radioactive Materials as well as to clarify jurisdiction concerns.

- Materials (including certain bulk materials) not previously regulated by NRC could fall under the Commission's jurisdiction, or become exempt depending on jurisdiction, which could lead to unnecessary public concern (MD01) (MD03) (MD04)

- Clarify that 10 CFR Part 71 focuses on regulating special nuclear source and by-product material, not naturally occurring materials. If NRC plans to regulate naturally occurring materials, then it must clarify its statutory authority to do so (MD01)

- Clarify whether, and how, the proposed rule would affect State Agencies regulating radioactive materials (MD02)

2.4 SCOPE CLARIFICATION

Commenters asked NRC to clarify the scope of the proposed changes to 10 CFR Part 71.

- Clarify whether all items listed in the issues paper are included in NRC's proposed rule (MD12)

- Clarify whether NRC and DOT intend to adopt all changes associated with TS-R-1 or just those contained in the issues paper (MD15).

- Publicize the full scope of the proposed regulations -- e.g., NRC's apparent intention to adopt new standards facilitating clearance or exemption of radioactive materials from regulatory control (which is contrary to public preferences), reducing "already inadequate requirements for Type B transport containers without fully informing or involving all communities along the transport routes" (MD06) (MD15)

Commenters asked for more information on the specific changes NRC proposes.

- Define all terms and provide background information in the next iteration, which will enable the public to understand and evaluate the context and rationale for NRC's proposed actions (AT22) (AT25)

- Provide the public with the full spectrum of ideas that the Commission is contemplating for incorporation (MD15) (MD16)

- NRC has not provided an adequate analysis of the impact of the proposed changes (MD05)

- Failed to identify the evidence on which NRC bases its suggestions to lower safety standards (0074)

- Provide route and transportation mode estimates of the acceptable risks inherent in the proposed changes, specifically, how many people can die legally under the proposed regulatory changes (MD16)

Commenters asked for additional details regarding the transportation process and the security arrangements associated with the proposed rulemaking's changes.

- Detail the links existing between this rulemaking process, the NRC, the DOT, and DOE's currently scheduled shipments of radioactive materials (AT22)

- Explain what security arrangements exist and what preparations NRC and DOT have made to deal with accidents and other such security breaches (AT24)

Commenters were concerned that NRC fully examine the impacts of the proposed changes on the DOE and industry.

- Need to provide a detailed analysis of the proposed changes on the DOE and whether relaxing NRC standards might result in relaxed standards at other federal agencies -- e.g., DOE, EPA (OA41)

- Detail the level and type of accountability industry has for its radioactive materials (AT25) (AT30)

2.5 HARMONIZATION WITH IAEA REGULATIONS

Commenters were concerned with the harmonization of NRC and IAEA regulations.

- Wondered whether the value of harmonization is sufficient when compared to the costs of implementation, especially when the magnitude of the safety benefits of such harmonization are considered (MD02) (MD05) (MD06) (MD10)

- Bottom line for the changes to 10 CFR Part 71 seem to be to enhance the bottom lines of the licensees (0050) (0073)

- NRC should explore what might happen if TS-R-1 is not adopted uniformly internationally and how that might affect international transport (MD02)

- Not adopting TS-R-1 standards risks stopping international commerce (MD06) (MD10)

- It is not incumbent for the U.S. to adopt international regulations simply because other countries are adopting them (MD15)

- Harmonization should not cause public harm -- e.g., restricting the ability to obtain some medical isotopes could cause greater public harm than allowing such shipments (MD01)

- Adopting parts of TS-R-1 will have minimum health and safety benefits but obvious costs (MD02) (MD19)

- The U.S. should have the right to adopt more stringent standards than those contained in TS-R-1, which should constitute a "minimum" set of requirements and not the highest applicable standard (MD05)

- Adopt a set of guiding principles to ensure that harmonization is done openly and in the best interest of public health and safety, such as the guiding principles used in the Transatlantic Consumer Dialogue (MD05) (0096)

- Ensure DOT and NRC regulations are consistent for all public shipments (0049)

- Revise 10 CFR 71 so that "IAEA requirements are the DOT regulations." (0049)

- Continue to first perform a safety check and ensure that safety levels are not diminished (MD06)

- Evaluate whether NRC faces regulatory incompatibility or simply interpretation issues (MD06)

- There is currently no urgency to harmonize because the world community is already harmonized using IAEA's Safety Series 6 (MD06) (MD15)

- NRC needs to address and/or modify parts of TS-R-1 before adopting it (MD20)

- NRC needs to compare TS-R-1 with recent science and engineering and not blindly adopt TS-R-1. Otherwise, revisions to 10 CFR Part 71 could be outdated before being finalized (0070)

2.6 OTHER ISSUES

A number of commenters were concerned with issues that were indirectly related to the proposed rulemaking.

Some commenters were interested in DOT's transition rule.

- Provide information on the timing of DOT's transition rule and whether United Nations (UN) numbering would be allowed under NRC's proposed changes (MD19)

- Clarify the meaning of DOT's transitional rulemaking for implementing the TS-R-1 standards domestically (MD15)

- U.S. agencies should not be encouraged to adopt regulations limiting the current review processes (MD16)

- DOT is acting arbitrarily and capriciously to move forward with preliminary changes in transport regulations and standards without due process (MD16)

Commenters addressed issues related to public exposure.

- NRC's proposed changes should not be allowed because public exposure rates are seemingly increased, and this is done without adequately informing the public of any risks associated with such an increase (AT22) (AT23) (AT27) (0075)

- Lowering containment standards, relaxing the testing requirements, and allowing air transport of plutonium (as well as overlooking DOE's plan to reverse the plutonium recycling ban) all conflict with regulations used in the 1970's, which were designed to limit the cumulative exposure to man-made radiation (0074)

Commenters also responded to issues that NRC had not addressed in its Issue Paper.

- Clarify and publicize the role, authority, and current U.S. interactions with the ICAO, IMO, and IAEA (MD15)

- Account for the long distances traveled in the U.S. -- i.e., estimated 2,400 mile trip to Nevada from eastern power plant locations -- especially when compared to the shorter distances traveled within and between European countries (0090)

- Assume the lowest level of training for emergency response. The rules should protect emergency responders and other personnel who could be expected to be around these types of shipments (0090)

- Because the complete chemistry of plutonium is not fully understood, NRC should neither minimize the criticality issue nor reduce regulatory stringency and should only allow changes in packaging if the packaging and transportation is made less dangerous and more protective of public health and safety (0096)

- NRC should limit the transport nuclear materials, discard ideas of using Mox fuel, consider deep sea storage of nuclear materials, and consider non-nuclear, non-polluting sources of energy, such as the sun, wind, water, and geothermal power (AT27)

3.0 CHANGING PART 71 TO SI UNITS ONLY

Commenters addressed the change from dual units to SI units only, with several stating their preference that NRC continue to allow the use both English and SI units.

- “Too soon” to switch to only SI units because some instruments are only calibrated to the “old” system (0059)

- English or curie units are required in FDA regulations and in new drug applications. FDA reluctant to move to SI units because the nuclear medicine community accustomed to curie unit (MD19) (AT28)

- TS-R-1 does not prohibit domestic use of dual-unit system by member countries (OA42)

- If switched to SI units only, licensee procedures and computer software would need to be changed throughout the industry, which would bring substantial cost and no safety benefit (0083)

- Keep using both units to eliminate confusion and increased human error that might come from unfamiliarity with a type of unit (OA46)

- Shipment paperwork and documentation are reported in both units (0081)

- The Agency should add another unit, such as calories because it might increase the public’s understanding of radiation (AT28)

- Should allow parenthetical equivalences in a familiar unit for each type of quantity mentioned; this will encourage thinking in SI units, while allowing for a gentle transition (0056)

- NRC and DOT do not need to lead on this issue; using only SI units would create a problem with industry and those who certify packages (MD12)
- NRC may be forced to use metric units because the U.S. government has a policy to adopt metric units but until such time, both sets of units should be used to avoid potential problems with industry and with those who certify packages (MD08) (MD12)

- Would be easier to deal in traditional as well as SI units (MD02)

- NRC should clarify the text so the values for fissile materials reflect the values listed in 10 CFR Part 71, and should add notes for uranium and plutonium (0049)

3.1 CONSIDERATION OF RISK

Commenters addressed consideration of risk in changing from dual to SI units only, and would like to use both units.

- Limited risk associated with switching to only SI units for international shipments, but for domestic shipments, dual units should be maintained (MD08) (MD20) (0051)

- In event of an accident, SI units might cause a response delay due to confusion with units (MD08) (MD20) (0051)

- Regulators would be more comfortable with both units because most think in terms of traditional units (MD02)

- Shippers think in English units, which could lead to errors in conversions if only SI units are used (0051)

- Would lead to increase in paperwork errors and an increase in situations that put the public at risk (0081)

- New drug applications and FDA regulations require use of English or Curie units, and FDA reluctant to remove Curie designation (MD19)

- Most packages currently marked with both 10 CFR Part 71 and SI units, which causes problems because the product and paperwork do not agree (MD19)

- If only SI units used in paperwork, when shipping papers are compared against what is labeled on the inside, there would be no correlation (MD19)

- Public transport would not be affected by the unit change (0049)

- Dual headings would be useful, though the change will not increase risk as long as intra-license shipments are allowed to maintain dual units (0049)

- A minimum ten-year transition period is necessary if NRC decides to change to SI units only (OA42)

- Little risk in changing to only SI units because these units are already used in shipping (0078)

- Unit confusion potentially caused the loss of the Mars Climate Orbiter spacecraft (0096)

Commenters said dual units are necessary to minimize the risk of inadvertently exposing workers to radiation.

- Packages have been received labeled only in SI units, and were incorrectly labeled as Type A rather than Type B material quantities (OA42)

- Inadequate carrier training has forced one commenter to essentially train common carriers, such as Federal Express, and trucking firms, in SI units (0048)

- Increased complexity in dealing with SI units greatly increases the possibility of conversion errors and unnecessary radiation exposure to workers (OA42)

3.2 ASSOCIATED COSTS

Commenters addressed the issue of costs associated with changing to SI units only.

- Possibly significant financial implications associated with changing documents for CoCs and licensing packages to SI units (OA42)

- Implementing the SI units only provision could impact all other Parts referenced in 10 CFR Part 71, and might require rewriting licenses and parts of the regulations (OA42)

- Changing to SI units only would result in high costs and numerous errors, with no benefit (MD12) (0066)

3.3 PROBLEMS WITH NON-ADOPTION

Several commenters addressed problems with non-adoption, stating their preference to use both units.

- New drug applications and FDA regulations require use of both units. FDA reluctant to remove curie designation and move to SI units (MD19)

- Most packages currently are marked with English and SI units, which causes problems because the product and paperwork do not agree (MD19)

- In a system with only SI units, there would be no correlation between shipping papers and interior labels (MD19)

3.4 ISSUE-SPECIFIC FACTORS FOR CONSIDERATION (INCLUDING BENEFITS)

Commenters addressed specific factors for consideration in changing to SI units only.

- Because Part 71 references other Parts, changing Part 71 to SI units only would require that every NRC region, licensing agency, and license adopt SI units (0042) (0051)
- Other agencies, including EPA and FDA, use English units in their regulations (0051)
- Until the U.S. adopts SI units, NRC should continue to allow use of dual units (MD08) (MD12) (MD20) (0051)
- In Nevada, majority of first responders are volunteers who will need SI unit training (OA43)
- Most people prefer using traditional units (OA42)
- Would seriously affect inventory records development and maintenance (0049)
- States and NRC should set authorization limits in Bq, and Part 71 and DOT regulations should reflect the unit changes. The Agency should revise RAMREG-01-98 immediately (0049)
- Using only SI units for shipping could cause confusion and safety issues because curie and mR units are currently used throughout the U.S. (MD17)
- Use dual units because in the event of an accident, both units are immediately available to emergency responders to assist in determining radiation risks and potential exposure (0086)
- Many HAZMAT employees do not use SI units on a daily basis, and dual units would improve their knowledge of the equivalency of the two different systems (0086)
- Be consistent throughout Title 10 and in regulations used by other government agencies (MD08) (MD20) (0051)

Several commenters suggested, or otherwise addressed, issues surrounding implementation of a transition period.

- Conversion to SI units could be accomplished within one year (MD08) (MD17) (MD20) (0051)
- Transition for radiological workers is uncertain (0051)
- Time is needed to train employees in carrier and distribution network (MD08)
- Recommends a transition time, where dual units are used, due to highly variable training budgets (OA43)
- Minimize the transition period and include half-life values in A1 and A2 table to avoid confusion and ensure compatibility with IATA rules (0049)
- Allow for three-year transition period (0078)

4.0 RADIONUCLIDE EXEMPTION VALUES

Commenters were concerned with the implications of changing the radionuclide exemption values to harmonize them with TS-R-1.

- Current standard is “reasonably simple” and new standards will disrupt the system and make compliance and enforcement more complex (0059)

- NRC should provide a breakdown of every isotope and whether harmonization would increase or decrease the threshold (MD05)

- Tension between rulemaking responsibilities of NRC and DOT and the science used by agencies in modifying exemption values (OA44)

- Incorporation of activity concentration and activity limits for exempt material and exempt consignment is positive and helpful (0048)

- Eleven of the listed values have DOT exemption values higher than NRC exemption values, but the magnitude of change is not consistent. Might create inconsistencies in transfer of material to other licensed or non-licensed facilities (0048)

- Clarify intent of activity limit for an exempt consignment (0048)

- Issue paper provides little objective basis for exemption values (OA44)

- NRC needs to scrutinize standards to determine whether values are justified to protect human health and the environment (OA44)

- Should incorporate TS-R-1 values into Part 71 for international shipments, but reference DOT exemption values for domestic shipments in 10 CFR Part 71, unless it can be shown that these values compromised public health and safety (OA42)

- To avoid burdensome and unnecessary costs, must set up protocol for adapting DOT values for non-transportation activities (OA42)

- Concerned that DOT would not review or question IAEA standards, and that the U.S., Agreement States, and environmental organizations have not had meaningful input into IAEA forums (OA44)

- Concerned that NRC could not analyze the effect of changes on radionuclide concentrations, and could not inform the public about which radionuclides would be affected (MD05)

One commenter expressed concerns related to the issue of Naturally Occurring Radioactive Materials

- Problem in determining what is exempt, is that when examining the specific activity of a natural material, there is a natural decay chain in a secular equilibrium with all its decay progeny (MD03)

- Clarify convention for evaluating the 70 becquerels per gram exclusion limit under 49 CFR and 10 CFR Part 71 (MD03)

- Review the report from the IAEA special working group on exemptions to understand what IAEA and the drafters of TS-R-1 intended (MD03)

- Document is not an enforceable rule anywhere until it is implemented through a legislative process or administrative procedure in a country or through a national agency (MD03)

- Public needs to know where numbers came from to understand why a standard is being adopted (MD03)

- Not understanding evaluating exemption language could result in uncertain and disharmonious situations worldwide when looking at implementation in other national organizations (MD03)

- If NRC and DOT harmonize the way exemption levels are applied, it should be done consistently with the intention of NRC drafters, which can be discerned by examining the supporting documentation (MD03)

Several commenters were opposed to an increase in exemption levels.

- Appreciates NRC's efforts to eliminate the "one-size-fits-all" approach, but questioned whether the Agency's approach is the best method (AT30)

- EPA's Safe Drinking Water Standards do not define a safe dose of ionizing radiation (AT22) (AT27)

- The Agency does not appear to be operating under the assumption that there is no consensus in medicine regarding a safe threshold for radionuclide contamination (AT23)

- Growing minority concern that lower levels of radiation impact the human body more per unit than higher doses do (AT23)

- Regulatory levels established in one arena are often generalized and improperly adopted in another arena (AT22)

- Concerned with increased, but not personally-approved, personal exposure rates due to NRC's proposed changes (AT27)

- Concerned that NRC is proposing a severe relaxation of exemption values for dangerous materials (OA41)

- Make exemption values as stringent as possible to protect the public (0096)

- Keep current exemption because unaware of a public safety issue associated with the current concentration (0083)

- Implementing radionuclide specific concentrations will require procedure and computer software changes with no apparent safety benefit (0083)

Several commenters had questions related to harmonizing the radionuclide exemptions.

- Has the standard been one millirem per pear per examination? (AT27)

- Explain in laymen's terms how the changes would impact daily life and link them to real-life context (AT37)

- How did NRC establish appropriate does levels and how does NRC decide a particular does may be problematic? (AT38)

4.1 CONSIDERATION OF RISK

Commenters addressed consideration of risk and unintended consequences of adoption of radionuclide exemption values.

- Questioned risks to public, workers, and emergency responders (0090)

- Would increase total number of shipments by requiring smaller quantities per shipment to meet the higher exemption values (MD04)

- Use/demand for oil and gas would increase (MD04)

- Significant impacts to certain industries (MD04)

- "Knock-on-effect" from NRC to DOT because States would not want to independently examine the technical aspects of the proposed rule (MD04)

- NRC should not promulgate regulations that result in decreased protection, and should not increase the amount of radioactivity allowed in packages (MD15)

- Hazards and risks must be equivalently recognized in all countries shipping radioactive material. Packaging standards should be consistent and afford required level of protection (MD17)

- Current DOT regulations protect transportation workers and the public under ordinary transport and incident/spill scenarios, and the proposed regulation does not present data to show it would significantly increase safety (0086)

- Move toward more stringent exemption values (OA41)

- Exemption values should not be increased because it might jeopardize public safety (0050) (0073)

- Clarify whole-health effects associated with the materials, and not just the cancer risk (OA46)

- Would greatly reduce the threshold definition of radioactive material, which would increase the number of radioactive shipments, and eventually lead to more accidents. Response personnel would be diverted from other tasks to respond to accidents involving shipments labeled as radioactive, that were previously considered non-hazardous (MD04)

- Concerned that personal exposure rates would increase (AT27) (0070)

- Would raise the threshold by approximately 25 percent (AT27)

- Will allow radiological materials with much higher concentrations than current exemptions to be shipped without regard to specific transportation regulations (0070)

- Opposes raising exemption values because, as acknowledged in EPA's Safe Drinking Water Standards, there is no safe dose of ionizing radiation (AT27)

- Exposure to several small doses of radiation from different sources has a cumulative, health-threatening effect (AT27)

4.2 ASSOCIATED COSTS

Commenters addressed the costs changing from current exemption values.

- Additional costs would be incurred for ensuring that activity concentrations are acceptable (MD12)

- Even with addition of exempt activity consignment approach, there would be increased characterizations costs, paperwork, and packaging processing time (MD12)

- Costs will be significant, even though low shipping volumes makes a detailed cost/benefit analysis difficult (MD12)

- Changing the definition of DOT Class 7 radioactive material could result in an additional $6 million of disposal costs (0086)

- Radionuclides, including Ra-228, Th-228, and Am-241 might become regulated resulting in regulation of some products (MD12)

- Would be possible for shippers of certain products to seek exemptions, but the process would likely be lengthy, burdensome, and may impact operations of the affected industries (MD12)

- Significant increase in cost to classify very low level radioactive material for transportation purposes because shipping personnel would need training and be required to develop methods for making exemption determinations (MD12)

4.3 PROBLEMS WITH NON-ADOPTION

A commenter addressed the problems with non-adoption of radionuclide exemption values.

- NRC should anticipate problems with overseas shipping due to differences in exemption values; a package under the limit in the U.S. might not be exempt under the A1, A2 values (MD08)

4.4 ISSUE-SPECIFIC FACTORS FOR CONSIDERATION (INCLUDING BENEFITS)

Commenters provided information on issues related to radionuclide exemption values.

- Adoption of specific exemption values could result in radioactive metals being sold to scrap dealers and then being recycled into consumer goods (OA41)

- Each radionuclide exemption value should be carefully examined because the values were not developed as a result of an enhanced public participation process (OA44)

- The Agency should allow for public comment on assumptions, data, and scientific analyses, and not simply accept the standards (OA44)

- Include possibility of ingestion and disbursement of radionuclides and their effects on the general public in establishing exemption values (OA42)

- Effects of radionuclide exposure include neurological degeneration, not just cancer (OA46)

- Radionuclide exemption values should apply to domestic shipments to avoid the confusion shippers would face if there were different requirements for exports and domestic shipments (0049)

- NRC and DOT should require all radioactive material be shipped to the address stated on the license or by the recipient, and should require that failure to do so be reported to the NRC (0049)

- Confusion raised by requirements for shipping, licensing, and disposal could be resolved by parenthetical explanations written on the regulations (0049)

- Exemption values should be uniform across the world to eliminate mistakes and delays in shipments (MD08) (MD17) (MD20)

- To prevent conflicts between DOT and NRC regulations, NRC should reference DOT regulations and not adopt unique exemptions for transportation or adopt a separate table (MD17) (0078)

- Should streamline 10 CFR Part 71 by eliminating duplicate requirements (MD08)

- Exemptions outside the transportation regulations should only be considered for the transportation aspect with just cause (MD08)

- DOT regulations and waste burial manifests already require knowledge of particular nuclides; little extra effort required to apply these methods to exemptions (0078)

- Exemptions should apply to all shipments to enhance compliance and make application easier (0078)

- Make a domestic exemption for low level materials, continuing to exclude materials with activity concentrations below 70 Bq/g provided they are only transported domestically (MD12)

- When the term “bulk” is equated with being unpackaged, it is inconsistent with 49 CFR definition for “bulk packaging” that refers to specific volume and mass ranges (0083)

- Proposal may eliminate certain disposal facilities from consideration without sufficient scientific or technical justification (0086)

Some commenters discussed the need for updates to reflect the new A1/A2 values.

- To identify, measure, and apply the mixture rule for radionuclides when determining the basic values for exempt material, the calculations and computer codes will need to be updated to reflect the new A1/A2 values; this will increase time to prepare a shipment (MD17)

- One year should be allotted for making appropriate updates (MD08) (MD20)

5.0 REVISION OF A_1 AND A_2 VALUES

Commenters addressed revisions to A1 and A2 values.

- Harmonization would not increase safety, but it would be expensive (MD05)

- Proposal is unfair because burden would fall largely on radiopharmaceutical manufacturers, while benefits primarily accrue to transporters (MD05)

- Revisions would increase allowable activity levels for many nuclides, violating the principle of increased safety by conforming with TS-R-1 (MD05)

- NRC should provide a breakdown of which radionuclides would have increased, decreased and unchanged levels (MD05)

- Should not revise values because would be introducing another inconsistency into NRC regulations if ICRP 61 were adopted (0083)

- Unclear why NRC would consider making regulations consistent with IAEA standards, but not with ICRP standards (0070)

- Risk eroding public confidence if accept and then ignore advice of international experts; need strong justification to discount ICRP recommendations (0070)

- Partial adoption of ICRP 61 by U.S. should not be a factor in transportation regulations because universal adoption of ICRP 61 is reflected in TS-R-1 A1/A2 values (MD08) (MD20)

- Models used to estimate the allowable levels have large uncertainties (MD06)

- Increasing A1 and A2 levels may not increase total risk, because of the underlying models' uncertainty (MD06)

- Opposes changes to dose projection because they would result in "dilution as the solution to pollution;" opposes changes that increase amount of radioactivity present in land, air, or water due to increasing the acceptable activity levels for existing dose levels (AT22)

- Existing values for exempt quantities are reasonable from a shipping standpoint, though there are problems with the implications beyond transportation (OA41)

- Revisions to A1 and A2 values would be a shift from an activity to a dose-based limit system, which is the same as the revisions to 10 CFR Part 20 (MD08)

- Opposed revisions in Part 20 and would oppose them in Part 71 for transportation because dose-based limits are more difficult to verify and enforce than activity levels (MD15)

- Any proposed rule should provide a detailed discussion of why A1 and A2 values are being changed for each affected nuclide (0050) (0073)

- Concerned that conforming with TS-R-1 would hinder use of molybdenum-99 generators (MD19)

- Encourages NRC and DOT to continue grandfathering effort (MD19)

- Opposed to proposal because it would reduce A1 value for Californium-252 (0058)

- Concerned with loosening definition of radioactive material (MD04)

- A1 and A2 values for some nuclides have gone up, suggesting overdue relaxing of a too-tight classification (MD04)
- Using assumptions that are too conservative, see thresholds for radioactive material lowered too far, for some materials by a factor of almost 10 (MD04)

5.1 CONSIDERATION OF RISK

Commenters addressed the risks associated with revising A1 and A2 values specified in 10 CFR Part 71.

- Opposes any revisions because they would substantially increase volume and amount of radiation, which would lead to increased risk (MD05)

- Because A values are based on models with large uncertainties, fluctuations in those values are likely subsumed within the models' uncertainties; thus overall risk would not necessarily increase (MD12)

- Opposes increase in allowable levels because it implies assumption of a "standard human being," but exposure to radionuclides might not affect each person identically (AT27)

- Little need to reduce A1 value for Cf-252 because there is little risk associated with use of "properly designed, constructed and maintained Type A packages" (0058)

- Issue needs additional thought because there may be risks besides cancer from exposure (0090)

- Questions if the change would increase or decrease public and worker protection, and what effect it would have on emergency responders (0090)

5.2 ASSOCIATED COSTS

Commenters discussed associated costs.

- Conforming with TS-R-1 will not likely increase or decrease safety, but will impose non-trivial costs on industry; therefore how the effort can be justified if a cost/benefit analysis is conducted? (MD05)

- Changing A1 value for Californium-252 could cost between $500,00 and $1.5 million; consumers' source costs therefore would increase (0058)

5.3 PROBLEMS WITH NON-ADOPTION

Commenters addressed issues related to non-adoption.

- Current grandfather clause specifies a 20 curie level for domestic uses only, and therefore no 20-curie generators can be shipped to Canada; important harmonization issue because 90 percent of the medical diagnostic and therapeutic studies completed are based on technetium generators (MD19)

- A1/A2 values in TS-R-1 are well documented and practical for transportation; appears to be no practical alternative to adoption of these values in Part 71 (OA42)

5.4 ISSUE-SPECIFIC FACTORS FOR CONSIDERATION (INCLUDING BENEFITS)

Commenters addressed issue-specific factors regarding the revision of A1 and A2 values.

- Continue to grandfather A2 values for molybdenum-99 to 20 curies. There is an industry trend to use larger generators in pharmacies (MD19)

- Explain how A1 values for Cf-252 were estimated in TS-R-1, and the note that further study be undertaken has not been adequately explained (0058)

- Although specific values of the A1/A2 table should not differ from those in TS-R-1, footnote "c" in 49 CFR 173.435 for molybdenum-99 should be retained; molybdenum-99 generators have been shipped safely for many years without risk or exposure to the public (MD08) (MD20)

- Would be useful for NRC to adopt the revised values because airlines and other carriers will likely use these values;

differing regulations for different shipments would cause confusion (0049)

Commenters endorsed the adoption of new A1 and A2 values.

- Change to use A1 and A2 values is an improvement over previous methods and provided a safety basis for the assigned values (0078)

- Exceptions for domestic use should not be granted (0078)

- Since ICRP 61 values are already reflected in A1 and A2 values, partial adoption of ICRP 61 values should not be a factor (MD17) (0078)

- Supports the adoption, with exceptions (MD12)

- Willing to assist the Agency in developing the appropriate Q-system parameters and performing the necessary calculations to determine numerical values for these radionuclides (MD12)

Commenters said A1 and A2 values should continue to be used for transportation because it is not practical to change systems unless the system is uniformly recognized around the world.

- No uniformly recognized system exists today (MD08) (MD20)

- Specific values of A1 and A2 table should not be different from those in TS-R-1, but should adopt DOE's proposed rule change to TS-R-1 to keep the A1 value for Cf-252 at 5 mg (MD17)

6.0 URANIUM HEXAFLUORIDE PACKAGE REQUIREMENTS

Commenters addressed issues related to uranium hexafluoride package requirements.

- Supports concept that certified packages meet or comply with performance requirements (MD20)

- Concerned with an exception allowing UF-6 packages to be evaluated for criticality without considering the in-leakage of water into the containment system. NRC should consider whether this is a change from current regulation, and whether it should be adopted (OA41)

- Need to conduct a study examining scenarios leading to an undesirable event, the likelihood of such an event, and the consequences, and then measure the event against a transportation safety goal (0052)

- There already have been instances of manufacturing defects with uranium hexafluoride packages; fatal accident in Tokaimura, Japan shows that worker mistakes can lead to inadvertent criticality or water inside an uranium hexafluoride package (0050) (0073)

- Sees little value in the proposed changes (MD12)

- Changes are result of two separate international initiatives and need not be integral part of regulations intended to minimize radiological hazards (MD12)

- Does not support TS-R-1 prohibition of pressure relief devices radiological hazards (MD12)

- Industry agrees with assessment that NRC-certified packages comply with the package performance requirements; industry working with DOT to address non-fissile UF6 packages (MD08)

- UF6 packages approved by DOT in 10 CFR Part 173.417 include fissile and non-fissile packages (0078)

- Instead of TS-R-1 guidance, NRC should do the following: clearly define the types of special design features that would be acceptable to ensure no single packaging error would permit leakage, issue the technical basis for accepting these features, and revise the existing rule to make the features part of the rule rather than an exception (0054)

- Opposes exceptions; packages should be required to meet all tests, including internal pressure, drop, and thermal (AT27)

6.1 CONSIDERATION OF RISK

Commenters addressed the risks associated with uranium hexafluoride packaging.

- Concerned with safety margins for uranium hexafluoride packaging (0050)

- Packages should be examined for criticality with the consideration of in-leakage of water (0050) (0073)

- The Agency should develop a risk assessment methodology for UF6 packages (0054)

- Without quantifying risk and estimating uncertainty and then comparing these results to a transportation safety goal, NRC cannot be assured of protecting public health and safety and the environment (0052)

6.2 ASSOCIATED COSTS

No comments were received.

6.3 PROBLEMS WITH NON-ADOPTION

Commenters addressed problems with non-adoption of uranium hexafluoride package requirements.

- Recognize ANSI N14.1 for UF6 packages and ISO7195 as equivalent standards (MD10) (MD20)

6.4 ISSUE-SPECIFIC FACTORS FOR CONSIDERATION (INCLUDING BENEFITS)

Commenters addressed issue-specific factors related to uranium hexafluoride package requirements.

- Proposed change would likely impact DOE and its sites (OA41)
- Proposed change is not expected to significantly impact the commenter's operations (MD12)
- Recognize ANSI 14.1 and ISO 7195 as equivalent standards for performance, safety, and compatibility with Protective Shipping Packages; this would allow manufacturer to dual rate/certify the UF6 cylinder and avoid confusion (0061)
- ANSI 14.1 and ISO 7195 are consistent in principle (MD08)

7.0 INTRODUCTION OF CRITICALITY SAFETY INDEX (CSI) REQUIREMENTS

Commenters addressed the introduction of criticality safety requirements.

- A labeling system for the index is a good idea (0090)
- Introducing a Criticality Safety Index (CSI) is an effective solution to the confusing double meaning for the current Transport Index (TI) (OA42) (0078)
- Use of the CSI should enhance shipment safety with a minimum burden on shippers (OA42)
- The CSI must be consistent with the TI; in general, NRC regulations must either be consistent with or match the DOT regulations (0083)
- The change provides clear separation of the reasons to limit the number of packages in a shipment (MD12)
- TI will give only an indication of the direct radiation hazard, and the CSI provides control of the criticality potential (MD12)
- With appropriate training, workers and managers in transport should be able to use the new system to control exposure risks more closely (MD12)
- Should not decrease separation distance requirements which are necessary to reduce the possibility of criticality occurring (AT27)
- Does not support adding CSI requirements because the TI already incorporates the more restrictive of the two values: dose and criticality. Adding the CSI requirements will not result in any added safety (MD20)
- Additional costs and efforts necessary to add the CSI to package labels and shipping paperwork outweigh any benefits (MD12)
- Amend 10 CFR Part 71 to include the CSI in order to control criticality (AT30) (AT31)
- The current practice, using the TI as the means to control criticality safety, does not provide responders with information on the undamaged condition of the package (AT30) (AT31)
- Use the TI to indicate the radiation level from the undamaged package (AT31)
- Do not allow transportation of plutonium by air, due to safety and terrorism concerns (AT25 *11 audience members agreed also) (AT27)
- Concerned with the lack of technical justification for the claim that adoption of the criticality safety requirements would result in "equivalent safety" (AT30)
- Safety far outweighs efficiency when considering relaxing regulations (AT30)
- If there are documents to show that increased efficiency will not jeopardize safety, the public needs to see them in order to comment effectively (AT30)

7.1 CONSIDERATION OF RISK

Commenters addressed risk considerations with CSI requirements.

- Should include the underlying technical justification for the term "equivalent safety" (AT30) (OA41)

- Concerned that the change would allow for more packages in a single shipment (OA41)

- How can NRC ensure the safety of criticality requirements? (AT30)

- Adding CSI requirements would create more opportunities for human error (MD20)

7.2 ASSOCIATED COSTS

Commenters addressed costs associated with the introduction of the CSI requirements.

- Benefits of adding the CSI requirements outweighed by the costs of additional labor, material, training, and administration (MD20)

- Introduction of the CSI requirements will impact training costs (MD12)

7.3 PROBLEMS WITH NON-ADOPTION

No comments were received.

7.4 ISSUE-SPECIFIC FACTORS FOR CONSIDERATION (INCLUDING BENEFITS)

Commenters addressed factors for consideration in introducing the CSI requirements.

- Industry supports the use of the new "CSI" label in conjunction with the TI label because separate labels are more meaningful, provide additional safety in transport, and may make some shipments more efficient by allowing an increase in the number of packages per conveyance or cargo hold (MD08)

- The only conceivable issue associated with using two different TI values for one shipment is if the two values are confused; should not happen, assuming people and organizations refer to them properly (MD08)

- Supports the adoption of the CSI because enforcement and compliance are greatly simplified by leaving TI as a value that can be determined largely by direct reading instruments (0059)

- Addition of CSI makes positive identification of fissile shipments much easier (0059)

8.0 TYPE C PACKAGES AND LOW DISPERSIBLE MATERIAL

Commenters provided information on Type C Packages and Low Dispersible Material requirements. Some commenters supported requirements for Type C Packages and Low Dispersible Material.

- Most air carriers follow ICAO regulations and will not accept goods unless shipped in accordance with TS-R-1 (MD10)
- Changes will not have a significant impact on operations (MD12)
- NRC should remove the plutonium-specific air requirements and replace them with the proposed requirements (MD08) (MD17) (MD20)

Some commenters did not support the proposed revisions.

- Supports current standard for plutonium air transport (OA41)
- Increase minimum standard to 129 meters per second to allow for the possibility of two airplanes colliding with one another (MD09) (AT27)
- Conduct testing sequentially to show cumulative effects on package (MD09)
- Postpone adoption of TS-R-1 requirements until questionable contents of TS-R-1 are resolved by the IAEA and the ICAO Dangerous Good Panel, and until ST-2 is finalized and released (MD09)
- Subject changes to packaging requirements to *de novo* technical review, and justify independently as protective of safety (OA44)
- Incorporate LDM concept into U.S. regulations (MD12)
- Reevaluate existing regulations for plutonium and clarify the relationship between Type C package requirements and any domestic requirements which are different (MD12)
- Increases in the number of shipments by a factor of between three and 10 (OA42)

Other commenters posed questions about the proposed requirements.

- How will NRC choose between 0360, IAEA standards, standards proposed by trade associations, or some other option (MD06)
- At what point to DOT and NRC consider the option of not permitting some types of transport? (MD15)
- What scenario did NRC base the value of “90 meters/second impact test) on? (AT30)
- Do Reavis 3300 containers meet Type C certification? (OA42)

8.1 CONSIDERATION OF RISK

Commenters provided information on risk considerations with Type C Packages and Low Dispersible Material. They provided the following recommendations.

- Consider what tests would be practical for demonstrating compliance with the Type C standards (AT27)
- Require that packages be able to be dropped from a plane in mid-air without the package being breached (AT27)
- Consider impacts on public safety (AT27) (OA42)

8.2 ASSOCIATED COSTS

Commenters provided information on the costs associated with Type C packages and Low Dispersible Material.

- Medical costs will increase to reflect higher transportation costs (OA47)

- Food safety costs will increase because of FDA-approved food irradiation (OA42)

- Total costs will increase by at least 25 percent due to replenishing units and excess transportation charges (OA42)

- Shipping costs will increase (OA42)

- Consider medical costs (such as Medicare costs and hospital costs), because process irradiators are needed for medical sterilization (OA42)

8.3 PROBLEMS WITH NON-ADOPTION

A commenter provided information on non-adoption problems regarding Type C packages and Low Dispersible Material.

- Plutonium would never be flown into the United States because TS-R-1 requires that all Type C packages and all Low Dispersible Materials need multilateral approval. Because of the MOU between DOT and NRC, DOT cannot approve these shipments without NRC approval (MD06)

8.4 ISSUE SPECIFIC FACTORS FOR CONSIDERATION (INCLUDING BENEFITS)

Commenters provided specific factors for consideration regarding Type C packages and Low Dispersible Material.

- If the activity content is limited to the thresholds specified, then the impact on air transport of currently certified Type B packages would be minimal (MD08) (MD17) (MD20)

- Efforts to develop the testing method or acceptance criteria should be pursued later, given that the need for the package is a number of years in the future (MD08) (MD20)

- Process irradiators ship approximately 50 million curies a year, probably by air, not boat or freight. If a limitation is placed on air transport for radioisotope quantities such as Cobalt-60, the number of air shipments would increase by a factor of three to seven (OA42)

9.0 DEEP IMMERSION TEST

Commenters provided information on the proposed changes to the deep immersion test. Some commenters supported the proposed requirement.

- The proposed changes would not have a significant impact on the commenter's program because their packages containing greater than 10^5 A2 are already evaluated for deep immersion or already have been grandfathered (MD12)

- U.S. and IAEA transportation regulations should be consistent, due to the international nature of transportation (MD08) (MD17) (MD20) (0078)

Some commenters opposed the proposed requirement.

- It is insufficient and unrealistic (AT27)

- Need definitions of "rupture" and "buckling" to know which term is more stringent (OA41)

- The language revision makes the exception level more conservative, and the criteria for meeting the requirement less specific. The current criteria for meeting the requirement should be used as a specific definition for the TS-R-1 language of "no rupture." (MD12)

- Suggested that the present criteria be maintained and extended to cover all packages with activity levels greater than or equal to 10^5 A2 quantities, with a note that this is more conservative than TS-R-1 requirements. This would eliminate the requirement for special review and certification of U.S. origin package designs. For non-irradiated fuel element shipments, there would be no impact on availability and shipping costs because there are few shipments of the required quantities of this material (OA42)

Some commenters responded to NRC's question about whether package designs originating from the U.S. have to be specifically reviewed and certified before shippers can export them.

- If the response is not specific to the deep immersion test but applies to all package design criteria, then the shipment of U.S. certified package designs for import/export use beginning in mid-2001 is entirely dependent upon approval of such designs to TS-R-1 performance standards (MD08) (MD17) (MD20)

- Failure to grant U.S. competent Authority Certifications for such designs would seriously hinder the industrial radiography industry, and place U.S. package designers and manufacturers at a strong competitive disadvantage (MD08) (MD17) (MD20)

Other commenters posed questions regarding the proposed requirements.

- What are the criteria for a special form A1 quantity, and is the deep immersion test necessary for BU packages for special form materials? (OA42)

- What technical justification exists to relax our test criteria for packages of irradiated nuclear fuel? (AT30)

- Will previously approved packages be grandfathered, or will they need to be re-certified by means of a deep immersion test? (0066)

- How does 105 A2 compare with 106 Ci? (AT30)

- Is it an oversight that BU packages containing A1 special form sources are exempt from this test? (OA42)

9.1 CONSIDERATION OF RISK

A commenter provided information regarding risk considerations of the deep immersion test.

- The proposed requirements do nothing to ensure the safety of the packages (AT30)

9.2 ASSOCIATED COSTS

No comments were received.

9.3 PROBLEMS WITH NON-ADOPTION

No comments were received.

9.4 ISSUE-SPECIFIC FACTORS FOR CONSIDERATION (INCLUDING BENEFITS)

Commenters provided information on specific factors for consideration on the deep immersion test.

- The Lawrence Livermore National Laboratory did not use the term “rupture” when a tritium-filled underground tank leaked into the ground and groundwater (OA41)

- Because very few packages exceed 10^5 A2, industry has not assessed the impact on availability of packages and shipping costs if all packages with an activity greater than 105 A2 are required to pass the immersion test (MD08) (MD17) (MD20)

10.0 GRANDFATHERING PREVIOUSLY APPROVED PACKAGES

Commenters provided information on the proposed rulemaking for grandfathering previously approved packages.

Several commenters support the proposed provision to grandfather previously approved packages.

- Supports the proposal, assuming new regulations would continue to be stricter (AT30)
- Provision is necessary otherwise NRC would have to set aside hundreds of long-term disposal sites for the various Type B quantity containers currently in use at hospitals and research institution (OA42)
- Older packages should be grandfathered unless safety deficiencies are identified (MD08) (MD17) (MD20) (0057) (0078) (0083)
- Grandfathering should be allowed for domestic shipments, even though it is not allowed for international shipments under TS-R-1 (MD08) (MD17) (MD20)
- Grandfathering should not be limited to the last two major revisions. Grandfathering provisions in the current 10 CFR Part 71.13 should be retained. The approval of fabrication should be revised to reflect TS-R-1 limitations of approval within the last two major revisions or re-certification prior to fabrication (0051)
- Existing packages (even older ones) are safe and durable, because they must be maintained in accordance with the heightened quality assurance regulations of TS-R-1 (MD08) (MD10)
- NRC may immediately withdraw a license if a particular package created a safety concern (MD08) (MD12) (MD17) (MD20)
- TS-R-1 allows for a phase-out of manufacturing of any packages that are not certified to the 1996 version of TS-R-1 by December 31, 2006 (MD08) (MD20) (0051)

Other commenters opposed the grandfathering provision.

- While it is important for more stringent requirements to apply to all existing containers, relaxed provisions would effectively make new containers less safe. In such instances, it is preferable that older provisions remain in effect, instead of the newer, relaxed provisions (AT22) (AT27)
- Opposed grandfathering existing packages, stating safety as a concern (MD05)

Several commenters provided recommendations to NRC regarding the grandfather requirement.

- Include a grandfathering provision for continued transportation of packages, such as fuel C-spec, Certification of Compliance (CoC) packages at NRC, and DOT spec packages (OA42)
- Incorporate specific requirements into the grandfathering provisions in order to maintain an effective package program. Manufacturers of CoC containers or packages should be allowed to show, by calculations or testing, that upgraded standards and TS-R-1 have been achieved (OA42)
- Fabrication of a new packaging to meet existing design approvals could only occur on a case-by-case basis (MD08) (MD20)

- Older packages should follow the 1967 edition of SS #6 that requires old packages to be re-certified, removed from service, or shipped via exemption (AT27)

- Perform a backfit analysis and add it as a requirement to Part 71 (0066)

- Incorporate “Packages that have been prepared for transport prior to (five-year effective date) may be offered for transport provided that the labeling, marking, and placarding provisions of the regulations in effect at the time of shipment are complied with.” (MD12)

- Create a system that would allow presently designed packages to be used for a reasonable amount of time after changes to the regulations are adopted (MD10)

Some commenters raised other issues related to grandfathering.

- TS-R-1 and its requirements allow the continued use of existing packages with a valid certification, however, the requirements do not allow the continued manufacturing of new packaging (MD08)

- Depending on the types and numbers of packages impacted, if older packages were removed from service, then their ability to transport radioactive material could be impacted (0083)

- Grandfathering should be based on technical significance of regulatory changes, and not on an arbitrary number of changes to regulations (MD12)

- Grandfathering should prohibit construction of new packages that do not meet regulatory conditions and should allow the continued use of packages proven safe and effective, making replacement necessary only under certain conditions (0059)

Many commenters raised issues regarding the time frame of the certification license.

- A three or five-year certification license is too short (MD08)

- The limited time period (proposed two-year cycle) could result in regulatory changes that affect a package in the middle or end of its design and licensing process because it takes two to three years to fully design and test a new package. The U.S. might adopt a different version of the regulation on a different schedule without knowing what standards they should be approving to (MD08) (MD10) (MD17) (MD20)

- Once a package is approved to the existing standard, its use should continue to be authorized as the packages does not become “unsafe” simply because of a regulatory wording change (MD17)

- The proposed program may be possible if it is conducted as a U.S. regulators update regulations -- i.e., with minor continuous change -- and with major change occurring only periodically (MD08) (MD17) (MD20)

- The two-year cycle would require re-certification at least every six years (MD12) (0051)

- As part of re-certification, every cask’s original design might also have to be re-certified, causing additional costs without significantly improving safety (0057) (0066)

- The shorter cycle would likely put pressure on cask designers to make safety a more important design element (AT30)

- A two-year cycle would create confusion on the part of the shippers and officials and thus interrupt shipments (MD20)

- Package designs should be issued for a fixed period, such as 20 years, to assure that they do not become obsolete before they are manufactured (MD08) (MD20)

Commenters posed questions to NRC.

- Who will be the party responsible for determining when a package is no longer certified? (0083)
- How many packages are currently available for shipping radioactive materials? (MD05)

- Can NRC clarify what requirements would be kept in the IAEA regulations and what requirements would be kept in the U.S. regulations? (AT27)

- Clarify "full compliance with TS-R-1 requirements." Will NRC consider partial compliance with TS-R-1? (AT30)

- What pressure would be put on industry or cask makers to bring grandfathered casks into compliance? What would be the time frame for bringing grandfathered casks into compliance? (AT30)

- If NRC does not change the regulatory 10-year time frame, would there be requirements to modify grandfathered casks? (AT30)

10.1 CONSIDERATION OF RISK

A commenter provided information on risk considerations regarding grandfathering previously approved packages.

- The proposed cycle would have a significant adverse impact on the ability of the Navy to refuel and de-fuel the nation's nuclear powered warships. All existing Naval Nuclear Propulsion Program shipping containers could become uncertifiable in as few as six years (MD12)

10.2 ASSOCIATED COSTS

Commenters provided information on the costs associated with grandfathering previously approved packages.

- Grandfathering all current CoCs would greatly reduce costs and administrative burdens (OA42)

- The expense of designing and fabricating large Type B and spent fuel packages cannot be justified if the potential lifetime of the cask is limited to a time period as short as six years (0051)

- The cost of recertifying existing casks would be prohibitive (0057)

- A 10- or 20-year certification license would be more cost-effective (MD08)

10.3 PROBLEMS WITH NON-ADOPTION

No comments were received.

10.4 ISSUE-SPECIFIC FACTORS FOR CONSIDERATION (INCLUDING BENEFITS)

A commenter provided issue-specific factors for consideration in grandfathering previously approved packages.

- There could be unintended consequences if grandfathering ever makes existing safe packages illegal. It is possible that instead of re-qualifying, changing, or replacing the package, the use might go completely out of compliance with the other transport regulations in order to avoid detection and inspection (0059)

11.0 CHANGES TO VARIOUS DEFINITIONS

Commenters provided information on changes to various definitions in the proposed rule.

- Adopt definitions to the extent the terms are used in the updated regulations (MD12)

- Clarify the terms “rupture,” “collapse,” “buckling,” and “in-leakage.” (OA41)

- Opposed to adopting the TS-R-1 definition identifying the specific types of packaging allowed for Class 7, and unless DOT revises its regulations, there will be a conflict domestically (MD08) (MD17) (MD20)

- Clarify the differences between “uniformly distributed,” “distributed throughout,” and “homogeneous.” (MD08) (MD17) (MD20) (0078)

- No conflict identified between TS-R-1 and other programs’ definitions (0078)

- Need additional knowledge of how the revised definitions will be used in order to estimate the impact of the changes to definitions (MD12)

- The proposed definitions of “confinement system” and “package” are indistinguishable for packages intended to transport fissile material. Use only one term, or clearly distinguish between the two. If the definition of “confinement system” is added, the term “competent authority” must also be defined. If the definition of “package” is incorporated, then definitions of “excepted” and “industrial” must also be added (MD12)

- Paragraph 225 introduces the term “low dispersible radioactive material” but fails to provide any guidance about what characteristics qualify the material (0083)

- The definition of “low dispersible radioactive material” should not refer to surface contamination, but rather activation of a solid material (0049)

- Retain the current 2000 picocuries per gram radioactive material definition for shipments within the U.S. and determine shipping categories based on external gamma flux readings (MD04)

- Add a definition for “sealed source.” It means “(for use of A1 values) encapsulated radioactive material that was designed and manufactured under a specific license and has been assigned a sealed source identification registry number.” (0049)

- The Confinement System definition should be revised to include fuel assemblies, the PWR Basket, and the Shipping Cask, since all three provide different levels and degrees of confinement (0066)

11.1 CONSIDERATION OF RISK

No comments were received.

11.2 ASSOCIATED COSTS

No comments were received.

11.3 PROBLEMS WITH NON-ADOPTION

No comments were received.

11.4 ISSUE-SPECIFIC FACTORS FOR CONSIDERATION (INCLUDING BENEFITS)

No comments were received.

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12.0 CRUSH TEST FOR FISSILE MATERIAL PACKAGE DESIGN

Commenters provided information on crush test requirements for packages containing fissile material. Several commenters supported the proposed requirements.

- Adopt the testing sequence to assure international uniformity (MD08) (MD20)

Other commenters opposed the proposed requirements.

- Keep the current regulations, requiring the crust test and free drop test (MD12) (AT22) (AT27) (0078)

- The crush test is especially useful for large packages (AT22)

- The proposed requirement is problematic because the two types of test have different results (OA41)

- Supports the crush test, especially for shipments that are transported by rail (MD12)

- The proposed requirements would require re-analyzing packages currently used for the Naval Nuclear Propulsion Program (NNPP), however, it would not significantly impact the NNPP because most of the packages weigh more than 1,100 pounds (MD12)

12.1 CONSIDERATION OF RISK

A commenter provided information regarding the risks associated with a crush test for fissile material package design. The commenter suggested the following.

- Increase the reliability of the crush test by: making it a physical test, rather than a computer test; using full-scale packages that are loaded with non-radioactive materials; including crush test for all package sizes; increasing test parameters to reflect real-world conditions (AT22)

12.2 ASSOCIATED COSTS

A commenter provided information on the costs associated with the crush tests.

- It would be an unfair and costly burden to eliminate the 1000A2 activity limit without providing flexibility in test sequencing (0066)

12.3 PROBLEMS WITH NON-ADOPTION

No comments were received.

12.4 ISSUE-SPECIFIC FACTORS FOR CONSIDERATION (INCLUDING BENEFITS)

Commenters provided information on issue-specific factors concerning crush tests for fissile material package design.

- The impact of the elimination of 1000A2 activity limit for fissile material packages having a mass not greater than 500 kg and overall density not greater than1000 kg/m^3 based on external dimensions is currently unknown (MD08) (MD20)

- Remove the 1000A2 threshold for fissile packages on the grounds that A2 levels are intended to be an index of radiological hazard rather than critically potential and that it is inconsistent with TS-R-1 (MD12)

13.0 FISSILE MATERIAL PACKAGE DESIGN FOR TRANSPORT BY AIRCRAFT

Commenters provided information regarding the proposed requirements for fissile material package design for transport by aircraft. Some commenters supported the proposed requirements.

- Supports the requirements, as they are generally parallel to those already in place for surface mode accidents (MD12)

- The regulations need to be understood consistently by the people who approve package designs for transport of fissile materials by air. Because ICAO will adopt TS-R-1 in early 2001, shipments must meet the requirements in TS-R-1 for fissile materials (MD08) (MD20)

- The impact on the Naval Nuclear Propulsion Program (NNPP) is likely to be minimal because more NNPP shipments of radioactive material via air transport are excepted packages (MD12)

- TS-R-1 tested should be adopted in total, to include fissile material package design for transport by aircraft (0078)

Other commenters opposed the proposed requirements.

- Concern for the comprehensibility of the regulations for Type B or below quantities of fissile materials (MD10)

- Consider a streamlined approval process for designs of air transport of fissile material (MD08) (MD20)

- Do not have any radioactive materials transported by air, and due to the case of a crash in a hard-to-reach area fire test requirements should specify at least a two-hour standard (AT27)

- Allowing the transport of plutonium by air is in conflict with the regulations used in the 1970s (0074)

A commenter posed a question regarding fissile material package design.

- When and in what situations will the transportation of fissile level material by air be required? (AT32)

13.1 CONSIDERATION OF RISK

No comments were received.

13.2 ASSOCIATED COSTS

No comments were received.

13.3 PROBLEMS WITH NON-ADOPTION

No comments were received.

13.4 ISSUE-SPECIFIC FACTORS FOR CONSIDERATION (INCLUDING BENEFITS)

No comments were received.

14.0 SPECIAL PACKAGE APPROVALS

Commenters provided information concerning special package approvals. Some commenters supported the special package approvals.

- Supports proposal to create a system for providing special package approvals without using the existing exemption requirements (MD06)

- Part 71 regulations should be consistent for Certificate of Compliance holders and licensees (0083)

Other commenters opposed special package approvals.

- NRC should review and grant each application on a case-by-case basis, and not use a generic regulation for special package approvals (MD16) (AT22) (AT27) (OA41) (0090)

- First responders, emergency management coordinators at the local level, and the people in transport corridor communities have a right to information that a specialized exemption process would provide (0090)

- Concerns for the public need to be given adequate weight in decision-making (0090)

- Eliminate special package approvals from the scope of the rulemaking effort, unless a correlation to IAEA's regulations can be clarified (0050) (0073)

- Adoption of a "Special Arrangement" provision may be more efficient than a special packages approval because of the various types of vessels that must be addressed (OA42)

- A special arrangement certificate would be beneficial to allow the transport of the damaged equipment for disposal when a Type B package has been damaged, continues to secure and shield the sources, but does not meet compliance standards (MD17)

- Category 3 packages should be excluded from this rulemaking. The many Cobalt-60 and Cesium-137 irradiators originally used for research, should be examined for future rulemaking (OA42)

- IAEA's special arrangement provision applies to shipments between countries in nonconforming packages, and does not lend itself to domestic shipments (MD07) (MD08) (MD20)

Commenters provided information on large objects.

- Concern for the definition of a "special large object" (MD12) (OA41)

- If special provisions are added then the term "large" must be defined with respect to both size and weight (MD12)

- Consider revisions to Part 71 to address large objects in general (including reactor vessels, steam generators and condensers, and components from reactors undergoing decommissioning activities) (MD07) (MD08) (MD20) (0066) (0078) (0083)

- Objects such as oil tubes and pipes, that are impossible to package due to their size, should be exempted from transportation requirements outside of the current requirements (MD04)

Commenters raised issues related to Type B quantities.

- Type B orphan sources should be included in a separate rule from the special large packages, because there could be an overlap between orphan sources and Type B quantities (OA42)

- NRC and DOT should collaborate to address the possibility of initiating a program that would minimize package review costs of decommissioning Type B quantities of cobalt-60 and cesium-137 (OA42)

Commenters raised issues related to the Trojan Reactor Vessel (TRV).

- The Trojan Reactor Vessel (TRV) shipment is not an adequate basis for determining whether or not to remove the requirement for exemptions for special packages and replace it with other provisions (MD05) (MD06) (MD16)

- If TRV shipment is the baseline for determining whether to revise the regulations, NRC should limit the scope of this special approval. Evaluation of river and barging conditions are, in reality, under the jurisdiction of the Coast Guard (MD06)

- Revise Part 71 to incorporate the risk-informed basis of the TRV package for other special package approvals (OA42) (0066)

- The special arrangement provisions should be included in TS-R-1 as the model under which shipments such as recent transport of the TRV could be accommodated (MD12)

One commenter posed a question to NRC.

- Will the special package approvals provision apply only to vessels and not to steam generators or reactor internals? (MD05)

14.1 CONSIDERATION OF RISK

Commenters provided information about risk considerations with special package approvals.

- Consider the mode of transportation and avoid letting unqualified person be transporters (AT22) (AT27)

- Transportation risks, in many cases, are much lower than the potential risk of transferring cells at a facility to legal shipping containers (OA42)

- Revising Part 71 to include Category 2 would be difficult because of the associated risks (OA42)

14.2 ASSOCIATED COSTS

A commenter provided information on the costs associated with special package approvals.

- A relaxation of the requirements of special package approvals would potentially reduce the cost of these shipments (MD12)

14.3 PROBLEMS WITH NON-ADOPTION

Commenters provided specific issues that NRC should consider when deciding whether to propose a special package approval process and how that process should be defined.

- With respect to special shipments, any change made to 10 CFR Part 71 will need to be specific to those items that are going to be regulated under NRC's MOU. Some large components such as steam generator and demineralizers and pressurizers, will likely fall under DOT’s jurisdiction, while NRC would regulate items like reactor pressure vessels (e.g., the Trojan reactor pressure vessel) (MD07) (MD08)

- A precedent has been established, and the possibility exists that the requirements placed on the shipment of the TRV might have been more restrictive than might have been determined as necessary at this particular point in time (MD07)

Commenters provided information on the issue of whether the risk-informed basis used specifically for the approval of the TRV shipment should be approved and adopted for other special package approvals.

- The Trojan shipment review is a point of reference for the basis of other similar shipments, but each case should be assessed on its own special circumstances (MD08) (MD17) (MD20)

15.0 EXPANSION OF PART 71 QUALITY ASSURANCE REQUIREMENTS TO HOLDERS OF, AND APPLICANTS FOR, A CERTIFICATE OF COMPLIANCE

Commenters provided information on the proposed rulemaking expansion of Part 71 (Quality Assurance Requirements to Holders of, and Applicants for, a Certificate of Compliance). Some commenters supported the proposed expansion.

- Cask designers and fabricators should be held responsible, as are parties on the reactor side (MD05)

- The proposed changes to expand the quality assurance requirements will not have a significant impact on the Naval Nuclear Propulsion Program (MD12)

Other commenters opposed the proposed expansion.

- Extending responsibility to fabricators or certificate holders would likely encourage fabricators to exit, because of the proposal's excessive regulatory and paper burden (MD08) (MD18)

- NRC might be regulating packages for which it is not responsible under NRC's MOU, resulting in issues when certificate holders do business with the Department of Energy (MD06)

- Issuing a notice of violation (NOV) instead of a notice of nonconformance (NONC) will not result in additional compliance. The current Quality Assurance control on the Part 71 packages under Subpart H is adequate (MD17)

Commenters provided recommendations to NRC.

- Clarify the current proposed provisions, specifically what is in the current regulations and what would be in the proposed regulations (MD12)

- Make publicly available the proposed rule language, and be certain NRC knows all cask producers, in order to ensure effective regulatory compliance (AT22) (AT27)

- Do not assume that "all folks will always conform with all aspects of Part 71 regulations given the abundant evidence of Part 72 conformance problems" (0050) (0073)

- Maintain consistency of quality assurance provisions between 10 CFR Parts 71 and 72 for dual purpose casks used for storage and transportation of spent nuclear fuel and high-level radioactive waste (MD08) (MD15) (MD20) (OA42) (0078)

- Establish the distinction between Part 71/72 packages used to transport/store spent fuel and Part 71 packages used to transport sealed radioactive sources. Also, specifically exempt 10 CFR Part 50 reactor licensees from participation in nuclear power-specific quality assurance activities (OA42)

15.1 CONSIDERATION OF RISK

A commenter provided the following suggestion regarding the consideration of risk for expanding the quality assurance program.

- Require revisions to a certificate of compliance for any safety-related design changes in order to achieve risk minimization (MD10)

15.2 ASSOCIATION COSTS

Commenters provided information regarding the costs associated with expanding the scope of the quality assurance requirements.

- The proposed requirements would cause suppliers to leave the business due to the additional paperwork and regulatory burden (MD08) (MD18)

- The provisions of 10 CFR Part 71 would lower costs for the owner of the certificate of compliance, as well as for the user community. Any change in a 10 CFR Part 71 package currently requires a complete revision to the certificate of compliance, thus necessitating sequential revisions to all international competent authority validations. As a consequence, even a change for a minor issue would result in a financial expenditure in excess of $100,000. (MD10) (0061)

15.3 PROBLEMS WITH NON-ADOPTION

No comments were received.

16.0 ADOPTION OF ASME CODE

Commenters provided information on the adoption of the ASME code. Some commenters supported the adoption of the ASME code.

- Use ASME Codes for all products which are used in transportation and storage of radioactive materials and provide an explanatory guideline in the Code that speaks to the subject of material categorization, whereby all manufacturers are using the same criteria when categorizing (0061)

- Using ASME standards would improve current problems with casks and the current lack of quality assurance (AT22) (AT27)

- Radioactive fuel elements should be required to follow ASME standards (AT22) (AT27)

- Incorporation of the Code by reference is the appropriate regulatory mechanism, following the precedent set by 10 CFR Part 50.55a rulemaking for the ASME Code Section III, Division 1. NRC should consider issuing guidance endorsing the use of Section III, Division 3 Code Cases and incorporation of the revised Division 3 through 10 CFR Part 72 (0080)

Other commenters opposed the adoption of ASME codes. They provided the following concerns.

- Effects on transportation (0061)

- Adoption of voluntary standards into regulations, specifically the inconsistency between industry standards and regulations (MD08)

- Difficulty in following ASME changes if made quickly (MD12)
- Endorsement of the ASME code as it applied to the design, certification and fabrication of packages (0051)

- Widespread impact of the adoption (MD10) (MD12) (MD18)

- Impact on existing Naval Nuclear Propulsion Program packages (MD12)

- Impacts to overseas markets (MD10)

- Any "unintended consequences" (MD18)

Commenters addressed other issues related to adoption of the ASME Code.

- Place standards in the regulatory guides, not codified in NRC regulations in order to better enforce them, keep them current with ASME standard changes, and satisfy the Congressional mandate to consider their use as consensus standards (MD08) (MD12) (MD17) (MD20) (0078)

- When an applicant commits to following Section 3, their compliance with that standard is reviewed -- i.e., it becomes part of NRC's approval process, and NRC can enforce its use in that process (MD12)

- If the ASME code is adopted, the development of it and the information involved must be publicly available (MD15)

- ASME code should not be applied to the smaller Type B packages such as industrial radiography devices (MD17)

- ASME codes for dual-use spent fuel packages should not be applied to other packages based on "risk analysis" comparing irradiated fuel elements with radioactive sources doubly encapsulated in SS with welded closures and certified to meet the "Special Form" requirements ASME welding specification should not be applied to shipping packages for sealed radioisotopic sources (0066)

Commenters expressed posed questions to NRC.

- Does the proposed change apply to dual use packages or to all Certificate of Compliance holders? (OA42)

- How will the requirement change if the industry standard changes in the future? (OA44)

- Clarify whether all packages are covered, or just spent fuel casks (MD17)

- Is NRC able to enforce the standard without placing in the regulations (MD05)

- Expressed confusion with the proposed changes (OA43)

16.1 CONSIDERATION OF RISK

Commenters provided information about the risk associated with adoption of the ASME Code.

- Questions whether its adoption will improve public safety (0090)

- Incorporation of the ASME code could have a catastrophic effect on parts of DOE and U.S. industry (MD12)

16.2 ASSOCIATED COSTS

Commenters provided information on the costs associated with adoption of the ASME code.

- Regulatory burden significantly increases when voluntary standards become regulations, due to the fact that ensuring regulatory compliance is difficult to accomplish (MD08)

- Adoption of ASME code into Parts 71 and 72 will be more costly due to increased fabrication costs for both storage and shipping casks and burdensome due to the final closure weld requirement (0066)

- Code stamps for all shipping containers would be very costly and would provide no benefit. Restructuring the design and procurement process to satisfy ASME requirements would be costly, would provide no additional assurance of product quality, and would force a separate process to be created that would be different from that used for other work (MD12)

- Cost increases without an equivalent increase in packaging safety (0051)

16.3 PROBLEMS WITH NON-ADOPTION

Commenters provided information about the issue-specific factors for consideration with respect to adopting the ASME code.

- NRC should study the international impacts of the proposal and consider a comparable international standard in conjunction with the proposed adoption of the ASME code (MD10)

- Some benefits of a third-party authorized Nuclear Inspector would accrue to the industry, specifically common standards will decrease complexity and interpretation, lower cost, and increase safety (0061)

17.0 ADOPTION OF CHANGES, TESTS, AND EXPERIMENTS AUTHORITY

Commenters provided information on the issue of change authority. Some commenters supported the effort to allow changes while other commenters asked that change authority be allowed for all packages, not just dual purpose packages.

- Expedite this change -- i.e., possibly on a schedule consistent with the proposed modifications to Part 72 (0066)
- As long as a cask is used for storage only, changes to the cask should not require our prior approval because doing so provides extra burden with little additional public protection (0083)

Commenters encouraged NRC to allow change authority to both domestic and international packages, as TS-R-1 does not have a specific change authority.

- Change authority has been proven in other countries and would allow time savings for both the regulatory reviewer and the package designer and/or manufacturer (MD17)
- Change authority should be extended to all packages, licensees, or users, however, each change should be submitted to the NRC/DOT and maintained in a master file so other users or licensees are aware of the changes (MD08) (MD20)
- Change TS-R-1 so that it allows change authority for all certificate holders (0078)

Several commenters did not support change authority.

- Because the definition of "minimal" has historically been ill-defined (0050)
- Proposed requirements would not result in 10 CFR Part 71 conforming with TS-R-1, specifically where the issues paper states, "the current IAEA standard ST-1 does not contain any equivalent provisions for changing a transportation package's design, without prior review by the competent authority." (0050) (0073)
- In the regulatory presumption that changes to cask design require approval, in the event of a technical debate, the applicant should seek approval (OA44)
- Comprehensively detail and define classes of changes that would be categorized as non-safety related and beneath review authority (OA44)
- Manufacturers and purveyors of transport containers should not be allowed to make changes of any kind without specific approval (MD16)
- Certificate holders should not be allowed to make changes in spent fuel storage cask designs without prior approval (MD05) (AT27)
- Be consistent and revoke the change, test, and experiment authority for 10 CFR Part 72 certificate holders (AT22) (AT27)
- The proposed requirement will result in radioactive waste leaks unless NRC performs a very tight review of the proposed changes (OA43)

- Relaxing testing requirements is in conflict with the regulations used in the 1970's (0074)

Many commenters expressed interest in receiving additional information from NRC about what changes might be allowable and highlighting that these allowable changes should only be for non-safety related activities (e.g., switching to non-reactive paints).

- NRC and DOT should be careful in determining allowable, non-safety changes with the effort to lengthen the certificate re-validation cycle, because it is conceivable that these changes would just be rolled into the new certification without review (MD06)

- An example of a non-safety related activity is ongoing consolidation within the electric power industry where companies that hold a license under one name are merging or being purchased by other companies (MD10)

Commenters posed questions to NRC.

- Will NRC extend the adoption of changes, test, and experimental authority to non-Part 71/72 spent fuel casks? (OA42)

- How will NRC address the issue of conformity with other nation's package and certificates (MD06)

17.1 CONSIDERATION OF RISK

Commenters provided information about issues related to risk and authority to make changes without NRC approval.

- References a GAO report that highlighted problems with transportation casks fabricated by Westinghouse, claiming that 20 out of 40 casks had been found to be defective (MD05)

- Opposes any action, such as moving to performance or risk-based management, that would increase the level and type of public risk (MD05)

- Encouraged NRC to pursue risk-informed decision-making (MD05) (MD08)

- Wants to ensure that NRC would continue to be able to monitor industry performance (i.e., maintain regulatory oversight capability) and be able to undo or revise changes or force amendments when necessary (MD08)

- What NRC believes is a safety issue may be different from what the public believes and what industry believes is a safety issue may be different from what NRC believes (MD15)

- Carefully and completely delineate what the authority is and what types of changes would be possible. Opposed to a case-by-case NRC review of licensee or manufacturer requested changes (MD16)

- Be consistent and revoke the authority from storage casks, and do not give it to transportation casks (AT22) (AT27)

- Certificate holders should not be allowed to make changes that are not reflected in the final safety analysis report or in other steps of the license approval process (AT22)

17.2 ASSOCIATED COSTS

Commenters encouraged NRC to move towards performance-based regulations, as seen in 10 CFR Parts 50, 70, and 76, in order to reduce economic and regulatory burden.

- Opportunity exists to allow small, non-safety related changes to be made to reduce burden without reducing overall safety -- e.g., painting a cask (MD08)

17.3 PROBLEMS WITH NON-ADOPTION

Commenters provided information regarding the problems with non-adoption of changes, tests, and experiments authority.

- Support expanding consideration to include materials that are not as dangerous as spent fuel (MD06)
- One problem with adopting change authority may be the inadequacy of design changes for transporting radioactive waste (OA43)
- Adopting change authority would eliminate the need to obtain NRC agreement with minor package design changes, thereby reducing future efforts (MD12)

18.0 FISSILE MATERIAL EXEMPTIONS AND GENERAL LICENSE PROVISIONS

Commenters addressed the fissile material exemptions and general license provisions.

- Agreed with the necessity for 62 FR 5907, but there are issues yet to be resolved for water moderated shipments (MD08)

- NUREG/CR-5342 is pertinent to NRC's plan to issue a proposed rulemaking (MD08)

- If NRC adopts Issue 16, it will be unable to conform with TS-R-1, as TS-R-1 does not currently contain provisions on general licenses for shipment of fissile material (0050) (0073)

- Who bears the responsibility for the cost of spent fuel removal? (AT27) (AT32)

- If companies must pay to obtain a license for a nuclear power plant, NRC should raise the costs of these licenses to fully cover the cost of transporting spent fuels (AT27) (AT32)

- If licensed corporations do not fully cover the costs of spent fuel removal, then the public will be responsible for bearing a future high cost when those fuels have to be removed (AT27) (AT32)

Other commenters spoke about NUREG/CR-5342.

- Concerned with how recommendations 3 and 4 (from NUREG/CR-5342) would introduce unnecessary complexity; concerns vanish if the ST-1 definitions for regulated material are adopted (MD12)

- Recommendation 17 could eliminate the fissile excepted category, which is something that should not be allowed to occur; if such a change is necessary, the Agency instead should revise the excepted packages definition to reduce the amount of fissile material present and ensure that 10 CFR Part 71.53 and 49 CFR 173.453 are consistent with TS-R-1 (MD12)

- Requests that all of the 16 sub-issues contained in NUREG/CR-5342 that focus on Fissile Material Exemptions and General License Provisions be addressed in the rulemaking (0078)

18.1 CONSIDERATION OF RISK

No comments were received.

18.2 ASSOCIATED COSTS

Commenters said there is no specific cost information available now on the cost impact of the implemented emergency rule or of the ORNL recommendations.

- A simple estimate indicates that during decommissioning, the shipments of contaminated soil or building rubble to a low-level waste disposal facility could double or triple due to the conveyance limit; this would lead to a doubling or tripling in the cost for that portion of the decommissioning (MD08) (MD20)

- In comparison to 49 CFR 173.453, the proposed change would add 22 waste shipments which would increase the public's exposure, as well as significantly increase the transportation costs for this material (0078)

18.3 PROBLEMS WITH NON-ADOPTION

Commenters addressed specific issues related to fissile material exemptions and general license provisions.

- Important to coordinate regulatory actions on fissile material exemptions with the international community (MD06)

- Listen to international counterparts at the next IAEA meeting to ensure that fissile material exempt in the rest of the world is exempt in the United States, and vice-versa (MD06)

- The consignment limit has yet to be justified and it appears that the concentration limits required for this classification are sufficient to ensure safety during transportation (0078)

- Because TS-R-1 includes a similar concentration limit to the limit in NUREG/CR-5342, industry recommends the Agency adopt this exemption (MD08) (MD20)

- While TS-R-1 has a total limit of fissile material, the Agency should not adopt it because there is no basis for the limit (MD08) (MD20)

Commenters responded to each of the 18 sub-issues or recommendations contained in NUREG/CR-5342.

- Industry supports recommendations 1, 2, and 5 (MD08) (MD20)

- Industry supports recommendations 10 and 12, but 12 should also include sec. 71.20, sec. 71.24, and the CSI with TI in this reformulation (MD08) (MD20)

- Industry supports recommendation 15, but it should include sec 71.18, sec. 71.22, and the CSI with TI in this reformulation (MD08) (MD20)

- In supporting recommendation 16 and 17, industry supports use of TI and CSI to limit conveyance. In determining the CSI for a package, special moderators and/or reflectors would be considered. Regarding recommendation 17, industry recognizes that a fissile material package that is exempt from the fissile marking may require a CSI of 0 to assure safe handling during transport (MD08) (MD20)

- Industry supports recommendation 18, but the definition should not be limited to materials having enrichments less than 1 wt% U-235 (MD08) (MD20)

- Industry does not support recommendation 3; fissile material under the appropriate conditions can be shipped in a Type A or industrial package, and there is no safety basis to establish minimal requirements for construction of the package simply because the material is fissile (MD08) (MD20)

- Industry does not support recommendation 4 and believes that CSI and exemption values for criticality need to be established (MD08) (MD20)

- Industry does not support recommendation 6, and use of TI and the CSI will address the concern (MD08) (MD20)

- Industry does not support recommendation 8, and sec. 71.18(e) provides a reasoned basis for considering the moderators, and therefore should be retained (MD08) (MD20)

- Industry does not support recommendation 9. TI and CSI need to be considered when shipping fissile material; however, sec. 71.18(e) and sec 71.20(c)(2-3) need to be harmonized (MD08) (MD20)

- Industry does not support recommendation 11; the combination of the TI and CSI will determine the package necessary to ship Pu-Be source in a package that contains up to 2500-g Pu-239. Controlled shipping conditions are not needed (MD08) (MD20)

- Industry does not support recommendation 13; sec. 71.22(e) provides a reasoned basis for considering the moderators and/or reflectors and should therefore be retained (MD08) (MD20)

- Industry does not support recommendation 14 due to the same objections as in recommendation 9 (MD08) (MD20)

- Industry does not currently have a position on recommendation 7 because little if any U-233 is being shipped by the commercial sector (MD08) (MD20)

19.0 DOUBLE CONTAINMENT OF PLUTONIUM (PRM-71-12)

Commenters provided information on the issue of double containment of plutonium. Some commenters supported eliminating the double containment requirement for plutonium.

- Already uses double containment when transporting plutonium, and anticipates continuing the practice (MD12)

- Eliminate the double containment requirement for plutonium because the additional regulatory requirement of a separate inner container for packages containing plutonium is not congruent with the requirements for all other radionuclides. There would be several benefits: decreased worker exposure if process time were reduced; reduced costs through more efficient handling and packaging; and internal harmonization of regulations (MD12)

- Eliminate double container requirement to be consistent with TS-R-1 concerning all shipments, including plutonium (MD08) (MD20)

Others objected to the relaxation of the double containment requirements.

- Consider that plutonium is shipped shorter distances in Europe than in the U.S. (MD18) (0090)

- Apply the requirement to all packages and shipments, not just plutonium (AT22) (AT27)

- The requirement is inappropriate because there will be significant increases in plutonium transportation in the future, specifically WIPP shipments (MD06) (MD15) (AT30) (OA41) (0050) (0059) (0073) (0077)

- No need has been demonstrated to justify eliminating double containment (0053) (0077)

- Consistency with TS-R-1 is not as important as internal consistency and consistency with the performance basis of the regulations. The proposed provision conflicts with the intent to have a performance based regulatory system (0051)

- Justify to the IAEA why double containment is necessary and revise TS-R-1 (0078)

- Elimination of the double containment requirement must be based on a sound, publicly available (i.e., not only on ADAMS) technical justification demonstrating that existing safety margins are retained (0050) (0053) (0073)

- A double container is required by Congress in the WIPP Land Withdrawal Act (MD05)

- The TRUPACT-II is not sufficient to protect the public and the design criteria are less than the real road conditions that it could endure (MD15)

- If DOE renounces its commitment to use double containment shipping containers, it would be a direct contradiction of the commitments made early in the WIPP program to ensure safe shipping of this material (MD16) (0053)

- Western states have traditionally opposed the relaxation of the requirements for plutonium transport. Plutonium transport is not usually undertaken for commercial reasons (OA44)

- Perform considerable safety analysis before finalized proposed revision (OA44)

- It was not the intent of the petition, PRM-71-12, to compare it with international standards (ST-1). The petition should be considered independently and on its own merits (MD21)

A commenter posed a question to NRC.

- Will adoption of TS-R-1 actually increase permissible concentration levels for approximately 44 percent of the radionuclides addressed? Is plutonium one of the 44 percent of radionuclides that would see an increase in permissible levels? (MD05)

19.1 CONSIDERATION OF RISK

Commenters provided information on risk consideration issues related to double containment of plutonium.

- The proposed revision would reduce the level of public protection (MD05) (MD15) (MD16) (AT22) (OA44)

- What additional protection does double containment provide when containment provisions already exist in the regulations that apply to all radioactive materials including those that are probably as hazardous or as radiotoxic as plutonium (MD12)

- Based on the Q system for the calculation of A1 and A2 values, an A2 quantity of any radionuclide has the same potential for damaging the environment and the human species as an A2 quantity of any other radionuclide (MD21)

- Citing the Environmental Evaluation Group report, double containment would reduce the expected quantity of radionuclides released from accidents to 28 percent of that with the current design. The double containment design would limit the curies released in the class VIII accident to 40 percent of that with the current design. Similar reductions were shown in radiation doses and in environmental contamination and cleanup costs (0053)

- Double containment would drastically reduce the latent cancer fatalities that would occur if a Severity Category VII or VIII accident were to occur. The expected number of radionuclide release accidents would drastically decrease (from 12 to 0.02) (0053)

- Citing a NIH report, there exists a strong correlation between the amounts of radiation and the number of cancer cases in various areas (AT27)

- There is no health or social benefit associated with removing current double containment requirements for plutonium (OA44)

- The existing requirements are overly conservative. The Q-system and the A1 and A2 values of 10 CFR Part 71 can adequately address the hazards associated with plutonium shipments. Special requirements for plutonium do not increase the safety of transportation (0051)

19.2 ASSOCIATED COSTS

Commenters provided information on the associated costs of requiring double containment for shipments of plutonium.

- Conduct a risk/cost analysis and if the cost savings, relative to the risk minimization that double containment affords, then NRC should not revise the current standards. As part of this effort, ask whether the public is willing to bear the added costs associated with double containment relative to the risk minimization (0070)

- Questioned NRC's approach, asking if a regulation costs a lot, is it wrong, and if it does not cost a lot, then is it right? (MD21)

- Unnecessary and burdensome requirement (MD07)

- Cited instances where double containment (i.e., TRUPACT-II containers) was less expensive than single containment -- i.e., $675,000 versus $760,00, respectively (0077)

19.3 PROBLEMS WITH NON-ADOPTION

A commenter provided information on problems with non-adoption, opposing the double containment requirement for shipments of plutonium.

- Double containment is already an overkill that has been brought on by Congress for a radionuclide that is safe in transport due to the A1 and A2 values that have been defined for that particular radionuclide (MD07)

20.0 CONTAMINATION LIMITS AS APPLIED TO SPENT FUEL AND HIGH LEVEL WASTE (HLW) PACKAGES

Commenters provided information regarding contamination limits as applied to spent fuel and High Level Waste (HLW) packages.

- Proposed rule will not result in a significant impact because containers are already inspected prior to shipment to ensure that surface contamination levels are less than 450 pCi/100 square cm (MD12)

- Contamination limits should apply equally to all packages in order to minimize regulatory confusion and ensure a higher rate of regulatory compliance (0078) (0090)

Other commenters opposed increasing package contamination limits.

- NRC should not increase exposure in any way (AT27)

- Increasing package contamination limits would allow an increased, ongoing release of radioactivity into the environment (AT22)

- External contamination on packages of radioactive material in transport is a significant problem and is the source of actual or perceived hazard that can cause damage to the nuclear industry (MD12)

- Do not change contamination limits (i.e., continue to use TS-R-1 limits) unless and until there is a sound technical basis for doing so (MD12)

- Clarify and elaborate the discussion of the 4 Bq/square cm limit (0066)

Commenters spoke to the issue of worker exposure rates.

- Worker exposure rates will conceivably increase by using the existing surface contamination limit (i.e., four becquerels per square centimeter) for large packages (MD06) (MD08) (MD12) (MD15)

- Regulations are designed to protect the public first and the workers second, therefore do not change the regulations (MD08)

- Worker exposure could increase by requiring double containment, thus raising is required, and expressed concern about how this issue with contamination limits impacts international shipments (MD06)

- Worker exposure rates are not likely to be reduced even if allowable surface contamination rates were significantly increased (MD12)

- Workers will be exposed to radiation while measuring the surface contamination level, regardless of the level of the package contamination limit (AT22) (AT27) (AT30) (0083)

- NRC should consider other ways to protect workers, including cask design and the use of robots (AT27) (AT30)

- If radiation levels are too great for workers to get close enough to measure it, it is too great to transport it (AT27)

- Contamination levels should not be reduced for larger packages handled by crane (AT27)

- NRC should consider developing an alternate contamination limit that results in adequate protection for both radiation workers and the public using risk based methodology (0083)

Commenters addressed the issue of public protection.

- Raising surface contamination limits, as applied to spend fuel and HLW packages, will effectively lower public protection, which would reduce public trust and confidence in NRC (AT22)

- The public is already adequately protected from external package contamination and the 4 Bq/square cm criterion should be applied to all packages, which would be consistent with TS-R-1 (MD08) (MD20)

Commenters were concerned with contamination limits as applied to spent fuel and HLW packages.

- Uncertain whether adding complexity to cask standards would help when responding to an accident (OA43)

- Assuming that the acceptable contamination level would be reduced, NRC needs to clarify how low its benchmark needs to be and where it should be taken from (OA46)

20.1 CONSIDERATION OF RISK

Commenters spoke to the risks associated with contamination limits as applied to spent fuel and HLW packages.

- Reducing the risk to nuclear workers with the possible cost of increasing the general public's exposure is unacceptable (AT27)

Some commenters requested that NRC not relax the contamination limits because of the increased exposure risk.

- Carefully consider the added exposure risk to truck and rail crews, intermodal workers, and hypothetically maximally exposed individuals along rail and highway routes before making any changes to the 4Bq/square cm contamination limit (0070)

- Higher external contamination levels on packages eventually stopped German waste shipments (AT22)

20.2 ASSOCIATED COSTS

No comments were received.

20.3 PROBLEMS WITH NON-ADOPTION

Commenters raised two issues associated with non-adoption of revised contamination limits.

- NRC should address work standards because U.S. worker dose rates are two and one-half times greater than those in the rest of the world but no effort has been made to harmonize on this point (MD05)

- If contamination limits are revised upwards, then the allowable revision should depend upon the total design of the package and transport system (i.e., totally enclosed packages might have lower limits that casks with accessible surfaces) (0059)

21.0 OTHER ISSUES

Commenters submitted comments on other issues related to the rulemaking.

- NRC should begin a proactive implementation and adoption of TS-R-1, similar to DOT's efforts with a transition rule, in order to avoid regulatory conflict (MD08) (MD19)

- Clarify whether high level waste is as highly route controlled (i.e., security is with the shipment at all times) as spent fuel (OA47)

- Clarify if and when IAEA/IATA regulations are in effect in NRC's and DOT's regulations (0049)

- Clarify when NRC's regulations supersede DOT's, and vice-versa, for domestic shipments (0049)

- Streamlining regulations may not serve the interests of public health and safety -- e.g., inappropriate design changes, reduced oversight (OA43)

- NRC could reduce public fears by posting signs on canisters of spent nuclear fuel while they are in transport that specify safe distances and lower exposure (ALARA) is desirable (0056)

- NRC needs to perform analyses to delineate increases, decreases, or neutral effects in radiation exposure to persons living in communities along transport routes (MD16)

21.1 CONSIDERATION OF RISK

No comments were received.

21.2 ASSOCIATED COSTS

No comments were received.

21.3 PROBLEMS WITH NON-ADOPTION

No comments were received.

21.4 ISSUE-SPECIFIC FACTORS FOR CONSIDERATION (INCLUDING BENEFITS)

No comments were received.

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 3.7×10^{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.

APPENDIX A

THE ISSUES PAPER

NUCLEAR REGULATORY COMMISSION

10 CFR Part 71

Major Revision to 10 CFR Part 71: Compatibility With ST-1--The IAEA Transportation Safety Standards--and Other Transportation Safety Issues, Issues Paper, and Notice of Public Meetings

AGENCY: Nuclear Regulatory Commission.

ACTION: Request for comment on issues paper, and notice of plans for public meetings.

SUMMARY: The Nuclear Regulatory Commission (NRC) is considering a rulemaking that would revise the Commission's regulations on packaging and transporting radioactive material to make it compatible with the International Atomic Energy Agency (IAEA) transportation safety standards as well as codify other requirements. The NRC is seeking early public input on the major issues associated with such a rulemaking. To aid in that process, the NRC is requesting comments on the issues paper included in this notice. Specifically, the NRC is interested in public and industry comments related to: Quantitative information on the costs and benefits resulting from consideration of the factors described in the issues paper, operational data on radiation exposures (increased or reduced) that might result from implementing the contemplated changes; whether the presented factors are appropriate; and whether other factors should be considered, including providing quantitative information for these factors. The Commission believes that the stakeholders' comments will help to quantify the potential impact of these changes and will assist the NRC, as the proposed rule is developed, in developing a risk-informed alternative as its preferred option. NRC also intends to conduct three public meetings in August and September of this year to discuss those issues and solicit public comments.

DATES: Submit comments at the public meetings, or in writing by September 30, 2000. Comments received after this date will be considered if it is practicable to do so, but the Commission is able to assure consideration only for comments received on or before this date.

In addition to providing opportunity for written (and electronic) comments, public meetings on the paper will be held as follows: August 10, 2000 NRC Headquarters, Washington, DC, 8:30 am-5pm; September 20, 2000, Atlanta, Georgia, J.W. Marriott, 3300 Lenox Road Northeast, Atlanta, GA 30326, 6-10 pm; September 26, 2000, Oakland, California, Oakland Federal Building, Edward R. Roybal Auditorium and Conference Center, 1301 Clay Street, Oakland, CA 94612, 6-10 pm.

ADDRESSES: Submit comments to: Secretary, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Attention: Rulemaking and Adjudications staff.

Deliver comments to 11555 Rockville Pike, Rockville, Maryland, between 7:30 a.m. and 4:15 p.m. on Federal workdays.

You may also provide comments via the NRC's interactive rulemaking web site at http://ruleforum.llnl.gov. This site provides the capability to upload comments as files (any format), if your web browser supports that function. For information about the interactive rulemaking web site, contact Ms. Carol Gallagher, (301) 415-5095 (e-mail: CAG@nrc.gov).

Copies of any comments received and documents related to this action may be examined at the NRC Public Document Room, 2120 L Street NW (Lower Level), Washington, DC Documents created or received at the NRC after November 1, 1999 are also available electronically at the NRC's Public Electronic Reading Room on the Internet at http://www.nrc.gov/NRC/ADAMS/index.html. From this site, the public can gain entry into the NRC's Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. For more information, contact the NRC Public Document Room (PDR) Reference staff at 1-800-397-4209, 202-634-3273 or email to pdr@nrc.gov.

FOR FURTHER INFORMATION CONTACT: Naiem S. Tanious, telephone: (301) 415-6103; e-mail: nst@nrc.gov, Office of Nuclear Material Safety and Safeguards, USNRC, Washington, DC 20555-0001. Specific comments on the public meeting process should be directed to Francis X. Cameron; e-mail fxc@nrc.gov, telephone: (301) 415-1642; Office of the General Counsel, USNRC, Washington, DC 20555-0001.

SUPPLEMENTARY INFORMATION:

I. Background

By international agreement and through Commission direction, the NRC staff is preparing an overall rulemaking effort that addresses the need to make 10 CFR Part 71 regulations, “Packaging and Transportation of Radioactive Material” compatible with the most current revision of the IAEA Safety Standards Series No. ST-1. Part 71 is based, in general, on the safety standards developed by the IAEA. The IAEA has been revising its transportation standards on approximately a 10-year cycle, with the last edition, ST-1, published in December 1996. Further, several additional issues related to other changes to 10 CFR Part 71 are being considered by NRC. These issues include the fissile material exemptions, general license provisions, and the current requirements for double containment of plutonium.

The NRC is supplementing its standard rulemaking process by conducting enhanced public participatory activities including facilitated public meetings before the start of any formal rulemaking process to solicit early and active public input on major issues with revision of 10 CFR Part 71. The NRC will also utilize its rulemaking web site to make the issues paper available to the public and to solicit public comments. To facilitate discussion and public comments, the NRC has prepared an issues paper that describes 18 rulemaking issues (IAEA and Non-IAEA-related) to be addressed in revisions to Part 71. These issues are described in more detail in Section III of this notice.

II. Request for Written and Electronic Comments and Plans for Public Meetings

The NRC is soliciting comments on the items presented in the issues paper in Section III of this notice. Comments may be submitted either in writing or electronically as indicated under the ADDRESSES heading. In addition to providing an opportunity for written comments, the NRC is holding facilitated public meetings at three different geographical locations on the issues discussed in Section III (see the DATES heading of this notice for the dates and locations of these meetings). In addition to the NRC staff, a representative from the Department of Transportation (DOT) will be available to answer any questions related to their concurrent rulemaking efforts.

In addition to inviting public comments on the issues presented in Section III, NRC is soliciting specific comments related to: (1) Quantitative information on the costs and benefits resulting from consideration of the factors described in the issues paper, (2) operational data on radiation exposures (increased or reduced) that might result from implementing the Part 71 changes; (3) whether the presented factors are appropriate; and (4) whether other factors should be considered, including providing quantitative information for these factors. The Commission believes that the stakeholders' comments will help to quantify the potential impact of these changes and will assist the NRC, as the proposed rule is developed, in developing a risk-informed alternative as its preferred option.

Based on the comments received in written or electronic form, and at the public meetings, the Commission will then be in a better position to evaluate options for Part 71 rulemaking, to decide on the preferred options, and to proceed with development of a proposed rule.

III. Issues Paper on Major Revision to 10 CFR Part 71: Compatibility with ST-1--the IAEA Transportation Safety Standards--and Other Transportation Safety Issues

A. Introduction

1. Background

In 1969, the International Atomic Energy Agency (IAEA), recognizing that its international regulations for the safe transportation of radioactive material should be revised from time to time because of scientific and technical advances, and accumulated experience, invited Member States (the U.S. is a Member State) to submit comments and suggest changes to its standards. As a result of this initiative, the IAEA issued revised standards in 1973 (Regulations for the Safe Transport of Radioactive Material, 1973 Edition, Safety Series (SS) No. 6). The IAEA has periodically reviewed its transportation regulations (about every ten years) to ensure that the regulations are kept current. Thus, a review of IAEA regulations was initiated in 1979 and resulted in the publication of revised regulations in 1985 (1985 Edition, SS No. 6).

The U.S. Nuclear Regulatory Commission (NRC) also periodically revises its regulations to make them compatible, to the extent appropriate, with those of the IAEA. On August 5, 1983 (48 FR 35600), the NRC published, in the Federal Register, a final revision to 10 CFR Part 71, “Packaging and Transportation of Radioactive Material.” That revision, in combination with a parallel revision of the hazardous materials transportation regulations of the U.S. Department of Transportation (DOT), brought U.S. domestic transport regulations into general accord with the 1973 edition of SS No. 6. The next IAEA revision of the transportation standards in SS No. 6 resulted in a revision to Part 71 that was published on September 28, 1995 (60 FR 50248), to make Part 71 compatible with the 1985 edition of SS No. 6. DOT published its corresponding revision to Title 49 of the Code of Federal Regulations on the same date.

In each case, the NRC coordinated its Part 71 revisions with the DOT. DOT is the U.S. Competent Authority for transportation of hazardous materials. “Radioactive Materials Regulations” is a subset of “Hazardous Materials Regulations” in Title 49. The DOT and the NRC co-regulate transport of radioactive material in the United States and have a Memorandum of Understanding to that effect.

The last revision to the IAEA SS No. 6 was titled Safety Standards Series No. ST-1, referred to hereafter as ST-1, and was published in December 1996.

2. Scope of Part 71 Rulemaking

The Commission has directed the NRC staff to begin rulemaking to revise Part 71 for compatibility with ST-1. The NRC staff compared ST-1 to SS No. 6 to identify changes made in ST-1, and then identified affected sections of Part 71. Based on this comparison, the NRC staff identified eleven Part 71 IAEA-compatibility issues to be addressed through the rulemaking process. These eleven issues (identified as issues 1 through 11) are discussed in greater detail in Section B. Seven additional issues were identified (issues 12 thru 18) for incorporation in the rulemaking process, through NRC staff identification and through Commission direction, and are also discussed in further detail in Section B.

The Part 71 rulemaking and this issues paper are being coordinated with DOT to ensure that consistent regulatory standards are maintained between NRC and DOT radioactive material transportation regulations, and to ensure coordinated publication of the final rules by each agency. Note that on December 28, 1999 (64 FR 72633), DOT published an Advance Notice of Proposed Rule regarding adoption of ST-1 in its regulations, and plans to proceed to develop a proposed rule for public comments and subsequently a final rule. In order to develop a final rule concurrent with the timing of the DOT final rule, the NRC staff developed the following schedule: (1) the NRC staff will submit to the Commission for approval, a proposed rule to revise Part 71 by March 1, 2001, (2) the proposed rule is expected to be published for public comment in April 2001, (3) the NRC staff is planning to hold public meetings during the public comment period, and (4) after the end of the public comment period, the staff will revise the rule and submit it for approval as a final rule by June 2002.

The NRC proposed rule will include a cost-benefit (regulatory analysis). Contrary to the NRC's rulemaking process under the Administrative Procedure Act, development of the IAEA ST-1 did not directly involve the public or include a cost-benefit analysis, to our knowledge. In contrast, NRC is bound to consider costs and benefits in its regulatory analysis, and is prepared to differ from the ST-1 standards, at least for domestic purposes, to the extent the standards cannot be justified from a cost-benefit perspective.

B. Issues Format

The following format is used in the presentation of the issues that follow. Each issue is assigned a tracking number with a short title, and includes an issue description paragraph and a listing of factors for consideration. The factors for consideration in this document are not meant to be a complete or final listing, but are included to help prompt consideration and discussion of the issue. In August and September 2000, through a series of public meetings and a summary workshop, the public and industry will be requested to (1) comment on and recommend additions, deletions, or modifications to the factors for consideration; (2) propose implementation options for each issue; and (3) provide estimated implementation cost information. Other venues for feedback will be made available through mailings and by internet through the NRC web site. This public feedback will then be used in developing implementation options for Commission consideration as the Part 71 rulemaking process proceeds. Comments received that are outside the scope of this rulemaking may be addressed in future rulemaking if warranted.

Factors for consideration that are common to most of the issues are stated here, rather than repeated in each issue. These include: (1) How should risk considerations (i.e., what can happen, how likely is it, what are the consequences) be factored into rulemaking on applicable issues, (2) costs (i.e., administrative, training, testing) to industry and/or Government agencies in adopting ST-1 requirements (issues 1-11) or the NRC-initiated changes (issues 12-18), and (3) potential problems that may occur as a result of adopting ST-1 requirements, or problems that may occur from partial or non-adoption of the ST-1 requirements resulting in dual standards between domestic (10 CFR 71) and international (ST-1) requirements. For issues 1-11, the "factors for consideration" noted under each issue are generally written in the context of adopting the ST-1 requirements into Part 71.

In the case of the eleven IAEA-compatibility issues, portions of the Safety Standards Series ST-1 are referenced by the corresponding paragraph number from the original IAEA document. The full text of the ST-1 references can be found in Appendix A of this issues paper.

Issue 1. Changing Part 71 to SI Units Only

Description

ST-1, Annex II, page 199 states: "This edition of the Regulations for the Safe Transport of Radioactive Material uses the International System of Units (SI)." The change to SI units exclusively is evident throughout ST-1. ST-1 also requires that activity values contained in shipping papers and displayed on package labels be expressed only in SI units (paragraphs 543 and 549). SS No. 6, 1985 Edition, used SI units as the primary controlling units, with subsidiary units in parentheses; either units were permissible on labels and shipping papers.

The ST-1 requirement regarding only the use of SI units conflicts with the NRC Metrication Policy issued on June 19, 1996 (61 FR 31169). This policy allows a dual-unit system to be used; SI units with English units in parentheses. According to the NRC's metrication policy, the following documents should be published in dual units: New regulations, major amendments to existing regulations, regulatory guides, NUREG-series documents, policy statements, information notices, generic letters, bulletins, and all written communications directed to the public. Documents specific to a licensee, such as inspection reports and docketed material dealing with a particular licensee, will be issued in the system of units employed by the licensee. Currently, Part 71 utilizes the dual unit scheme in accordance with the NRC Metrication Policy.

Factors for Consideration

- What changes would licensees and Certificate of Compliance holders have to make to relevant documents if NRC revised 10 CFR Part 71 to require SI units only?
- What risks and safety impacts might occur in shipments because of possible confusion or erroneous conversion between the currently utilized English units and SI units?
- What sort of transition period would be needed to allow for the conversion to exclusive use of SI units?
- What other conforming changes would have to be made to Title 10?

Issue 2. Radionuclide Exemption Values

Description

Exempt materials are those which are of such low potential hazard that they may not be required to be shipped in accordance with specific transportation regulations. In ST-1, the IAEA adopted a new approach to specifying these materials by developing radionuclide-specific activity concentration values for exempt materials and activity limits for exempt consignments. These new values are found in ST-1, Tables I and II, and Section IV. Related information is provided in paragraphs 401 through 406 of ST-1. Exempt materials are those that fall below the listed activity concentration values. Exempt consignments are packages or loads that have a total activity less than the listed activity values.

The exempt materials activity concentration values range from 0.1 to 1,000,000 Bq/g, with most radionuclides in the 1 to 100 Bq/g range. This IAEA requirement does not currently exist in Part 71. Appendix A to Part 71--Determination of A_1 and A_2, does not contain exemption values for each radionuclide because the exemption for low-level radioactive material as contained in 10 CFR 71.10(a) is 70 Bq/g (2000 picoCuries per gram) or less.

Some materials, such as ores containing naturally occurring radionuclides, would be brought into the scope of the regulations for the first time; however, provisions are included in ST-1 that reduce the potential impact on natural materials containing radionuclides at these low levels. The provisions continue to exempt natural material and ores containing naturally occurring radionuclides, that are not intended to be processed for the use of these radionuclides, provided the activity concentration of the material does not exceed 10 times the values [ST-1 paragraph 107(e)]. Additionally, for materials that may appear in the scope of the regulations for the first time, but which have activity concentrations not exceeding 30 times the exempt activity concentrations, provisions exist in ST-1 to allow them to be transported as LSA-I materials that may be transported unpackaged (in bulk). However, there may be unintended consequences in implementing the ST-1 concentration values where applied to non-transportation activities. The DOT current exempt material standard of 70 Bq/g (2000 picoCuries per gram), based on previous IAEA transportation standards, has application by cross reference outside the domain of transportation.

Factors for Consideration

- In some cases, would shippers have to expend resources to: (1) Identify the radionuclides in a material; (2) measure the activity concentration of each radionuclide; and, (3) apply the method for mixtures of radionuclides when determining the basic radionuclide values for exempt material?
- Should the exemption values apply to domestic as well as export shipments?
- If the exemption values only applied to export shipments, would the resulting standard be practical to implement?
- If DOT specifies the exemption values in its regulations (49 CFR 173), should the NRC incorporate those same exemption values in Part 71, or simply make reference to the exemption values in the DOT regulations?
- There may be unintended consequences to adoption of specific exemption values as the current exemption value is used for non-transportation related activities. To what extent and in what manner would a change to specific exemption values affect entities whose non-transportation activities are linked to the current exemption value?

Issue 3. Revision of A_1 and A_2

Description

The A_1 and A_2 values specified in Part 71, Appendix A, are basic dose-based values used in several areas of the regulations, including determining the type of package that must be used for transporting radioactive material. For example, the A_1 values are the maximum activity of special-form materials allowed in a Type A package, and the A_2 values are the maximum activity of non-special-form material allowed in a Type A package. The A_1 and A_2 values are also used for several other quantitative limits including Type B-package activity release limits, low-specific activity material specifications, and excepted package content limits.

The ST-1 revised A_1 and A_2 values are primarily based on dosimetric models that use the IAEA's Q system for dose determination. The Q system includes consideration of a broad range of specific exposure pathways consisting of: External photon dose, external beta dose, inhalation dose, skin and ingestion dose because of contamination, and dose from submersion in gaseous isotopes. The main changes in the Q system resulted from making the dosimetric models consistent with those used in International Commission on Radiation Protection (ICRP) Publication 61. The lung model and dose conversion factors were updated to the latest ICRP models and the radionuclide values were recalculated. The Q system reference doses and exposure pathways were not changed.

Factors for Consideration

- Is there a practical alternative to adoption of the A_1 and A_2 values?
- Are there specific values that should be modified for domestic use only? What would be the justification for doing so?
- To what extent should the US partial adoption of ICRP 61 be considered for revising the A_1 and A_2 values?

Issue 4. Uranium Hexafluoride Package Requirements

Description

ST-1 introduces detailed requirements for uranium hexafluoride (UF_6) packages designed for more than 0.1 kg UF_6 NRC certifies Type B and fissile (i.e., enriched uranium) UF_6 packages under 10 CFR Part 71. Although most of these issues are under DOT in 49 CFR Part 173, the new ST-1 provisions relevant to 10 CFR Part 71 are summarized as follows (see Appendix A for a listing of the specific ST-1 provisions):

Para 629: Packages shall be packaged and transported in accordance with an international standard, ISO 7195, “Packaging of Uranium Hexafluoride (UF_6) for Transport.” ST-1 also allows [para 632(a)] for use of equivalent national standards (e.g., ANSI N14.1); provided that approval by all countries involved in the shipment is obtained (i.e., multilateral approval).

Para 630: ST-1 requires that packages must withstand: (a) A minimum internal pressure test to 2.8 MPa (1.4 MPa for multilateral approval), (b) the “normal conditions of transport” drop test, and (c) the hypothetical accident condition thermal test (except that packages containing grater than 9000 kg are exempt from this test if given multilateral approval).

Para 631: ST-1 prohibits packages from utilizing pressure relief devices.

Para 677(b): ST-1 includes an exception that allows UF_6 packages to be evaluated for criticality without considering the in-leakage of water into the containment system. This provision means that a single fissile UF_6 package does not have to be subcritical assuming that water leaks into the containment system. This provision only applies when there is no physical contact of the cylinder valve to any other component of the packaging after the hypothetical accident tests, the valve remains leak-tight, and when there is a high degree of quality control in the manufacture, maintenance, and repair of packaging coupled with tests to demonstrate closure of each package before each shipment.

Factors for Consideration

- NRC practice has been to certify fissile UF_6 packages (including the cylinder which is the containment vessel and a protective overpack) that are shown to be leaktight when subject to the hypothetical accident tests and to specify that the cylinder meets ANSI N14.1 (ANSI N14.1 has the domestic pressure test requirement in 630(a), not the regulations). For this reason, it is believed that NRC-certified UF_6 packages already comply with the above package performance requirements (para 630 and 677(b)). However, these changes appear to have significant ramifications for non-fissile UF_6 packaging that are under the purview of DOT.
- NRC practice has been to reference the ANSI N14.1 standard in the certification, but not to reference the standard in the rule. Although the ISO-7195-2000 standard (in draft) has been drafted taking into account ANSI N14.1, a detailed confirmation of the compatibility of the two standards has not been performed. NRC has representation on the ANSI N14.1 revision panel.

Issue 5. Introduction of Criticality Safety Index (CSI) Requirements

Description

For fissile material packages, ST-1 defines a new term, “criticality safety index” (CSI) (paragraph 218), that applies in addition to the traditional package transport index (TI). In current domestic regulations and in the previous IAEA regulations, the overall package TI was determined based upon the more limiting of a “TI based upon criticality considerations” and a “TI based on package radiation levels.” Both NRC and DOT regulations define and rely on the TI to determine appropriate safety requirements.

The CSI is determined in the same manner as the current TI “based upon criticality considerations,” but it now must be displayed on shipments of fissile material (paras 544-545) using a new “fissile material” label. A package TI is still determined in the same way as the “TI based on package radiation levels” and continues to be displayed on the traditional “radioactive material” label.

Factors for Consideration

- Under the new approach, it is believed that some shipments of fissile material packages might be made more efficiently (equivalent safety but more packages allowed in a single shipment), due to avoiding the situation where separation distance requirements (radiological safety) restrict package accumulation (criticality safety), or vice versa.
- Are any issues envisioned in the use of two TI values for shipments?

Issue 6. Type C Packages and Low Dispersible Material

Description

IAEA has adopted the concept of a new category of package, the Type C package (paragraphs 230, 667-670, 730, 734-737) that could withstand severe accident conditions in air transport without loss of containment or significant increase in external radiation levels. At the same time, ST-1 introduced a new category of material, Low Dispersible Material (LDM), which due to its limited radiation hazard and low dispersibility could continue to be transported by aircraft in Type B packages. U.S. regulations have no Type C package or LDM category, but do have specific requirements for the air transport of plutonium. These specific NRC requirements for the air transportation of plutonium (10 CFR 71.64 and 71.74) continue to apply, and will not be addressed in this rulemaking.

The Type C requirements apply to packages destined for air transport that contain a total activity above the following thresholds: for special form material--3,000 A_1 or 100,000 A_2, whichever is lesser, and for all other radioactive material--3,000 A_2. Below these thresholds, Type B packages would be permitted to be used in air transport.

The Type C package performance requirements are significantly more stringent than those for Type B packages. For example, a 90 m/s impact test is required instead of the 9 m-drop test. A 60-minute fire test is required instead of the 30-minute Type B requirement. Other additional tests, such as a puncture/tearing test are also imposed. These tests are more stringent and are expected to result in package designs that will survive more severe aircraft accidents than Type B package designs.

The LDM specification was added to account for materials (package contents) that have inherently limited dispersibility, solubility, and external radiation levels. The test requirements for LDM are a subset of the Type C package requirements (90 m/s impact and 60 minute thermal test) with an added solubility test, and must be performed on the material without packaging. Specific acceptance criteria are established for evaluating the performance of the material during and after the tests (less than 100 A_2 in gaseous or particulate form of less than 100 micrometer aerodynamic equivalent diameter and less than 100 A_2 in solution). These stringent performance and acceptance requirements are intended to ensure that these materials can continue to be transported safely in Type B packages aboard aircraft.

Factors for Consideration

- What would be the impact on air transport of currently certified Type B packages if the activity content is limited to the activity content thresholds specified above?
- What tests and analyses would be a practical method for demonstrating compliance with the type C package standards?

Issue 7. Deep Immersion Test

Description

The IAEA performance requirement for deep water immersion contained in ST-1 (para. 657 and 730) is an expansion of the requirement contained in SS No. 6. Previously, the deep immersion test was only required for packages of irradiated fuel exceeding 37 PBq (1,000,000 Ci). The ST-1 requirements apply to all Type B(U) and B(M) packages containing more than 10^5A_2 and to Type C packages.

10 CFR 71.61 requires a deep immersion test for packages of irradiated nuclear fuel with activity greater than 10^6 Ci. Currently, 10 CFR 71.61 is more conservative than SS No. 6, with respect to irradiated fuel package design requirements because it requires that a package for irradiated nuclear fuel must be designed such that its undamaged containment system can withstand an external water pressure of 2 MPa for a period of not less than one hour without collapse, buckling, or in leakage of water. The conservatism lies in the test criteria of no collapse, buckling, or in leakage as compared to the “no rupture” criteria found in SS No. 6 and ST-1.

To be consistent with ST-1, the NRC would have to revise 10 CFR Part 71.61 to apply to all packages with activity greater than 10^5A_2 and adopt the ST-1 test criteria.

Factors for Consideration

- How should the differences in the acceptance standards be addressed?
- What would be the impact on availability of packages and shipping costs if all packages with an activity greater than 10^5A_2 are required to pass the immersion test requirements?
- Would US origin package designs have to be specially reviewed and certified before shippers could export them in accordance with international regulations if ST-1 requirements were not adopted?

Issue 8. Grandfathering Previously Approved Packages

Description

Historically, IAEA, DOT, and NRC regulations have included transitional arrangements or “grandfathering” provisions whenever the regulations have undergone major revision. The purpose of grandfathering is to minimize the costs and impacts of implementing changes in the regulations. Package designs and packagings compliant with the existing regulations do not become “unsafe” when the regulations are amended (unless a significant safety issue is corrected in the revision).

Grandfathering typically includes provisions that allow for: (1) Continued use of existing package designs and packagings already fabricated, although some additional requirements may be imposed, (2) completion of packagings in the process of being fabricated or that may be fabricated within a given time period after the regulatory change; and (3) limited modifications to package designs and packagings without the need to demonstrate full compliance with the revised regulations, provided that the modifications do not significantly affect the safety of the package.

A major change in ST-1 is that “grandfathering” should be limited to only those package designs that have been certified under the last two major revisions of the regulations. Packages approved under an earlier revision would either be removed from service or be required to be re-certified under the revised regulations that result from this rulemaking.

As revised in 1996, IAEA regulations in ST-1 only recognize the “grandfathering” of package designs certified under the 1973 and 1985 editions of IAEA regulations (SS No. 6). Package designs approved under the 1967 edition of SS No. 6 would be required to be re-certified, removed from service, or shipped via exemption (i.e., special arrangement). If this approach to “grandfathering” is adopted in DOT and NRC regulations, package designs approved to earlier versions of DOT and NRC regulations (i.e., those based on 1967 IAEA regulations) would be required to be re-certified, removed from service, or shipped via exemption.

Factors for Consideration

- Should the “grandfathering” of previously approved packages be limited to those approved under the last two major revisions of the regulations? If not, on what basis should the “grandfathering” of previously approved packages be allowed?
- How long should “grandfathered” packages be allowed to be fabricated or used?
- What type and magnitude of package design changes should be allowed for “grandfathered” packages, before re-certification to the current set of regulations is required?
- IAEA has initiated a process to review and update ST-1 on a two-year frequency and does this new process raise any issues on the grandfathering limitations to the last two major revisions?

Issue 9. Changes to Various Definitions

Description

The NRC is contemplating changes to various definitions in Part 71 to provide internal consistency and improve correlation with ST-1. 10 CFR 71.4 includes defined terms used throughout Part 71. These terms require clear definition so that they can be used to accurately communicate requirements to licensees. The NRC would add the following definitions from ST-1: (1) Confinement system (paragraph 209), (2) Criticality safety index (paragraph 218; reference issue 5), (3) Low dispersible radioactive material (paragraph 225; reference issue 6), and (4) Quality assurance (paragraph 232). Additionally, the NRC would propose to revise the definition of “package” in 10 CFR 71.4 to be consistent with ST-1. For reference, the ST-1 definitions are contained in Appendix A and provided below.

Para. 209. “Confinement System shall mean the assembly of fissile material and packaging components specified by the designer and agreed to by the competent authority as intended to preserve criticality safety.”

Para. 218. “Criticality safety index (CSI) assigned to a package, overpack or freight container containing fissile material shall mean a number which is used to provide control over the accumulation of packages, overpacks or freight containers containing material.”

Para. 225. “Low dispersible radioactive material shall mean either a solid radioactive material or a solid radioactive material in a sealed capsule, that has limited dispersibility and is not in powdered form.”

Para. 232. “Quality assurance shall mean a systematic programme of controls and inspections applied by an organization or body involved in the transport of radioactive material which is aimed at providing adequate confidence that the standard of safety prescribed in these Regulations is achieved in practice.”

Factors for Consideration

- Do the definitions conflict with existing programs, or introduce other issues or concerns?
- Are there other definitions of terms that are recommended for incorporation in Part 71?

Issue 10. Crush Test for Fissile Material Package Design

Description

Under requirements for packages containing fissile material, ST-1 682(b) requires tests specified in paragraphs 719-724 followed by whichever of the following is the more limiting: the drop test onto a bar as identified in paragraph 727(b) and, either the crush test listed in paragraph 727(c) for packages having a mass not greater than 500 kg and an overall density not greater than 1000 kg/m\3\ based on external dimensions, or the nine meter drop test listed in paragraph 727(a) for all other packages; or the water immersion test of paragraph 729.

SS No.6 and Part 71 presently require the crush test for fissile material packages having a mass not greater than 500 kg and an overall density not greater than 1000 kg/m^3 based on external dimensions, and radioactive contents greater than 1000 A_2 not as special form radioactive material. Under ST-1, the crush test is no longer limited to fissile material packages containing an activity greater than 1000 A_2 because ST-1 has extended the crush test requirement to include fissile material package designs regardless of the activity of the contents. This was done in recognition that the crush environment was a potential accident force that should be protected against for both radiological safety purposes (packages containing more than 1000 A_2 in normal form) and criticality safety purposes (fissile material package designs).

To be consistent with ST-1, the NRC would have to revise 10 CFR Part 71 wording to recognize removal of the 1000 A_2 activity limit with respect to the crush test requirement for fissile material package designs. However, full compliance with ST-1 requirements for fissile material packages would also require changes to the hypothetical accident conditions test sequencing of 10 CFR 71.73 and would require performance of the nine-meter free drop test or the crush test, but not both as presently required by Sec. 71.73.

Factors for Consideration

- How should the differences in the test sequencing and required tests be addressed? Would the test sequencing requirements be applied to Type B packages as well?
- What would be the impact on availability of packages and shipping costs due to elimination of the 1000 A_2 activity limit for fissile material packages having a mass not greater than 500 kg and an overall density not greater than 1000 kg/m^3 based on external dimensions?
- If Part 71 is changed to only eliminate the 1000 A_2 activity limit for fissile material packages, but all other tests and the testing sequence remains unchanged, what implications would this have for US origin packages for export?

Issue 11. Fissile Material Package Design for Transport by Aircraft

Issue Description

For shipment of fissile material by air, ST-1 requires that packages with quantities greater than excepted amounts (that would include all the NRC certified packages) require an additional criticality evaluation. Specifically, the requirements are:

Para 680(a): Packages must remain subcritical, assuming 20 centimeters water reflection but not inleakage (i.e., moderation) when subjected to the tests for Type C packages (see Issue 6). The specification of no water ingress is given as the objective of this requirement is protection from criticality events resulting from mechanical or physical rearrangement of the geometry of the package (i.e., fast criticality).

Para 680(b) This provision states that if a package takes credit for "special features," this package can only be presented for air transport if it is shown that these features remain effective even under the Type C test conditions followed by a water immersion test. "Special features" are specified in ST-1 Para 677, and include features that provide moderator exclusion.

The application of the paragraph 680 requirement to fissile-by-air packages is in addition to the normal condition tests (and possibly accident tests) that the package already must meet. Thus:

- A Type IF or AF package by air must: 1) Withstand incident-free conditions of transport with respect to release, shielding, and maintaining subcriticality (single package and array of packages), (2) withstand accident condition tests with respect to maintaining subcriticality (single package and array of packages), and (3) comply with para 680 with respect to maintaining subcriticality (single package).
- A Type BF package by air must: (1) Withstand incident-free conditions of transport and Type B tests with respect to release, shielding, and maintaining subcriticality (single package and array of packages); and (2) comply with para 680 with respect to maintaining subcriticality (single package).
- A Type C fissile material package must withstand: incident-free conditions of transport (single package and array of packages), Type B tests (single package and array of packages), and Type C tests (single package) with respect to release, shielding, and maintaining subcriticality.

Factors for Consideration

- Certain factors need to be considered in determining the practical impacts of domestic adoption of ST-1 paragraph 680. First, all uranium can be shipped in non-Type C package (IF, AF) due to its A_1 and A_2 values. The paragraph 680(a) requirements appear to be readily satisfied by low-enriched uranium, because low enriched uranium (less than approximately 5% enrichment) would typically require moderation (e.g., by water) to achieve nuclear criticality, but the test specifies no water ingress. Secondly, there are statutory restrictions on air transport of plutonium in the U.S. Finally, packaging for air transportation may follow International Civil Aviation Organization Technical Instructions that are also being revised for compatibility with ST-1.

Issue 12: Special Package Approvals

Description

The transport of large objects that are too large for certified packagings and cannot satisfy the packaging requirements was not considered in the development of Part 71. However, as decommissioning activities increase, the need to transport large objects is rising. For example, in 1997, Portland General Electric Company (PGE) requested approval of the Trojan Reactor Vessel Package (TRVP) (including internals) for transport to the disposal facility operated by US Ecology on the Hanford Nuclear Reservation near Richland, Washington. The TRVP contained approximately 74 petabequerels (2 million curies) in the form of activated metal and 5.7 terabequerels (155 curies) in the form of internal surface contamination; was filled with low-density concrete; and weighed approximately 900 metric tons (1000 tons).

The Commission approved the Trojan shipment under exemptions issued through 10 CFR Part 71.8. Also, the U.S. Department of Transportation's (DOT's) regulations that govern radioactive material shipments do not recognize packages approved via NRC exemption, so DOT also had to consider and issue an exemption for the Trojan shipment.

Because it is the Commission's policy to avoid the use of exemptions for recurring licensing actions, the NRC staff is considering adding regulatory provisions to Part 71 to address special package approvals. If adopted, these provisions would provide a mechanism for review of special packages under the regulations without the need for exemptions.

Factors for Consideration

- Should Part 71 be revised to address reactor vessels specifically or to address large objects in general?
- Should NRC consider adopting an analogue of IAEA's special arrangement provision modified to address packaging?
- What (additional) determinations should be included in an application for a special package approval?
- Should the risk-informed basis used specifically for the Trojan approval be adopted for other special package approvals?

Issue 13. Expansion of Part 71 Quality Assurance Requirements to Holders of, and Applicants for, a Certificate of Compliance

Description

The NRC has observed problems with the performance of 10 CFR Part 72 Certificate of Compliance (CoC) holders in implementing the Part 72 quality assurance (QA) requirements. Problems have occurred in design, design control, fabrication, and corrective action areas. Although CoCs are legally binding documents, certificate holders or applicants for a CoC and their contractors and subcontractors have not clearly been brought within the scope of Part 72 requirements. Therefore, because the terms “certificate holder” and “applicant for a certificate of compliance” do not appear in the Part 72, Subpart G regulations, the NRC has not had a clear basis to cite these persons for violations of Part 72 requirements in the same way it treats licensees.

The NRC Enforcement Policy[1] and its implementing program were established to support the NRC's overall safety mission in protecting public health and safety and the environment. Consistent with this purpose, enforcement actions are used as a deterrent to emphasize the importance of compliance with requirements and to encourage prompt identification and comprehensive correction of the violations. Enforcement sanctions consist of Notices of Violation (NOVs), civil penalties, and orders of various types. In addition to formal enforcement actions, the NRC also uses related administrative actions such as Notices of Nonconformance (NONs), Confirmatory Action Letters, and Demands for Information to supplement its enforcement program. The NRC expects licensees, certificate holders, and applicants for a CoC to adhere to any obligations and commitments that result from these actions and will not hesitate to issue appropriate orders to ensure that these obligations and commitments are met. The nature and extent of the enforcement action are intended to reflect the seriousness of the violation involved. An NOV is a written notice setting forth one or more violations of a legally binding requirement.

However, when the NRC has identified a failure to comply with Part 72 QA requirements by certificate holders or applicants for a CoC, it has issued an NON rather than an NOV. Although an NON and an NOV appear to be similar, the Commission prefers the issuance of an NOV because: (1) The issuance of an NOV effectively conveys to both the person violating the requirement and the public that a violation of a legally binding requirement has occurred; (2) the use of graduated severity levels associated with an NOV allows the NRC to effectively convey to both the person violating the requirement and the public a clearer perspective on the safety and regulatory significance of the violation; and (3) violation of a regulation reflects the NRC's conclusion that potential risk to public health and safety could exist. Therefore, the NRC believed that limiting the available enforcement sanctions to administrative actions was insufficient to address the performance problems observed in industry.

In response to this problem, the NRC staff submitted a rulemaking plan to revise Part 72 to the Commission in SECY-97-214.[2] In a Staff Requirements Memorandum (SRM) to SECY-97-214, the Commission approved the staff's rulemaking plan and directed the staff to also consider whether conforming changes to the quality assurance (QA) regulations in Part 71 would be necessary, because of dual purpose cask designs. Dual purpose cask designs are intended for both the storage of spent fuel under Part 72 and the transportation of spent fuel under Part 71. In a memorandum from the EDO to the Commission, dated December 3, 1997, the NRC staff indicated that expansion of the Part 71 QA provisions to include certificate holders and applicants for a Certificate of Compliance (CoC) would be made as part of the rulemaking to conform Part 71 to IAEA standard ST-1.

The Commission recently issued a final rule expanding QA regulations in Part 72, Subpart G, to specifically include certificate holders and applicants for a CoC. Consequently, the NRC is now considering similarly expanding the QA regulations in Part 71, Subpart H, to specifically include certificate holders and applicants for a CoC. The NRC believes that this change is necessary to ensure consistency between the QA provisions of Parts 71 and 72, particularly in light of NRC approval of dual purpose cask designs. As with the Part 72 final rule, this issue

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[1] NUREG-1600, “General Statement of Policy and Procedures for NRC Enforcement Actions,” May 2000.

[2] SECY-97-214, “Changes to 10 CFR Part 72, Expand Applicability to Include Certificate Holders and Applicants and Their Contractors and Subcontractors,” dated September 24, 1997. This rulemaking plan expanded the applicability of the QA provision of Part 72, Subpart G, to specifically include Part 72 certificate holders and applicants for a Certificate of Compliance.

would provide explicit notice to certificate holders and applicants for a CoC of their QA responsibilities; and would provide the NRC staff with additional enforcement sanction--should violations of the Part 71 QA requirements occur.

Factors for Consideration

- Should consistency be maintained between the QA provisions of Parts 71 and 72, in light of the existence of dual purpose cask designs?

Issue 14. Adoption of ASME Code

Description

The NRC staff proposes that the ASME (American Society of Mechanical Engineers) Code, Section III, Division 3, be incorporated by reference in 10 CFR Part 71 via rulemaking. This rule will ensure implementation of the ASME Code in cask fabrication, including all QA aspects of the code, such as the presence of an authorized nuclear inspector (ANI) during the fabrication to ensure that the code requirements are met, and stamping of components after fabrication is complete. This approach would be similar to how the ASME Code is endorsed for power reactors under 10 CFR 50.55(a) and would make the fabrication process for transportation cask containments commensurate with that used for nuclear power plant components.

NRC inspections of vendors'/fabricators' shops (for fabrication of spent fuel storage canisters and transportation casks) have identified, over the past several years, quality control (QC) and quality assurance (QA) problems in these fabricated systems. A major reason for these problems is that these fabricators/vendors do not fully use a code for QA in the fabrication process of these systems. These QA problems have in some instances continued in spite of repeated adverse NRC and licensee findings.

The NRC staff intends to incorporate two recent developments. First, ASME issued a consensus code in May 1997 entitled: “Containment Systems and Transport Packages for Spent Fuel and High Level Radioactive Waste,” ASME B&PV Code Section III, Division 3, that would require stamping of components constructed to it (i.e., the transportation cask's containment). Second, Public Law 104-113 “National Technology Transfer and Advancement Act” was enacted in 1996 to require that Federal agencies use consensus standards (e.g., the ASME B&PV Code), except when there are justified reasons for not doing so. These two developments support efforts to initiate rulemaking in this area.

Factors for Consideration

- Can other regulatory vehicles for NRC endorsement of Code be used or should this only be done by rulemaking?
- Are there other voluntary consensus standards that should be considered in addition to, or in lieu of, ASME code?

Issue 15. Adoption of Changes, Tests, and Experiments Authority

Description

The Commission recently approved a final rule to expand the provisions of 10 CFR 72.48, “Changes, Tests, and Experiments,” to include Part 72 certificate holders (October 4, 1999; 64 FR 53582). 10 CFR Part 72 Certificate holders are allowed to make changes to a spent fuel storage cask design or conduct tests and experiments, without prior NRC review and approval, if certain requirements are met. However, Part 71 contains no similar provisions to permit a certificate holder to change the design of a Part 71 transportation package. The NRC has issued Certificates of Compliance (CoC) under Parts 71 and 72 for dual purpose casks [packages] (i.e., containers intended for both the storage and transportation of spent fuel). This has created the situation where a 10 CFR Part 72 certificate holder is authorized to change a storage design feature of a dual-purpose storage/transportation cask without obtaining NRC prior approval; however, the 10 CFR Part 71 certificate holder is not authorized to modify transportation package design without obtaining NRC prior approval, even when the same physical component and change is involved.

In SECY-99-130[3] and SECY-99-054.[4] The staff indicated that comments had been received on the proposed rule that requested that authority similar to 10 CFR 72.48 be created in Part 71, particularly with respect to dual purpose casks. Staff indicated that this issue would be addressed in the subsequent rulemaking to conform Part 71 with IAEA standard ST-1. The Commission adopted the staff's recommendations in a Staff Requirements Memorandum (SRM) dated June 22, 1999.

In SECY-99-054 staff recommended that a similar authority to 10 CFR 72.48 be created for spent fuel transportation packages intended for domestic use only. Staff also recommended that this authority be limited to Part 50 and 72 licensees shipping spent fuel and the Part 71 certificate holder. Furthermore, other supporting changes to Part 71 would be required to ensure consistency with the process contained in 10 CFR 72.48. These changes would include using common terminology such as “changes to the cask design, as described in the final safety analysis report” (FSAR) and a process for requesting amendments to a CoC. Requirements for periodically updating a transportation package FSAR would also be required to ensure an accurate “licensing” basis is available for evaluating future proposed changes, and requirements for package users to have a copy of the FSAR, and the updated FSAR.

The current IAEA standard ST-1 does not contain any equivalent provisions for changing a transportation package's design, without prior review by the competent authority.

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[3] SECY-99-130, “Final Rule--Revisions to Requirements of 10 CFR Parts 50 and 72 Concerning Changes, Tests, and Experiments,” dated May 12, 1999.

[4] SECY-99-054, “Plans for Final Rule--Revisions to Requirements of 10 CFR Parts 50, 52, and 72 Concerning Changes, Tests, and Experiments,” dated February 22, 1999.

Factors for Consideration

- Should this change authority apply to spent fuel packages involved in domestic commerce only?
- Should this change authority be expanded to include all types of transportation packages, licensees, or users?
- Should the change authority apply to all domestic transportation packages?
- Should the change authority apply to dual purpose spent fuel packages?

Issue 16. Fissile Material Exemptions and General License Provisions

Discussion

The NRC published an emergency final rule on February 10, 1997 (62 FR 5907), amending Part 71 regulations that deal with shipments of exempt quantities of fissile material and shipments of fissile material under a general license. An NRC licensee had identified that a shipment of waste material (beryllium oxide containing a low concentration of high-enriched uranium) that met the fissile exemption provisions of 10 CFR 71.53 had the potential for an accidental criticality in certain specific circumstances. Packages shipped under the provisions of 10 CFR 71.53 were considered inherently safe for criticality-safety purposes. These regulations assumed that only ordinary water (H_2O) could be present as a moderating material. The regulations did not contemplate the presence of special moderating materials (e.g., beryllium, graphite, or deuterium). Because of this criticality safety issue, the NRC published a rule that was immediately effective with no opportunity for pre-promulgation public comment. The NRC did solicit comments after the rule was effective. All public comments supported the need for the emergency final rule when the shipments contained special moderators (moderators other than water); however, the commenters stated that the rule had gone too far for water moderated shipments, that it was excessively restrictive and costly to licensees, and that further rulemaking was necessary.

Based on these comments, NRC staff contracted with Oak Ridge National Laboratory (ORNL) to thoroughly review fissile material exemptions and general license provisions. ORNL performed computer model calculations of k_{eff} (k-effective) for various combinations of fissile material and moderating material--including beryllium, carbon, deuterium, silicon-dioxide, and water--to verify the accuracy of minimum critical mass values. These minimum critical mass values were then applied to the regulatory structure contained in Part 71, and revised mass limits for both the general license and exemption provisions to Part 71 were determined. Also, ORNL researched the historical bases for the fissile material exemption and general license regulations in Part 71 and discussed the impact of the emergency final rule's restrictions on NRC licensees. The ORNL study was issued as NUREG/CR-5342 in July 1998 (available via the following NRC web site: http://www.nrc.gov/NRC/NUREGS/CR5342/index.html). The ORNL study confirmed that the emergency rule was needed to provide safe transportation of packages with special moderators that are shipped under the general license and fissile material exemptions, but may be excessive for water-moderated shipments.

NUREG/CR-5342 identified 16 recommended actions for additional rulemaking. Additionally, the Commission's SRM on SECY-96-268 approving the emergency final rule directed the staff to issue guidance for instances where fissile materials may be mixed in the same shipping container with different moderators. The staff indicated that this issue would be addressed in a forthcoming rulemaking (memorandum from the EDO to the Commission, dated September 8, 1998). On October 27, 1999, the NRC published Federal Register Notice 64 FR

57769 responding to public comments on the emergency final rule, and also requesting information on the cost impact of the final rule from the public, industry, and the DOE, because the NRC staff had not been successful in obtaining this information. The requirements for the fissile material general licenses are provided in 10 CFR 71.18, 71.20, 71.22, and 71.24, and the fissile material exemptions are provided in 71.53.

IAEA standard ST-1 contains language on fissile exemptions and restrictions on the use of special moderators. However, ST-1 does not presently contain provisions on general licenses for shipment of fissile material; previous version did contain general license conditions.

Factors for Consideration

- Should all, or only some, of the 16 sub-issues (i.e., the recommendations contained in NUREG/CR-5342) be included in this rulemaking on this issue?
- Should additional issues or alternative approaches on the fissile exemptions or general license provisions be included in this rulemaking?
- Is there available cost data that may help to understand the cost impact of the implemented emergency rule; or help to better understand the possible cost impact of the ORNL recommendations?

Issue 17. Double Containment of Plutonium (PRM-71-12)

Description

The NRC received a Petition for Rulemaking from International Energy Consultants, Inc. (IEC), dated September 25, 1997. The petition was docketed as PRM-71-12 and was published for public comment on February 19, 1998. The comment period was extended to July 31, 1998. The petitioner requested that regulations in 10 CFR 71.63 be eliminated. The petitioner argued that the double containment requirement in 71.63(b) was not consistent with the basis for other packaging standards (i.e., the Q-value system for identifying the A_1 and A_2 values for each nuclide). The petitioner also argued that the use of double containment for shipments of plutonium imposed unnecessary costs (i.e., fabrication of shipping packages and a weight penalty). As an option, the petitioner requested that 71.63 be entirely eliminated.

In 1974, the Atomic Energy Commission (AEC) issued 10 CFR 71.63 which imposed special requirements on the shipment of plutonium in excess of 0.74 terabecquerels (20 curies). These requirements specify that plutonium must be in solid form (71.63(a)) and that packages used to ship plutonium must provide a separate inner containment (i.e., the “double containment” requirement) (71.63(b)). In adopting these requirements, the AEC specifically excluded plutonium in the form of reactor fuel elements, metal or metal alloys, and other plutonium-bearing solids that the Commission determines, on a case-by-case basis, do not require double containment. These regulations have remained essentially unchanged since 1974, except for the addition in 1998 of vitrified high-level waste in sealed canisters to the list of exempt forms of plutonium. Double containment is in addition to Type B packaging standards and is not required for any other nuclides that are listed in Part 71. Additionally, IAEA standard ST-1 does not contain a double containment requirement for any nuclide.

The AEC issued this regulation at a time when wide-spread reprocessing of commercial spent fuel was anticipated. The AEC expected increases in the quantities of plutonium to be shipped and the number of shipments of plutonium. In addition, the specific activity of the plutonium was expected to increase with increased burnup, resulting in higher gamma and neutron radiation levels, greater heat generation, and greater pressure generation potential from plutonium nitrate solutions in shipping containers. Because of these expected changes and because of the susceptibility of liquids to leakage, the AEC believed that safety would be significantly enhanced if the basic form for shipments of plutonium were changed from liquid to solid, and if the solid form of plutonium were required to be shipped in a package providing double containment of the contents.

The AEC indicated that "The arguments for requiring a solid form of plutonium for shipment are largely subjective, in that there is no hard evidence on which to base statistical probabilities or to assess quantitatively the incremental increase in safety which is expected."[5] The AEC also indicated that the double containment provision compensates for the fact that the plutonium may not be in a "nonrespirable" form. Notwithstanding these rationales, some of the underlying assumptions for this rule were altered in 1979 when the U.S. government decided that reprocessing of civilian spent fuel and reuse of plutonium was not desirable. Consequently, the expected plutonium reprocessing economy and wide-spread shipments never materialized.

With respect to PRM-71-12, eight public comments were received on the petition; of those, three supported the petition and five opposed the petition. The supporting comments essentially stated that the IAEA's Q-System accurately reflects the dangers of nuclides, including plutonium, and that elimination of 10 CFR 71.63(a) and (b) would make the regulations more performance based, reduce costs and personnel exposures, and be consistent with the IAEA standards.

The five opposing comments essentially stated that plutonium is very dangerous, especially in liquid form, and therefore additional regulatory requirements are warranted, that existing regulations are not overly burdensome, especially in light of the total expected transportation cost, that TRUPACT-II package meets 71.63(b) requirement, that a commenter (i.e., the Western Governors Association) has worked for over 10 years to ensure a safe transportation system for WIPP, including educating the public about the TRUPACT-II package, and that any change now would erode public confidence and be detrimental to the entire transportation system for WIPP shipments, and that additional personnel exposure due to double containment is insignificant.

Factors for Consideration

- Should NRC change any of the special requirements for the transportation of plutonium?
- Should the double containment requirement in 71.63(b) be eliminated?
- Should both the solid form and the double containment requirements of 71.63(a) and (b) be eliminated?
- Is consistency with IAEA standard ST-1 important on this issue?

Issue 18. Contamination Limits as Applied to Spent Fuel and High Level Waste (HLW) Packages

Description

As part of the NRC's upcoming public meetings on proposed changes to 10 CFR Part 71, the Commission will consider the issue of removable package contamination limits for transportation (i.e., radioactive material that can be removed from the surface of a package

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[5] SECY-R-74-5, dated July 6, 1973.

prior to shipment). This issue involves contamination limits for all transportation packages, including spent fuel and HLW packages, contained in DOT regulations which are based on the international transportation standards for contamination limits. The NRC staff requests public and stakeholder views on whether different contamination limits should be considered for spent fuel and HLW packages, and recommendations for future interactions that NRC has with DOT and IAEA on this issue. NRC staff is aware that the IAEA is starting a review of contamination models and limits, and this review will be conducted over the next few years.

The removable contamination limit of 4 Becquerels per square centimeter (4Bq/cm2) is contained in IAEA Safety Series 6, in ST-1, in U.S. DOT regulations (49 CFR 173.443), and by reference to DOT's regulations in NRC's 10 CFR Part 71. The limit applies to the transportation of all packages, regardless of size. Thus, the 4 Bq/cm2 contamination limit applies to shipment of spent fuel and HLW packages, even though the unique aspects of these packages were not explicitly considered in the modeling assumptions used in developing the contamination limit. Specifically, the contamination limit was designed to reduce delivery worker exposure from external contamination on small packages during frequent manual handling of these packages in freight facilities; however, unlike small packages moved by delivery workers, handling of spent fuel and HLW packages is done by cranes and other manipulation equipment, due to the large weights involved, and does not involve extensive personnel contact, thereby reducing worker exposure from external package contamination.

Irrespective of remote handling, workers must obtain contamination readings on a spent fuel or HLW package's external surfaces to ensure compliance with the 4 Bq/cm2 limit prior to release for shipment. Due to the large surface areas involved in the contamination checks, and the prolonged time that workers are in the vicinity of a loaded package while performing these checks, they receive exposure from radiation emanating through the package walls. Further, should the contamination checks reveal contamination above 4 Bq/cm2, then additional worker exposure occurs during decontamination activities and subsequent checks of contamination levels to achieve the 4 Bq/cm2 limit. It should be noted that if the contamination limit for spent fuel and HLW packages was changed, workers would still be required to check the packages for contamination (under the changed limit) and thus receive exposure while performing this activity and any required decontamination activities.

Factors for Consideration

- Should the 4 Bq/cm2 limit continue to apply to spent fuel and HLW packages or should an alternative limit be developed? Is there an alternate contamination limit or alternative approach that will result in lowered exposure to workers, yet ensure that the rail and truck workers as well as the public are adequately protected from external package contamination?
- If alternative contamination limits are established for spent fuel and HLW packages, is there any concern with the possible resulting difference in US domestic regulations and international standards?

Appendix A--Paragraphs Referenced from IAEA ST-1

Appendix A contains the full text of specific paragraphs from ST-1 referenced in the eleven IAEA-compatibility issues. Paragraphs are listed numerically in ascending order, with the corresponding issue identified in bold text at the end of the reference.

107. The Regulations do not apply to:

(e) natural material and ores containing naturally occurring radionuclides which are not intended to be processed for use of these radionuclides provided the activity concentration of the material does not exceed 10 times the values specified in paras 401-406. (Issue 2)

209. Confinement system shall mean the assembly of fissile material and packaging components specified by the designer and agreed to by the competent authority as intended to preserve criticality safety. (Issue 9)

218. Criticality safety index (CSI) assigned to a package, overpack or freight container containing fissile material shall mean a number which is used to provide control over the accumulation of packages, overpacks or freight containers containing fissile material. (Issue 9)

225. Low dispersible radioactive material shall mean either a solid radioactive material or a solid radioactive material in a sealed capsule, that has limited dispersibility and is not in powder form. (Issue 9)

230. Package shall mean the packaging with its radioactive contents as presented for transport. The types of packages covered by these Regulations, which are subject to the activity limits and material restrictions of Section IV and meet the corresponding requirements, are:

(a) Excepted package;

(b) Industrial package Type 1 (Type IP-1);

(c) Industrial package Type 2 (Type IP-2);

(d) Industrial package Type 3 (Type IP-3);

(e) Type A package;

(f) Type B(U) package;

(g) Type B(M) package;

(h) Type C package.

Packages containing fissile material or uranium hexafluoride are subject to additional requirements. (Issue 6)

232. Quality assurance shall mean a systematic programme of controls and inspections applied by any organization or body involved in the transport of radioactive material which is aimed at providing adequate confidence that the standard of safety prescribed in these Regulations is achieved in practice. (Issue 9)

401. The following basic values for individual radionuclides are given in Table I:

(a) A_1 and A_2 in TBq;

(b) activity concentration for exempt material in Bq/g; and

(c) activity limits for exempt consignments in Bq. (Issue 2)

402. For individual radionuclides which are not listed in Table I the determination of the basic radionuclide values referred to in para. 401 shall require competent authority approval or, for international transport, multilateral approval. Where the chemical form of each radionuclide is known, it is permissible to use the A_2 value related to its solubility class as recommended by the International Commission on Radiological Protection, if the chemical forms under both normal and accident conditions of transport are taken into consideration. Alternatively, the radionuclide values in Table II may be used without obtaining competent authority approval. (Issue 2)

403. In the calculations of A_1 and A_2 for a radionuclide not in Table I, a single radioactive decay chain in which the radionuclides are present in their naturally occurring proportions, and in which no daughter nuclide has a half-life either longer than 10 days or longer than that of the parent nuclide, shall be considered as a single radionuclide; and the activity to be taken into account and the A_1 or A_2 value to be applied shall be those corresponding to the parent nuclide of that chain. In the case of radioactive decay chains in which any daughter nuclide has a half-life either longer than 10 days or greater than that of the parent nuclide, the parent and such daughter nuclides shall be considered as mixtures of different nuclides. (Issue 2)

404. For mixtures of radionuclides, the determination of the basic radionuclide values referred to in para. 401 may be determined as follows:

$$X_m = \frac{1}{\sum_i \frac{f(i)}{X(i)}}$$

Table I. BASIC RADIONUCLIDE VALUES

Radionuclide (atomic number)	A_1	A_2	Activity concentration for exempt material	Activity limit for an exempt consignment
	(TBq)	(TBq)	(Bq/g)	(Bq)
Actinium (89)				
Ac-225 (a)	8×10^{-1}	6×10^{-3}	1×10^{1}	1×10^{4}
Ac-227 (a)	9×10^{-1}	9×10^{-5}	1×10^{-1}	1×10^{3}
Ac-228	6×10^{-1}	5×10^{-1}	1×10^{1}	1×10^{6}
Silver (47)				
Ag-105	2×10^{0}	2×10^{0}	1×10^{2}	1×10^{6}
Ag-108m (a)	7×10^{-1}	7×10^{-1}	1×10^{1} (b)	1×10^{6} (b)
Ag-110m (a)	4×10^{-1}	4×10^{-1}	1×10^{1}	1×10^{6}
Ag-111	2×10^{0}	6×10^{-1}	1×10^{3}	1×10^{6}
Aluminium (13)				
Al-26	1×10^{-1}	1×10^{-1}	1×10^{1}	1×10^{5}
Americium (95)				
Am-241	1×10^{1}	1×10^{-3}	1×10^{0}	1×10^{4}
Am-242m (a)	1×10^{1}	1×10^{-3}	1×10^{0} (b)	1×10^{4} (b)
Am-243 (a)	5×10^{0}	1×10^{-3}	1×10^{0} (b)	1×10^{3} (b)
Argon (18)				
Ar-37	4×10^{1}	4×10^{1}	1×10^{6}	1×10^{8}
Ar-39	2×10^{1}	4×10^{1}	1×10^{7}	1×10^{4}
Ar-41	3×10^{-1}	3×10^{-1}	1×10^{2}	1×10^{9}
Arsenic (33)				
As-72	3×10^{-1}	3×10^{-1}	1×10^{1}	1×10^{5}
As-73	4×10^{1}	4×10^{1}	1×10^{3}	1×10^{7}
As-74	1×10^{0}	9×10^{-1}	1×10^{1}	1×10^{6}
As-76	3×10^{-1}	3×10^{-1}	1×10^{2}	1×10^{5}
As-77	2×10^{1}	7×10^{-1}	1×10^{3}	1×10^{6}
Astatine (85)				
At-211 (a)	2×10^{1}	5×10^{-1}	1×10^{3}	1×10^{7}
Gold (79)				
Au-193	7×10^{0}	2×10^{0}	1×10^{2}	1×10^{7}

Table I. BASIC RADIONUCLIDE VALUES (continued)

Radionuclide (atomic number)	A_1	A_2	Activity concentration for exempt material	Activity limit for an exempt consignment
	(TBq)	(TBq)	(Bq/g)	(Bq)
Au-194	1×10^{0}	1×10^{0}	1×10^{1}	1×10^{6}
Au-195	1×10^{1}	6×10^{0}	1×10^{2}	1×10^{7}
Au-198	1×10^{0}	6×10^{-1}	1×10^{2}	1×10^{6}
Au-199	1×10^{1}	6×10^{-1}	1×10^{2}	1×10^{6}
Barium (56)				
Ba-131 (a)	2×10^{0}	2×10^{0}	1×10^{2}	1×10^{6}
Ba-133	3×10^{0}	3×10^{0}	1×10^{2}	1×10^{6}
Ba-133m	2×10^{1}	6×10^{-1}	1×10^{2}	1×10^{6}
Ba-140 (a)	5×10^{-1}	3×10^{-1}	1×10^{1} (b)	1×10^{5} (b)
Beryllium (4)				
Be-7	2×10^{1}	2×10^{1}	1×10^{3}	1×10^{7}
Be-10	4×10^{1}	6×10^{-1}	1×10^{4}	1×10^{6}
Bismuth (83)				
Bi-205	7×10^{-1}	7×10^{-1}	1×10^{1}	1×10^{6}
Bi-206	3×10^{-1}	3×10^{-1}	1×10^{1}	1×10^{5}
Bi-207	7×10^{-1}	7×10^{-1}	1×10^{1}	1×10^{6}
Bi-210	1×10^{0}	6×10^{-1}	1×10^{3}	1×10^{6}
Bi-210m (a)	6×10^{-1}	2×10^{-2}	1×10^{1}	1×10^{5}
Bi-212 (a)	7×10^{-1}	6×10^{-1}	1×10^{1} (b)	1×10^{5} (b)
Berkelium (97)				
Bk-247	8×10^{0}	8×10^{-4}	1×10^{0}	1×10^{4}
Bk-249 (a)	4×10^{1}	3×10^{-1}	1×10^{3}	1×10^{6}
Bromine (35)				
Br-76	4×10^{-1}	4×10^{-1}	1×10^{1}	1×10^{5}
Br-77	3×10^{0}	3×10^{0}	1×10^{2}	1×10^{6}
Br-82	4×10^{-1}	4×10^{-1}	1×10^{1}	1×10^{6}
Carbon (6)				
C-11	1×10^{0}	6×10^{-1}	1×10^{1}	1×10^{6}
C-14	4×10^{1}	3×10^{0}	1×10^{4}	1×10^{7}

Table I. BASIC RADIONUCLIDE VALUES (continued)

Radionuclide (atomic number)	A_1	A_2	Activity concentration for exempt material	Activity limit for an exempt consignment
	(TBq)	(TBq)	(Bq/g)	(Bq)
Calcium (20)				
Ca-41	Unlimited	Unlimited	1×10^{5}	1×10^{7}
Ca-45	4×10^{1}	1×10^{0}	1×10^{4}	1×10^{7}
Ca-47 (a)	3×10^{0}	3×10^{-1}	1×10^{1}	1×10^{6}
Cadmium (48)				
Cd-109	3×10^{1}	2×10^{0}	1×10^{4}	1×10^{6}
Cd-113m	4×10^{1}	5×10^{-1}	1×10^{3}	1×10^{6}
Cd-115 (a)	3×10^{0}	4×10^{-1}	1×10^{2}	1×10^{6}
Cd-115m	5×10^{-1}	5×10^{-1}	1×10^{3}	1×10^{6}
Cerium (58)				
Ce-139	7×10^{0}	2×10^{0}	1×10^{2}	1×10^{6}
Ce-141	2×10^{1}	6×10^{-1}	1×10^{2}	1×10^{7}
Ce-143	9×10^{-1}	6×10^{-1}	1×10^{2}	1×10^{6}
Ce-144 (a)	2×10^{-1}	2×10^{-1}	1×10^{2} (b)	1×10^{5} (b)
Californium (98)				
Cf-248	4×10^{1}	6×10^{-3}	1×10^{1}	1×10^{4}
Cf-249	3×10^{0}	8×10^{-4}	1×10^{0}	1×10^{3}
Cf-250	2×10^{1}	2×10^{-3}	1×10^{1}	1×10^{4}
Cf-251	7×10^{0}	7×10^{-4}	1×10^{0}	1×10^{3}
Cf-252	5×10^{-2}	3×10^{-3}	1×10^{1}	1×10^{4}
Cf-253 (a)	4×10^{1}	4×10^{-2}	1×10^{2}	1×10^{5}
Cf-254	1×10^{-3}	1×10^{-3}	1×10^{0}	1×10^{3}
Chlorine (17)				
Cl-36	1×10^{1}	6×10^{-1}	1×10^{4}	1×10^{6}
Cl-38	2×10^{-1}	2×10^{-1}	1×10^{1}	1×10^{5}
Curium (96)				
Cm-240	4×10^{1}	2×10^{-2}	1×10^{2}	1×10^{5}
Cm-241	2×10^{0}	1×10^{0}	1×10^{2}	1×10^{6}
Cm-242	4×10^{1}	1×10^{-2}	1×10^{2}	1×10^{5}

Table I. BASIC RADIONUCLIDE VALUES (continued)

Radionuclide (atomic number)	A_1	A_2	Activity concentration for exempt material	Activity limit for an exempt consignment
	(TBq)	(TBq)	(Bq/g)	(Bq)
Cm-243	9×10^{0}	1×10^{-3}	1×10^{0}	1×10^{4}
Cm-244	2×10^{1}	2×10^{-3}	1×10^{1}	1×10^{4}
Cm-245	9×10^{0}	9×10^{-4}	1×10^{0}	1×10^{3}
Cm-246	9×10^{0}	9×10^{-4}	1×10^{0}	1×10^{3}
Cm-247 (a)	3×10^{0}	1×10^{-3}	1×10^{0}	1×10^{4}
Cm-248	2×10^{-2}	3×10^{-4}	1×10^{0}	1×10^{3}
Cobalt (27)				
Co-55	5×10^{-1}	5×10^{-1}	1×10^{1}	1×10^{6}
Co-56	3×10^{-1}	3×10^{-1}	1×10^{1}	1×10^{5}
Co-57	1×10^{1}	1×10^{1}	1×10^{2}	1×10^{6}
Co-58	1×10^{0}	1×10^{0}	1×10^{1}	1×10^{6}
Co-58m	4×10^{1}	4×10^{1}	1×10^{4}	1×10^{7}
Co-60	4×10^{-1}	4×10^{-1}	1×10^{1}	1×10^{5}
Chromium (24)				
Cr-51	3×10^{1}	3×10^{1}	1×10^{3}	1×10^{7}
Caesium (55)				
Cs-129	4×10^{0}	4×10^{0}	1×10^{2}	1×10^{5}
Cs-131	3×10^{1}	3×10^{1}	1×10^{3}	1×10^{6}
Cs-132	1×10^{0}	1×10^{0}	1×10^{1}	1×10^{5}
Cs-134	7×10^{-1}	7×10^{-1}	1×10^{1}	1×10^{4}
Cs-134m	4×10^{1}	6×10^{-1}	1×10^{3}	1×10^{5}
Cs-135	4×10^{1}	1×10^{0}	1×10^{4}	1×10^{7}
Cs-136	5×10^{-1}	5×10^{-1}	1×10^{1}	1×10^{5}
Cs-137 (a)	2×10^{0}	6×10^{-1}	1×10^{1} (b)	1×10^{4} (b)
Copper (29)				
Cu-64	6×10^{0}	1×10^{0}	1×10^{2}	1×10^{6}
Cu-67	1×10^{1}	7×10^{-1}	1×10^{2}	1×10^{6}
Dysprosium (66)				
Dy-159	2×10^{1}	2×10^{1}	1×10^{3}	1×10^{7}

Table I. BASIC RADIONUCLIDE VALUES (continued)

Radionuclide (atomic number)	A_1	A_2	Activity concentration for exempt material	Activity limit for an exempt consignment
	(TBq)	(TBq)	(Bq/g)	(Bq)
Dy-165	9×10^{-1}	6×10^{-1}	1×10^{3}	1×10^{6}
Dy-166 (a)	9×10^{-1}	3×10^{-1}	1×10^{3}	1×10^{6}
Erbium (68)				
Er-169	4×10^{1}	1×10^{0}	1×10^{4}	1×10^{7}
Er-171	8×10^{-1}	5×10^{-1}	1×10^{2}	1×10^{6}
Europium (63)				
Eu-147	2×10^{0}	2×10^{0}	1×10^{2}	1×10^{6}
Eu-148	5×10^{-1}	5×10^{-1}	1×10^{1}	1×10^{6}
Eu-149	2×10^{1}	2×10^{1}	1×10^{2}	1×10^{7}
Eu-150(short lived)	2×10^{0}	7×10^{-1}	1×10^{3}	1×10^{6}
Eu-150(long lived)	7×10^{-1}	7×10^{-1}	1×10^{1}	1×10^{6}
Eu-152	1×10^{0}	1×10^{0}	1×10^{1}	1×10^{6}
Eu-152m	8×10^{-1}	8×10^{-1}	1×10^{2}	1×10^{6}
Eu-154	9×10^{-1}	6×10^{-1}	1×10^{1}	1×10^{6}
Eu-155	2×10^{1}	3×10^{0}	1×10^{2}	1×10^{7}
Eu-156	7×10^{-1}	7×10^{-1}	1×10^{1}	1×10^{6}
Fluorine (9)				
F-18	1×10^{0}	6×10^{-1}	1×10^{1}	1×10^{6}
Iron (26)				
Fe-52 (a)	3×10^{-1}	3×10^{-1}	1×10^{1}	1×10^{6}
Fe-55	4×10^{1}	4×10^{1}	1×10^{4}	1×10^{6}
Fe-59	9×10^{-1}	9×10^{-1}	1×10^{1}	1×10^{6}
Fe-60 (a)	4×10^{1}	2×10^{-1}	1×10^{2}	1×10^{5}
Gallium (31)				
Ga-67	7×10^{0}	3×10^{0}	1×10^{2}	1×10^{6}
Ga-68	5×10^{-1}	5×10^{-1}	1×10^{1}	1×10^{5}
Ga-72	4×10^{-1}	4×10^{-1}	1×10^{1}	1×10^{5}
Gadolinium (64)				
Gd-146 (a)	5×10^{-1}	5×10^{-1}	1×10^{1}	1×10^{6}

Table I. BASIC RADIONUCLIDE VALUES (continued)

Radionuclide (atomic number)	A_1	A_2	Activity concentration for exempt material	Activity limit for an exempt consignment
	(TBq)	(TBq)	(Bq/g)	(Bq)
Gd-148	2×10^{1}	2×10^{-3}	1×10^{1}	1×10^{4}
Gd-153	1×10^{1}	9×10^{0}	1×10^{2}	1×10^{7}
Gd-159	3×10^{0}	6×10^{-1}	1×10^{3}	1×10^{6}
Germanium (32)				
Ge-68 (a)	5×10^{-1}	5×10^{-1}	1×10^{1}	1×10^{5}
Ge-71	4×10^{1}	4×10^{1}	1×10^{4}	1×10^{8}
Ge-77	3×10^{-1}	3×10^{-1}	1×10^{1}	1×10^{5}
Hafnium (72)				
Hf-172 (a)	6×10^{-1}	6×10^{-1}	1×10^{1}	1×10^{6}
Hf-175	3×10^{0}	3×10^{0}	1×10^{2}	1×10^{6}
Hf-181	2×10^{0}	5×10^{-1}	1×10^{1}	1×10^{6}
Hf-182	Unlimited	Unlimited	1×10^{2}	1×10^{6}
Mercury (80)				
Hg-194 (a)	1×10^{0}	1×10^{0}	1×10^{1}	1×10^{6}
Hg-195m (a)	3×10^{0}	7×10^{-1}	1×10^{2}	1×10^{6}
Hg-197	2×10^{1}	1×10^{1}	1×10^{2}	1×10^{7}
Hg-197m	1×10^{1}	4×10^{-1}	1×10^{2}	1×10^{6}
Hg-203	5×10^{0}	1×10^{0}	1×10^{2}	1×10^{5}
Holmium (67)				
Ho-166	4×10^{-1}	4×10^{-1}	1×10^{3}	1×10^{5}
Ho-166m	6×10^{-1}	5×10^{-1}	1×10^{1}	1×10^{6}
Iodine (53)				
I-123	6×10^{0}	3×10^{0}	1×10^{2}	1×10^{7}
I-124	1×10^{0}	1×10^{0}	1×10^{1}	1×10^{6}
I-125	2×10^{1}	3×10^{0}	1×10^{3}	1×10^{6}
I-126	2×10^{0}	1×10^{0}	1×10^{2}	1×10^{6}
I-129	Unlimited	Unlimited	1×10^{2}	1×10^{5}
I-131	3×10^{0}	7×10^{-1}	1×10^{2}	1×10^{6}
I-132	4×10^{-1}	4×10^{-1}	1×10^{1}	1×10^{5}

Table I. BASIC RADIONUCLIDE VALUES (continued)

Radionuclide (atomic number)	A_1	A_2	Activity concentration for exempt material	Activity limit for an exempt consignment
	(TBq)	(TBq)	(Bq/g)	(Bq)
I-133	7×10^{-1}	6×10^{-1}	1×10^{1}	1×10^{6}
I-134	3×10^{-1}	3×10^{-1}	1×10^{1}	1×10^{5}
I-135 (a)	6×10^{-1}	6×10^{-1}	1×10^{1}	1×10^{6}
Indium (49)				
In-111	3×10^{0}	3×10^{0}	1×10^{2}	1×10^{6}
In-113m	4×10^{0}	2×10^{0}	1×10^{2}	1×10^{6}
In-114m (a)	1×10^{1}	5×10^{-1}	1×10^{2}	1×10^{6}
In-115m	7×10^{0}	1×10^{0}	1×10^{2}	1×10^{6}
Iridium (77)				
Ir-189 (a)	1×10^{1}	1×10^{1}	1×10^{2}	1×10^{7}
Ir-190	7×10^{-1}	7×10^{-1}	1×10^{1}	1×10^{6}
Ir-192	1×10^{0}(c)	6×10^{-1}	1×10^{1}	1×10^{4}
Ir-194	3×10^{-1}	3×10^{-1}	1×10^{2}	1×10^{5}
Potassium (19)				
K-40	9×10^{-1}	9×10^{-1}	1×10^{2}	1×10^{6}
K-42	2×10^{-1}	2×10^{-1}	1×10^{2}	1×10^{6}
K-43	7×10^{-1}	6×10^{-1}	1×10^{1}	1×10^{6}
Krypton (36)				
Kr-81	4×10^{1}	4×10^{1}	1×10^{4}	1×10^{7}
Kr-85	1×10^{1}	1×10^{1}	1×10^{5}	1×10^{4}
Kr-85m	8×10^{0}	3×10^{0}	1×10^{3}	1×10^{10}
Kr-87	2×10^{-1}	2×10^{-1}	1×10^{2}	1×10^{9}
Lanthanum (57)				
La-137	3×10^{1}	6×10^{0}	1×10^{3}	1×10^{7}
La-140	4×10^{-1}	4×10^{-1}	1×10^{1}	1×10^{5}
Lutetium (71)				
Lu-172	6×10^{-1}	6×10^{-1}	1×10^{1}	1×10^{6}
Lu-173	8×10^{0}	8×10^{0}	1×10^{2}	1×10^{7}
Lu-174	9×10^{0}	9×10^{0}	1×10^{2}	1×10^{7}

Table I. BASIC RADIONUCLIDE VALUES (continued)

Radionuclide (atomic number)	A_1	A_2	Activity concentration for exempt material	Activity limit for an exempt consignment
	(TBq)	(TBq)	(Bq/g)	(Bq)
Lu-174m	2×10^{1}	1×10^{1}	1×10^{2}	1×10^{7}
Lu-177	3×10^{1}	7×10^{-1}	1×10^{3}	1×10^{7}
Magnesium (12)				
Mg-28 (a)	3×10^{-1}	3×10^{-1}	1×10^{1}	1×10^{5}
Manganese (25)				
Mn-52	3×10^{-1}	3×10^{-1}	1×10^{1}	1×10^{5}
Mn-53	Unlimited	Unlimited	1×10^{4}	1×10^{9}
Mn-54	1×10^{0}	1×10^{0}	1×10^{1}	1×10^{6}
Mn-56	3×10^{-1}	3×10^{-1}	1×10^{1}	1×10^{5}
Molybdenum (42)				
Mo-93	4×10^{1}	2×10^{1}	1×10^{3}	1×10^{8}
Mo-99 (a)	1×10^{0}	6×10^{-1}	1×10^{2}	1×10^{6}
Nitrogen (7)				
N-13	9×10^{-1}	6×10^{-1}	1×10^{2}	1×10^{9}
Sodium (11)				
Na-22	5×10^{-1}	5×10^{-1}	1×10^{1}	1×10^{6}
Na-24	2×10^{-1}	2×10^{-1}	1×10^{1}	1×10^{5}
Niobium (41)				
Nb-93m	4×10^{1}	3×10^{1}	1×10^{4}	1×10^{7}
Nb-94	7×10^{-1}	7×10^{-1}	1×10^{1}	1×10^{6}
Nb-95	1×10^{0}	1×10^{0}	1×10^{1}	1×10^{6}
Nb-97	9×10^{-1}	6×10^{-1}	1×10^{1}	1×10^{6}
Neodymium (60)				
Nd-147	6×10^{0}	6×10^{-1}	1×10^{2}	1×10^{6}
Nd-149	6×10^{-1}	5×10^{-1}	1×10^{2}	1×10^{6}
Nickel (28)				
Ni-59	Unlimited	Unlimited	1×10^{4}	1×10^{8}
Ni-63	4×10^{1}	3×10^{1}	1×10^{5}	1×10^{8}
Ni-65	4×10^{-1}	4×10^{-1}	1×10^{1}	1×10^{6}

Table I. BASIC RADIONUCLIDE VALUES (continued)

Radionuclide (atomic number)	A_1	A_2	Activity concentration for exempt material	Activity limit for an exempt consignment
	(TBq)	(TBq)	(Bq/g)	(Bq)
Neptunium (93)				
Np-235	4×10^{1}	4×10^{1}	1×10^{3}	1×10^{7}
Np-236(short-lived)	2×10^{1}	2×10^{0}	1×10^{3}	1×10^{7}
Np-236(long-lived)	9×10^{0}	2×10^{-2}	1×10^{2}	1×10^{5}
Np-237	2×10^{1}	2×10^{-3}	1×10^{0} (b)	1×10^{3} (b)
Np-239	7×10^{0}	4×10^{-1}	1×10^{2}	1×10^{7}
Osmium (76)				
Os-185	1×10^{0}	1×10^{0}	1×10^{1}	1×10^{6}
Os-191	1×10^{1}	2×10^{0}	1×10^{2}	1×10^{7}
Os-191m	4×10^{1}	3×10^{1}	1×10^{3}	1×10^{7}
Os-193	2×10^{0}	6×10^{-1}	1×10^{2}	1×10^{6}
Os-194 (a)	3×10^{-1}	3×10^{-1}	1×10^{2}	1×10^{5}
Phosphorus (15)				
P-32	5×10^{-1}	5×10^{-1}	1×10^{3}	1×10^{5}
P-33	4×10^{1}	1×10^{0}	1×10^{5}	1×10^{8}
Protactinium (91)				
Pa-230 (a)	2×10^{0}	7×10^{-2}	1×10^{1}	1×10^{6}
Pa-231	4×10^{0}	4×10^{-4}	1×10^{0}	1×10^{3}
Pa-233	5×10^{0}	7×10^{-1}	1×10^{2}	1×10^{7}
Lead (82)				
Pb-201	1×10^{0}	1×10^{0}	1×10^{1}	1×10^{6}
Pb-202	4×10^{1}	2×10^{1}	1×10^{3}	1×10^{6}
Pb-203	4×10^{0}	3×10^{0}	1×10^{2}	1×10^{6}
Pb-205	Unlimited	Unlimited	1×10^{4}	1×10^{7}
Pb-210 (a)	1×10^{0}	5×10^{-2}	1×10^{1} (b)	1×10^{4} (b)
Pb-212 (a)	7×10^{-1}	2×10^{-1}	1×10^{1} (b)	1×10^{5} (b)
Palladium (46)				
Pd-103 (a)	4×10^{1}	4×10^{1}	1×10^{3}	1×10^{8}
Pd-107	Unlimited	Unlimited	1×10^{5}	1×10^{8}

Table I. BASIC RADIONUCLIDE VALUES (continued)

Radionuclide (atomic number)	A_1	A_2	Activity concentration for exempt material	Activity limit for an exempt consignment
	(TBq)	(TBq)	(Bq/g)	(Bq)
Pd-109	2×10^{0}	5×10^{-1}	1×10^{3}	1×10^{6}
Promethium (61)				
Pm-143	3×10^{0}	3×10^{0}	1×10^{2}	1×10^{6}
Pm-144	7×10^{-1}	7×10^{-1}	1×10^{1}	1×10^{6}
Pm-145	3×10^{1}	1×10^{1}	1×10^{3}	1×10^{7}
Pm-147	4×10^{1}	2×10^{0}	1×10^{4}	1×10^{7}
Pm-148m (a)	8×10^{-1}	7×10^{-1}	1×10^{1}	1×10^{6}
Pm-149	2×10^{0}	6×10^{-1}	1×10^{3}	1×10^{6}
Pm-151	2×10^{0}	6×10^{-1}	1×10^{2}	1×10^{6}
Polonium (84)				
Po-210	4×10^{1}	2×10^{-2}	1×10^{1}	1×10^{4}
Praseodymium (59)				
Pr-142	4×10^{-1}	4×10^{-1}	1×10^{2}	1×10^{5}
Pr-143	3×10^{0}	6×10^{-1}	1×10^{4}	1×10^{6}
Platinum (78)				
Pt-188 (a)	1×10^{0}	8×10^{-1}	1×10^{1}	1×10^{6}
Pt-191	4×10^{0}	3×10^{0}	1×10^{2}	1×10^{6}
Pt-193	4×10^{1}	4×10^{1}	1×10^{4}	1×10^{7}
Pt-193m	4×10^{1}	5×10^{-1}	1×10^{3}	1×10^{7}
Pt-195m	1×10^{1}	5×10^{-1}	1×10^{2}	1×10^{6}
Pt-197	2×10^{1}	6×10^{-1}	1×10^{3}	1×10^{6}
Pt-197m	1×10^{1}	6×10^{-1}	1×10^{2}	1×10^{6}
Plutonium (94)				
Pu-236	3×10^{1}	3×10^{-3}	1×10^{1}	1×10^{4}
Pu-237	2×10^{1}	2×10^{1}	1×10^{3}	1×10^{7}
Pu-238	1×10^{1}	1×10^{-3}	1×10^{0}	1×10^{4}
Pu-239	1×10^{1}	1×10^{-3}	1×10^{0}	1×10^{4}
Pu-240	1×10^{1}	1×10^{-3}	1×10^{0}	1×10^{3}
Pu-241 (a)	4×10^{1}	6×10^{-2}	1×10^{2}	1×10^{5}

Table I. BASIC RADIONUCLIDE VALUES (continued)

Radionuclide (atomic number)	A_1	A_2	Activity concentration for exempt material	Activity limit for an exempt consignment
	(TBq)	(TBq)	(Bq/g)	(Bq)
Pu-242	1×10^{1}	1×10^{-3}	1×10^{0}	1×10^{4}
Pu-244 (a)	4×10^{-1}	1×10^{-3}	1×10^{0}	1×10^{4}
Radium (88)				
Ra-223 (a)	4×10^{-1}	7×10^{-3}	1×10^{2} (b)	1×10^{5} (b)
Ra-224 (a)	4×10^{-1}	2×10^{-2}	1×10^{1} (b)	1×10^{5} (b)
Ra-225 (a)	2×10^{-1}	4×10^{-3}	1×10^{2}	1×10^{5}
Ra-226 (a)	2×10^{-1}	3×10^{-3}	1×10^{1} (b)	1×10^{4} (b)
Ra-228 (a)	6×10^{-1}	2×10^{-2}	1×10^{1} (b)	1×10^{5} (b)
Rubidium (37)				
Rb-81	2×10^{0}	8×10^{-1}	1×10^{1}	1×10^{6}
Rb-83 (a)	2×10^{0}	2×10^{0}	1×10^{2}	1×10^{6}
Rb-84	1×10^{0}	1×10^{0}	1×10^{1}	1×10^{6}
Rb-86	5×10^{-1}	5×10^{-1}	1×10^{2}	1×10^{5}
Rb-87	Unlimited	Unlimited	1×10^{4}	1×10^{7}
Rb(nat)	Unlimited	Unlimited	1×10^{4}	1×10^{7}
Rhenium (75)				
Re-184	1×10^{0}	1×10^{0}	1×10^{1}	1×10^{6}
Re-184m	3×10^{0}	1×10^{0}	1×10^{2}	1×10^{6}
Re-186	2×10^{0}	6×10^{-1}	1×10^{3}	1×10^{6}
Re-187	Unlimited	Unlimited	1×10^{6}	1×10^{9}
Re-188	4×10^{-1}	4×10^{-1}	1×10^{2}	1×10^{5}
Re-189 (a)	3×10^{0}	6×10^{-1}	1×10^{2}	1×10^{6}
Re(nat)	Unlimited	Unlimited	1×10^{6}	1×10^{9}
Rhodium (45)				
Rh-99	2×10^{0}	2×10^{0}	1×10^{1}	1×10^{6}
Rh-101	4×10^{0}	3×10^{0}	1×10^{2}	1×10^{7}
Rh-102	5×10^{-1}	5×10^{-1}	1×10^{1}	1×10^{6}
Rh-102m	2×10^{0}	2×10^{0}	1×10^{2}	1×10^{6}
Rh-103m	4×10^{1}	4×10^{1}	1×10^{4}	1×10^{8}

Table I. BASIC RADIONUCLIDE VALUES (continued)

Radionuclide (atomic number)	A_1	A_2	Activity concentration for exempt material	Activity limit for an exempt consignment
	(TBq)	(TBq)	(Bq/g)	(Bq)
Rh-105	1×10^{1}	8×10^{-1}	1×10^{2}	1×10^{7}
Radon (86)				
Rn-222 (a)	3×10^{-1}	4×10^{-3}	1×10^{1} (b)	1×10^{8} (b)
Ruthenium (44)				
Ru-97	5×10^{0}	5×10^{0}	1×10^{2}	1×10^{7}
Ru-103 (a)	2×10^{0}	2×10^{0}	1×10^{2}	1×10^{6}
Ru-105	1×10^{0}	6×10^{-1}	1×10^{1}	1×10^{6}
Ru-106 (a)	2×10^{-1}	2×10^{-1}	1×10^{2} (b)	1×10^{5} (b)
Sulphur (16)				
S-35	4×10^{1}	3×10^{0}	1×10^{5}	1×10^{8}
Antimony (51)				
Sb-122	4×10^{-1}	4×10^{-1}	1×10^{2}	1×10^{4}
Sb-124	6×10^{-1}	6×10^{-1}	1×10^{1}	1×10^{6}
Sb-125	2×10^{0}	1×10^{0}	1×10^{2}	1×10^{6}
Sb-126	4×10^{-1}	4×10^{-1}	1×10^{1}	1×10^{5}
Scandium (21)				
Sc-44	5×10^{-1}	5×10^{-1}	1×10^{1}	1×10^{5}
Sc-46	5×10^{-1}	5×10^{-1}	1×10^{1}	1×10^{6}
Sc-47	1×10^{1}	7×10^{-1}	1×10^{2}	1×10^{6}
Sc-48	3×10^{-1}	3×10^{-1}	1×10^{1}	1×10^{5}
Selenium (34)				
Se-75	3×10^{0}	3×10^{0}	1×10^{2}	1×10^{6}
Se-79	4×10^{1}	2×10^{0}	1×10^{4}	1×10^{7}
Silicon (14)				
Si-31	6×10^{-1}	6×10^{-1}	1×10^{3}	1×10^{6}
Si-32	4×10^{1}	5×10^{-1}	1×10^{3}	1×10^{6}
Samarium (62)				
Sm-145	1×10^{1}	1×10^{1}	1×10^{2}	1×10^{7}
Sm-147	Unlimited	Unlimited	1×10^{1}	1×10^{4}

Table I. BASIC RADIONUCLIDE VALUES (continued)

Radionuclide (atomic number)	A_1	A_2	Activity concentration for exempt material	Activity limit for an exempt consignment
	(TBq)	(TBq)	(Bq/g)	(Bq)
Sm-151	4×10^{1}	1×10^{1}	1×10^{4}	1×10^{8}
Sm-153	9×10^{0}	6×10^{-1}	1×10^{2}	1×10^{6}
Tin (50)				
Sn-113 (a)	4×10^{0}	2×10^{0}	1×10^{3}	1×10^{7}
Sn-117m	7×10^{0}	4×10^{-1}	1×10^{2}	1×10^{6}
Sn-119m	4×10^{1}	3×10^{1}	1×10^{3}	1×10^{7}
Sn-121m (a)	4×10^{1}	9×10^{-1}	1×10^{3}	1×10^{7}
Sn-123	8×10^{-1}	6×10^{-1}	1×10^{3}	1×10^{6}
Sn-125	4×10^{-1}	4×10^{-1}	1×10^{2}	1×10^{5}
Sn-126 (a)	6×10^{-1}	4×10^{-1}	1×10^{1}	1×10^{5}
Strontium (38)				
Sr-82 (a)	2×10^{-1}	2×10^{-1}	1×10^{1}	1×10^{5}
Sr-85	2×10^{0}	2×10^{0}	1×10^{2}	1×10^{6}
Sr-85m	5×10^{0}	5×10^{0}	1×10^{2}	1×10^{7}
Sr-87m	3×10^{0}	3×10^{0}	1×10^{2}	1×10^{6}
Sr-89	6×10^{-1}	6×10^{-1}	1×10^{3}	1×10^{6}
Sr-90 (a)	3×10^{-1}	3×10^{-1}	1×10^{2} (b)	1×10^{4} (b)
Sr-91 (a)	3×10^{-1}	3×10^{-1}	1×10^{1}	1×10^{5}
Sr-92 (a)	1×10^{0}	3×10^{-1}	1×10^{1}	1×10^{6}
Tritium (1)				
T(H-3)	4×10^{1}	4×10^{1}	1×10^{6}	1×10^{9}
Tantalum (73)				
Ta-178(long-lived)	1×10^{0}	8×10^{-1}	1×10^{1}	1×10^{6}
Ta-179	3×10^{1}	3×10^{1}	1×10^{3}	1×10^{7}
Ta-182	9×10^{-1}	5×10^{-1}	1×10^{1}	1×10^{4}
Terbium (65)				
Tb-157	4×10^{1}	4×10^{1}	1×10^{4}	1×10^{7}
Tb-158	1×10^{0}	1×10^{0}	1×10^{1}	1×10^{6}
Tb-160	1×10^{0}	6×10^{-1}	1×10^{1}	1×10^{6}

Table I. BASIC RADIONUCLIDE VALUES (continued)

Radionuclide (atomic number)	A_1	A_2	Activity concentration for exempt material	Activity limit for an exempt consignment
	(TBq)	(TBq)	(Bq/g)	(Bq)
Technetium (43)				
Tc-95m (a)	2×10^{0}	2×10^{0}	1×10^{1}	1×10^{6}
Tc-96	4×10^{-1}	4×10^{-1}	1×10^{1}	1×10^{6}
Tc-96m (a)	4×10^{-1}	4×10^{-1}	1×10^{3}	1×10^{7}
Tc-97	Unlimited	Unlimited	1×10^{3}	1×10^{8}
Tc-97m	4×10^{1}	1×10^{0}	1×10^{3}	1×10^{7}
Tc-98	8×10^{-1}	7×10^{-1}	1×10^{1}	1×10^{6}
Tc-99	4×10^{1}	9×10^{-1}	1×10^{4}	1×10^{7}
Tc-99m	1×10^{1}	4×10^{0}	1×10^{2}	1×10^{7}
Tellurium (52)				
Te-121	2×10^{0}	2×10^{0}	1×10^{1}	1×10^{6}
Te-121m	5×10^{0}	3×10^{0}	1×10^{2}	1×10^{5}
Te-123m	8×10^{0}	1×10^{0}	1×10^{2}	1×10^{7}
Te-125m	2×10^{1}	9×10^{-1}	1×10^{3}	1×10^{7}
Te-127	2×10^{1}	7×10^{-1}	1×10^{3}	1×10^{6}
Te-127m (a)	2×10^{1}	5×10^{-1}	1×10^{3}	1×10^{7}
Te-129	7×10^{-1}	6×10^{-1}	1×10^{2}	1×10^{6}
Te-129m (a)	8×10^{-1}	4×10^{-1}	1×10^{3}	1×10^{6}
Te-131m (a)	7×10^{-1}	5×10^{-1}	1×10^{1}	1×10^{6}
Te-132 (a)	5×10^{-1}	4×10^{-1}	1×10^{2}	1×10^{7}
Thorium (90)				
Th-227	1×10^{1}	5×10^{-3}	1×10^{1}	1×10^{4}
Th-228 (a)	5×10^{-1}	1×10^{-3}	1×10^{0} (b)	1×10^{4} (b)
Th-229	5×10^{0}	5×10^{-4}	1×10^{0} (b)	1×10^{3} (b)
Th-230	1×10^{1}	1×10^{-3}	1×10^{0}	1×10^{4}
Th-231	4×10^{1}	2×10^{-2}	1×10^{3}	1×10^{7}
Th-232	Unlimited	Unlimited	1×10^{1}	1×10^{4}
Th-234 (a)	3×10^{-1}	3×10^{-1}	1×10^{3} (b)	1×10^{5} (b)
Th(nat)	Unlimited	Unlimited	1×10^{0} (b)	1×10^{3} (b)

Table I. BASIC RADIONUCLIDE VALUES (continued)

Radionuclide (atomic number)	A_1	A_2	Activity concentration for exempt material	Activity limit for an exempt consignment
	(TBq)	(TBq)	(Bq/g)	(Bq)
Titanium (22)				
Ti-44 (a)	5×10^{-1}	4×10^{-1}	1×10^{1}	1×10^{5}
Thallium (81)				
Tl-200	9×10^{-1}	9×10^{-1}	1×10^{1}	1×10^{6}
Tl-201	1×10^{1}	4×10^{0}	1×10^{2}	1×10^{6}
Tl-202	2×10^{0}	2×10^{0}	1×10^{2}	1×10^{6}
Tl-204	1×10^{1}	7×10^{-1}	1×10^{4}	1×10^{4}
Thulium (69)				
Tm-167	7×10^{0}	8×10^{-1}	1×10^{2}	1×10^{6}
Tm-170	3×10^{0}	6×10^{-1}	1×10^{3}	1×10^{6}
Tm-171	4×10^{1}	4×10^{1}	1×10^{4}	1×10^{8}
Uranium (92)				
U-230 (fast lung absorption)(a)(d)	4×10^{1}	1×10^{-1}	1×10^{1} (b)	1×10^{5} (b)
U-230 (medium lung absorption)(a)(e)	4×10^{1}	4×10^{-3}	1×10^{1}	1×10^{4}
U-230 (slow lung absorption)(a)(f)	3×10^{1}	3×10^{-3}	1×10^{1}	1×10^{4}
U-232 (fast lung absorption)(d)	4×10^{1}	1×10^{-2}	1×10^{0} (b)	1×10^{3} (b)
U-232 (medium lung absorption)(e)	4×10^{1}	7×10^{-3}	1×10^{1}	1×10^{4}
U-232 (slow lung absorption)(f)	1×10^{1}	1×10^{-3}	1×10^{1}	1×10^{4}
U-233 (fast lung absorption)(d)	4×10^{1}	9×10^{-2}	1×10^{1}	1×10^{4}
U-233 (medium lung absorption)(e)	4×10^{1}	2×10^{-2}	1×10^{2}	1×10^{5}
U-233 (slow lung absorption)(f)	4×10^{1}	6×10^{-3}	1×10^{1}	1×10^{5}
U-234 (fast lung absorption)(d)	4×10^{1}	9×10^{-2}	1×10^{1}	1×10^{4}
U-234 (medium lung absorption)(e)	4×10^{1}	2×10^{-2}	1×10^{2}	1×10^{5}
U-234 (slow lung absorption)(f)	4×10^{1}	6×10^{-3}	1×10^{1}	1×10^{5}
U-235 (all lung absorption types)(a),(d),(e),(f)	Unlimited	Unlimited	1×10^{1} (b)	1×10^{4} (b)
U-236 (fast lung absorption)(d)	Unlimited	Unlimited	1×10^{1}	1×10^{4}
U-236 (medium lung absorption)(e)	4×10^{1}	2×10^{-2}	1×10^{2}	1×10^{5}
U-236 (slow lung absorption)(f)	4×10^{1}	6×10^{-3}	1×10^{1}	1×10^{4}

Table I. BASIC RADIONUCLIDE VALUES (continued)

Radionuclide (atomic number)	A_1	A_2	Activity concentration for exempt material	Activity limit for an exempt consignment
	(TBq)	(TBq)	(Bq/g)	(Bq)
U-238 (all lung absorption types)(d),(e),(f)	Unlimited	Unlimited	1×10^{1} (b)	1×10^{4} (b)
U (nat)	Unlimited	Unlimited	1×10^{0} (b)	1×10^{3} (b)
U (enriched to 20% or less)(g)	Unlimited	Unlimited	1×10^{0}	1×10^{3}
U (dep)	Unlimited	Unlimited	1×10^{0}	1×10^{3}
Vanadium (23)				
V-48	4×10^{-1}	4×10^{-1}	1×10^{1}	1×10^{5}
V-49	4×10^{1}	4×10^{1}	1×10^{4}	1×10^{7}
Tungsten (74)				
W-178 (a)	9×10^{0}	5×10^{0}	1×10^{1}	1×10^{6}
W-181	3×10^{1}	3×10^{1}	1×10^{3}	1×10^{7}
W-185	4×10^{1}	8×10^{-1}	1×10^{4}	1×10^{7}
W-187	2×10^{0}	6×10^{-1}	1×10^{2}	1×10^{6}
W-188 (a)	4×10^{-1}	3×10^{-1}	1×10^{2}	1×10^{5}
Xenon (54)				
Xe-122 (a)	4×10^{-1}	4×10^{-1}	1×10^{2}	1×10^{9}
Xe-123	2×10^{0}	7×10^{-1}	1×10^{2}	1×10^{9}
Xe-127	4×10^{0}	2×10^{0}	1×10^{3}	1×10^{5}
Xe-131m	4×10^{1}	4×10^{1}	1×10^{4}	1×10^{4}
Xe-133	2×10^{1}	1×10^{1}	1×10^{3}	1×10^{4}
Xe-135	3×10^{0}	2×10^{0}	1×10^{3}	1×10^{10}
Yttrium (39)				
Y-87 (a)	1×10^{0}	1×10^{0}	1×10^{1}	1×10^{6}
Y-88	4×10^{-1}	4×10^{-1}	1×10^{1}	1×10^{6}
Y-90	3×10^{-1}	3×10^{-1}	1×10^{3}	1×10^{5}
Y-91	6×10^{-1}	6×10^{-1}	1×10^{3}	1×10^{6}
Y-91m	2×10^{0}	2×10^{0}	1×10^{2}	1×10^{6}
Y-92	2×10^{-1}	2×10^{-1}	1×10^{2}	1×10^{5}
Y-93	3×10^{-1}	3×10^{-1}	1×10^{2}	1×10^{5}
Ytterbium (79)				

Table I. BASIC RADIONUCLIDE VALUES (continued)

Radionuclide (atomic number)	A_1	A_2	Activity concentration for exempt material	Activity limit for an exempt consignment
	(TBq)	(TBq)	(Bq/g)	(Bq)
Yb-169	4×10^{0}	1×10^{0}	1×10^{2}	1×10^{7}
Yb-175	3×10^{1}	9×10^{-1}	1×10^{3}	1×10^{7}
Zinc (30)				
Zn-65	2×10^{0}	2×10^{0}	1×10^{1}	1×10^{6}
Zn-69	3×10^{0}	6×10^{-1}	1×10^{4}	1×10^{6}
Zn-69m (a)	3×10^{0}	6×10^{-1}	1×10^{2}	1×10^{6}
Zirconium (40)				
Zr-88	3×10^{0}	3×10^{0}	1×10^{2}	1×10^{6}
Zr-93	Unlimited	Unlimited	1×10^{3} (b)	1×10^{7} (b)
Zr-95 (a)	2×10^{0}	8×10^{-1}	1×10^{1}	1×10^{6}
Zr-97 (a)	4×10^{-1}	4×10^{-1}	1×10^{1} (b)	1×10^{5} (b)

Table I. BASIC RADIONUCLIDE VALUES (continued)

FOOTNOTES:

(a) A_1 and/or A_2 values include contributions from daughter nuclides with half-lives less than 10 days
(b) Parent nuclides and their progeny included in secular equilibrium are listed in the following:

Sr-90	Y-90
Zr-93	Nb-93m
Zr-97	Nb-97
Ru-106	Rh-106
Cs-137	Ba-137m
Ce-134	La-134
Ce-144	Pr-144
Ba-140	La-140
Bi-212	Tl-208 (0.36), Po-212 (0.64)
Pb-210	Bi-210, Po-210
Pb-212	Bi-212, Tl-208 (0.36), Po-212 (0.64)
Rn-220	Po-216
Rn-222	Po-218, Pb-214, Bi-214, Po-214
Ra-223	Rn-219, Po-215, Pb-211, Bi-211, Tl-207
Ra-224	Rn-220, Po-216, Pb-212, Bi-212, Tl-208 (0.36), Po-212 (0.64)
Ra-226	Rn-222, Po-218, Pb-214, Bi-214, Po-214, Pb-210, Bi-210, Po-210
Ra-228	Ac-228
Th-226	Ra-222, Rn-218, Po-214
Th-228	Ra-224, Rn-220, Po-216, Pb212, Bi-212, Tl208 (0.36), Po-212 (0.64)
Th-229	Ra-225, Ac-225, Fr-221, At-217, Bi-213, Po-213, Pb-209
Th-nat	Ra-228, Ac-228, Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208 (0.36), Po-212 (0.64)
Th-234	Pa-234m
U-230	Th-226, Ra-222, Rn-218, Po-214
U-232	Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208 (0.36), Po-212 (0.64)
U-235	Th-231
U-238	Th-234, Pa-234m
U-nat	Th-234, Pa-234m, U-234, Th-230, Ra-226, Rn-222, Po-218, Pb-214, Bi-214, Po-214, Pb-210, Bi-210, Po-210
U-240	Np-240m
Np-237	Pa-233
Am-242m	Am-242
Am-243	Np-239

(c) The quantity may be determined from a measurement of the rate of decay or a measurement of the radiation level at a prescribed distance from the source.
(d) These values apply only to compounds of uranium that take the chemical form of UF_6, UO_2F_2 and $UO_2(NO_3)_2$ in both normal and accident conditions of transport.
(e) These values apply only to compounds of uranium that take the chemical form of UO_3, UF_4, UCl_4 and hexavalent compounds in both normal and accident conditions of transport.
(f) These values apply to all compounds of uranium other than those specified in (d) and (e) above.
(g) These values apply to *unirradiated uranium* only.

where,

f(i) is the fraction of activity or activity concentration of radionuclide i in the mixture;
X(i) is the appropriate value of A_1 or A_2, or the activity concentration for exempt material or the activity limit for an exempt consignment as appropriate for the radionuclide i; and
X_m is the derived value of A_1 or A_2, or the activity concentration for exempt material or the activity limit for an exempt consignment in the case of a mixture. **(Issue 2)**

Table II. BASIC RADIONUCLIDE VALUES FOR UNKNOWN RADIONUCLIDES OR MIXTURES

Radioactive contents	A_1	A_2	Activity concentration for exempt material	Activity limits for exempt consignments
	TBq	TBq	Bq/g	Bq
Only beta or gamma emitting nuclides are known to be present	0.1	0.02	1×10^{1}	1×10^{4}
Only alpha emitting nuclides are known to be present	0.2	9×10^{-5}	1×10^{-1}	1×10^{3}
No relevant data are available	0.001	9×10^{-5}	1×10^{-1}	1×10^{3}

405. When the identity of each radionuclide is known but the individual activities of some of the radionuclides are not known, the radionuclides may be grouped and the lowest radionuclide value, as appropriate, for the radionuclides in each group may be used in applying the formulas in paras and . Groups may be based on the total alpha activity and the total beta/gamma activity when these are known, using the lowest radionuclide values for the alpha emitters or beta/gamma emitters, respectively. **(Issue 2)**

406. For individual radionuclides or for mixtures of radionuclides for which relevant data are not available, the values shown in Table II shall be used. **(Issue 2)**

543. Each label conforming to the models in Fig. 2, Fig. 3 and Fig. 4 shall be completed with the following information:

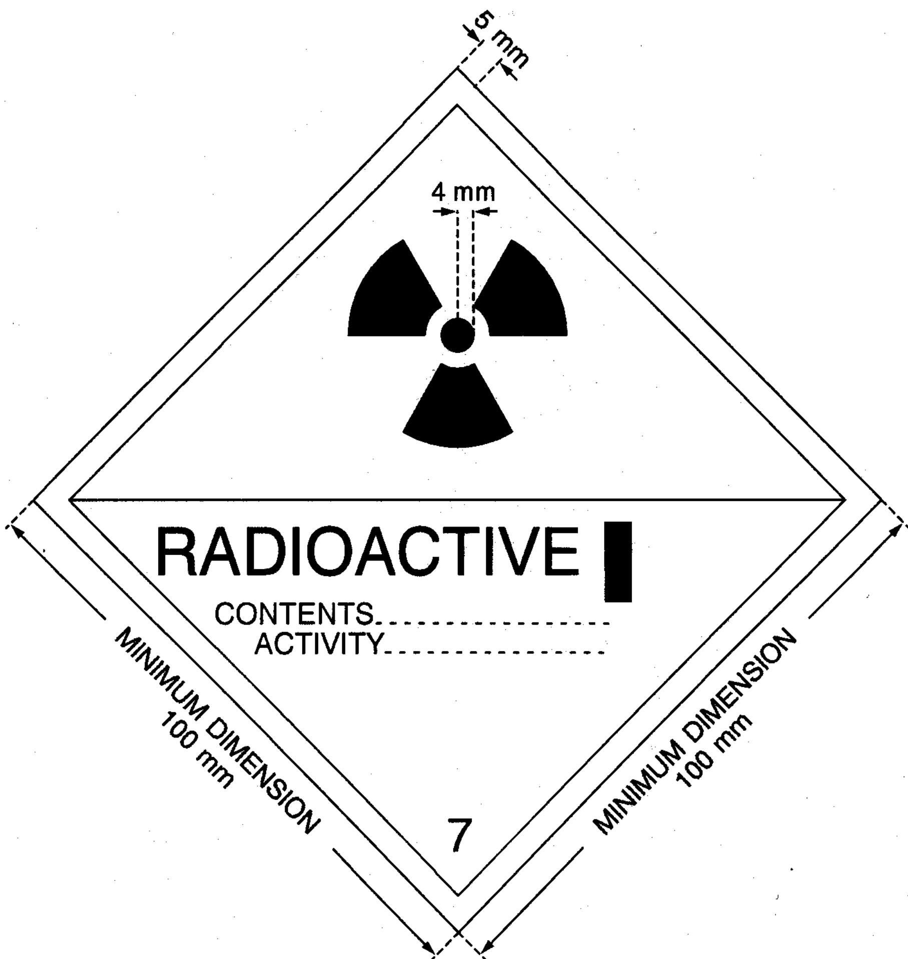

FIG. 2. Category I-WHITE label. The background colour of the label shall be white, the colour of the trefoil and the printing shall be black, and the colour of the category bar shall be red.

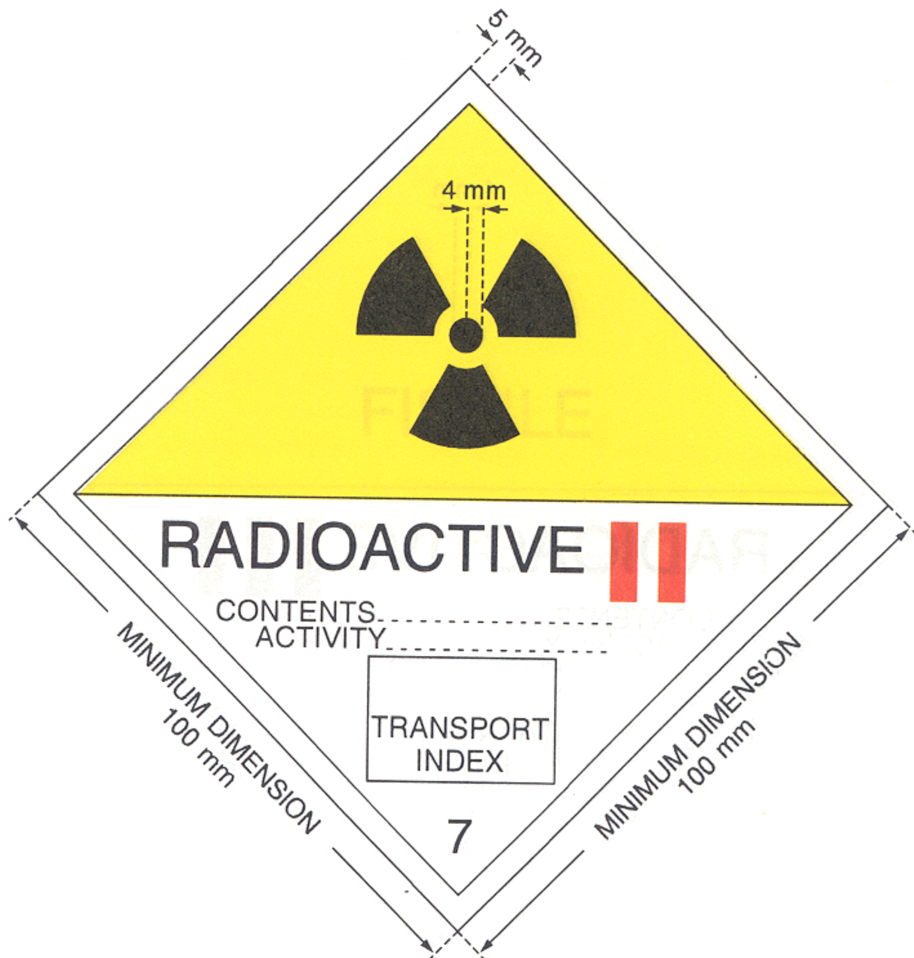

FIG. 3. Category II-YELLOW label. The background colour of the upper half of the label shall be yellow and the lower half white, the colour of the trefoil and the printing shall be black, and the colour of the category bars shall be red.

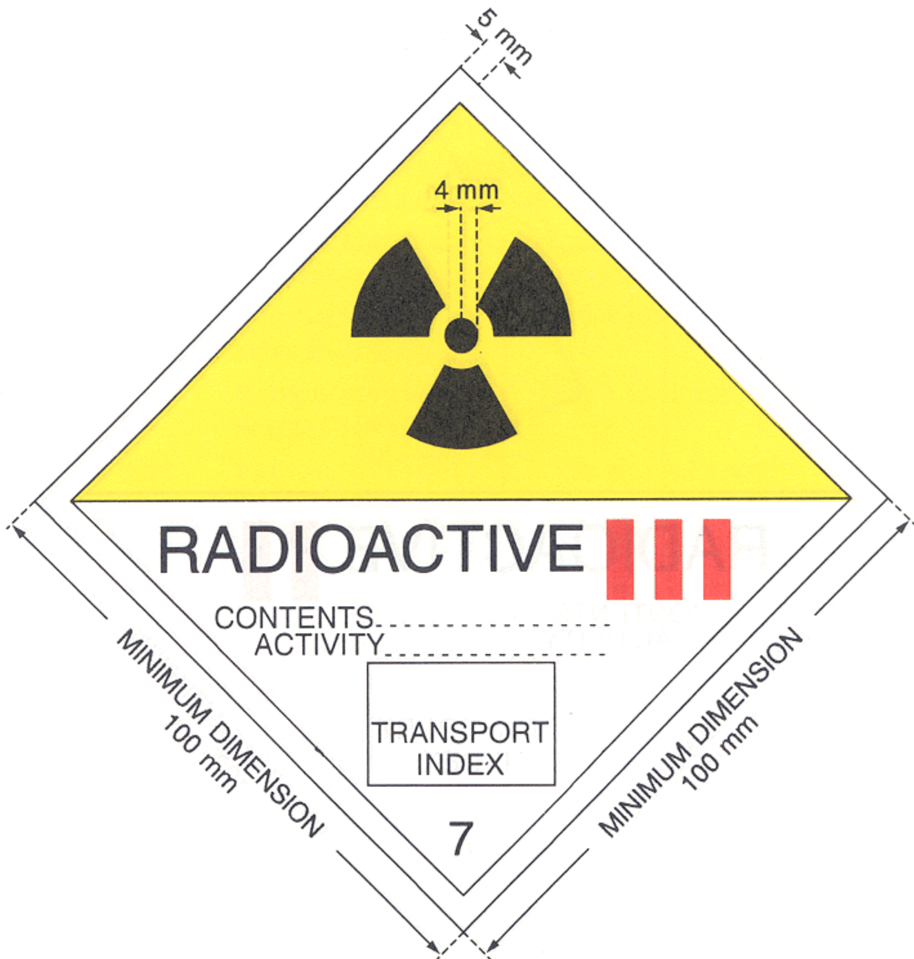

FIG. 4. Category III-YELLOW label. The background colour of the upper half of the label shall be yellow and the lower half white, the colour of the trefoil and the printing shall be black, and the colour of the category bars shall be red.

(a) Contents:

(i) Except for LSA-I material, the name(s) of the radionuclide(s) as taken from Table I, using the symbols prescribed therein. For mixtures of radionuclides, the most restrictive nuclides must be listed to the extent the space on the line permits. The group of LSA or SCO shall be shown following the name(s) of the radionuclide(s). The terms “LSA-II”, “LSA-III”, “SCO-I” and “SCO-II” shall be used for this purpose.

(ii) For LSA-I material, the term “LSA-I” is all that is necessary; the name of the radionuclide is not necessary.

(b) Activity: The maximum activity of the radioactive contents during transport expressed in units of becquerels (Bq) with the appropriate SI prefix (see Annex II). For fissile material, the mass of fissile material in units of grams (g), or multiples thereof, may be used in place of activity.

(c) For overpacks and freight containers the “contents” and “activity” entries on the label shall bear the information required in subparas 543(a) and 543(b), respectively, totalled together for the entire contents of the overpack or freight container except that on labels for overpacks or freight containers containing mixed loads of packages containing different radionuclides, such entries may read “See Transport Documents”.

(d) Transport index: See paras 526 and 527. (No transport index entry is required for category I-WHITE.) (Issue 1)

544. Each label conforming to the model in Fig. 5 shall be completed with the criticality safety index (CSI) as stated in the certificate of approval for special arrangement or the certificate of approval for the package design issued by the competent authority. (Issue 5)

545. For overpacks and freight containers, the criticality safety index (CSI) on the label shall bear the information required in para. 544 totalled together for the fissile contents of the overpack or freight container. (Issue 5)

549. The consignor shall include in the transport documents with each consignment the following information, as applicable in the order given:

(a) The proper shipping name, as specified in Table VIII;

(b) The United Nations Class number “7";

(c) The United Nations number assigned to the material as specified in Table VIII, preceded by the letters “UN”;

(d) The name or symbol of each radionuclide or, for mixtures of radionuclides, an appropriate general description or a list of the most restrictive nuclides;

(e) A description of the physical and chemical form of the material, or a notation that the material is special form radioactive material or low dispersible radioactive material. A generic chemical description is acceptable for chemical form;

(f) The maximum activity of the radioactive contents during transport expressed in units of becquerels (Bq) with an appropriate SI prefix (see Annex II). For fissile material, the mass of fissile material in units of grams (g), or appropriate multiples thereof, may be used in place of activity.

(g) The category of the package, i.e. I-WHITE, II-YELLOW, III-YELLOW;

(h) The transport index (categories II-YELLOW and III-YELLOW only);

(i) For consignments including fissile material other than consignments excepted under para. 672, the criticality safety index;

(j) The identification mark for each competent authority approval certificate (special form radioactive material, low dispersible radioactive material, special arrangement, package design, or shipment) applicable to the consignment;

(k) For consignments of packages in an overpack or freight container, a detailed statement of the contents of each package within the overpack or freight container and, where appropriate, of each overpack or freight container in the consignment. If packages are to be removed from the overpack or freight container at a point of intermediate unloading, appropriate transport documents shall be made available;

(l) Where a consignment is required to be shipped under exclusive use, the statement "EXCLUSIVE USE SHIPMENT"; and

(m) For LSA-II, LSA-III, SCO-I and SCO-II, the total activity of the consignment as a multiple of A_2 (Issue 1)

629. Except as allowed in para. 632, uranium hexafluoride shall be packaged and transported in accordance with the provisions of the International Organization for Standardization document ISO 7195:, "Packaging of uranium hexafluoride (UF_6;) for transport" [1] and the requirements of paras 630-631. The package shall also meet the requirements prescribed elsewhere in these Regulations which pertain to the radioactive and fissile properties of the material. (Issue 4)

630. Each package designed to contain 0.1 kg or more of uranium hexafluoride shall be designed so that it would meet the following requirements:

(a) withstand without leakage and without unacceptable stress, as specified in the International Organization for Standardization document ISO 7195\10\, the structural test as specified in para. 718;

(b) withstand without loss or dispersal of the uranium hexafluoride the test specified in para. 722; and

(c) withstand without rupture of the containment system the test specified in para. 728. (Issue 4)

631. Packages designed to contain 0.1 kg or more of uranium hexafluoride shall not be provided with pressure relief devices. (Issue 4)

632. Subject to the approval of the competent authority, packages designed to contain 0.1 kg or more of uranium hexafluoride may be transported if:

(a) the packages are designed to requirements other than those given in ISO 7195[10] and paras 630-631 but, notwithstanding, the requirements of paras 630-631 are met as far as practicable. (Issue 4)

657. A package for radioactive contents with activity greater than 10^5 A_2 shall be so designed that if it were subjected to the enhanced water immersion test specified in para. 730, there would be no rupture of the containment system. (Issue 7)

667. Type C packages shall be designed to meet the requirements specified in paras 606-619, and of paras 634-647, except as specified in para. 646(a), and of the requirements specified in paras 651-654, paras 658-664, and, in addition, of paras 668-670. (Issue 6)

668. A package shall be capable of meeting the assessment criteria prescribed for tests in paras 656(b) and 660 after burial in an environment defined by a thermal conductivity of 0.33 W/m.K and a temperature of 38 deg.C in the steady state. Initial conditions for the assessment shall assume that any thermal insulation of the package remains intact, the package is at the maximum normal operating pressure and the ambient temperature is 38 deg.C. (Issue 6)

669. A package shall be so designed that, if it were at the maximum normal operating pressure and subjected to:

(a) the tests specified in paras 719-724, it would restrict the loss of radioactive contents to not more than 10^{-6} A_2 per hour; and

(b) the test sequences in para. 734, it would meet the following requirements:

(i) retain sufficient shielding to ensure that the radiation level at 1 m from the surface of the package would not exceed 10 mSv/h with the maximum radioactive contents which the package is designed to contain; and

(ii) restrict the accumulated loss of radioactive contents in a period of 1 week to not more than 10 A_2 for krypton-85 and not more than A_2 for all other radionuclides.

Where mixtures of different radionuclides are present, the provisions of paras 404-406 shall apply except that for krypton-85 an effective A_2 (i) value equal to 10 A_2 may be used. For case (a) above, the assessment shall take into account the external contamination limits of para. 508. (Issue 6)

670. A package shall be so designed that there will be no rupture of the containment system following performance of the enhanced water immersion test specified in para. 730. (Issue 6)

677. For a package in isolation, it shall be assumed that water can leak into or out of all void spaces of the package, including those within the containment system. However, if the design incorporates special features to prevent such leakage of water into or out of certain void spaces, even as a result of error, absence of leakage may be assumed in respect of those void spaces. Special features shall include the following:

(a) Multiple high standard water barriers, each of which would remain watertight if the package were subject to the tests prescribed in para. 682(b), a high degree of quality control in the manufacture, maintenance and repair of packagings and tests to demonstrate the closure of each package before each shipment; or

(b) For packages containing uranium hexafluoride only:

(i) packages where, following the tests prescribed in para. 682(b), there is no physical contact between the valve and any other component of the packaging other than at its original point of attachment and where, in addition, following the test prescribed in para. 728 the valves remain leaktight; and

(ii) a high degree of quality control in the manufacture, maintenance and repair of packagings coupled with tests to demonstrate closure of each package before each shipment. (Issue 4 and issue 11)

680. For packages to be transported by air:

(a) the package shall be subcritical under conditions consistent with the tests prescribed in para. 734 assuming reflection by at least 20cm of water but no water inleakage; and

(b) allowance shall not be made for special features of para. 677 unless, following the tests specified in para. 734 and, subsequently, para. 733, leakage of water into or out of the void spaces is prevented. (Issue 11)

682. A number “N” shall be derived, such that two times “N” shall be subcritical for the arrangement and package conditions that provide the maximum neutron multiplication consistent with the following:

(a) Hydrogenous moderation between packages, and the package arrangement reflected on all sides by at least 20 cm of water; and

(b) The tests specified in paras 719-724 followed by whichever of the following is the more limiting:

(i) the tests specified in para. 727(b) and, either para. 727(c) for packages having a mass not greater than 500 kg and an overall density not greater than 1000 kg/m3 based on the external dimensions, or para. 727(a) for all other packages; followed by the test specified in para. 728 and completed by the tests specified in paras 731-733; or

(ii) the test specified in para. 729; and

(c) Where any part of the fissile material escapes from the containment system following the tests specified in para. 682(b), it shall be assumed that fissile material escapes from each package in the array and all of the fissile material shall be arranged in the configuration and moderation that results in the maximum neutron multiplication with close reflection by at least 20 cm of water. (Issue 10)

719. The tests are: the water spray test, the free drop test, the stacking test and the penetration test. Specimens of the package shall be subjected to the free drop test, the stacking test and the penetration test, preceded in each case by the water spray test. One

specimen may be used for all the tests, provided that the requirements of para. 720 are fulfilled. (Issue 10)

720. The time interval between the conclusion of the water spray test and the succeeding test shall be such that the water has soaked in to the maximum extent, without appreciable drying of the exterior of the specimen. In the absence of any evidence to the contrary, this interval shall be taken to be two hours if the water spray is applied from four directions simultaneously. No time interval shall elapse, however, if the water spray is applied from each of the four directions consecutively. (Issue 10)

721. Water spray test: The specimen shall be subjected to a water spray test that simulates exposure to rainfall of approximately 5 cm per hour for at least one hour. (Issue 10).

722. Free drop test: The specimen shall drop onto the target so as to suffer maximum damage in respect of the safety features to be tested.

(a) The height of drop measured from the lowest point of the specimen to the upper surface of the target shall be not less than the distance specified in Table XIII for the applicable mass. The target shall be as defined in para. 717.

(b) For rectangular fibreboard or wood packages not exceeding a mass of 50 kg, a separate specimen shall be subjected to a free drop onto each corner from a height of 0.3 m.

(c) For cylindrical fibreboard packages not exceeding a mass of 100 kg, a separate specimen shall be subjected to a free drop onto each of the quarters of each rim from a height of 0.3 m. (Issue 10)

723. Stacking test: Unless the shape of the packaging effectively prevents stacking, the specimen shall be subjected, for a period of 24 h, to a compressive load equal to the greater of the following:

(a) The equivalent of 5 times the mass of the actual package; and

(b) The equivalent of 13 kPa multiplied by the vertically projected area of the package.

The load shall be applied uniformly to two opposite sides of the specimen, one of which shall be the base on which the package would typically rest. (Issue 10)

724. Penetration test: The specimen shall be placed on a rigid, flat, horizontal surface which will not move significantly while the test is being carried out.

(a) A bar of 3.2 cm in diameter with a hemispherical end and a mass of 6 kg shall be dropped and directed to fall, with its longitudinal axis vertical, onto the centre of the weakest part of the specimen, so that, if it penetrates sufficiently far, it will hit the containment system. The bar shall not be significantly deformed by the test performance.

(b) The height of drop of the bar measured from its lower end to the intended point of impact on the upper surface of the specimen shall be 1 m. (Issue 10)

727. Mechanical test: The mechanical test consists of three different drop tests. Each specimen shall be subjected to the applicable drops as specified in para. 656 or para. 682. The

order in which the specimen is subjected to the drops shall be such that, on completion of the mechanical test, the specimen shall have suffered such damage as will lead to the maximum damage in the thermal test which follows.

(a) For drop I, the specimen shall drop onto the target so as to suffer the maximum damage, and the height of the drop measured from the lowest point of the specimen to the upper surface of the target shall be 9 m. The target shall be as defined in para. 717.

(b) For drop II, the specimen shall drop so as to suffer the maximum damage onto a bar rigidly mounted perpendicularly on the target. The height of the drop measured from the intended point of impact of the specimen to the upper surface of the bar shall be 1 m. The bar shall be of solid mild steel of circular section, (15.0 ± 0.5) cm in diameter and 20 cm long unless a longer bar would cause greater damage, in which case a bar of sufficient length to cause maximum damage shall be used. The upper end of the bar shall be flat and horizontal with its edges rounded off to a radius of not more than 6 mm. The target on which the bar is mounted shall be as described in para. 717.

(c) For drop III, the specimen shall be subjected to a dynamic crush test by positioning the specimen on the target so as to suffer maximum damage by the drop of a 500 kg mass from 9 m onto the specimen. The mass shall consist of a solid mild steel plate 1 m by 1 m and shall fall in a horizontal attitude. The height of the drop shall be measured from the underside of the plate to the highest point of the specimen. The target on which the specimen rests shall be as defined in para. 717. (Issue 10)

729. Water immersion test: The specimen shall be immersed under a head of water of at least 15 m for a period of not less than eight hours in the attitude which will lead to maximum damage. For demonstration purposes, an external gauge pressure of at least 150 kPa shall be considered to meet these conditions. (Issue 10)

730. Enhanced water immersion test: The specimen shall be immersed under a head of water of at least 200 m for a period of not less than one hour. For demonstration purposes, an external gauge pressure of at least 2 MPa shall be considered to meet these conditions. (Issue 7)

734. Specimens shall be subjected to the effects of each of the following test sequences in the orders specified:

(a) the tests specified in paras 727(a), 727(c), 735 and 736; and

(b) the test specified in para. 737.

Separate specimens are allowed to be used for each of the sequences (a) and (b). (Issue 6)

735. Puncture/tearing test: The specimen shall be subjected to the damaging effects of a solid probe made of mild steel. The orientation of the probe to the surface of the specimen shall be as to cause maximum damage at the conclusion of the test sequence specified in para. 734(a).

(a) The specimen, representing a package having a mass less than 250 kg, shall be placed on a target and subjected to a probe having a mass of 250 kg falling from a height of 3 m above the intended impact point. For this test the probe shall be a 20 cm diameter cylindrical bar with the striking end forming a frustum of a right circular cone with the following dimensions: 30 cm height and 2.5 cm in diameter at the top. The target on which the specimen is placed shall be as specified in para. 717.

(b) For packages having a mass of 250 kg or more, the base of the probe shall be placed on a target and the specimen dropped onto the probe. The height of the drop , measured from the point of impact with the specimen to the upper surface of the probe shall be 3 m. For this test the probe shall have the same properties and dimensions as specified in (a) above, except that the length and mass of the probe shall be such as to incur maximum damage to the specimen. The target on which the base of the probe is placed shall be as specified in para. 717. (Issue 6)

736. Enhanced thermal test: The conditions for this test shall be as specified in para. 728, except that the exposure to the thermal environment shall be for a period of 60 minutes. (Issue 6)

737. Impact test: The specimen shall be subject to an impact on a target at a velocity of not less than 90 m/s, at such an orientation as to suffer maximum damage. The target shall be as defined in para. 717. (Issue 6)

Dated at Rockville, Maryland, this 11th day of July, 2000.

For the Nuclear Regulatory Commission.
William F. Kane,
Director, Office of Nuclear Material Safety and Safeguards.

APPENDIX B

CROSS REFERENCE OF COMMENTERS
BY COMMENTER NAME

Commenter	Commenter Number	Sections
Action for a Clean Environment	AT33	2.2
AEA Technology QSA, Inc.	MD17, 0055	2.1, 3.1, 3.4, 4.1, 4.4, 5.4, 8.0, 8.4, 9.0, 9.4, 10.0, 11.0, 14.0, 14.3, 15.0, 16.0, 17.0
Airline Pilots Association	MD09	8.0
American Petroleum Institute	MD04, 0087	2.1, 2.3, 3.0, 4.1, 5.0, 11.0, 14.0
ASME International	0080	16.0
Attorney General's Office, State of New Mexico	0053	19.0, 19.1
Barrowes, Steven C.	0056	3.0, 21.0
Bastin, Clinton	AT28	3.0
Blue Ridge Environmental Defense League	MD16, 0068	2.1, 2.2, 2.4, 2.6, 14.0, 17.0, 17.1, 19.0, 19.1, 21.0
Calvert Cliffs Nuclear Power Plant, Inc.	MD18	2.1, 15.0, 15.2, 16.0, 19.0
Chem-Nuclear Systems/Nuclear Energy Institute	MD07	14.0, 14.3, 19.2, 19.3
Clark County Department of Comprehensive Planning	OA43, 0092	2.1, 2.2, 2.3, 3.4, 7.0, 16.0, 17.0, 17.3, 20.0, 21.0
Columbiana Boiler Company	0061	6.4, 15.2, 16.0, 16.3
Connecticut Department of Environmental Protection	MD01	2.1, 2.3, 2.5
Environmentalists, Inc.	0074	2.4, 2.6, 13.0, 17.0, 19.0
Eureka County, Yucca Mountain Information Office	0090	2.2, 2.3, 2.6, 3.0, 4.0, 4.1, 5.1, 7.0, 9.0, 12.0, 14.0, 16.1, 19.0, 20.0
Fabilli, Virginia	0075	2.6
Falchi, Frank	MD21	19.0, 19.1, 19.2
Ferguson, Tom	AT34	2.3
Flemming, Bill	AT32	13.0, 18.0
Florida Department of Health, Bureau of Radiation	MD02	2.1, 2.3, 2.5, 3.0, 3.1
Frontier Technology Corporation	0058	5.0, 5.1, 5.2, 5.4
Fulk, Marion	OA46	3.0, 4.1, 4.4, 20.0
GA/DNR/EPD	AT31	7.0
General Atomics	0057	10.0, 10.2

Commenter	Commenter Number	Sections
Georgia Public Service Commission	0059	2.2, 3.0, 4.0, 7.4, 10.0, 10.4, 16.0, 19.0, 20.3
GTS Duratek	0051	2.1, 3.0, 3.1, 3.4, 10.0, 10.2, 14.3, 16.0, 16.2, 19.0, 19.1
Human Race	AT24	2.4
J.L. Shepherd & Associates	OA42, OA45, 0067	2.4, 3.0, 3.1, 3.2, 3.4, 4.0, 4.4, 5.3, 7.0, 8.0, 8.1, 8.2, 8.4, 9.0, 10.0, 10.2, 14.0, 14.1, 15.0, 16.0, 17.0
League of Women Voters of South Carolina	0096	2.1, 2.3, 2.5, 2.6, 3.1, 4.0
Lincoln County/City of Caliente	0070	2.5, 3.0, 4.1, 5.0, 19.2, 20.0, 20.1
Mallinckrodt Inc.	MD19, AT26	2.5, 2.6, 3.0, 3.1, 3.3, 5.0, 5.3, 5.4, 21.0
Member of Audience	AT35	2.2
Member of Audience	AT36	2.2
Member of Audience	AT37	2.2, 4.0
Member of Audience	AT38	4.0
Member of Audience	AT39	2.3
Member of Audience	AT40	2.2
N/A	0048	3.1, 4.0
N/A	0094	2.1
N/A	0095	2.1, 2.2
New England Coalition on Nuclear Pollution	0073	2.1, 2.2, 2.3, 2.5, 3.0, 4.1, 5.0, 6.0, 6.1, 14.0, 15.0, 17.0, 18.0, 19.0
New Mexico Environmental Evaluation Group	0077	19.0, 19.2
NIRS Southeast	AT22	2.2, 2.3, 2.4, 2.6, 4.0, 4.1, 5.0, 10.0, 12.0, 12.1, 14.0, 14.1, 15.0, 16.0, 17.0, 17.1, 18.0, 19.0, 19.1, 20.0, 20.1

Commenter	Commenter Number	Sections
Nuclear Energy Institute	MD08, 0084	2.1, 2.5, 3.0, 3.1, 3.3, 3.4, 4.3, 4.4, 5.0, 5.4, 6.0, 6.4, 7.4, 8.0, 8.4, 9.0, 9.4, 10.0, 10.2, 11.0, 12.0, 12.4, 13.0, 14.0, 14.3, 15.0, 15.2, 16.0, 16.2, 17.0, 17.1, 17.2, 18.0, 18.2, 18.3, 19.0, 20.0, 21.0
Nuclear Fuel Services	0078	2.3, 3.1, 3.4, 4.4, 5.4, 6.0, 7.0, 9.0, 10.0, 10.4, 11.0, 12.0, 13.0, 14.0, 15.0, 16.0, 17.0, 18.0, 18.2, 18.3, 19.0, 20.0
Nuclear Information and Resource Service	MD15, 0069, 0072	2.1, 2.2, 2.4, 2.5, 2.6, 5.0, 8.0, 15.0, 16.0, 17.0, 17.1, 19.0, 19.1, 20.0
Oregon State University	0054	2.1, 6.0, 6.1
Ortinger, Pat	0063	2.2
PECO Nuclear	0081	3.0, 3.1
Physicians for Social Responsibility Atlanta	AT23	2.2, 2.3, 2.6, 4.0
Port of Oakland	OA47	8.2, 21.0
Portland General Electric	0066	3.0, 3.2, 9.0, 10.0, 11.0, 12.2, 14.0, 15.0, 16.0, 16.2, 17.0, 20.0
Public Citizen	MD05, 0060, 0062	2.1, 2.2, 2.3, 2.4, 2.5, 4.0, 5.0, 5.1, 5.2, 6.0, 10.0, 14.0, 15.0, 16.0, 17.0, 17.1, 19.0, 19.1, 20.3
Shundahai Network	AT25	2.2, 2.4, 7.0
The Pennsylvania State University	0049	2.5, 3.0, 3.1, 3.4, 4.4, 5.4, 11.0, 21.0
Transport Logistics International/Columbiana Boiler Company	MD10	2.5, 6.3, 8.0, 10.0, 13.0, 15.0, 15.1, 15.2, 16.0, 16.3, 17.0, 17.1, 17.2
Tri-Valley CARES	OA41	2.2, 2.3, 2.4, 4.0, 4.1, 4.4, 5.0, 6.0, 6.4, 7.1, 8.0, 9.0, 9.4, 11.0, 12.0, 14.0, 19.0, 21.0
U.S. Department of Energy	MD12, MD13, MD14, 0065, 0091	2.1, 2.4, 3.0, 3.2, 3.3, 3.4, 4.0, 4.1, 4.2, 4.4, 5.1, 5.4, 6.0, 6.4, 7.0, 7.2, 8.0, 9.0, 10.0, 10.1, 11.0, 12.0, 12.4, 13.0, 14.0, 14.2, 15.0, 16.0, 16.1, 16.2, 17.0, 17.3, 18.0, 19.0, 19.1, 20.0

Commenter	Commenter Number	Sections
U.S. Department of the Army	0086	3.0, 3.4, 4.1, 4.2, 4.4
U.S. Department of Transportation	MD06, MD11, 0088, 0089	2.2, 2.4, 2.5, 5.0, 8.0, 8.3, 14.0, 15.0, 17.0, 17.3, 18.3, 19.0, 20.0
U.S. Nuclear Regulatory Commission	0052	2.1, 6.0, 6.1
Union of Concerned Scientists	0050, 0076	2.1, 2.2, 2.5, 3.0, 4.1, 5.0, 6.0, 6.1, 14.0, 15.0, 17.0, 18.0, 19.0
United States Enrichment Corporation	MD20, 0071	2.1, 2.5, 3.0, 3.1, 3.4, 4.4, 5.0, 5.4, 6.0, 6.3, 7.0, 7.1, 7.2, 8.0, 8.4, 9.0, 9.4, 10.0, 11.0, 12.0, 12.4, 13.0, 14.0, 14.3, 15.0, 16.0, 17.0, 18.2, 18.3, 19.0, 20.0
Virginia Power	0083	2.1, 3.0, 4.0, 4.4, 5.0, 7.0, 10.0, 11.0, 14.0, 17.0, 20.0
WAND	AT27, AT29, 0064	2.1, 2.2, 2.3, 2.6, 3.0, 4.0, 4.1, 5.1, 6.0, 7.0, 8.0, 8.1, 9.0, 10.0, 12.0, 13.0, 14.0, 14.1, 15.0, 16.0, 17.0, 17.1, 18.0, 19.0, 19.1, 20.0, 20.1
Western States Legal Foundation	OA44, 0085	2.2, 4.0, 4.4, 8.0, 16.0, 17.0, 19.0, 19.1
Womens Active for New Orleans/Women's Action for New Directions	AT30, 0082	2.2, 2.3, 2.4, 2.5, 4.0, 7.0, 7.1, 8.0, 9.0, 9.1, 10.0, 19.0, 20.0
World Nuclear Transport Institute	0079, 0093	2.1
Zirconium Environmental Committee	MD03	2.2, 2.3, 4.0

APPENDIX C

CROSS REFERENCE OF COMMENTERS
BY COMMENTER NUMBER

Commenter Number	Commenter	Organization Type
Rockville, Maryland Public Meeting (August 10, 2000)		
MD01	Connecticut Dept. Environmental Protection	State Government
MD02	Florida Dept. of Health, Bureau of Radiation	State Government
MD03	Zirconium Environmental Committee	Nuclear Industry
MD04	American Petroleum Institute	Citizen/Environmental Group
MD05	Public Citizen	Citizen/Environmental Group
MD06	U.S. Department of Transportation	Federal Government
MD07	Chem-Nuclear Systems/Nuclear Energy Institute	Nuclear Industry
MD08	Nuclear Energy Institute	Nuclear Industry
MD09	Airline Pilots Association	Professional Association
MD10	Transport Logistics International/Columbiana Boiler Company	Nuclear Industry
MD11	U.S. Department of Transportation	Federal Government
MD12	U.S. Department of Energy	Federal Government
MD13	U.S. Department of Energy	Federal Government
MD14	U.S. Department of Energy	Federal Government
MD15	Nuclear Information and Resource Service	Citizen/Environmental Group
MD16	Blue Ridge Environmental Defense League	Citizen/Environmental Group
MD17	AEA Technology QSA, Inc.	Nuclear Industry
MD18	Calvert Cliffs Nuclear Power Plant, Inc.	Nuclear Industry
MD19	Mallinckrodt Inc.	Nuclear Industry
MD20	United States Enrichment Corporation	Nuclear Industry
MD21	Falchi, Frank	Private Citizen
Atlanta, Georgia Public Meeting (September 20, 2000)		
AT22	NIRS Southeast	Citizen/Environmental Group
AT23	Physicians for Social Responsibility Atlanta	Citizen/Environmental Group
AT24	Human Race	Citizen/Environmental Group
AT25	Shundahai Network	Citizen/Environmental Group
AT26	Mallinckrodt Inc.	Nuclear Industry
AT27	WAND	Citizen/Environmental Group
AT28	Bastin, Clinton	Private Citizen
AT29	WAND	Citizen/Environmental Group
AT30	Womens Active for New Orleans	Citizen/Environmental Group

Commenter Number	Commenter	Organization Type
AT31	GA/DNR/EPD	State Government
AT32	Flemming, Bill	Private Citizen
AT33	Action for a Clean Environment	Citizen/Environmental Group
AT34	Ferguson, Tom	Private Citizen
AT35	Member of Audience	Private Citizen
AT36	Member of Audience	Private Citizen
AT37	Member of Audience	Private Citizen
AT38	Member of Audience	Private Citizen
AT39	Member of Audience	Private Citizen
AT40	Member of Audience	Private Citizen
Oakland, California Public Meeting (September 26, 2000)		
OA41	Tri-Valley CARES	Citizen/Environmental Group
OA42	J.L. Shepherd & Associates	Nuclear Industry
OA43	Clark County Department of Comprehensive Planning	Local Government
OA44	Western States Legal Foundation	Professional Association
OA45	J.L. Shepherd & Associates	Nuclear Industry
OA46	Fulk, Marion	Private Citizen
OA47	Port of Oakland	Local Government
NRC-Received Electronic and Hard Copy Comments		
0048	N/A	N/A
0049	The Pennsylvania State University	Educational Institution
0050	Union of Concerned Scientists	Citizen/Environmental Group
0051	GTS Duratek	Nuclear Industry
0052	U.S. Nuclear Regulatory Commission	Federal Government
0053	Attorney General's Office, State of New Mexico	State Government
0054	Oregon State University	Educational Institution
0055	AEA Technology QSA, Inc.	Nuclear Industry
0056	Barrowes, Steven C.	Private Citizen
0057	General Atomics	Nuclear Industry
0058	Frontier Technology Corporation	Nuclear Industry
0059	Georgia Public Service Commission	State Government
0060	Public Citizen	Citizen/Environmental Group

Commenter Number	Commenter	Organization Type
0061	Columbiana Boiler Company	Nuclear Industry
0062	Public Citizen	Citizen/Environmental Group
0063	Ortinger, Pat	Private Citizen
0064	WAND	Private Citizen
0065	U.S. Department of Energy	Federal Government
0066	Portland General Electric	Nuclear Industry
0067	J. L. Shepherd & Associates	Nuclear Industry
0068	Blue Ridge Environmental Defense League	Citizen/Environmental Group
0069	Nuclear Information and Resource Service	Citizen/Environmental Group
0070	Lincoln County/City of Caliente	Local Government
0071	United States Enrichment Corporation	Nuclear Industry
0072	Nuclear Information and Resource Service	Citizen/Environmental Group
0073	New England Coalition on Nuclear Pollution	Citizen/Environmental Group
0074	Environmentalists, Inc.	Citizen/Environmental Group
0075	Fabilli, Virginia	Private Citizen
0076	Union of Concerned Scientists	Citizen/Environmental Group
0077	New Mexico Environmental Evaluation Group	Citizen/Environmental Group
0078	Nuclear Fuel Services	Nuclear Industry
0079	World Nuclear Transport Institute	Nuclear Industry
0080	ASME International	Nuclear Industry
0081	PECO Nuclear	Nuclear Industry
0082	Women's Action for New Directions	Citizen/Environmental Group
0083	Virginia Power	Utility
0084	Nuclear Energy Institute	Nuclear Industry
0085	Western States Legal Foundation	Professional Association
0086	U.S. Department of the Army	Federal Government
0087	American Petroleum Institute	Citizen/Environmental Group
0088	U.S. Department of Transportation	Federal Government
0089	U.S. Department of Transportation	Federal Government
0090	Eureka County, Yucca Mountain Information Office	Local Government
0091	U.S. Department of Energy	Federal Government
0092	Clark County Department of Comprehensive Planning	Local Government

Commenter Number	Commenter	Organization Type
0093	World Nuclear Transport Institute	Nuclear Industry
0094	N/A	Private Citizen
0095	N/A	Private Citizen
0096	League of Women Voters of South Carolina	Citizen/Environmental Group

NUREG/CR-6713

Draft Regulatory Analysis of Major Revision of 10 CFR Part 71

Proposed Rule

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U.S. Nuclear Regulatory Commission
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ABSTRACT

This report presents the regulatory analysis 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 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. This report includes: (1) a summary of the findings, (2) a discussion of the regulatory options analyzed, (3) an assessment of the estimate values (benefits) and impacts (costs) 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 impacts. In fact, the analysis indicates that most of the changes will have negligible effects or result in slight increases in values.

TABLE OF CONTENTS

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	Indpendent 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 3.7×10^{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 Regulatory Analysis of the U.S. Nuclear Regulatory Commission's (NRC or Commission'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 International Atomic Energy Agency (IAEA) (*Regulations for the Safe Transport of Radioactive Material*, IAEA Safety Standards Series No. TS-R-1, June 2000), and the U.S. Department of Transportation's 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 safe 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 preferred option for each of the 19 individual issues described in Chapter 2 and analyzed in Chapter 3 of this document. In the paragraphs following this table, further description of the values and impacts of the options is provided. Chapters 2 and 3 provide additional detail on the changes and associated values and impacts.

For purposes of this analysis, the proposed rulemaking has been grouped into 19 different potential changes to Part 71, which could be adopted either all together as one list or independently in a partial list. None of the 19 potential changes, which are described and evaluated in turn in the remainder of this report, are expected to result in significant impacts (costs), whether promulgated individually or together. In fact, most of the changes would have negligible effects or result in slight increases in values (benefits). In particular, the following changes are primarily administrative in nature 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 2.1.1 and 3.3.1);

- Revision of A_1 and A_2 (see Sections 2.1.3 and 3.3.3);

- A new requirement to display the Criticality Safety Index on shipping packages of fissile material (see Sections 2.1.5 and 3.3.5);

Table ES-1. Summary of Preferred Options

Technical Issue		Preferred Option
1.	Changing Part 71 to the International System of Units (SI) Only	Option 1 (No Action)
2.	Radionuclide Exemption Values	Option 2
3.	Revision of A_1 and A_2	Option 2
4.	Uranium Hexafluoride Package Requirements	Option 2
5.	Introduction of the Criticality Safety Index Requirements	Option 2
6.	Type C Packages and Low Dispersible Material	Option 1 (No Action)
7.	Deep Immersion Test	Option 2
8.	Grandfathering Previously Approved Packages	Option 2
9.	Changes to Various Definitions	Option 2
10.	Crush Test for Fissile Material Package Design	Option 2
11.	Fissile Material Package Designs for Transport by Aircraft	Option 2
12.	Special Package Authorizations	Option 2
13.	Expansion of Part 71 Quality Assurance Requirements to Certificate of Compliance (CoC) Holders	Option 2
14.	Adoption of ASME Code	Option 1 (No Action)
15.	Change Authority	Option 2
16.	Fissile Material Exemptions and General License Provisions	Option 2
17.	Double Containment of Plutonium (PRM-71-12)	Option 2
18.	Contamination Limits as Applied to Spent Fuel and High Level Waste (HLW) Packages	For information only. No options identified.
19.	Modifications of Event Reporting Requirements	Option 2

- A provision to “grandfather” older shipping packages under the Part 71 requirements in existence when their Certificates of Compliance (CoC) were issued (see Sections 2.1.8 and 3.3.8);

- Procedures for approval of special arrangements for shipment of special packages (see Sections 2.2.1 and 3.4.1);

- Modifications to Event Reporting Requirements (see Sections 2.2.8 and 3.4.8).

IAEA-Related Changes

The proposed changes to harmonize Part 71 with TS-R-1 are expected to result in a net benefit in terms of regulatory efficiency, which will result in reduced costs. In addition, the change to various definitions would result in clarification of the requirements, thus slightly reducing burden for licensees. In whole, however, each potential change will result in mixed, but overall minor, effects. Due to a lack of quantitative data it is not possible to describe the net value or impact of each potential change in terms of costs. The following paragraphs describe the preferred option for each issue, and further provide a qualitative summary of the values and impacts associated with the changes.

Changing Part 71 to the International System of Units (SI) Only. The preferred option is Option 1, the No-Action alternative. As described in section 3.3.1, the change to the use of SI units only would result in minor values and impacts. While regulatory efficiency would be increased, the change could result in additional exposure of workers and the public to radiation due to possible flawed conversions from SI units to customary units. However, the frequency to which these individuals are exposed to radiation is not expected to increase because transportation accident frequency would not increase as a result of this change. Finally, additional costs would be incurred by licensees, the NRC, and other government agencies to implement the change.

Radionuclide Exemption Values. The preferred option is Option 2. Under this option, NRC would adopt the radionuclide exemption values contained in TS-R-1. Adoption of the TS-R-1 radionuclide exemption values is expected to have minor benefits as well as impacts (see Section 3.3.2). Licensees may incur some minor administrative costs as well as costs to determine whether exemption levels are met. However, these costs are outweighed by the increase in regulatory efficiency between regulatory agencies and the facilitation of international shipments of exempted packages.

Revision of A_1 and A_2. The preferred option is Option 2. Option 2 recommends the adoption of the newly revised A_1 and A_2 values in TS-R-1, with the exception of the values for ^{99}Mo and ^{252}Cf. Overall, it is expected that there would be a slight benefit in terms of potential exposure as a result of changing to the more refined values contained in TS-R-1 (see Section 3.3.3). Minor costs could be realized by licensees, the NRC, and other government agencies as a result of this change. In particular, licensees could incur implementation costs if licensees must revise various aspects of shipping programs or modify shipping processes to assure compliance with the proposed A_1 and A_2 values. These one-time costs, however, are expected to be minimal and are outweighed by the benefit of reduction in potential exposure.

Uranium Hexafluoride (UF_6) Package Requirements. Option 2 is the preferred option. NRC would promulgate a new section 71.55(g), consistent with the UF_6 exception requirements contained in TS-R-1, while restricting the use of this exception to packages with a maximum enrichment of 5 weight percent ^{235}U. Adoption of Option 2 (see Section 3.3.4) is expected to have mixed effects. Risk of exposure is expected to decrease slightly, while implementation and operational costs for licensees are expected to increase. Regulatory efficiency also would show a slight increase with respect to international shipments, and thus provide a slight net reduction in costs to the NRC. Further, damage to the environment will be less likely to occur due to radiation in the event of a vehicular accident that results in a fire. Overall, the net reduction in risk, potential exposure, and environmental damage is expected to be greater than the additional implementation and operational costs for licensees.

Introduction of the Criticality Safety Index Requirements. Option 2, the preferred option, would require labels indicating both the Transportation Index (TI) and the Criticality Safety Index

(CSI) for transport of fissile material packages. The addition of the CSI in transport (see Section 3.3.5) is expected to result in minor implementation and operational costs for licensees, while providing a benefit to emergency responders in the case of transportation accidents. Additional benefits would be realized by the NRC for international shipments because regulatory efficiency would be increased.

Type C Packages and Low Dispersible Material. The preferred option is Option 1, the no-action alternative. Under this option, NRC would not adopt the Type C package or low dispersible radioactive material concepts for air transportation contained in TS-R-1. Incorporation of these concepts would result in an increase in regulatory efficiency as a result of the adoption of the TS-R-1 requirements, which would facilitate international shipments (see Section 3.3.6). Additional resource costs would, however, be incurred by NRC and the licensees. These additional costs to licensees would include implementation costs for the design of new packages to meet the Type C requirements rather using existing Type B packages. However, NRC currently has in place, requirements governing domestic shipments of plutonium by air (which would be shipped in the new Type C packages), and because there are very few shipments of this nature, there is little need for this new type of package design in domestic commerce. As a result, the potential impacts outweigh the benefits of adopting these concepts.

Deep Immersion Test. Option 2 is the preferred option. Option 2 recommends revising Part 71 to require an enhanced water immersion test for transporting packages containing radioactive materials with activity greater than 10^5 A_2. Requiring an enhanced deep immersion test (see Section 3.3.7) would improve regulatory efficiency by bringing U.S. regulations in harmony with the standards contained in TS-R-1. This would improve the efficiency of handling imports and exports and would make U.S. standards compatible with other IAEA member states. However, the requirement could result in costs to licensees as they test and certify packages to the proposed standard. The NRC also may incur costs for developing procedures, reviewing and approving test results, and recertifying packages. Alternatively, the proposed change may reduce impacts to public health in the case of an accident. Adoption of the change would prevent the possible expenses of restricting the accident area (to prevent users such as boaters or fishers from entering the vicinity) and remediating any contamination of the marine environment. The net effect is that the values of adopting Option 2 outweigh the potential costs to licensees.

Grandfathering Previously Approved Packages. The preferred option is Option 2. Option 2 would modify Part 71 to phase out packages approved under IAEA Safety Series 6 (1967). This option would include a 3-year transition period for the grandfathering provision on packages approved under Safety Series 6. In addition, packages approved under Safety Series 6 (1985) would not be allowed to be fabricated after December 31, 2006. The purpose of grandfathering is to minimize the costs and impacts of implementing changes in the regulations on existing package designs and packagings. The proposed revisions related to grandfathering of previously approved packages (see Section 3.3.8) would result in enhanced regulatory efficiency by bringing NRC's requirements in harmony with those contained in TS-R-1. The proposed change would, however, result in implementation costs to the NRC because the Agency would have to revise regulatory guides and NUREG-series documents. The change could result in implementation and operation costs to Agreement States if they adopt and implement parallel requirements. While minimal costs may be realized by licensees, it is expected that, the overall expected benefits outweigh the additional potential costs.

Changes to Various Definitions. Option 2 is the preferred option. Under Option 2, NRC would add various definitions to 10 CFR 71.4 and modify existing definitions to ensure compatibility with definitions found in TS-R-1, and to improve clarity in NRC regulations. These changes would provide greater internal consistency with other NRC regulations and greater compatibility with TS-R-1, thus improving regulatory efficiency (see Section 3.3.9). By modifying existing definitions and adding new definitions, licensees also will benefit through more effective understanding of the requirements of Part 71. The changes would result in implementation costs to the NRC, with respect to revisions necessary to regulatory guides and NUREG-series documents. The changes could affect Agreement States in a similar fashion. However, the increased regulatory efficiency and greater clarification for licensees outweigh the costs to NRC.

Crush Test for Fissile Material Package Design. The preferred option is Option 2. Option 2 recommends adoption, in part, of the TS-R-1 requirement for a crush test for radioactive contents of Type B packages greater than 1000 A_2. In addition, Option 2 would extend the crush test requirement to fissile material package designs regardless of the level of radioactive contents. Adoption of Option 2 (see Section 3.3.10) would result in enhanced regulatory efficiency by correcting inconsistencies between Part 71 requirements and TS-R-1. However, further information on the impact of the TS-R-1 requirement for fissile material package testing is required. The change also would result in implementation costs imposed on licensees to demonstrate compliance and may lead to the redesign of packages. Lastly, the change would result in NRC implementation costs associated with modifying the regulations and revising guidance documents.

Fissile Material Package Designs for Transport by Aircraft. Option 2, the preferred option, would result in the adoption of the TS-R-1 criticality evaluation requirements for shipment of fissile packages by aircraft. Option 2 would provide the NRC with the regulatory framework for approving package designs that will be used internationally (see Section 3.3.11). NRC costs would be reduced while maintaining consistency with international requirements, thus enhancing regulatory efficiency. Shippers will be required to meet these requirements even if the NRC does not adopt them, because the International Civil Aviation Organization (ICAO) is adopting regulations consistent with TS-R-1 effective July 1, 2001; thus, no additional costs are imposed on licensees. Further, some U.S. domestic air carriers are already requiring compliance with the ICAO regulations even for domestic shipments.

NRC-Initiated Changes

Special Package Authorizations. Option 2 is the preferred option. Under this option, NRC would incorporate new regulations in Part 71 that address approval for shipment of special packages and that demonstrate an acceptable level of safety. Incorporation of the new regulations (see Section 3.4.1) would result in enhanced regulatory efficiency by standardizing the requirements for special package approval to provide greater regulatory certainty and clarity. It also would ensure consistent treatment among licensees requesting authorization for shipment of special packages. Since the change is expected to streamline the process for handling nonstandard packages, considerable savings would be realized, both in NRC staff time and licensee staff time. Further, the regulations would require a demonstration of an acceptable level of safety for shipment of these packages, and the result is expected to be a decreased risk of radiation exposure to the public and workers as opposed to the shipment alternatives.

Expansion of Part 71 Quality Assurance Requirements to Certificate of Compliance (CoC) Holders. The preferred option is Option 2. Option 2 recommends that NRC explicitly subject CoC holders and CoC applicants to the requirements contained in 10 CFR Part 71. NRC also would add recordkeeping and reporting requirements for CoC holders and CoC applicants. Adoption of the change for bringing CoC holders and applicants under authority of Part 71 (see Section 3.4.2) would ensure that Part 71 is more consistent with other NRC regulations (thus enhancing regulatory efficiency) in that certificate holders and applicants for a CoC would be responsible for the behavior of their contractors and subcontractors. CoC holders and applicants for a CoC will incur costs associated with understanding and implementing the new regulations, as well as in preparing and submitting reports. NRC will incur costs associated with supervising certificate holders and applicants for a CoC and maintaining and reviewing the records for certificate holder submittals. Overall, the increased efficiency and improved consistency with other NRC regulations outweigh the potential costs to CoC holders and applicants.

Adoption of ASME Code. Option 1, the No-Action alternative, is the preferred option. The adoption of the changes to incorporate the ASME Code (see Section 3.4.3) would result in additional implementation and operational costs to licensees. Adoption of this code is expected to result in some benefit with respect to public health. However, because of the potential for the ASME code to be revised over the next several years, adoption at this time could result in additional costs to both NRC and licensees should the regulations need to be revised in the future.

Change Authority. Option 2 is the preferred option. Option 2 would revise Part 71 to add a new general license section for dual-purpose packages (i.e., packages designed for both shipment and storage of spent nuclear fuel) and a new subpart which provides requirements for submission, approval, and amendment of these new packages. In addition to providing a new process for approving dual-purpose transportation packages, the new requirements would provide authority for certificate holders to make changes to a dual-purpose package design without prior NRC approval. The subpart also would include new requirements for submitting and updating a final safety analysis report describing the package’s design. Adoption of this change authority would result in implementation and operational costs to licensees associated with understanding and implementing this change in licensing requirements. Licensees and CoC holders also will incur costs when submitting reports every 24 months. However, the licensees and CoC holders will realize cost savings associated with preparing license amendments and paying fees to NRC that are required under current regulations. NRC will incur some costs in reviewing reports submitted by licensees and CoC holders, but these costs will be offset by increased regulatory efficiency resulting from a clearer and more consistent interpretation between NRC, licensees, and CoC holders. As a result, NRC would be able to better direct resources that would be spent reviewing license amendments to areas where measurable improvements in safety can be made.

Fissile Material Exemptions and General License Provisions. The preferred option is Option 2. Option 2 recommends adoption of a subset of the 17 recommendations contained in NUREG/CR-5342, *Assessment and Recommendations for Fissile-Material Packaging Exemptions and Licenses Within 10 CFR Part 71*. The effects of adoption of the recommended changes would be both positive and negative, depending on the specific recommendation (see Section 3.4.5). Recommendations 1, 2, and 5 would enhance regulatory efficiency due to increased clarity of NRC regulations. Recommendations 3, 4, 6, 9, and 12 would increase costs to licensees. Recommendations 7, 8, 10, 13, 14, 15, and 16 would eliminate the potential for criticality accidents, which would, in turn, yield environmental and public health and safety benefits. Finally, recommendations 11 and 17 would result in savings to licensees.

Double Containment of Plutonium. Option 2 is the preferred option. Under Option 2, NRC would adopt, in part, the recommended action of Petition PRM-71-12. Specifically, 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. Adoption of the change for the double containment of plutonium (see Section 3.4.6) would result in implementation and operational savings for licensees and other government agencies (DOE). However, because the NRC believes that the current Type B package requirements are sufficient to protect human health and safety, the change is not expected to result in increased costs as a result of exposure to radiation during an accident and may result in decreased worker exposure.

Contamination Limits as Applied to Spent Fuel and High Level Waste (HLW) Packages. No options have been identified for this issue. 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 in this document and no regulatory analysis conducted.

Modification of Event Reporting Requirements. The preferred option is Option 2. Option 2 recommends revising 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. The NRC also would 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. Adoption of the conforming change to Part 71 for event reporting requirements (see Section 3.4.8) would result in an increase in regulatory efficiency within NRC. There would be a one-time implementation cost for licensees for revising procedures and for training. Additionally, licensees would benefit due to a reduction in the recurring annual reporting burden as a result of reducing the efforts associated with reporting events of little or no risk or safety significance. It is anticipated that the NRC's recurring annual review efforts for telephone notifications and written reports will not be significantly reduced.

1. Introduction

The U.S. Nuclear Regulatory Commission (NRC or Commission) has initiated a proposed rulemaking to: (1) harmonize its transportation regulations found in 10 CFR Part 71 with the most recent transportation standards established by the International Atomic Energy Agency (IAEA) in TS-R-1 and the U.S. DOT's regulations at 49 CFR; and (2) address the Commission's goals for risk-informed regulations and eliminating inconsistencies with other regulatory approaches.

This document presents ICF's Regulatory Analysis of the regulatory options being considered by NRC. The purpose of this regulatory analysis is to evaluate the costs and benefits associated with the regulatory changes being considered by NRC. Although no statutory mandates exist for the NRC to conduct regulatory analyses, the Commission voluntarily began performing these types of studies in 1976 to ensure that all regulatory burdens will achieve intended regulatory objectives with minimal impacts to licensees. Hence, the NRC considers the regulatory analysis process an integral part of its statutory mission to ensure the protection of public health and safety, property, environmental quality, and national defense and security from civilian uses of nuclear materials.

The remainder of the introduction is divided into two sections. Section 1.1 provides background information on the history, extent, and relationship of this problem; and Section 1.2 states the objectives of the rulemaking.

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, 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 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.

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 in response to the issue paper by mail, and electronic comments received on the NRC web site, were considered in preparing this Regulatory Analysis.

IAEA Transportation Standards

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, 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 transport 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 the revision of the DOT Hazmat Regulations in 1968. This revision also was the basis for a major revision to the NRC's 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. The designation of ST-1 was changed, along with minor revisions to the document, to TS-R-1 in June 2000. NRC is currently working to harmonize 10 CFR Part 71 with the latest IAEA TS-R-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 October 19, 1998, the Commission decided in SRM-SECY-98-168 to promulgate a rule to conform 10 CFR Part 71 with TS-R-1. Accordingly, the NRC staff prepared a draft rulemaking plan to be supported by a Regulatory Analysis and an Environmental Assessment.

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[1] Grella, A. "Summary of the Regulations Governing Transport of Radioactive Materials in the USA." RAMTRANS, Volume 9. No.4, pp. 279-292 (1999).

[2] Ibid.

Fissile Material Shipments and Exemptions

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. Available knowledge of radioactive material transport and historic practice have indicated that little or no regulatory oversight is 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 allow licensees to make shipments without first seeking approval from the NRC.

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 exemptions appearing in 10 CFR 71.53 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 highly-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.

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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."

[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 next 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.

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 the revisions to 10 CFR Part 71, with the objectives of:

- 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:

<u>Packaging and Transportation</u>

- 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.

<u>Other Regulations</u>

- 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.

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[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.

- 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.

1.2 Objectives of the Proposed Rulemaking

The objectives of the rulemaking are to both (1) harmonize NRC's transportation regulations with other regulatory agencies (DOT, IAEA), and (2) implement 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.

2. Identification of Alternative Regulatory Options

NRC is considering 19 changes to its radioactive material transportation regulations. The first 11 changes 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 eight changes are regulatory 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 19 changes [Section 2.2.5] is based in part on the specific recommendations presented in NUREG/CR-5342.)

For each of the 19 changes, this Regulatory Analysis considers two regulatory options. Option 1 is the No-Action Alternative. Option 2 is based in part on TS-R-1, Safe Transportation Standards. The discussion that follows assumes a familiarity with and understanding of TS-R-1. Option 2 also is based on Commission direction for staff to evaluate additional changes to reduce regulatory burden on licensees.

For the changes to fissile material license provisions, Option 2 is 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 Regulatory Analysis 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.

The potential changes to 10 CFR Part 71 are summarized in Table 2-1 below and are described in more detail in the sections that follow.

Table 2-1. List and Summary Description of Potential 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 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.

Table 2-1. List and Summary Description of Potential Changes to 10 CFR Part 71 (continued)

Technical Issue	Summary Description of Potential Requirements
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.
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.

2.1 Actions to Harmonize NRC Transportation Regulations with IAEA Safe Transport Standards

2.1.1 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 United States 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.

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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.

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.

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.

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[9] Based on telephone conversations with Mr. Felix Killar, NEI on August 30, 1999 and Ms. Lynette Hendricks, NEI on August 31, 1999.

Option 2: Amendments to 10 CFR Part 71

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.

2.1.2 Radionuclide Exemption Values

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 1×10^{-1} to 1×10^{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 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:

Option 1: No-Action Alternative

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: Amendments to 10 CFR Part 71

Under Option 2, NRC would adopt, in 10 CFR Part 71, IAEA's radionuclide-specific exemption values for all radionuclides.

2.1.3 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 also are 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: Amendments to 10 CFR Part 71

Under Option 2, NRC would revise Part 71 to incorporate the TS-R-1 A_1 and A_2 values maintaining the current exceptions for ^{252}Cf and ^{99}Mo.

2.1.4 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 also are 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

[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.

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 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.

Option 1: No-Action Alternative

Under the No-Action Alternative (Option 1), NRC would not modify Part 71 to incorporate the TS-R-1 UF_6 requirements.

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[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.

Option 2: Amendments to 10 CFR Part 71

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.

2.1.5 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.

Option 1: No-Action Alternative

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.

Option 2: Amendment to 10 CFR Part 71

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.

2.1.6 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 also must 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: Amendments to 10 CFR Part 71</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.

2.1.7 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_2, 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_2 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: Amendments to 10 CFR Part 71

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.

2.1.8 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: Amendments to 10 CFR Part 71

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.

2.1.9 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 modifications addressed in the proposed rule.

Option 2: Amendments to 10 CFR Part 71

Under Option 2, NRC proposes to 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 proposal would add or modify the following:

- 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

2.1.10 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: (1) having 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 also is 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: Amendments to 10 CFR Part 71

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.

2.1.11 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

[13] The ST-1 imposition of Type C and LDM requirements (see Section 2.1.6) 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 also was a need to consider how these more severe test conditions should be applied to fissile package designs transported by air.

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

  (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.

Option 2: Amendments to 10 CFR Part 71

Under Option 2, the NRC would include this new TS-R-1 require for an additional criticality evaluation, in a new paragraph 71.55(f), that only applies to air transport.

2.2 NRC-Specific Changes

2.2.1 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: Amendments to 10 CFR Part 71</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 also would address regulatory and environmental conditions and requirements that are characteristic to the nuclear industry in the U.S.

2.2.2 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 also will 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 action, they also would 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: Amendments to 10 CFR Part 71</u>

Under Option 2, NRC would explicitly subject CoC Holders and CoC applicants to the requirements contained in 10 CFR Part 71. NRC also would add recordkeeping and reporting requirements for CoC Holders and CoC applicants.

2.2.3 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 also is 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.

Option 1: No-Action Alternative

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.

Option 2: Amendments to 10 CFR Part 71

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 also would apply to dual-purpose casks.

2.2.4 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) the subsequent Commission 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.

<u>Option 1: No-Action Alternative</u>

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.

<u>Option 2: Amendments to 10 CFR Part 71</u>

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 also would include new requirements for submitting and updating a final safety analysis report describing the package's design.

2.2.5 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 (section71.18)
    - 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.

Option 1: No-Action Alternative

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.

Option 2: Amendments to 10 CFR Part 71

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 changes to Part 71, which are summarized in Table 2-2 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.

2.2.6 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 2-2. 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.

Table 2-2. Recommendations and Changes Related to Fissile Material Packaging Exemptions and General Licenses (Continued)

NUREG/CR-5342 Recommendation	Summary of Recommended Action
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 also would 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.
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 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 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 action also will be revised such that packages shipped under these sections should use the criticality control transport index determined by those sections. The action would revise the phrase “[n]ot in excess of 10” instead of the phrase “[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.

Table 2-2. Recommendations and Changes Related to Fissile Material Packaging Exemptions and General Licenses (Continued)

NUREG/CR-5342 Recommendation	Summary of Recommended Action
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.
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.

Table 2-2. Recommendations and Changes Related to Fissile Material Packaging Exemptions and General Licenses (Continued)

NUREG/CR-5342 Recommendation	Summary of Recommended Action
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 this 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) also should 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 (20 Ci) 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: Amendments to 10 CFR Part 71

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.

2.2.7 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.

No regulatory change is proposed at this time. Therefore, no regulatory options have been identified. The above discussion is for information purposes only.

2.2.8 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>

[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: Amendments to 10 CFR Part 71

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 also would 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.

3. Analysis of Values and Impacts

This chapter examines the values and impacts expected to result from NRC's proposed rulemaking. It is divided into four main sections. Section 3.1 identifies attributes that are and are not expected to be affected by the rulemaking. Section 3.2 describes how values and impacts were analyzed. Section 3.3 examines the projected values and impacts associated with the actions to harmonize NRC's transportation regulations with the IAEA's latest safety standards. Finally, Section 3.4 examines the projected values and impacts associated with the NRC-specific actions.

NRC's proposed rulemaking would modify 10 CFR Part 71 to incorporate the IAEA safe transportation standards contained in TS-R-1 and other changes, in addition to the recommendations contained in NUREG/CR-5342. Each of the actions would result in certain values and/or impacts. Thus, the values and impacts of NRC's proposed rulemaking as a whole consist of the sum of all values and impacts associated with each of the actions. For many of the affected attributes, the values and impacts are expected to be negligible. These values and impacts, therefore, are difficult to estimate, and have not been quantified in this analysis.

3.1 Identification of Affected Attributes

This section identifies and describes the factors within the public and private sectors that the regulatory alternatives (discussed in Section 2) are expected to affect. These factors were classified as "attributes," using the list of attributes provided by NRC in Chapter 5 of its *Regulatory Analysis Technical Evaluation Handbook*.[15] Each attribute listed in Chapter 5 was evaluated, and the basis for selecting those attributes expected to be affected by the action is presented in the balance of this section.

Affected Attributes

- Public Health (Accident) -- Changes to radiation exposures to the public, due to changes in accident frequencies and accident consequences, could result from the proposed rule. The regulatory options could both alter the number of shipments (thereby altering the accident frequency) and reduce the likelihood of occurrences of criticality (thereby reducing accidental consequences).

- Occupational Health (Accident) -- Changes to radiation exposures to workers, due to changes in accident frequencies and accident consequences, could result from the proposed rule. The regulatory options could both alter the number of shipments (thereby altering the accident frequency) and reduce the likelihood of occurrences of criticality (thereby reducing accidental consequences).

- Occupational Health (Routine) -- Changes to radiation exposures to workers during normal packaging and transportation operations could result from the proposed rule. The regulatory options could alter the number of packages or shipments, thereby altering the number of workers exposed or the duration of the exposure.
- Offsite Property -- Effects on offsite property, due to changes in accident frequencies and accident consequences, could result from the action. The regulatory options could

[15] *Regulatory Analysis Technical Evaluation Handbook*, *Final Report*, NUREG/BR-0184, Office of Nuclear Regulatory Research, January 1997.

both alter the number of shipments (thereby altering the accident frequency) and reduce the likelihood of occurrences of criticality (thereby reducing accidental consequences).

- Onsite Property -- Effects on onsite property (direct and indirect), due to changes in accident frequencies and accident consequences, could result from the action. The regulatory options could both alter the number of shipments (thereby altering the accident frequency) and reduce the likelihood of occurrences of criticality (thereby reducing accidental consequences).

- Industry Implementation -- The regulatory options would result in implementation costs and savings to industry if industry must evaluate and/or enact changes to ensure that its operating procedures will comply with the actions.

- Industry Operation -- The regulatory options would result in industry operation costs and savings to industry if industry must alter its current packaging and shipping procedures to comply with the action.

- NRC Implementation -- The regulatory options would result in NRC implementation costs and savings to put the actions into operation. Specifically, NRC would incur implementation costs to revise guidance documents, and where applicable, develop new guidance.

- NRC Operation -- The regulatory options would result in NRC operation costs or savings if the number of shipments requiring specific NRC approval changes (e.g., the number of shipments that fail to qualify for the fissile exemption and the general licenses).

- Regulatory Efficiency -- The requirements would be expected to result in enhanced regulatory efficiency by clarifying the meaning and applicability of specific terms and requirements, increasing the level of consistency among different regulations, and reducing the potential for noncompliance.

- Environmental Considerations -- Effects on the environment, due to changes in accident frequencies and accident consequences, could result from the action. The regulatory options could both alter the number of shipments (thereby altering the accident frequency) and reduce the likelihood of occurrences of criticality (thereby reducing accidental consequences). These environmental effects are being addressed separately in the Environmental Assessment being developed in support of the proposed rulemaking.

- Other Government -- The regulatory options could affect implementation and operation costs of DOE, to the extent that its material shipments must comply with NRC regulations. The regulatory options also could affect implementation and operation costs of Agreement States if they must enact and implement parallel requirements. The regulatory options would not be expected to affect implementation or operation costs of DOT.

- Improvements in Knowledge -- The regulatory options could result in improved data collection that may ultimately result in more robust risk assessments and safety evaluations (i.e., less uncertainty) and, consequently, in improvements in regulatory policy and regulatory requirements.

Attributes *Not* Affected

- Public Health (Routine) -- No significant changes are expected with respect to routine radiation exposures to the public. Even if the number of shipments of radioactive materials significantly increases or decreases as a result of the rule, the change in exposure to members of the public as a result of routine shipments would be negligible.

- Safeguards and Security Considerations -- The regulatory options, if they alter the costs associated with accepting or downblending weapons-grade uranium from the former Soviet Union, could have some effect on security considerations. The magnitude of this effect is likely to be small, however, due to the U.S. government's role in funding these operations.

- General Public -- The action is not expected to have any effects on the general public.

- Antitrust Considerations -- The action is not expected to have any antitrust effects.

3.2 Analytical Methodology

This section describes the process used to evaluate values and impacts associated with the regulatory options. The *values* (benefits) of the rule include any desirable changes in affected attributes (e.g., improved public health due to a reduced potential for criticality) while the *impacts* (costs) include any undesirable changes in affected attributes (e.g., increased staff requirements to conduct NRC operations). As described in Section 3.1, the attributes expected to be affected include the following:

- Public Health (Accident)
- Occupational Health (Accident)
- Occupational Health (Routine)
- Offsite Property
- Onsite Property
- Industry Implementation
- Industry Operation
- NRC Implementation
- NRC Operation
- Regulatory Efficiency
- Environmental Considerations
- Other Government
- Improvements in Knowledge

For many of these attributes, the nature or cause of a value or impact is straightforward. For example, values and impacts associated with the attribute "NRC operations" should result from, respectively, either a decrease or increase in the number of NRC staff hours (or other NRC resources) required to oversee the Part 71 requirements on a day-to-day basis. Similarly, values and impacts associated with the attribute "regulatory efficiency" should result from changes to the overall clarity, consistency, or level of consolidation of applicable regulations.

The overall value or impact for some attributes, however, results from the interaction of several influencing factors. For example, a regulatory option that increases the number of packages and/or shipments required of licensees could simultaneously (1) reduce the potential for criticality and (2) increase the potential for routine radiological exposure. In this case, it would be the *net effect* of the influencing factors (i.e., criticality potential and radiological exposure) that would govern whether an overall value or impact would result for several affected attributes, including public health, occupational health, on- and off-site property, and environmental considerations. Similarly, a single regulatory option could affect licensee costs in multiple ways (e.g., it might conceivably increase packaging and shipping costs but decrease costs associated with making transport index calculations).

Ideally, a value-impact analysis quantifies these net effects and calculates the overall values and impacts of each regulatory option. This requires a baseline characterization of the transportation universe, including factors such as the number of licensees affected, the number of shipments and packages affected, the types of packaging used, the transportation method, and the transportation distance. Data availability is a severely limiting factor for the purposes of establishing a baseline characterization of the affected universe.

<u>Data Collection Activities</u>

In support of the development of the value-impact analysis, ICF undertook a significant data collection effort. The first step in the data collection was to determine specific data needs to support the analysis of values and impacts for each of the actions that, in total, make up each of the regulatory options. Specifically, ICF identified the following types of information necessary to develop the value-impact analysis:

Baseline Information

- Number of exempt packages
- Number of non-exempt packages
- Number of exempt shipments
- Number of non-exempt shipments
- Cost per exempt package
- Cost per non-exempt package
- Cost per exempt shipment
- Cost per non-exempt shipment
- Average number of packages per exempt shipments
- Average number of packages per non-exempt shipment

Information for Each action

- 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
- Change in cost per exempt package
- Change in cost per non-exempt package
- Change in cost per exempt shipment
- Change in cost per non-exempt shipment
- Average number of packages per exempt shipment

- Average number of packages per non-exempt shipment
- Cost to clean up and repair criticality accidents
- Cost to clean up and repair traffic accidents
- Change in time required for record-keeping/reporting
- Change in time for regulatory determinations/calculations
- Change in time for regulatory review

ICF conducted numerous 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 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 change in the number of packages/shipments and associated costs for each of the actions. The questions developed for this survey are listed in Appendix C. 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 actions for fissile materials.

3.3 Values and Impacts of actions to Harmonize 10 CFR Part 71 with IAEA TS-R-1

3.3.1 Changing Part 71 to the International System of Units (SI) Only

Values and Impacts of Option 1

Under the No-Action alternative (Option 1), NRC licensees and applicants would continue to use their preferred system of measurement for completing shipping papers and SI units for completing labels used in the transportation of radioactive materials. Thus, no values or impacts would result from Option 1.

Although an increase in the current number of flawed conversions or accident rates within the U.S. is not expected under Option 1, 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.

Values and Impacts of Option 2

Under Option 2, NRC would require the use of the International System of Units (SI), also known as the metric system, in shipping papers and labels used in the transportation of radioactive materials. By doing this, the units in shipping papers and labels associated with the packaging and transportation of radioactive materials would be consistent with the units used in the IAEA and guidance documents associated with IAEA.

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)).

The following attributes are expected to be affected by adoption of this action:

- Public Health (Accident) – Changes in radiation exposures to the public, due to changes in accident frequencies and accident consequences, could result from the change. The change would require, in some instances, conversion from customary units to SI units in order to satisfy Part 71 reporting requirements. Thus, radiation exposure to the public may change due to possible flawed unit conversions. In addition, the use of SI units only may be a safety issue in an emergency if responders are unfamiliar with the SI system. An estimation of the values/impacts associated with this attribute will be completed in concurrence with the Environmental Assessment being developed in support of this rulemaking.

- Occupational Health (Accident) – Changes in radiation exposures to workers, due to changes in accident frequencies and accident consequences, could result from the change. The change would require, in some instances, conversion from customary units to SI units in order to satisfy Part 71 reporting requirements. Thus, radiation exposure to workers may change due to possible flawed unit conversions. In addition, the use of SI units only may be a safety issue in an emergency if responders are unfamiliar with the SI system. An estimation of the values/impacts associated with this attribute will be completed in concurrence with the Environmental Assessment being developed in support of this rulemaking.

- Offsite Property – Effects on offsite property, due to changes in accident frequencies and accident consequences, could result from the change. The change would require, in some instances, conversion from customary units to SI units in order to satisfy Part 71 reporting requirements. Thus, accident frequencies and offsite property consequences resulting from the occurrence of an accident may increase due to possible flawed unit conversions. An estimation of the values/impacts associated with this attribute will be completed in concurrence with the Environmental Assessment being developed in support of this rulemaking.

- Onsite Property – Effects on onsite property, due to changes in accident frequencies and accident consequences, could result from the change. The change would require, in some instances, conversion from customary units to SI units in order to satisfy Part 71 reporting requirements. Thus, accident frequencies and onsite property consequences resulting from the occurrence of an accident may increase due to possible flawed unit conversions. An estimation of the values/impacts associated with this attribute will be completed in concurrence with the Environmental Assessment being developed in support of this rulemaking.
- Industry Implementation -- The change would result in implementation costs to industry sectors currently using customary units (e.g., companies who ship spent fuel, regular fuel, and/or low-specific activity material to destination sites within the U.S.).

- Industry Operation – The change would result in additional operational costs to sectors of industry currently using customary units. These sectors would have to convert from customary units to SI units, altering their current procedures in completing shipping papers and labels used in the packaging and transportation of radioactive materials.

- Other Government – The change could affect implementation and operation costs of Agreement States because they would have to adopt and implement parallel requirements. The change also could affect DOE if it currently submits information in customary units. It is expected, however, that DOE submits data in SI units. In addition, the change could affect DOT's implementation costs, if regulations in 49 CFR 172.202 (shipping papers) were revised to be consistent with this change. However, the change is not expected to affect DOT's operation costs.

- Regulatory Efficiency – The change is expected to result in enhanced regulatory efficiency because the units in shipping papers and labels associated with the packaging and transportation of radioactive materials would be consistent with international standards groups (e.g., IAEA).

- Environmental Considerations -- Effects on the environment, due to changes in accident frequencies and accident consequences, could result from the change. The change would require, in some instances, conversion from customary units to SI units in order to satisfy Part 71 reporting requirements. Thus, effects on the environment could result due to possible flawed unit conversions. In addition, the use of SI units only may be a safety issue in an emergency if responders are unfamiliar with the SI system.

Due to data limitations, only a portion of the values and impacts described above can be quantified. The results that can be quantified based on available data are described below.

Estimated Costs to Industry

In the U.S., approximately 2.8 million shipments of radioactive materials are made annually by nuclear power reactor licensees and materials licensees.[16] ICF estimates that approximately 70 to 90 percent of these licensees currently use customary units in their daily operations, including completion of shipping papers and preparation of labels for shipments sent off site.[17] Thus, the annual number of shipments with shipping papers and labels in Customary units ranges between approximately 1.96 million to 2.52 million.

Licensees who currently complete shipping papers and prepare labels in customary units may have to revise their procedural and administrative activities to convert from customary units to SI units. ICF assumes that unit conversions would be done once, and would be used to complete the shipping paper and label for the corresponding shipment. On average, the time needed to make unit conversions is estimated to be 0.05 hours (or 3 minutes) per shipment.[18] Therefore, at a rate of $129 per hour of professional staff, the annual cost for making unit conversions would range between approximately $12.6 million and $16.3 million per year (see Table 3-1).

Table 3-1. Implementation Costs to Industry Sectors Currently Using Customary Units

[16] U.S. Department of Transportation, Office of Hazardous Materials Safety, Research and Special Programs Administration, *Hazardous Materials Shipments*, October 1998.

[17] ICF estimated a lower (70 percent) and upper (90 percent) bound of the number of licensees using Customary units. ICF believes that users of SI units primarily include those licensees involved in international shipments (i.e., exports and/or imports).

[18] Based on best professional judgment.

Estimate	Annual number of shipments with shipping papers and labels in customary units (million)	Annual cost for licensees converting from customary to SI units ($ million)
Low	1.96	12.6
High	2.52	16.3

Estimated Costs to Other Government

As noted above, it is expected that DOE already uses SI units. If this were not the case, however, DOE would incur implementation costs for creating a system to convert from customary units to SI units. DOE makes approximately 5,500 shipments of radioactive material per year.[19] Assuming a rate of $129 per hour for professional staff and 0.05 hours per package to make unit conversions (as used above for industry), DOE also could incur costs of up to $35,475 per year.

3.3.2 Radionuclide Exemption Values

Values and Impacts of Option 1

Under the No-Action alternative (Option 1), NRC would continue to use one specific activity limit for exemption of any type of radionuclide. Thus, no values or impacts would result for domestic shipments from Option 1.

Option 1 would 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 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 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.

Values and Impacts of Option 2

Under Option 2, NRC would adopt, in 10 CFR Part 71, IAEA's radionuclide-specific exemption values for all materials. The nature of the changes under Option 2 makes it difficult to quantify the values or impacts. 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 changes 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." Appendix B provides additional detail regarding the

[19] U.S. Department of Transportation, Office of Hazardous Materials Safety, Research and Special Programs Administration, *Hazardous Materials Shipments*, October 1998.

estimation of the values and impacts of this action, based on ICF's review of this report. The values and impacts are summarized below:

- Industry Implementation – Minor administrative and procedural changes would be necessary to provide the framework for operation under radionuclide-specific exemptions.

- Industry Operation – In some cases, shippers would have to expend resources to identify the isotopes in material to ensure that it is exempt instead of verifying that it is less than 70 Bq/g.

- NRC Implementation -- Under this option, NRC would incur costs to revise guidance documents and related materials.

- Regulatory Efficiency – Implementing this change would make U.S. regulations more consistent with international regulations. International shipment could be affected by the differences in national regulations.

Due to data limitations, only a portion of the values and impacts described above can be quantified. The results that can be quantified based on available data are described below.

Estimated Costs to NRC

NRC would be required to make revisions to guidance documents and related materials. It is estimated that these revisions would take approximately two staff-months to complete. Assuming a cost of $129 per hour for staff, and 20 days per month at 8 hours each, this results in a total cost of approximately $41,280.

3.3.3 Revision of A_1 and A_2

Values and Impacts of Option 1

Under the No-Action Alternative (Option 1), NRC would retain the current A_1 and A_2 values promulgated in 10 CFR Part 71. Thus, no significant values or impacts would result from Option 1. There would be an impact in that NRC regulations would not be consistent with TS-R-1, but the overall impact of this inconsistency is estimated to be minimal.

Values and Impacts of Option 2

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.

In general, the new A_1 and A_2 values are within a factor of about 3 of the earlier values; there are a few radionuclides where the new A_1 and A_2 values are outside this range. Nearly 40 radionuclides have new A_1 values higher than previous values by factors ranging between 10 and 100. This is due mainly to improved modeling for beta emitters. There are no new A_1 or A_2 values that are lower than the previous figures by more than a factor of 10. A few radionuclides previously listed are now excluded but two additional ones have been added, both isomers of ^{150}Eu and ^{236}Np.

In order to gain some insight into how the revisions could affect packages in the U.S., ICF examined a 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.” Appendix B provides additional information on the estimated values and impacts associated with this action, which are summarized below:

- Public Health (Accident) – Changes to radiation exposure to the public due to accident consequences could result from the change. The A_1 and A_2 values were revised by IAEA based on refined modeling of possible doses from radionuclides. It is unclear whether the change for each individual radionuclide would slightly increase or decrease the total risk to public health, but the change to the refined values would be an overall value to public health by ensuring that the A_1 and A_2 values are more precisely based on risk. Analysis of the change showed no significant change in the number of shipments per year; therefore, accident frequency would not be affected.

- Occupational Health (Accident) – Changes to radiation exposure to workers due to accident consequences could result from the change. The A_1 and A_2 values were revised by IAEA based on refined modeling of possible doses from radionuclides. It is unclear whether the change for each individual radionuclide would slightly increase or decrease the total risk to workers, but the change to the refined values would be an overall value to worker health. Analysis of the change showed no significant change in the number of shipments per year; therefore, accident frequency would not be affected.

- Occupational Health (Routine) – Changes to radiation exposure to workers due to normal transportation conditions could result from the change. The A_1 and A_2 values were revised by IAEA based on refined modeling of possible doses from radionuclides. It is unclear whether the change for each individual radionuclide would slightly increase or decrease the total risk to workers, but the change to the refined values would be an overall value to worker health. Analysis of the change showed no significant change in the number of shipments per year; therefore, shipment frequency and routine worker dose would not be affected.

- Industry Implementation – The action could result in implementation costs to industry if industry must revise various aspects of shipping programs or modify shipping processes to assure compliance with the proposed A_1 and A_2 values. However, the cost is expected to be negligible since industry already has programs in place that use A_1 and A_2 values.
- NRC Implementation – The change is expected to result in implementation costs to the NRC to revise the A_1 and A_2 values.

- Other Government – The action could affect implementation and operation costs of DOE to the extent that its shipments must comply with NRC regulations. The action also could affect implementation and operation costs of Agreement States if they must enact and implement parallel requirements. There is not enough available information about the costs to DOE and Agreement States to quantify the resultant impact. The action also would affect the DOT in that DOT A_1 and A_2 values would need to be revised to be consistent with those in Part 71. DOT costs are expected to be similar to those of the NRC.

- Regulatory Efficiency – The action would improve regulatory efficiency by bringing U.S. regulations in compliance with the standards of the IAEA. This would improve the efficiency of handling imports and exports and would make U.S. standards compatible with other IAEA members.

- Environmental Considerations – Effects on the environment due to accident consequences could result from the change. The A_1 and A_2 values were revised by IAEA based on refined modeling of possible doses from radionuclides. It is unclear how the change for each individual radionuclide would affect the total risk to the environment, but the change to the refined values would be an overall value to environmental protection. Analysis of the change showed no significant change in the number of shipments per year; therefore, accident frequency would not be affected.

Due to data limitations, only a portion of the values and impacts described above can be quantified. The results that can be quantified based on available data are described below.

Estimated Costs to NRC

The changes to the A_1 and A_2 values are estimated to require approximately two staff-months of effort. Assuming a cost of $129 per hour for staff, and 20 days per month at 8 hours each, this results in a total cost of approximately $41,280. This cost is expected to consist mostly of development costs, such as preparing documents. This estimation of staff time is consistent with that estimated by the NRC during the last revision of the A_1 and A_2 values.

Estimated Costs to Other Government

The changes to the A_1 and A_2 values are estimated to require approximately two staff-months of effort for DOT. Assuming a cost of $129 per hour for staff, and 20 days per month at 8 hours each, this results in a total cost of approximately $41,280.

3.3.4 Uranium Hexafluoride (UF_6) Package Requirements

Values and Impacts of Option 1

Under the No-Action Alternative (Option 1), the TS-R-1 requirements regarding the packaging of UF_6 would not be included in 10 CFR Part 71. Thus, no values or impacts would result from Option 1.

Values and Impacts of Option 2

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. This would make Part 71 consistent with TS-R-1, enhance NRC regulatory efficiency and provide a uniform approval basis for designs which are used internationally. The following attributes are likely to be affected by this option:

- Public Health (Accident) – Under the action, cylinders containing UF_6 that meet the hypothetical fire test (a measure of resistance to release in the event of a fire) may cause less public health damage in the event of a vehicular accident. That is, residents along trucking routes will have a lower risk of exposure to radiation in the event of a fire following a vehicular accident.

- Occupational Health (Accident) – Similarly, cylinders containing UF_6 that meet the hypothetical fire test (a measure of resistance to release in the event of a fire) may cause less occupational health damage in the event of a vehicular accident. That is, truck operators will have a lower risk of exposure to radiation in the event of a fire following a vehicular accident.

- Offsite Property – Offsite property will be less likely to be exposed to and damaged from radiation in the event of a vehicular accident that results in a fire.

- Industry Implementation -- Industry might need to provide training to workers on how to handle the overpacks (e.g., proper loading of cylinders into overpacks, proper methods to secure the overpacked cylinder to tie down points on trailers, etc.).

- Industry Operation – Industry operations are likely to be affected through an increase in cost of either proving current cylinders would pass the hypothetical fire test or, more likely, overpacking the existing cylinders. This impact would be spread between private sector conversion facilities that produce UF_6 from yellow cake and the USEC facilities for any occasional shipment of depleted UF_6 between sites. In addition, when a depleted UF_6 conversion facility comes online at one or more sites, there will be an additional cost of shipping the stockpiled UF_6 cylinders.

- Regulatory Efficiency – Under the action, regulatory efficiency is likely to increase as a result of U.S. regulations being consistent with the international community.

- Environmental Considerations – Damage to the environment will be less likely to occur due to radiation in the event of a vehicular accident that results in a fire.

Due to data limitations, only a portion of the values and impacts described above can be quantified. The results that can be quantified based on available data are described below.

Estimated Costs to Industry

In developing this analysis, it was determined that there is no substantial difference between the ANSI N14.1 standard and the ISO 7195 standard for UF_6 packaging, and therefore, there would be no cost impacts, provided older cylinders that are stockpiled at sites are not required to be repackaged. Similarly, if the thermal test is waived for cylinders containing more than 9,000 kg UF_6, there will be little to no cost impact on industry. This is because only small cylinders, which are typically not used for natural or depleted UF_6, would be the only types of cylinders that have to meet the thermal test requirements, and it is believed that many of these small cylinders are already overpacked. (Smaller cylinders are typically used to transport enriched UF_6, but these cylinders are already believed to be overpacked.)

If, however, NRC did not waive the thermal test requirement for cylinders containing more than 9,000 kg UF_6, between 2,000 and 2,500 cylinders per 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 50,000+ cylinders could be affected. The costs to industry would be two-fold. First, there would be a one-time cost of $9 million to $13 million to design overpacks, purchase overpacks, and purchase additional trailers with the proper tie-down locations. Second, ongoing costs based on a cost of approximately $1,480 per shipment could result in an annual cost of $3.0 million to $3.7 million for routine operations, and $350,000 to $3.7 million per year to ship stockpiled cylinders to a conversion facility over a 20-year period.[20]

Most of the impact of adopting the TS-R-1 UF_6 provisions will fall on the 30-inch and 48-inch bare cylinders which are within the purview of the DOT and for which there is a "multilateral" approval option that could be used to mitigate most of this potential impact. Therefore, the adoption of the TS-R-1 requirements are not expected to have significant impact on fissile package designs for UF_6.

3.3.5 Introduction of the Criticality Safety Index Requirements

<u>Values and Impacts of Option 1</u>

Under the No-Action Alternative (Option 1), NRC would not require labels or modify definitions for CSI. Thus, no values or impacts would result from Option 1.

<u>Values and Impacts of Option 2</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. The values and impacts are summarized below:

- Public Health (Accident) – Emergency responders would benefit from additional information upon arrival at the accident scene. However, this additional information would only affect their actions in the most severe and unusual accident circumstances.

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[20] These costs were based on the April 18, 1985, *Draft U.S. Position Paper on Proposed Changes to the IAEA Regulatory Requirements for the Packaging of Uranium Hexafluoride*, R. Pope, F. Kovac, and R. Michelhaugh.

- Industry Implementation – Minor administrative and procedural changes would be necessary to provide the framework for marking packages for both criticality and radiation.

- Industry Operation – The action would result in minor additional effort for marking to ensure that packages are marked with both transportation indices.

- NRC Implementation – Under the option, NRC would incur costs to revise guidance documents and related materials.

- Other Government – Emergency responders would have to be notified of the changes to the information on the labels, and references would be provided. In addition, DOE would incur implementation and operation costs in complying with the new requirements.

- Regulatory Efficiency – Implementing this change would make U.S. regulations more consistent with international regulations. International shipment could be affected by the differences in national regulations.

Due to data limitations, only a portion of the values and impacts described above can be quantified. The results that can be quantified based on available data are described below.

Estimated Costs to Industry

In the U.S., approximately 2.8 million shipments of radioactive materials are made annually by nuclear power reactor licensees and materials licensees.[21] A very conservative estimate would be that 10 percent of these shipments (or 280,000) contain fissile material requiring labels indicating the CSI and TI. Assuming 5 packages per shipment and $1 per package for labeling, the total annual costs to licensees would be approximately $1.4 million.

Estimated Costs to NRC

NRC would be required to make revisions to guidance documents and related materials. It is estimated that these revisions would take approximately two staff-months to complete. Assuming a cost of $129 per hour for staff, and 20 days per month at 8 hours each, this results in a total cost of approximately $41,280. These costs have already been accounted for in this analysis.

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[21] U.S. Department of Transportation, Office of Hazardous Materials Safety, Research and Special Programs Administration, *Hazardous Materials Shipments*, October 1998.

Estimated Costs to Other Government

DOE makes approximately 22 fissile material shipments per year.[22] Assuming increased costs of $5 per shipment to comply with the labeling requirement, DOE would incur annual costs of $110.

3.3.6 Type C Packages and Low Dispersible Material

Values and Impacts of Option 1

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. Thus, no values or impacts would result from Option 1.

Values and Impacts of Option 2

Under Option 2, NRC would revise 10 CFR Part 71 to incorporate the Type C Packages and low dispersible radioactive material concepts for air transportation but retain section 71.74, the accident conditions for air transport of plutonium. There would be an increase in regulatory efficiency as a result of the nonadoption of the TS-R-1 requirements, which would enhance international shipments. Additional resource costs would be incurred by NRC. Costs also would be incurred by industry. These additional costs to industry would include implementation costs for the design of new packages to meet the Type C requirements rather using existing Type B packages.

The following attributes are expected to be affected:

- Public Health (Accident) – The accident risk of air shipments is higher than the accident risk of ground shipments.

- Public Health (Routine) -- The public receives lower routine exposures from an air shipment than from an overland shipment. People in their homes and on the highway do not receive measurable exposure from air shipments, and Type C packages would not be carried on passenger aircraft.

- Occupational Health (Routine) – Workers receive additional exposure using air transportation. Although the en route exposure is about the same, air transportation leads to additional handling since the originating and receiving facilities do not have air strips. Packages will normally be trucked to an airport, requiring more loading and unloading than a ground shipment.

- Offsite Property – The consequences to offsite property increase in proportion to the increased radiological accident consequences.

- Industry Implementation -- Industry would need to develop and certify Type C packages.
- Industry Operation – DOE was the only user for Type C packages identified. (See Other Government.)

[22] The estimated annual number of fissile material shipments by DOE is based on the number of such shipments that occurred in fiscal years 1995 and 1996, as reported in DOE's *Transportation Activities Summary Report for Fiscal Years 1995 and 1996*.

- NRC Implementation -- NRC development costs would include such activities as preparation of documents, publishing notices of rulemakings, holding public hearings, and responding to public comments.

- Other Government – Several foreign research reactor spent fuel casks have been shipped by air to port cities and loaded onto a ship for delivery to the U.S. DOE would realize operational cost savings if the aircraft were allowed to fly directly to the U.S.

- Regulatory Efficiency – Under the action, regulatory efficiency is likely to increase as a result of U.S. regulations being consistent with the international community.

Due to data limitations, only a portion of the values and impacts described above can be quantified. The results that can be quantified based on available data are described below.

Estimated Costs to NRC

NRC would be required to prepare documents and conduct other activities (such as publishing notices of rulemakings, holding public hearings, and responding to public comments) as a result of the action. It is estimated that these activities would take approximately two staff-months to complete. Assuming a cost of $129 per hour for staff, and 20 days per month at 8 hours each, this results in a total cost of approximately $41,280.

3.3.7 Deep Immersion Test

Values and Impacts of Option 1

Under Option 1, the No-Action Alternative, NRC would not require design of a package with radioactive contents greater than 10^5 A_2 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. Thus, no values or impacts would result from Option 1.

Values and Impacts of Option 2

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.

Appendix B provides additional information on the estimation of the values and impacts associated with the action. The affected attributes are described below:

- Public Health (Accident) – The action may reduce the impact to public health 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 action were implemented.
- Occupational Health (Accident) – The action could 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 without rupturing, thereby keeping the radioactive materials enclosed.

- Offsite Property – The action is intended to prevent the containment system from rupturing and possibly releasing radioactive material if a package was lost in deep water. Retaining package integrity would prevent the possible expenses of restricting the area (to prevent users such as boaters or fishers from entering the vicinity) and remediating any contamination of the marine environment.

- Industry Implementation -- Implementation of the action could result in costs to licensees as they test and certify packages to the standard.

- NRC Implementation -- NRC development costs would include such activities as preparation of documents, publishing notices of rulemakings, holding public hearings, and responding to public comments. It also is anticipated that NRC staff may incur costs for developing procedures, reviewing and approving test results, and recertifying packages.

- NRC Operation – NRC could incur recurring costs to ensure continued compliance with the proposed rule, although these costs are not expected to be significant.

- Other Government – The action could affect implementation and operation costs of the DOE to the extent that its shipments must comply with NRC regulations. There is not enough available information to quantify the resultant costs, but it is expected to be similar to those of industry.

- Regulatory Efficiency – The action would improve regulatory efficiency by bringing U.S. regulations in compliance with the standards of the IAEA. This would improve the efficiency of handling imports and exports and would make U.S. standards compatible with other IAEA members.

- Environmental Considerations – Effects on the environment due to changes in accident consequences could result from the change. The revised testing requirement would prevent the rupture of package containment at deeper depths, thereby preventing possible contamination of the marine environment.

Due to data limitations, only a portion of the values and impacts described above can be quantified. The results that can be quantified based on available data are described below.

Estimated Costs to Industry

Implementation of the action could result in costs to licensees as they test and certify packages to the standard. This total cost to industry is estimated to range from $245,000 to $2,928,000, with the expected total cost to be near $734,000. (See Appendix B for additional information on how these costs were estimated.)

Estimated Costs to NRC

NRC would be required to prepare documents and conduct other activities (such as publishing notices of rulemakings, holding public hearings, and responding to public comments) as a result of the action. It is estimated that these activities would take approximately two staff-months to complete. Assuming a cost of $129 per hour for staff, and 20 days per month at 8 hours each, this results in a total cost of approximately $41,280.

The estimated costs for NRC review and recertification of cask designs is estimated to be approximately $20,640 per cask design or $495,360 for all casks. (See Appendix B for additional information on how these costs were estimated.)

3.3.8 Grandfathering of Previously Approved Packages

<u>Values and Impacts of Option 1</u>

Under the No-Action Alternative (Option 1), NRC would not adopt the new grandfathering provisions contained in TS-R-1. Thus, no values or impacts would result from Option 1.

<u>Values and Impacts of Option 2</u>

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. The affected attributes are described below:

- NRC Implementation -- The change would result in implementation costs to the NRC. The NRC would have to revise regulatory guides and NUREG-series documents in order to indicate which packages are covered by the “grandfathering of older packages” provision.

- Other Government – The change could affect implementation and operation costs of Agreement States if they adopt and implement parallel requirements. (The change is not expected to affect implementation or operation costs of DOT.) If Agreement States adopt the “grandfathering of older packages” provision, they would only need to revise documents that they have developed specifically for their licensees (e.g., application materials).

Due to data limitations, only a portion of the values and impacts described above can be quantified. The results that can be quantified based on available data are described below.

Estimated Costs to NRC

The NRC estimates that it would need to revise approximately 30 documents. On average, the time needed to make the necessary revisions is estimated to be 0.5 hours per document. Thus, the total burden for revising the documents is approximately 15 hours. At a rate of $129 per hour for professional staff, the cost for revising regulatory guides and NUREG-series documents to include the “grandfathering of older packages” provision is estimated to be $1,935.

Estimated Costs to Other Government

The number of documents that Agreement States would need to revise is estimated to be approximately 15. On average, the time needed to make the necessary revisions is estimated to be 0.5 hours per document. Thus, the total burden for revising the documents is approximately 7.5 hours. At a rate of $129 per hour for professional staff, the cost for revising Agreement State documents to include the “grandfathering of older packages” provision is estimated to be $968.

3.3.9 Changes to Various Definitions

Values and Impacts of Option 1

Under the No-Action Alternative (Option 1), NRC would not add or make changes to definitions in 10 CFR Part 71.4. Thus, no values or impacts would result from Option 1.

Values and Impacts of Option 2

Under Option 2, NRC would add and change various definitions to 10 CFR 71.4 to ensure compatibility with definitions found in IAEA’s TS-R-1. The affected attributes are expected to include:

- Industry Implementation -- The change would result in implementation cost savings to industry. By modifying existing definitions and adding new definitions, licensees will benefit through more effective understanding of the requirements of Part 71.

- NRC Implementation -- The change would result in implementation costs to the NRC. The NRC would have to revise regulatory guides and NUREG-series documents in order to include the new or revised definitions of 10 CFR 71.4.

- Other Government – The change could affect implementation and operation costs of Agreement States because they would have to adopt the revision to the various definitions in 10 CFR 71.4. (The change is not expected to affect implementation or operation costs of DOT.) It is assumed that Agreement States use regulatory guides and NUREG-series documents published by the NRC. Thus, Agreement States would only need to revise documents that they have developed specifically for their licensees (e.g., application materials).

- Regulatory Efficiency – The change is expected to improve regulatory efficiency by achieving consistency with international standards groups (e.g., IAEA).

Due to data limitations, only a portion of the values and impacts described above can be quantified. The results that can be quantified based on available data are described below.

Estimated Costs to NRC

It is estimated that approximately 30 documents would require revision. On average, the time needed to make the necessary revisions to the various definitions is estimated to be 0.5 hours per document. Thus, the total burden for revising the various definitions included in the 30 documents is approximately 15 hours. At a rate of $129 per hour for professional staff, the cost for revising the definitions in regulatory guides and NUREG-series documents is estimated to be $1,935.

Estimated Costs to Other Government

The number of documents that Agreement States would need to revise is estimated to be approximately 15. On average, the time needed to make the necessary revisions to the various definitions is estimated to be 0.5 hours per document. Thus, the total burden for revising the various definitions included in the 15 documents is approximately 7.5 hours. At a rate of $129 per hour for professional staff, the cost for revising the various definitions in Agreement State documents is estimated to be $968.

3.3.10 Crush Test for Fissile Material Package Design

Values and Impacts of Option 1

Under the No-Action Alternative (Option 1), the NRC would not modify Part 71 to incorporate the crush test requirement for fissile material packages. Thus, no values or impacts would result from Option 1.

Values and Impacts of Option (2)

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. The affected attributes are described below:

- Regulatory Efficiency – The requirement would result in enhanced regulatory efficiency by correcting inconsistencies between Part 71 requirements and TS-R-1. However, further information on the impact of the TS-R-1 requirement for fissile material package testing is required.

- Industry Implementation -- The change would result in implementation costs imposed to demonstrate compliance and may lead to the redesign of packages.

- NRC Implementation – The regulatory change would result in NRC implementation costs associated with modifying the regulations and revising guidance documents.

Due to data limitations, only a portion of the values and impacts described above can be quantified. The results that can be quantified based on available data are described below.

Estimated Costs to NRC

NRC would be required to prepare documents and conduct other activities (such as publishing notices of rulemakings, holding public hearings, and responding to public comments) as a result of the action. It is estimated that these activities would take approximately two staff-months to complete. Assuming a cost of $129 per hour for staff, and 20 days per month at 8 hours each, this results in a total cost of approximately $41,280.

3.3.11 Fissile Material Package Designs for Transport by Aircraft

Values and Impacts of Option 1

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. Thus, no values or impacts would result from Option 1.

Values and Impacts of Option (2)

Under Option 2, the this new TS-R-1, additional criticality evaluation would be included in a new proposed paragraph 71.55(f) that only applies to air transport. The affected attributes are described below:

- Industry Implementation – The regulatory change would result in implementation savings to industry by eliminating the need for two different package designs.
- NRC Implementation – The change would result in NRC implementation costs associated with revising guidance manuals.
- NRC Operation – The change would result in NRC operation savings by eliminating the need for two different package designs and evaluations.
- Regulatory Efficiency – The requirement would result in enhanced regulatory efficiency by eliminating dual requirements for package design.

Due to data limitations, only a portion of the values and impacts described above can be quantified. The results that can be quantified based on available data are described below.

Estimated Costs to NRC

NRC would be required to prepare documents and conduct other activities (such as publishing notices of rulemakings, holding public hearings, and responding to public comments) as a result of the action. It is estimated that these activities would take approximately two staff-months to complete. Assuming a cost of $129 per hour for staff, and 20 days per month at 8 hours each, this results in a total cost of approximately $41,280.

3.4 Values and Impacts of NRC-Specific actions

3.4.1 Special Package Authorizations

The December 1996 revision of the safe transport standards (TS-R-1) developed by the IAEA, provides specific procedures for demonstrating the level of safety for shipment of special packages.

Values and Impacts of Option 1

Under the No-Action Alternative (Option 1), NRC would continue to address approval of special packages using exemptions under 10 CFR 71.8. Thus, no values or impacts would result from Option 1.

Values and Impacts of Option 2

Under Option 2, NRC would incorporate new regulations into 10 CFR Part 71 that similarly address shipment of special packages and demonstrate an acceptable level of safety. These requirements would essentially be equivalent to Paragraph 312 of TS-R-1 and would contain specific requirements for licensees to (1) demonstrate that the object/material is not readily packageable using available packages and that other shipment options are not preferable, (2) demonstrate that the special package generally complies with regulations, (3) specify the to-be-shipped configuration, (4) identify all deviations from regulations, and (5) identify measures that compensate for deviations from the regulations, commit to the use of these measures, and demonstrate the effectiveness of these measures in assuring shipment safety. The requirements would permit NRC staff review and authorization of special packages without issuing exemptions. The following attributes are expected to be affected:

- Public Health (Accident) – The action would provide added safeguards against radiation exposure to humans. Special package shipments are likely to increase regardless of the outcome of this rulemaking, as a result of future decommissioning activities. The justification for authorizing special packages for shipment is a decreased risk of radiation exposure to the public and workers as opposed to the shipment alternatives. Standardizing the health and safety collection requirements for these shipments will benefit human health by reducing the need to dispose of reactors and components in multiple shipments. In contrast, a failure to provide consistent health and safety information could lead to increased risk to health and property in some instances.

- Occupational Health (Accident) – See discussion for Public Health (Accident) above.

- Occupational Health (Routine) – See discussion for Public Health (Accident) above.

- Industry Implementation and Operation – Although licensees would realize savings by not having to prepare exemptions for special packages, the information collection requirements for shipment of special packages require the demonstration of a level of safety. Providing a consistent standard for the health and safety information collection is not expected to reduce this burden on licensees.

- NRC Implementation and Operation – The action would result in savings to NRC by reducing the burden of case-by-case review in authorizing packaging and shipping procedures for licensed material in excess of Type A quantities. Specifically, the action would eliminate the need for evaluating the health and safety information collection requirements for shipment of every special package.

- Regulatory Efficiency – The action would result in enhanced regulatory efficiency by standardizing the requirements to provide greater regulatory certainty and clarity than the no-action option, and would ensure consistent treatment among licensees requesting authorization for shipment of special packages. This increase in regulatory efficiency, however, would depend in part on modifications to DOT's regulations to recognize NRC special package exemptions.

Due to data limitations, only a portion of the values and impacts described above can be quantified. The results that can be quantified based on available data are described below.

Estimated Costs to Industry

The information collection requirements for shipment of special packages require the demonstration of a level of safety. Providing a consistent standard for the health and safety information collection is not expected to reduce this burden on licensees.

The Supporting Statement for 10 CFR Part 71, Revision to the Extension, discusses information collection requirements for packaging, preparation for shipment, and transport of licensed material. The burden estimates for 10 CFR Part 71 information collection requirements include a rate of $125 per hour for professional staff for preparation of the reports prepared in response to the 10 CFR Part 71 information collection requirements.[23] The annual burden for complying with the information collection requirements in Part 71 is estimated to be about 180 hours per licensee.[24] However, the licensee staff hours per submittal for 10 CFR 71.31, application for package approval, is estimated to be 300.

In estimating the additional preparation of health and safety information for shipment of special packages it was assumed that an additional 75 staff hours (25 percent of 300) would be required. At the rate of $129 per hour for professional staff, this additional cost amounts to $9,675 per shipment. Also, there may be some inherent cost savings to industry with respect to preparing health and safety information, but they are not expected to be significant.

Estimated Costs and Savings to NRC

The action would benefit NRC in that NRC would realize savings by reducing the number of case-by-case reviews for shipment of special packages. Due to limited data availability, the values of this change to the NRC have not been quantified in this analysis. The change under Option 2 would result in other values that are not quantified in this analysis. In particular, the change would result in enhanced regulatory efficiency because it would provide greater regulatory certainty and clarity than the no-action option and would ensure consistent treatment among all licensees requesting authorization for shipment of special packages.

The annual cost for the NRC to process and review the records and reports required by 10 CFR Part 71 is estimated to be approximately $3,182,585.[25] This estimate is based on 20,800 staff review hours for a total of 350 licensees (approximately 60 hours per licensee). It was assumed that the additional review of health and safety information for each shipment of special packages would result in an additional 30 staff hours (50 percent of 60). Assuming decommissioning efforts result in 5 shipments per year under special arrangement, this additional cost to NRC amounts to $19,350 annually. A reduced burden given the elimination of case-by-case evaluation of health and safety requirements is expected. However, the increase in the number of special arrangement shipments due to anticipated decommissioning efforts is likely to offset any savings.

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[23] This rate is based on NRC's fully recoverable fee rate and includes both salaries and overhead.

[24] This estimate is based upon a total annual burden to 350 licensees of 63,537 hours.

[25] These costs are fully recovered through fee assessments to NRC licensees pursuant to 10 CFR Parts 170 and/or 171.

3.4.2 Expansion of Part 71 Quality Assurance Requirements to Certificate of Compliance (CoC) Holders

<u>Values and Impacts of Option 1</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. Thus, no values or impacts would result from Option 1.

<u>Values and Impacts of Option 2</u>

Under Option 2, NRC would explicitly subject CoC Holders and CoC applicants to the requirements contained in 10 CFR Part 71. NRC also would add recordkeeping and reporting requirements for CoC Holders and CoC applicants. The attributes expected to be affected by this action are described below:

- Public Health, Onsite and Offsite Property -- By incorporating CoCs and CoC applicants in Part 71, any deficiencies noted by NRC will result in a notice of violation (NOV). This enforcement action will allow NRC to issue orders or take other enforcement actions necessary to ensure compliance with Part 71 requirements. This will ultimately lead to safer transportation casks, although this benefit is small and impossible to quantify relative to the current safety levels of transportation casks.

- Industry Implementation and Operation – CoCs and CoC applicants will incur costs associated with understanding and implementing the new regulations. They also will have to submit reports under Part 71 that they were not submitting previously. These reports are described in SECY 99-174; it is assumed that similar reports will be required if CoCs and CoC applicants are incorporated in the Part 71 applicability. SECY 99-174 states that "Additional requirements for recordkeeping and reporting for certificate holders are needed, to include records required to be kept as a condition of the CoC [certificate of compliance]. This will provide an enforcement basis equivalence to the record keeping and reporting regulations for licensees."

- NRC Implementation and Operation – NRC will incur costs associated with supervising CoCs and CoC applicants, and maintaining and reviewing the records for submittals.

- Regulatory Efficiency – NRC's ability to issue NOVs to CoCs and CoC applicants will improve the regulatory efficiency of NRC enforcement actions. NRC can follow up the issuance of NOVs with more strict regulatory enforcement actions. This is not currently possible under Part 71, because CoCs and CoC applicants are not explicitly subject to the regulations of Part 71.

Due to data limitations, only a portion of the values and impacts described above can be quantified. The results that can be quantified based on available data are described below.

Estimated Costs to Industry

For the 31 CoC Holders, the burden associated with recordkeeping and reporting was determined to be 100 hours per year, from the Part 72 rulemaking. Assuming a cost of $129 per hour for staff, the estimated total cost to these entities is therefore approximately $400,000 per year.

Estimated Costs to NRC

NRC will incur costs associated with tracking submissions to the agency. It was assumed that NRC will spend approximately 20 hours per year per CoC Holder for these activities. Assuming a cost of $129 per hour, the total cost to the NRC is estimated at approximately $80,000.

3.4.3 Adoption of ASME Code

<u>Values and Impacts of Option 1</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. Thus, no values or impacts would result from Option 1.

NRC notes that, if the ASME code is not implemented for spent fuel 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.

<u>Values and Impacts of Option 2</u>

Under Option 2, NRC would adopt the ASME B&PV Code Section III, Division 3, for spent fuel transportation casks in 10 CFR Part 71. This action would eventually apply to spent fuel storage canister confinement and spent fuel transportation cask containment for all applications, including dual-purpose casks. The attributes expected to be affected by this action include:

- Public Health, Onsite and Offsite Property -- Transportation and dual-purpose casks manufactured under the ASME B&PV Code, Section III, Division 3 will be manufactured using QA/QC procedures that are more complete than those presently in place. The casks are therefore less likely to fail during a transportation accident and are less likely to contain a design flaw that would lead to a leak of radioactive material. For these reasons, the ASME-certified casks provide a lesser risk to public health and property. Although this is clearly a benefit of the proposed rule, the likelihood of a flawed cask being involved in an accident or leak is so remote that the public health/property benefits of the ASME QA/QC program relative to the current licensee/NRC program are impossible to quantify.

- Industry Implementation and Operation – CoC Holders and manufacturers will incur additional costs due to: (1) conducting a site survey of the production facility, (2) the review of cask design plans by a professional engineer, and (3) the employment of an on-site authorized nuclear inspector (ANI). CoC Holders and manufacturers will save costs associated with fabrication errors, such as having to repair faulty casks, and lost sales during faulty cask repair. They also will save the costs associated with employing an onsite QA/QC inspector. However, because of the potential for the ASME code to be revised over the next several years, adoption at this time could result in additional costs to licensees should the regulations be revised in the future.

- NRC Implementation and Operation – NRC will save some costs, by reducing the need for full-time inspectors who periodically inspect CoC Holders and fabricators. This on-site inspection function will be carried out by the authorized nuclear inspector (ANI). However, because of the potential for the ASME code to be revised over the next several years, adoption at this time could result in additional costs to NRC should the regulations need to be revised in the future.

Due to data limitations, only a portion of the values and impacts described above can be quantified. The results that can be quantified based on available data are described below.

Estimated Costs and Savings to Industry

Currently, there are six transportation cask fabricators.[26] On-site, one-time ASME survey costs will total approximately $440,000. Costs for ASME certification and the on-site authorized nuclear inspector (ANI) will total approximately $765,000 per year, although the fabricators will save approximately $450,000 per year because they will not have to employ an on-site QA/QC inspector (this function is filled by the ANI). Thus, the net yearly cost increase to the fabricators is $315,000.

In addition, industry will save costs associated with avoiding fabrication errors that will be discovered by the ANI. Although these savings are impossible to quantify on a per year basis, NRC documented one case in which a fabricator and NRC spent $570,000 inspecting and repairing flawed casks. The fabricator was estimated to have lost $2.1 million in sales during this time, because its resources were directed at affecting repairs to the flawed casks and not to cask production. It is assumed that an on-site ANI would have discovered the production flaw.

3.4.4 Change Authority

Values and Impacts of Option 1

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. Thus, no values or impacts would result from Option 1.

[26] Personal communication with Ron Parkhill, U.S. Nuclear Regulatory Commission, October, 1999.

Values and Impacts of Option 2

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 the authority for CoCs to make changes to a dual purpose package design without prior NRC approval. The section also would include new requirements for submitting and updating a final safety analysis report describing the package's design. A discussion of the attributes expected to be affected by the action is provided below:

- Industry Implementation and Operation – Licensees and CoC Holders will have to spend time and incur costs associated with understanding and implementing the new requirements. CoC Holders will incur costs when submitting an FSAR detailing minor changes, tests, and experiments they make with regard to transportation package design. The CoC Holders will save costs associated with preparing license amendments and paying fees to NRC that are required under the current regulations (i.e., because these will no longer be required if provisions similar to 10 CFR 72.48 are implemented in Part 71).

- NRC Implementation and Operation – NRC will realize cost savings associated with no longer having to review license amendments for CoC Holders making minimal changes to their procedures. These cost savings will be partially offset in that NRC will need to review reports that are required to be submitted CoC Holders making minor changes.

- Regulatory Efficiency – There would be a clearer and more consistent interpretation between the NRC, licensees, and CoC Holders regarding requirements necessitated by changes in procedures. It will therefore be possible to direct NRC resources that would be spent reviewing license amendments to areas where measurable improvements in safety can be achieved.

Due to data limitations, only a portion of the values and impacts described above can be quantified. The results that can be quantified based on available data are described below.

Estimated Costs and Savings to Industry

For the 350 record-keeping licensees listed in the Part 71 Supporting Statement, professional judgment was used to assume that in any given year 50 percent of licensees will perform a "minimal change." Submittals under section 72.48 are required every two years (as is the case with the proposed Part 71 requirements) and therefore, approximately 88 submittals are expected per year. The total cost savings of reporting the "minimal changes" versus preparing license amendments is estimated at approximately $2.4 million per year. However, the 350 licensees will incur a one-time recordkeeping cost of approximately $2.3 million in the first year the proposed rule is implemented.

Estimated Cost Savings to NRC

NRC costs are projected to decline slightly under the proposed rule, because the agency will not have to review as many license amendments each year. This cost savings was determined to be negligible in the section 72.48 regulatory analysis and will be offset by the agency having to adopt new document controls to handle the “minimal change” submissions required every two years for licensees making “minimal changes.”

3.4.5 Fissile Material Exemptions and General License Provisions

Values and Impacts of Option 1

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. Thus, no values or impacts would result from Option 1.

Values and Impacts of Option 2

Under Option 2, NRC would modify the 10 CFR Part 71 regulations as necessary to implement the entire set of 17 recommendations contained in NUREG/CR-5342. The attributes expected to be affected by the actions include:

- Public Health (Accident) – Changes to radiation exposures to the public, due to changes in accident frequencies and accident consequences, could result from the action. The regulatory options could both alter the number of fissile shipments (thereby altering the accident frequency) and reduce the likelihood of occurrences of criticality (thereby reducing accidental consequences).

- Occupational Health (Accident) – Changes to radiation exposures to workers, due to changes in accident frequencies and accident consequences, could result from the action. The regulatory options could both alter the number of fissile shipments (thereby altering the accident frequency) and reduce the likelihood of occurrences of criticality (thereby reducing accidental consequences).

- Occupational Health (Routine) – Changes to radiation exposures to workers during normal packaging and transportation operations could result from the action. The regulatory options could alter the number of fissile packages or shipments, thereby altering the number of workers exposed or the duration of the exposure.

- Offsite Property – Effects on offsite property, due to changes in accident frequencies and accident consequences, could result from the action. The regulatory options could both alter the number of fissile shipments (thereby altering the accident frequency) and reduce the likelihood of occurrences of criticality (thereby reducing accidental consequences).

- Onsite Property – Effects on onsite property (direct and indirect), due to changes in accident frequencies and accident consequences, could result from the action. The regulatory options could both alter the number of fissile shipments (thereby altering the accident frequency) and reduce the likelihood of occurrences of criticality (thereby reducing accidental consequences).

- Industry Implementation -- The action would result in implementation costs or savings to industry if industry must evaluate and/or enact changes to ensure that its operating procedures will comply with the action.

- Industry Operation – The action would result in industry operation costs or savings if industry must alter its current packaging and shipping procedures to comply with the action.

- NRC Implementation -- The action would result in NRC implementation costs or savings to put the action into operation. Specifically, NRC would incur implementation costs to revise guidance documents and possibly to establish a data collection system and database infrastructure.

- NRC Operation – The action would result in NRC operation costs or savings if the number of shipments requiring specific NRC approval changes (i.e., the number of shipments that fail to qualify for the fissile exemption and the general licenses) and possibly to operate and maintain a data collection system and database.

- Regulatory Efficiency – The action would be expected to result in enhanced regulatory efficiency by clarifying the meaning and applicability of specific terms and requirements, and by reducing noncompliance.

- Environmental Considerations – Effects on the environment, due to changes in accident frequencies and accident consequences, could result from the action. The regulatory options could both alter the number of fissile shipments (thereby altering the accident frequency) and reduce the likelihood of occurrences of criticality (thereby reducing accidental consequences).

- Other Government – The action could affect implementation and operation costs of the U.S. Department of Energy, to the extent that its fissile material shipments must comply with NRC regulations. The action also could affect implementation and operation costs of Agreement States if they must enact and implement parallel requirements. (The action would not be expected to affect implementation or operation costs of DOT.)

- Improvements in Knowledge – The action, if it includes a data collection requirement, could result in improved knowledge that may ultimately result in more robust risk assessments and safety evaluations (i.e., less uncertainty) and, consequently, in improvements in regulatory policy and regulatory requirements.

As discussed previously, ICF has been seeking detailed information from industry to assist in developing a quantitative estimate of the values and impacts associated with the changes to the fissile material packaging and transportation requirements. In order to develop these estimates, significant data needs must be met, including the following:

- Number/types of packages/shipments containing the radionuclide ^{238}Pu.

- Number of packages/shipments of fissile material having a specific activity greater than 43 Bq/g but less than 70 Bq/g.

- Number/type of packages/shipments containing Pu-Be sources, including the quantity of plutonium.

- Number of packages/shipments falling under each of sections 71.18, 71.20, 71.22, and 71.24, and the TI and/or aggregate TI further distinguished by exclusive use versus non-exclusive use.

- Number/types of packages/shipments per conveyance.

- Number/type of packages/shipments currently falling under sections 71.20 and 71.24 that contain ^{235}U broken out by (1) the number of grams for each ^{235}U enrichment weight percentage, and (2) whether the fissile radionuclides are distributed uniformly and cannot form a lattice arrangement within the packaging.

- Number/types of packages/shipments currently shipped under sections 71.18(e) and 71.22(e) containing Be, C, and D_2O, and how much Be, C, and D_2O is contained (in grams and as a percent of fissile material mass).

- Number/types of packages/shipments of fissile materials with high-density hydrogenous moderators exceeding 15% of the mass of hydrogenous moderator in the package.

- Number/types of packages/shipments of fissile materials broken out by the 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_2O.

- Number/type of packages/shipments that both currently fall under section 71.53 and contain Be, C, and D_2O.

- Number/type of package/shipments broken out by TI.

- Number/type of package/shipments that currently fall under the section 71.53(c) exemption for uranyl nitrite solutions transport.

- Number/type of additional packages/shipments that would fall under section 71.53(b) absent the requirement that the fissile material were distributed homogeneously throughout the package contents and that the material not form a lattice arrangement within the package.

- To the extent not determinable based on the above information, the number/types of such packages meeting section 71.53, and currently shipped under sections 71.18, 71.20, 71.22, 71.24, and/or under Subparts E and F.

Such data are not readily available, and much of the data may not exist at all.[27] Consequently, this study analyzes values and impacts on a qualitative basis taking into account the regulatory option, each individual affected attribute, other factors influencing these attributes (e.g., potential for criticality, potential for radiation exposure, number of required packages and/or shipments, efforts required to make regulatory determinations or calculations, recordkeeping and reporting requirements), and applicable discussion and analysis contained in NUREG/CR-5342. Values and impacts reported for several attributes are based on analysis presented in a related environmental assessment prepared for this rulemaking.

Each of the 17 recommendations would, if implemented, result in certain values and/or impacts. Thus, the values and impacts of Option 2 as a whole consist of the sum of all values and impacts associated with the 17 recommendations.

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[27] Survey data on radioactive material shipments are not specific enough for use in the present analysis and, moreover, are more than a decade old (*Transport of Radioactive Material in the United States,* SRI International, April 1985).

Table 3-2 summarizes the values and impacts associated with each of the 17 recommendations contained in NUREG/CR-5342.

- Recommendation 1 – The action would result in enhanced regulatory efficiency due to increases in the clarity of NRC's regulations and improvements in the consistency between 10 CFR Part 71, 49 CFR Part 173, and IAEA No. TS-R-1. It also is conceivable that the action could result in a reduced potential for criticality due to the increased understanding of the regulations that would likely result.

- Recommendation 2 – The action would result in enhanced regulatory efficiency due to increases in the clarity of NRC's regulations and improvements in the consistency between 10 CFR Part 71 and IAEA No. TS-R-1. Also, licensees potentially could incur lower costs primarily due to reduced fissile shipments. As a result of the reduction in total fissile shipments, the potential for radiological exposures also would be reduced, yielding environmental, health, safety, and avoided offsite and onsite property damage benefits.

- Recommendation 3 – The action would increase costs to licensees, but would reduce the potential for criticality and thus would yield environmental, health, safety, and avoided offsite and onsite property damage benefits.

- Recommendation 4 – The action would most likely increase the regulatory burden on licensees and could result in increased costs to licensees due to necessary increases in the number of fissile material shipments. An increase in total fissile shipments would, in turn, increase the potential for radiological exposures, yielding possible negative impacts on the environment, health, safety, and offsite and onsite property. The net effect is uncertain, however, because of the potential for reductions in criticality.

- Recommendation 5 – The action would result in enhanced regulatory efficiency by consolidating the sections of 10 CFR Part 71 that pertain to exemptions into a single subpart.

- Recommendation 6 – The action would impose a recordkeeping and reporting burden on licensees, and would impose a recordkeeping and review burden on NRC. The added burden would consist of both initial costs (e.g., development of reporting formats, establishment of a fissile shipment database) and ongoing costs (e.g., periodic preparation and review of reports, maintenance of the database) to licensees and NRC. The action also would lead to improvements in knowledge for both licensees and NRC, and would enable NRC to better understand and regulate the shipment of fissile materials.

Table 3-2. Values and Impacts Associated with Actions Related to NUREG/CR-5342 Recommendations

ATTRIBUTE	ACTION																
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
Public Health (Accident)		V(X)	V(C)	?			V(C) I(X)	V(C)	?(X)	V(C) ?(X)	?(X)	V(X)	V(C) I(X)	V(C) ?(X)	V(C)?(X)	V(C,X)	
Occupational Health (Accident)		V(X)	V(C)	?			V(C) I(X)	V(C)	?(X)	V(C) ?(X)	?(X)	V(X)	V(C) I(X)	V(C) ?(X)	V(C)?(X)	V(C,X)	
Occupational Health (Routine)		V(X)		?			I(X)		I(X)	?(X)	V(X)	?(X)	V(X)	?(X)	?(X)		
Offsite Property		V(X)	V(C)	?			V(C) I(X)	V(C)	?(X)	V(C) ?(X)	?(X)	V(X)	V(C) ?(X)	V(C) ?(X)	V(C)?(X)	V(C,X)	
Onsite Property		V(X)	V(C)	?			V(C) I(X)	V(C)	?(X)	V(C) ?(X)	?(X)	V(X)	V(C) ?(X)	V(C) ?(X)	V(C)?(X)	V(C,X)	
Industry Implementation		V(S,G)	I(S)	I(S,G)		I(R)	I(S,G)	I(G)	V(G) ?(S)	V(G) ?(S)	V(G)	I(S)	V(S)	V(G) ?(S)	V(G) ?(S)	I(S)	V(G)
Industry Operation		V(S,G)	I(S)	I(S,G)		I(R)	I(S,G)	I(G)	?(S)	V(G) ?(S)	V(S,G)	I(S)	V(S)	V(G) ?(S)	V(G)?(S)	I(S)	V(G)
NRC Implementation	I	I	I	I	I	I	I	I	I	I	I	I	I	I	I	I	I
NRC Operation		V(G)	?			I				?	V(G)			?	?		V(G)
Regulatory Efficiency	V	V			V		V	V		V	V	V	V	V	V	V	V
Environmental Considerations		V(X)	V(C)	?			V(C) I(X)	V(C)	?(X)	V(C) ?(X)	?(X)	V(X)	V(C) ?(X)	V(C) ?(X)	V(C)?(X)	V(C,X)	
Other Government		V(S,G)	I(S)	I(S,G)			I(S,D)	I(G)	V(G) I(S)	V(G) ?(S)	V(S,G)	I(S)	V(S)	V(G) ?(S)	V(G) ?(S)	I(S)	V(G)
Improvements in Knowledge						V											

KEY:
Values/Impacts: V = Value; I = Impact; ? = Direction of effect is uncertain due to data limitations
Factors influencing attributes: C = Criticality potential; X = Radiological exposure; S = number (or cost) of packages and/or shipments; G = Regulatory determinations/ calculations; R = Recordkeeping/reporting

- Recommendation 7 – The action would eliminate the potential for criticality and thus would yield environmental, health, safety, and avoided offsite and onsite property damage benefits. The action also would impose costs on licensees through added packaging requirements, increased shipments, and increased regulatory burden. The increase in shipments could, in turn, increase the potential for radiological exposures during shipping. However, the reduction in criticality risk would largely outweigh the risks from these exposures. The recommendation also would result in enhanced regulatory efficiency by creating a separate general license for Pu-Be sources, thus increasing the clarity of NRC's regulations.

- Recommendation 8 – The action would eliminate the potential for criticality and thus would yield environmental, health, safety, and avoided offsite and onsite property damage benefits. The action would impose an increased regulatory burden on licensees, however, in that it would require licensees to perform additional calculations related to the aggregate transport index. This recommendation also would result in enhanced regulatory efficiency by consolidating certain sections of 10 CFR Part 71 and by increasing the clarity of NRC's regulations.

- Recommendation 9 – The action would affect licensees' costs and may have, potentially, minor effects on radiological exposures. The action also would reduce the regulatory burden on licensees by reducing their administrative implementation costs (i.e., it would reduce the number of calculations licensees would need to make in determining permissible masses).

- Recommendation 10 – The action would eliminate the potential for criticality and thus would yield environmental, health, safety, and avoided offsite and onsite property damage benefits. Also, by modifying the Be, C, and D_2O quantity restrictions to incorporate a mass-based limit rather than a percentage-based limit, the action would reduce the number of calculations licensees would need to make in order to determine compliance with the regulations, thus reducing regulatory burden. The action also would result in enhanced regulatory efficiency by simplifying and clarifying NRC's regulations.

- Recommendation 11 – The action would reduce regulatory burden on licensees by simplifying the calculation of fissile material quantities and the categorization of mass limits. The action also would result in enhanced regulatory efficiency by simplifying and clarifying NRC's regulations.

- Recommendation 12 – The action would result in licensees incurring higher costs in meeting the added packaging requirements for shipments under the general licenses. As a result of these requirements, however, the potential for radiological exposures would be reduced, yielding environmental, health, safety, and avoided offsite and onsite property damage benefits. (The potential for criticality would not be affected by this recommendation.) The action also would result in enhanced regulatory efficiency due to increases in consistency within NRC's regulations.

- Recommendation 13 – The action would eliminate the potential for criticality and thus would yield environmental, health, safety, and avoided offsite and onsite property damage benefits. Also, the action would reduce regulatory burden on licenses by simplifying the calculation of fissile material quantities and the categorization of mass limits. The action also would result in enhanced regulatory efficiency due to increases in consistency within NRC's regulations.

- Recommendation 14 – The action would eliminate the potential for criticality and thus would yield environmental, health, safety, and avoided offsite and onsite property damage benefits. Also, the action would reduce regulatory burden on licenses by simplifying certain calculations that would need to be made in order to comply with the regulations. The action also would result in enhanced regulatory efficiency due to increases in consistency within NRC's regulations.

- Recommendation 15 – The action would eliminate the potential for criticality and thus would yield environmental, health, safety, and avoided offsite and onsite property damage benefits. Also, the action would reduce regulatory burden on licenses by simplifying certain calculations that would need to be made in order to comply with the regulations. The action also would result in enhanced regulatory efficiency due to increases in consistency within NRC's regulations.

- Recommendation 16 – The action would eliminate the potential for criticality and thus would yield environmental, health, safety, and avoided offsite and onsite property damage benefits. However, some licensees would incur higher costs under this action in meeting the added packaging requirements for transport of uranyl nitrite solutions. The action also would result in enhanced regulatory efficiency by simplifying NRC's regulations.

- Recommendation 17 – The action would result in savings to licensees with respect to determining whether package contents are homogeneous and form a lattice arrangement within the package. The action also would result in enhanced regulatory efficiency by simplifying NRC's regulations.

Given the severe data limitations, this analysis provides only minimal quantitative analysis of values and impacts associated with the changes to the fissile material requirements. ICF is continuing its data collection efforts, and is evaluating ways to develop surrogate data should actual industry data not be made available.

Estimated Costs to Industry

For the action associated with Recommendation 6, industry would incur additional costs to submit recordkeeping/reporting information electronically. (This analysis assumes that NRC will bear the costs for development, implementation, and maintenance of the database system.) It is estimated that an additional 0.2 hours per submission would be required to submit these data. The Supporting Statement for Part 71 indicates that approximately 122 submissions are made annually by licensees. At a cost of $129 per hour, this results in an estimated cost to licensees of $3,150 to submit data to the NRC electronically for input into the database.

Estimated Costs to NRC

For the action associated with Recommendation 6, NRC would incur capital and O&M costs to develop a database system. NRC also would incur costs associated with review of data submitted by industry. It is estimated that approximately 0.75 FTE would be required initially to establish the database. This would result in a cost to NRC of approximately $201,240. Annual maintenance and data review are estimated to cost NRC an additional $268,320 (or one FTE). Capital and O&M costs for the computer hardware are difficult to analyze without specific information concerning the type of system to be developed and, therefore, have not been quantified.

3.4.6 Double Containment of Plutonium (PRM-71-12)

Values and Impacts of Option 1

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. Thus, no values or impacts would result from Option 1.

Values and Impacts of Option 2

Under Option 2, NRC would delete section 71.63 special 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 to determine if a Type B package is required). The attributes expected to be affected are described below:

- Public Health (Accident) – Removing a layer of packaging (protection) increases the probability and consequences of accidents that can breach the Type B package. It is anticipated, therefore, that an increase in exposure could result during an accident. The additional costs that might be incurred as a result will be developed with the preparation of the Environmental Assessment supporting this proposed rulemaking.

- Occupational Health (Routine) – Workers receive additional exposure while sealing the second layer of packaging. Eliminating this step and the associated radiation exposure results in a reduction in possible exposure. The cost savings that might be incurred as a result will be developed with the preparation of the Environmental Assessment supporting this proposed rulemaking.

- Offsite Property – The consequences to offsite property increase in proportion to the increase radiological accident consequences. The costs/savings that might be incurred as a result will be developed with the preparation of the Environmental Assessment supporting this proposed rulemaking.

- Industry Implementation -- Removing the requirement for double containment could reduce packaging costs. However, much of DOE's plutonium is stored in containers qualified as one level of containment and thus, would meet the double containment criteria when shipped whether or not it is required. Packages being used for plutonium shipments and packages that are planned for plutonium shipments in the next decade, such as packages that carry DOE-STD-3013 containers and SAFKEG packages, meet the double containment requirement. It would cost DOE more to redesign to a lower level of safety than to continue to use double containment. After this next decade, the major plutonium transportation affected by this regulation will be the continued repository shipments in TRUPAC-II packaging systems. Since these packages are already being produced and handling and shipping fixtures are designed around these packages, it is unlikely that DOE would change these operations. Therefore, future DOE shipments of plutonium in single containment packages cannot be predicted at this time.

- Industry Operation – Essentially all anticipated plutonium shipments would be done by DOE. (See Other Government.)

- NRC Implementation – Under the options, NRC would incur costs to revise guidance documents and related materials.

- Other Government – Removing the requirement for double containment could reduce operational costs. However, DOE has already spent a great deal developing transportation and storage containers that can be used under double containment.

Due to data limitations, only a portion of the values and impacts described above can be quantified. The results that can be quantified based on available data are described below.

Estimated Costs to NRC

NRC would be required to make revisions to guidance documents and related materials. It is estimated that these revisions would take approximately two staff-months to complete. Assuming a cost of $129 per hour for staff, and 20 days per month at 8 hours each, this results in a total cost of approximately $41,280. These costs, however, have already been accounted for previously in the analysis.

3.4.7 Contamination Limits as Applied to Spent Fuel and High Level Waste (HLW) Packages

No regulatory changes are being proposed. Therefore, no regulatory options have been identified. As a result, no analysis was conducted.

3.4.8 Modifications of Event Reporting Requirements

<u>Values and Impacts of Option 1</u>

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 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. Thus, no values or impacts would result from Option 1.

<u>Values and Impacts of Option (2)</u>

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 also would 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. The affected attributes are described below:

- Regulatory Efficiency – The change would result in enhanced conformity among Parts 50, 71, and 72.

- NRC Implementation – The change would result in NRC implementation costs for licensees for revising procedures and for training. A key benefit of the proposed amendments would be a reduction in the recurring annual reporting burden on licensees, as a result of reducing the efforts associated with reporting events of little or no risk or safety significance. It is anticipated that the NRC's recurring annual review efforts for telephone notifications and written reports will not be significantly reduced.

Due to data limitations, only a portion of the values and impacts described above can be quantified. The results that can be quantified based on available data are described below

Estimated Costs to NRC

NRC would be required to prepare documents and conduct other activities (such as publishing notices of rulemakings, holding public hearings, and responding to public comments) as a result of the action. It is estimated that these activities would take approximately two staff-months to complete. Assuming a cost of $129 per hour for staff, and 20 days per month at 8 hours each, this results in a total cost of approximately $41,280.

4. Backfit Analysis

The regulatory options examined in this regulatory analysis do not involve any provisions that would require backfits as defined in 10 CFR Part 50.109(a)(1). Consequently, a backfit analysis is not necessary.

5. Decision Rationale

As discussed earlier in this analysis, NRC's regulatory action consists of 19 individual changes that are intended to (1) harmonize the radioactive transportation regulations in 10 CFR Part 71 with the IAEA's TS-R-1, and (2) simplify NRC's regulations, while maintaining the safety standards for containers used to ship and store radioactive waste, and reduce paperwork and burden for licensees seeking to make minor changes in their operations. For each of the 19 issues addressed by the proposed rulemaking, the values and impacts associated with modifying its transportation regulations in 10 CFR Part 71 (as proposed under Option 2) and with adopting the No-Action alternative (Option 1) have been considered.

Due to severe data limitations on radioactive material shipments and other factors related to the rulemaking, ICF was unable to quantify a number of the values and impacts that are expected to occur as a result of Option 2. Nevertheless, given that the amendments described in Option 2 for each issue would simplify the Part 71 requirements applicable to licensees shipping radioactive materials, increase consistency with other regulatory programs, relax certain restrictions on radioactive material packages and shipments that are not justified based on plausible criticality concerns, and ensure adequate criticality safety for a number of newly-considered plausible transportation and packaging situations, these options are generally preferable to Option 1. For some issues, however, it was determined that revising the regulations would not result in any net economic or safety-related benefits to licensees, NRC, other government agencies (e.g., DOE, DOT), or the public.

For each of the 19 changes under consideration, Table 5-1 below summarizes the options determined to be most preferable based on professional judgment and limited quantitative analysis.

Table 5-1. Summary of Preferred Options

Technical Issue	Preferred Option
1. Changing Part 71 to the International System of Units (SI) Only	Option 1 (No-Action)
2. Radionuclide Exemption Values	Option 2
3. Revision of A_1 and A_2	Option 2
4. Uranium Hexafluoride Package Requirements	Option 2
5. Introduction of the Criticality Safety Index Requirements	Option 2
6. Type C Packages and Low Dispersible Material	Option 1 (No-Action)
7. Deep Immersion Test	Option 2
8. Grandfathering Previously Approved Packages	Option 2
9. Changes to Various Definitions	Option 2
10. Crush Test for Fissile Material Package Design	Option 2
11. Fissile Material Package Designs for Transport by Aircraft	Option 2
12. Special Package Authorizations	Option 2
13. Expansion of Part 71 Quality Assurance Requirements to Certificate of Compliance (CoC) Holders	Option 2
14. Adoption of ASME Code	Option 1 (No-Action)
15. Change Authority	Option 2
16. Fissile Material Exemptions and General License Provisions	Option 2
17. Double Containment of Plutonium (PRM-71-12)	Option 2
18. Contamination Limits as Applied to Spent Fuel and High Level Waste (HLW) Packages	For information only. No options identified.
19. Modifications of Event Reporting Requirements	Option 2

6. Implementation

Any action would be enacted through a Proposed Rule Notice, a public comment period, and a Final Rule. Implementation can begin immediately following the enactment of the final rule. No impediments to implementation of the recommended alternatives have been identified. Regulatory Guides for licensees would be required to provide an explanation of the regulatory requirements and methods for complying with the revised packaging and transport requirements for fissile material shipments.

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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 unclear 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 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 also is 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 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	1:2,000
350 g	1:200[a]

[a]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 percent ^{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 percent 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
Estimation of Values and Impacts for Proposed Actions

Technical Issue 2: Revision to Radionuclide Exemption Values

The nature of the proposed change makes it difficult to quantify the safety impacts or benefits. Because exempt packages are not required to adhere to the reporting requirements of 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." 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 TS-R-1. The results are shown in the Table B-1 below.

Table B-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 Sandia report
[2] - Derived from Sandia report

Out of the six radionuclides examined, two (Tc-99m and Mo-99) would have a higher exemption level than the current 70 Bq/g, and 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 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 TS-R-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 was handled, since it is generally at or near a level that would affect contaminated waste handling, not product distribution.

The following isotopes have 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 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. 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 that 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 (DOE, 1997) 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.

No public health or safety problems were identified for the current exemption level of 70 Bq/g for all radionuclides. In the hundreds of thousands of shipments that span five decades, no public health or safety impact attributable to the current exemption value provision has been identified.

The proposed exemption values do not provide a significant improvement in safety. The draft provisions would impose new complexity and economic burdens to the transportation industry. The new use of a formula to determine the exemption of mixtures of radionuclides would be a burden on licensees and may lead to errors in use. The draft provisions may decrease harmony between IAEA and member states' regulations if the lack of economic merit for the proposed changes leads to the U.S. and other member states adopting provisions different from those in TS-R-1.

Technical Issue 3: Revision of A_1 and A_2 Values

In general, the new A_1 and A_2 values are within a factor of about 3 of the earlier values; there are a few radionuclides where the new A_1 and A_2 values are outside this range. Approximately 40 radionuclides have new A_1 values higher than previous values by factors ranging between 10 and 100. This is due mainly to improved modeling for beta emitters. There are no new A_1 or A_2 values that are lower than the previous figures by more than a factor of 10. A few radionuclides previously listed are now excluded, but two additional ones have been added, i.e., both isomers of Eu-150 and Np-236.

In order to gain some insight into how the proposed revisions could affect packages in the U.S., ICF examined a 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." 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 new IAEA A_1 and A_2 values for the radionuclide. The results are shown in Table B-2.

Table B-2. A_1 and A_2 Values for Commonly Shipped Radionuclides

Radionuclide	Number of Packages Shipped Annually	Part 71 A_1 Values (TBq)	TS-R-1 A_1 Values (TBq)	Part 71 A_2 Values (TBq)	TS-R-1 A_2 Values (TBq)
Am-241	395,000	2	10	2×10^{-4}	1×10^{-3}
Tc-99m	570,000	8	10	8	4
I-125	267,000	20	20	2	3
Mo-99	219,000	0.6	1	0.5[a]	0.6
Ir-192	80,500	1	1	0.5	0.6
Cs-137	196,000	2	2	0.5	0.6

a. Part 71 allows 0.74 TBq (20 Ci) for domestic shipping of Mo-99.

For these six radionuclides, all of the A_1 values either increased or stayed the same. Five of the A_2 values increased and one A_2 value decreased. These proposed A_1 and A_2 values were compared to the average activity per package to determine whether the proposed change would have much impact on shippers. Without detailed information on the distribution of material quantities in packages actually transported, this average value is used for evaluation of impacts.

> <u>Americium-241</u>: The A_1 and A_2 values for Am-241 increased by a factor of 6.75 during the last revision of Part 71 in 1995 (60 FR 50248). ICF evaluated these changes to the A_1 and A_2 values using the same data available for this analysis. ICF found that practically all Am-241 was shipped in special form in packages with average curie values that were well below the proposed A_1 limit. Therefore, ICF concluded that the revised A_1 and A_2 values would not lead to changes in the amount of material transported per package, the number of packages transported per year, the type of package used for these shipments, or the risk impact for Am-241 shipments.
>
> The proposed A_1 and A_2 values are 10 TBq and 1×10^{-3} TBq, respectively, which are both higher by a factor of 5 than those currently in Part 71. The average curie quantity per package of Am-241 is 0.153 Ci (5.66×10^{-3} Tbq). Since the average value is well

below the proposed A_1 limit, and the fact that these changes would be smaller than the 1994 changes, it is concluded that the proposed change would have no impact on these shipments.

Technicium-99m: Under the proposed action, the A_1 value for Tc-99m would increase from 8 TBq to 10 TBq and the A_2 value would decrease from 8 TBq to 4 TBq. The average curie quantity per package of Tc-99m is 0.123 Ci (4.55 x 10^{-3} TBq) (Javitz et. al., 1985). This value is well below the proposed A_1 or A_2 value; therefore, it is concluded that the proposed change would have no impact on these shipments.

Iodine-125: Under the proposed action, the A_1 value for I-125 would stay the same while the A_2 value would increase from 2 to 3 TBq. The average curie quantity per package of I-125 is 0.001 Ci (3.7 x 10^{-5} TBq) (Javitz et. al., 1985). This value is well below the proposed A_1 or A_2 value; therefore, it is concluded that the proposed change would have no impact on these shipments.

Molybdenum-99: Under the proposed action, the A_1 value for Mo-99 would increase from 0.6 to 1 TBq. The current A_2 value for Mo-99 is 0.5 TBq per package. Adoption of the proposed regulation would increase that limit to 0.6 TBq per package. The average quantity of Mo-99 currently being transported is 5.53 curies (0.20 TBq) per package (Javitz et. al., 1985). This average value is below both the current and proposed A_2 limits. There may be, however, specific cases in which the quantity currently being shipped exceeds the proposed A_2 limit. A specific example is the shipment of Mo-99/Tc-99m radiopharmaceutical generators. The NRC allows up to 0.74 TBq (20 Ci) for domestic shipments in a Type A package.

Since, on the average, neither the quantity shipped per package nor the package type used is likely to be affected, no change in the consequences of a hypothesized accident can be expected. Therefore, no significant change in the risk impact of these shipments is expected.

Iridium-192: Under the proposed action, the A_1 value for Ir-192 would remain at 1 TBq while the A_2 value would increase from 0.5 to 0.6 TBq. The average curie quantity per package of Ir-192 is 61.5 Ci (2.28 Tbq). This average value is already above the current limits on shipment in Type A packages; therefore, Type B packages would be used for these shipments. Since the A_1 value would stay the same and the A_2 value would increase only slightly, and the fact that the average Ir-192 shipment is already above the Type A package limit, it is concluded that the proposed change would have little impact on these shipments.

Cesium-137: Under the proposed action, the A_1 value for Cs-137 would remain at 2 TBq while the A_2 value would increase from 0.5 to 0.6 TBq. The average curie quantity per package of Cs-137 is 0.268 Ci (0.01 TBq). This average value is well below the proposed A_1 or A_2 value; therefore, it is concluded that the proposed change would have no impact on these shipments.

The A_1 and A_2 values in Part 71 were last revised in 1995. A regulatory and environmental impact analysis was developed for these revisions and concluded that there was no significant impact from adjusting the A_1 and A_2 values. This conclusion is still valid for these proposed changes.

For mixtures for which relevant data are not available, Table II of TS-R-1 provides A_1 and A_2 values. Unlike Part 71, the new Table II separates mixtures of alpha emitters from mixtures of unknown radionuclides. The current and proposed values are shown in Table B-3.

Table B-3. A_1 and A_2 Values for Mixtures of Unknown Radionuclides

Contents	Part 71 A_1 Values (TBq)	TS-R-1 A_1 Values (TBq)	Part 71 A_2 Values (TBq)	TS-R-1 A_2 Values (TBq)
Only beta or gamma emitting nuclides known to be present	0.2	0.1	0.02	0.02
Only alpha emitting nuclides are known to be present	0.10	0.2	2×10^{-5}	9×10^{-5}
No relevant data are available	0.10	0.001	2×10^{-5}	9×10^{-5}

The A_1 values have increased for alpha emitters and decreased for beta and gamma emitters and unknown radionuclides. The A_2 values have stayed the same for beta and gamma emitters and increased for alpha emitters and unknown radionuclides. There are no data available that estimate the number of packages of unknown radionuclides shipped each year for each of these categories; however, the number is believed to be small. Because the number is believed to be small, and because the above analysis shows little or no impact for changes in A_1 and A_2 values for packages with known radionuclides, it is concluded that changes in A_1 and A_2 for packages of unknown radionuclides also would have negligible impact.

Estimated Costs to Licensees

Licensee resources will have to be spent to evaluate changes reflected in Tables I and II of TS-R-1. As a result of the review of the changes, the licensees will have to expend varying levels of resources to update various aspects of their shipping programs. The licensees also may have to modify their shipping processes to assure compliance with new A_1 and A_2 values. However, these costs are expected to be small since shippers already have programs in place that use the A_1 and A_2 limits. Additionally, the analysis performed for the 1995 revision of the A_1 and A_2 values concluded that the cost to licensees would be negligible. This conclusion is still considered to be valid.

The revised A_1 and A_2 values may change the package types that must be used by a shipper; for example, an increased A_1 value may allow a shipper to use a Type A package rather than a more expensive Type B package. However, as discussed in the previous paragraphs, the six isotopes that are most commonly shipped were evaluated and it was determined that the proposed changes would not have a significant impact or cost to shippers due to changes in package types.

Estimated Costs to NRC

The changes to the A_1 and A_2 values are estimated to require approximately two staff-months of effort. Assuming a cost of $129 per hour for staff, and 20 days per month at 8 hours each, this results in a total cost of approximately $41,280. This estimation of staff time is consistent with that estimated by the NRC during the last revision of the A_1 and A_2 values.

Technical Issue 7: Deep Immersion Test

The proposed scope expansion to all packages containing more than 10^5 A_2 and all Type C packages may increase the number of shipments that are required to use packages that can successfully pass the enhanced deep immersion test. Under current Part 71 requirements, only some shipments of irradiated nuclear fuel are required to pass the deep immersion test. For the revised A_2 values in TS-R-1, 10^5 A_2 is a number ranging from 9 TBq (243 Ci) for Ac-227 to 4,000,000 TBq (1.08 x 10^8 Ci) for Ar-37, As-73, Co-58m, Fe-55, Ge-71, Kr-81, Np-235, Pd-103, Pt-193, Rh-103m, T (H-3), Tb-157, Tm-171, V-49, and Xe-131m.

Of approximately 2.9 million commercial packages of radioactive material shipped per year, only about 129,000 packages (4.4 percent) contain a curie content of more than 100 Ci. About 397 packages, or 0.01 percent, contain a curie content of more than 1,000 Ci. Out of a total of 32,000 DOE packages shipped annually, it is estimated that approximately 833 packages (2.6 percent) contain a curie content of more than 100 Ci; approximately 409 packages (1.3 percent) contain more than 1,000 Ci; approximately 124 packages (0.4 percent) contain more than 10,000 Ci; approximately 25 packages (0.08 percent) contain more than 100,000 Ci; and 17 packages (0.05 percent) contain more than 1,000,000 Ci. Most, if not all, of these high activity DOE packages are probably shipments of spent fuel, which may already be shipped in packages that meet the deep immersion requirement. These small percentages indicate that a very small number of packages would be affected by the proposed change in the testing requirement.

The package types that would certainly be affected by the proposed change would be those for spent fuel. The two largest shipping campaigns for spent fuel are expected to be the movement of commercial and DOE spent fuel to Yucca Mountain (or other approved repository) and the importation of foreign research reactor fuel for storage and disposal in the U.S. The typical inventories for these spent fuels were determined and the appropriate A_2 for the mixture of radionuclides was calculated, using the equation given in TS-R-1. The A_2 value was then multiplied by 10^5 and compared to the total activity of the package to assess whether the package would be required to meet the deep immersion test if Part 71 is revised. The formula for calculating the A_2 value for a mixture is:

$$A_2 = 1/\sum[f(i)/A_2(i)]$$

where A_2 is the A_2 for the mixture, f(i) is the fraction of activity of radionuclide i in the mixture, and $A_2(i)$ is the appropriate value of A_2 from TS-R-1.

The typical radionuclide activity for a pressurized water reactor fuel assembly that would be shipped to Yucca Mountain would have decayed for approximately 26 years. The total activity of the package would not exceed 1,000,000 Ci unless more than 12 assemblies were placed in a package. Therefore, under the current Part 71 regulations, the package would not have to meet the deep immersion test if less than 12 assemblies were shipped in one package. Evaluation against the proposed requirement revealed that, regardless of how many assemblies were shipped in a package, the activity of the average package would be higher than 10^5 A_2; therefore, the package would have to meet the deep immersion test if Part 71 is revised. These results are shown in Table B-4, assuming six fuel assemblies in a package.

Table B-4. Steps in Calculation of A_2 for PWR Spent Fuel

Isotope (i)	Curies per Assembly[a]	Curies per Package	TBq per Package	f(i)	A_2 from TS-R-1 (TBq)	f(i)/A_2
H-3	98.0	588.0	21.8	1.25E-03	40	3.13E-05
Co-60	150.0	900.0	33.3	1.92E-03	0.4	4.80E-03
Ni-59	1.3	7.8	0.3	1.66E-05	unlimited	0.00E+00
Ni-63	180.0	1,080.0	40.0	2.30E-03	30	7.68E-05
Kr-85	930.0	5,580.0	206.5	1.19E-02	10	1.19E-03
Sr-90	21,000.0	126,000.0	4,662.0	2.69E-01	0.3	8.96E-01
Zr-93	1.2	7.2	0.3	1.54E-05	unlimited	0.00E+00
Tc-99	7.1	42.6	1.6	9.08E-05	0.9	1.01E-04
Cs-134	16.0	96.0	3.6	2.05E-04	0.7	2.92E-04
Cs-137	31,000.0	186,000.0	6,882.0	3.97E-01	0.6	6.61E-01
Sm-151	190.0	1,140.0	42.2	2.43E-03	10	2.43E-04
Pu-238	1,700.0	10,200.0	377.4	2.17E-02	0.001	2.17E+01
Pu-239	180.0	1,080.0	40.0	2.30E-03	0.001	2.30E+00
Pu-240	270.0	1,620.0	59.9	3.45E-03	0.001	3.45E+00
Pu-241	20,000.0	120,000.0	4,440.0	2.56E-01	0.06	4.26E+00
Am-241	1,700.0	10,200.0	377.4	2.17E-02	0.001	2.17E+01
Am-242/242m	11.0	66.0	2.4	1.41E-04	0.001	1.41E-01
Am-243	13.0	78.0	2.9	1.66E-04	0.001	1.66E-01
Cm-242	8.7	52.2	1.9	1.11E-04	0.01	1.11E-02
Cm-243	8.3	49.8	1.8	1.06E-04	0.001	1.06E-01
Cm-244	700.0	4,200.0	155.4	8.96E-03	0.002	4.48E+00
Total	**78,164.6**	**468,987.6**	**17,352.5**	**1.0**		**60.0**

a. Obtained from DOE 1999.

B-5. Steps in Calculation of A_2 for BR-2 Spent Fuel (Continued)

Using the f(i)/A_2 value obtained from Table B-4, the A_2 for the mixture equals 0.017 TBq and 10^5 A_2 equals 1,700 TBq. Therefore, the activity of the spent fuel (17,352.5 TBq) is higher than the 10^5 A_2 value for the mixture (1,700 Tbq), and the fuel would need to be shipped in a cask that can pass the deep immersion test.

The bounding fuel type for foreign research reactors would be BR-2 fuel that had decayed for less than one year. The total activity of the package would be slightly less than 1,000,000 Ci. Therefore, under the current Part 71 requirements, the package would not be required to meet the deep immersion test. Additionally, the calculation of A_2 for this fuel type revealed that the total activity of the package would not exceed 10^5 A_2; therefore, the package also would not have to meet the deep immersion test if Part 71 was revised. However, since these packages are coming from foreign countries that may already have regulations that are consistent with IAEA standards, any such packages that exceed the 10^5 A_2 limit may already be certified to the deep immersion test. A shipment of research reactor fuel from South Korea was in fact placed in a cask that meets the deep immersion test. Additionally, shipments of these foreign fuel types are only slightly below the limits of 1,000,000 Ci or 10^5 A_2. These results are shown in Table B-5.

Table B-5. Steps in Calculation of A_2 for BR-2 Spent Fuel

Isotope (i)	Curies per package[b]	TBq per package	f(i)	A_2 from TS-R-1 (TBq)	f(i)/A_2
H-3	86.4	3.2	9.28E-05	40.0	2.32E-06
Kr-85	2,470.0	91.4	2.65E-03	10.0	2.65E-04
Sr-89	40,800.0	1,509.6	4.38E-02	0.6	7.30E-02
Sr-90	20,800.0	769.6	2.23E-02	0.3	7.45E-02
Y-90	20,800.0	769.6	2.23E-02	0.3	7.45E-02
Y-91	73,000.0	2,701.0	7.84E-02	0.6	1.31E-01
Zr-95	107,000.0	3,959.0	1.15E-01	0.8	1.44E-01
Nb-95	220,000.0	8,140.0	2.36E-01	1.0	2.36E-01
Ru-103	8,900.0	329.3	9.56E-03	2.0	4.78E-03
Rh-103m	8,900.0	329.3	9.56E-03	40.0	2.39E-04
Ru-106	21,500.0	795.5	2.31E-02	0.2	1.15E-01
Sn-123	427.0	15.8	4.59E-04	0.6	7.64E-04
Sb-125	890.0	32.9	9.56E-04	1.0	9.56E-04
Te-125m	212.0	7.8	2.28E-04	0.9	2.53E-04
Te-127m	887.0	32.8	9.53E-04	0.5	1.91E-03
Te-129m	189.0	7.0	2.03E-04	0.4	5.07E-04
Cs-134	16,400.0	606.8	1.76E-02	0.7	2.52E-02
Cs-137	20,600.0	762.2	2.21E-02	0.6	3.69E-02
Ce-141	5,740.0	212.4	6.16E-03	0.6	1.03E-02

B-5. Steps in Calculation of A_2 for BR-2 Spent Fuel (Continued)

Isotope (i)	Curies per package[b]	TBq per package	f(i)	A_2 from TS-R-1 (TBq)	f(i)/A_2
Ce-144	312,000.0	11,544.0	3.35E-01	0.2	1.68E+00
Pm-147	48,300.0	1,787.1	5.19E-02	2.0	2.59E-02
Pm-148m	75.6	2.8	8.12E-05	0.7	1.16E-04
Eu-154	620.0	22.9	6.66E-04	0.6	1.11E-03
Eu-155	130.0	4.8	1.40E-04	3.0	4.65E-05
U-234	0.0	0.0	9.82E-10	0.006	1.64E-07
U-235	0.0	0.0	1.48E-08	unlimited	0.00E+00
U-238	0.0	0.0	3.66E-10	unlimited	0.00E+00
Pu-238	64.2	2.4	6.90E-05	0.001	6.90E-02
Pu-239	1.8	0.1	1.98E-06	0.001	1.98E-03
Pu-240	1.2	0.0	1.29E-06	0.001	1.29E-03
Pu-241	284.0	10.5	3.05E-04	0.06	5.08E-03
Am-241	0.4	0.0	4.25E-07	0.001	4.25E-04
Am-242m	0.0	0.0	1.13E-09	0.001	1.13E-06
Am-243	0.0	0.0	4.65E-09	0.001	4.65E-06
Cm-242	1.8	0.1	1.88E-06	0.01	1.88E-04
Cm-244	1.3	0.0	1.43E-06	0.002	7.14E-04
Total	**931,081.7**	**34,450.0**	**1.0**		**2.7**

b. Obtained from DOE 1996.

Using the f(i)/A_2 value obtained from Table B-5, the A_2 for the mixture equals 0.369 TBq and 10^5 A_2 equals 36,900 TBq. Therefore, the activity of the spent fuel (34,450 TBq) is lower than the 10^5 A_2 value for the mixture (36,900 TBq), and the fuel would not need to be shipped in a cask that can pass the deep immersion test.

The fact that some packages for spent fuel are currently required to pass the deep immersion test indicates that some spent fuel casks already meet this requirement. However, large quantities of other types of materials, which are currently shipped in other types of Type B packages, also may need to use a package that passes this requirement. Therefore, a new Type B package that meets the proposed standard may have to be designed, developed, and certified.

The radionuclides that comprise the largest number of commercial shipments (more than 100,000 packages per year), as well as radionuclides that had the highest average activity per package shipped (more than 1 Ci per package), were identified. The applicable 10^5 A_2 value was then compared to the average curie value per package. The results are shown in Table B-6.

B-5. Steps in Calculation of A_2 for BR-2 Spent Fuel (Continued)

Table B-6 shows that the average activity value per package is much lower than the 10^5 A_2 value for each of these radionuclides. This indicates that these packages would not be affected by the proposed change and would not have to meet the deep immersion test.

Table B-6. Comparison of 10^5 A_2 with Average Commercial Shipping Values

Radionuclide	10^5 A_2 from TS-R-1 (TBq)	Average TBq per Package[a]
Most Frequently Shipped Packages		
Am-241	100	0.004
Co-60[b]	40,000	0.3
Cs-137	60,000	0.01
I-123	300,000	0.0002
I-125	300,000	0.00004
I-131	70,000	0.0006
Mo-99[b]	60,000	0.2
Tc-99m	400,000	0.005
Tl-201	400,000	0.0006
Xe-133	1,000,000	0.01
Packages with Highest Average Curies per Package		
Au-198	60,000	0.2
Ce-144	20,000	0.04
Fe-55	4,000,000	0.05
Ir-192	60,000	2.3
Rb-86	50,000	0.08
U (natural)	Unlimited	0.07

a. From Javitz et. al.
b. Also has high average curies per package shipped

The DOE packages with high activity levels (more than 1,000 Ci per package) also were compared to the corresponding values of 10^5 A_2. The results are shown in Table B-7.

Table B-7. Comparison of 10^5 A_2 with Average DOE Shipping Values

Radionuclide	10^5 A_2 from TS-R-1 (TBq)	Average TBq per Package[a]
Ce-144	20,000	212
Cm-244	200	70
Cs-137	60,000	216
H-3	4,000,000	105
Ir-192	60,000	360
Kr-85	1,000,000	1,473
Sr-90	30,000	127
U-234	600[b]	2,675

a. From Javitz et. al.
b. Assumes the most conservative A_2 value for slow lung absorption.

For DOE packages, all the reviewed radionuclides would have an activity much less than 10^5 A_2, except for U-234. Although the packages containing U-234 have high activity, the number of packages shipped represents only 0.04 percent of the 32,000 packages shipped annually. Therefore, the number of packages affected would be small.

Occupational Health (Accident)

The deep immersion test would be for packages containing activity of more than 10^5 A_2, 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. 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 regulatory limits for dose rate.

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 is 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. 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.

Estimated Costs to Industry

The implementation of additional deep immersion testing will require manufacturers to evaluate testing procedures and container designs. This may require significant amounts of time. Some spent fuel packages already meet the requirement for the deep immersion test, although it is unclear how many. Therefore, it was assumed that all 24 currently licensed spent fuel casks would be tested.

Most container models certified for spent fuel have metal-on-metal seals and heavy closure devices. External pressure will help seal the cask unless a pressure level is reached at which significant deformation of the closure mechanism or the seals or lids occurs.

Cask designs are currently evaluated by the use of air pressure tests, computer simulations, and material strength calculations. The added need to evaluate cask designs for the possibility of loss of containment integrity could considerably increase the time required for certification. At a minimum, the manufacturers could expend one month to reevaluate designs and apply for recertification. A month of such work has been estimated to cost approximately $8,300 in 1994. Assuming an escalation of 4 percent per year would increase this cost to $10,200 in 1999. More typically, a cask design would be evaluated with special attention to seals and closures. This is expected to take approximately three months and cost about $30,600. At a maximum, some cask designers would find it necessary to review test calculations, check seals and closure mechanisms, and modify the designs to withstand the deep immersion test. This effort may require up to one full year for each design and could cost as much as $122,000.

If each of the 24 casks is required to undergo the reevaluation, the total costs to industry could range from $245,000 to $2,928,000, with the expected typical total cost to be near $734,000. These costs are an upper bound, because some casks are already certified as meeting the deep immersion test.

It is possible that packages of materials other than spent fuel may exceed the 10^5 A_2 limit. In this case, licensee resources may be expended to design and develop a new Type B package. Additional licensee resources may have to be expended if the enhanced Type B package must be used for shipments that previously would have been acceptable in another Type B package, assuming the new package is more expensive. However, the number of packages exceeding 10^5 A_2 that are not spent fuel is estimated to be exceedingly small, and thus licensees may be inclined to ship multiple packages containing less material rather than design a new package.

Estimated Costs to the NRC

NRC development costs would include such activities as preparation of documents, publishing notices of rulemaking, holding public hearings, and responding to public comments. It is estimated that the revision of the limits for the deep immersion test would require approximately two staff-months to complete. Assuming a cost of $129 per hour for staff, and 20 days per month at 8 hours each, this results in a total cost of approximately $41,280.

If the proposed action is adopted, the NRC will incur costs to implement the revised requirements. This may consist of such activities as developing procedures, reviewing and approving test results, recertifying packages, and taking other actions to assure compliance. It is expected that the revision of limits for the deep immersion test would require approximately one month per cask design. Assuming a cost of $129 per hour for staff, and 20 days per month at 8 hours each, this results in a total cost of approximately $20,640 per cask design and $495,360 for all casks.

The NRC also may incur operation costs. These are the recurring costs that are necessary to ensure continued compliance with the proposed rule. It is expected that implementation of the revised deep immersion testing limits will not create any significant change to current NRC operating costs.

APPENDIX C

APPENDIX C
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 D

Table D-1. A_1 and A_2 Values for Radionuclides

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Ac-225 (a)	Actinium (89)	$8.0X10^{-1}$	$2.2X10^{1}$	$6.0X10^{-3}$	$1.6X10^{-1}$	$2.1X10^{3}$	$5.8X10^{4}$
Ac-227 (a)		$9.0X10^{-1}$	$2.4X10^{1}$	$9.0X10^{-5}$	$2.4X10^{-3}$	2.7	$7.2X10^{1}$
Ac-228		$6.0X10^{-1}$	$1.6X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$8.4X10^{4}$	$2.2X10^{6}$
Ag-105	Silver (47)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$1.1X10^{3}$	$3.0X10^{4}$
Ag-108m (a)		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$9.7X10^{-1}$	$2.6X10^{1}$
Ag-110m (a)		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.8X10^{2}$	$4.7X10^{3}$
Ag-111		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$5.8X10^{3}$	$1.6X10^{5}$
Al-26	Aluminum (13)	$1.0X10^{-1}$	2.7	$1.0X10^{-1}$	2.7	$7.0X10^{-4}$	$1.9X10^{-2}$
Am-241	Americium (95)	$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$1.3X10^{-1}$	3.4
Am-242m (a)		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$3.6X10^{-1}$	$1.0X10^{1}$
Am-243 (a)		5.0	$1.4X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$7.4X10^{-3}$	$2.0X10^{-1}$
Ar-37	Argon (18)	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$3.7X10^{3}$	$9.9X10^{4}$
Ar-39		$2.0X10^{1}$	$5.4X10^{2}$	$4.0X10^{1}$	$1.1X10^{3}$	1.3	$3.4X10^{1}$
Ar-41		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.5X10^{6}$	$4.2X10^{7}$
As-72	Arsenic (33)	$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$6.2X10^{4}$	$1.7X10^{6}$
As-73		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$8.2X10^{2}$	$2.2X10^{4}$
As-74		1.0	$2.7X10^{1}$	$9.0X10^{-1}$	$2.4X10^{1}$	$3.7X10^{3}$	$9.9X10^{4}$
As-76		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$5.8X10^{4}$	$1.6X10^{6}$
As-77		$2.0X10^{1}$	$5.4X10^{2}$	$7.0X10^{-1}$	$1.9X10^{1}$	$3.9X10^{4}$	$1.0X10^{6}$
At-211 (a)	Astatine (85)	$2.0X10^{1}$	$5.4X10^{2}$	$5.0X10^{-1}$	$1.4X10^{1}$	$7.6X10^{4}$	$2.1X10^{6}$
Au-193	Gold (79)	7.0	$1.9X10^{2}$	2.0	$5.4X10^{1}$	$3.4X10^{4}$	$9.2X10^{5}$
Au-194		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$1.5X10^{4}$	$4.1X10^{5}$
Au-195	Gold (79)	$1.0X10^{1}$	$2.7X10^{2}$	6.0	$1.6X10^{2}$	$1.4X10^{2}$	$3.7X10^{3}$
Au-198		1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$9.0X10^{3}$	$2.4X10^{5}$
Au-199		$1.0X10^{1}$	$2.7X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$7.7X10^{3}$	$2.1X10^{5}$
Ba-131 (a)	Barium (56)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$3.1X10^{3}$	$8.4X10^{4}$
Ba-133		3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	9.4	$2.6X10^{2}$
Ba-133m		$2.0X10^{1}$	$5.4X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.2X10^{4}$	$6.1X10^{5}$
Ba-140 (a)		$5.0X10^{-1}$	$1.4X10^{1}$	$3.0X10^{-1}$	8.1	$2.7X10^{3}$	$7.3X10^{4}$
Be-7	Beryllium (4)	$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{1}$	$5.4X10^{2}$	$1.3X10^{4}$	$3.5X10^{5}$
Be-10		$4.0X10^{1}$	$1.1X10^{3}$	$6.0X10^{-1}$	$1.6X10^{1}$	$8.3X10^{-4}$	$2.2X10^{-2}$
Bi-205	Bismuth (83)	$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$1.5X10^{-3}$	$4.2X10^{4}$
Bi-206		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$3.8X10^{3}$	$1.0X10^{5}$
Bi-207		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	1.9	$5.2X10^{1}$
Bi-210		1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$4.6X10^{3}$	$1.2X10^{5}$
Bi-210m (a)		$6.0X10^{-1}$	$1.6X10^{1}$	$2.0X10^{-2}$	$5.4X10^{-1}$	$2.1X10^{-5}$	$5.7X10^{-4}$
Bi-212 (a)		$7.0X10^{-1}$	$1.9X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$5.4X10^{5}$	$1.5X10^{7}$

Table D-1. A_1 and A_2 Values for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Bk-247	Berkelium (97)	8.0	$2.2X10^{2}$	$8.0X10^{-4}$	$2.2X10^{-2}$	$3.8X10^{-2}$	1.0
Bk-249 (a)		$4.0X10^{1}$	$1.1X10^{3}$	$3.0X10^{-1}$	8.1	$6.1X10^{1}$	$1.6X10^{3}$
Br-76	Bromine (35)	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$9.4X10^{4}$	$2.5X10^{6}$
Br-77		3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$2.6X10^{4}$	$7.1X10^{5}$
Br-82		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{4}$	$1.1X10^{6}$
C-11	Carbon (6)	1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.1X10^{7}$	$8.4X10^{8}$
C-14		$4.0X10^{1}$	$1.1X10^{3}$	3.0	$8.1X10^{1}$	$1.6X10^{-1}$	4.5
Ca-41	Calcium (20)	Unlimited	Unlimited	Unlimited	Unlimited	$3.1X10^{-3}$	$8.5X10^{-2}$
Ca-45		$4.0X10^{1}$	$1.1X10^{3}$	1.0	$2.7X10^{1}$	$6.6X10^{2}$	$1.8X10^{4}$
Ca-47 (a)		3.0	$8.1X10^{1}$	$3.0X10^{-1}$	8.1	$2.3X10^{4}$	$6.1X10^{5}$
Cd-109	Cadmium (48)	$3.0X10^{1}$	$8.1X10^{2}$	2.0	$5.4X10^{1}$	$9.6X10^{1}$	$2.6X10^{3}$
Cd-113m		$4.0X10^{1}$	$1.1X10^{3}$	$5.0X10^{-1}$	$1.4X10^{1}$	8.3	$2.2X10^{2}$
Cd-115 (a)		3.0	$8.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.9X10^{4}$	$5.1X10^{5}$
Cd-115m		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$9.4X10^{2}$	$2.5X10^{4}$
Ce-139	Cerium (58)	7.0	$1.9X10^{2}$	2.0	$5.4X10^{1}$	$2.5X10^{2}$	$6.8X10^{3}$
Ce-141		$2.0X10^{1}$	$5.4X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.1X10^{3}$	$2.8X10^{4}$
Ce-143		$9.0X10^{-1}$	$2.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.5X10^{4}$	$6.6X10^{5}$
Ce-144 (a)		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$1.2X10^{2}$	$3.2X10^{3}$
Cf-248	Californium (98)	$4.0X10^{1}$	$1.1X10^{3}$	$6.0X10^{-3}$	$1.6X10^{-1}$	$5.8X10^{1}$	$1.6X10^{3}$
Cf-249		3.0	$8.1X10^{1}$	$8.0X10^{-4}$	$2.2X10^{-2}$	$1.5X10^{-1}$	4.1
Cf-250		$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{-3}$	$5.4X10^{-2}$	4.0	$1.1X10^{2}$
Cf-251		7.0	$1.9X10^{2}$	$7.0X10^{-4}$	$1.9X10^{-2}$	$5.9X10^{-2}$	1.6
Cf-252		$5.0X10^{-2}$	1.4	$3.0X10^{-3}$	$8.1X10^{-2}$	$2.0X10^{1}$	$5.4X10^{2}$
Cf-253 (a)		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{-2}$	1.1	$1.1X10^{3}$	$2.9X10^{4}$
Cf-254		$1.0X10^{-3}$	$2.7X10^{-2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$3.1X10^{2}$	$8.5X10^{3}$
Cl-36	Chlorine (17)	$1.0X10^{1}$	$2.7X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.2X10^{-3}$	$3.3X10^{-2}$
Cl-38		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$4.9X10^{6}$	$1.3X10^{8}$
Cm-240	Curium (96)	$4.0X10^{1}$	$1.1X10^{3}$	$2.0X10^{-2}$	$5.4X10^{-1}$	$7.5X10^{2}$	$2.0X10^{4}$
Cm-241		2.0	$5.4X10^{1}$	1.0	$2.7X10^{1}$	$6.1X10^{2}$	$1.7X10^{4}$
Cm-242	Curium (96)	$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-2}$	$2.7X10^{-1}$	$1.2X10^{2}$	$3.3X10^{3}$
Cm-243		9.0	$2.4X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$1.9X10^{-3}$	$5.2X10^{1}$
Cm-244		$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{-3}$	$5.4X10^{-2}$	3.0	$8.1X10^{1}$
Cm-245		9.0	$2.4X10^{2}$	$9.0X10^{-4}$	$2.4X10^{-2}$	$6.4X10^{-3}$	$1.7X10^{-1}$
Cm-246		9.0	$2.4X10^{2}$	$9.0X10^{-4}$	$2.4X10^{-2}$	$1.1X10^{-2}$	$3.1X10^{-1}$
Cm-247 (a)		3.0	$8.1X10^{1}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$3.4X10^{-6}$	$9.3X10^{-5}$
Cm-248		$2.0X10^{-2}$	$5.4X10^{-1}$	$3.0X10^{-4}$	$8.1X10^{-3}$	$1.6X10^{-5}$	$4.2X10^{-3}$

Table D-1. A_1 and A_2 Values for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Co-55	Cobalt (27)	$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$1.1X10^{5}$	$3.1X10^{6}$
Co-56		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.1X10^{3}$	$3.0X10^{4}$
Co-57		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$3.1X10^{2}$	$8.4X10^{3}$
Co-58		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$1.2X10^{3}$	$3.2X10^{4}$
Co-58m		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$2.2X10^{5}$	$5.9X10^{6}$
Co-60		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$4.2X10^{1}$	$1.1X10^{3}$
Cr-51	Chromium (24)	$3.0X10^{1}$	$8.1X10^{2}$	$3.0X10^{1}$	$8.1X10^{2}$	$3.4X10^{3}$	$9.2X10^{4}$
Cs-129	Cesium (55)	4.0	$1.1X10^{2}$	4.0	$1.1X10^{2}$	$2.8X10^{4}$	$7.6X10^{5}$
Cs-131		$3.0X10^{1}$	$8.1X10^{2}$	$3.0X10^{1}$	$8.1X10^{2}$	$3.8X10^{3}$	$1.0X10^{5}$
Cs-132		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$5.7X10^{3}$	$1.5X10^{5}$
Cs-134		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$4.8X10^{1}$	$1.3X10^{3}$
Cs-134m		$4.0X10^{1}$	$1.1X10^{3}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.0X10^{5}$	$8.0X10^{6}$
Cs-135		$4.0X10^{1}$	$1.1X10^{3}$	1.0	$2.7X10^{1}$	$4.3X10^{-5}$	$1.2X10^{-3}$
Cs-136		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$2.7X10^{3}$	$7.3X10^{4}$
Cs-137 (a)		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	3.2	$8.7X10^{1}$
Cu-64	Copper (29)	6.0	$1.6X10^{2}$	1.0	$2.7X10^{1}$	$1.4X10^{5}$	$3.9X10^{6}$
Cu-67		$1.0X10^{1}$	$2.7X10^{2}$	$7.0X10^{-1}$	$1.9X10^{1}$	$2.8X10^{4}$	$7.6X10^{5}$
Dy-159	Dysprosium (66)	$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{1}$	$5.4X10^{2}$	$2.1X10^{2}$	$5.7X10^{3}$
Dy-165		$9.0X10^{-1}$	$2.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.0X10^{5}$	$8.2X10^{6}$
Dy-166 (a)		$9.0X10^{-1}$	$2.4X10^{1}$	$3.0X10^{-1}$	8.1	$8.6X10^{3}$	$2.3X10^{5}$
Er-169	Erbium (68)	$4.0X10^{1}$	$1.1X10^{3}$	1.0	$2.7X10^{1}$	$3.1X10^{3}$	$8.3X10^{4}$
Er-171		$8.0X10^{-1}$	$2.2X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$9.0X10^{4}$	$2.4X10^{6}$
Eu-147	Europium (63)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$1.4X10^{3}$	$3.7X10^{4}$
Eu-148		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$6.0X10^{2}$	$1.6X10^{4}$
Eu-149		$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{1}$	$5.4X10^{2}$	$3.5X10^{2}$	$9.4X10^{3}$
Eu-150 (short lived)		2.0	$5.4X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$6.1X10^{4}$	$1.6X10^{6}$
Eu-150 (long lived)		2.0	$5.4X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$6.1X10^{4}$	$1.6X10^{6}$
Eu-152		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	6.5	$1.8X10^{2}$
Eu-152m		$8.0X10^{-1}$	$2.2X10^{1}$	$8.0X10^{-1}$	$2.2X10^{1}$	$8.2X10^{4}$	$2.2X10^{6}$
Eu-154		$9.0X10^{-1}$	$2.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	9.8	$2.6X10^{2}$
Eu-155		$2.0X10^{1}$	$5.4X10^{2}$	3.0	$8.1X10^{1}$	$1.8X10^{1}$	$4.9X10^{2}$
Eu-156		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$2.0X10^{3}$	$5.5X10^{4}$
F-18	Fluorine (9)	1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.5X10^{6}$	$9.5X10^{7}$

Table D-1. A_1 and A_2 Values for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Fe-52 (a)	Iron (26)	$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$2.7X10^{5}$	$7.3X10^{6}$
Fe-55		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$8.8X10^{1}$	$2.4X10^{3}$
Fe-59		$9.0X10^{-1}$	$2.4X10^{1}$	$9.0X10^{-1}$	$2.4X10^{1}$	$1.8X10^{3}$	$5.0X10^{4}$
Fe-60 (a)		$4.0X10^{1}$	$1.1X10^{3}$	$2.0X10^{-1}$	5.4	$7.4X10^{-4}$	$2.0X10^{-2}$
Ga-67	Gallium (31)	7.0	$1.9X10^{2}$	3.0	$8.1X10^{1}$	$2.2X10^{4}$	$6.0X10^{5}$
Ga-68		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$1.5X10^{6}$	$4.1X10^{7}$
Ga-72		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.1X10^{5}$	$3.1X10^{6}$
Gd-146 (a)	Gadolinium (64)	$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$6.9X10^{2}$	$1.9X10^{4}$
Gd-148		$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{-3}$	$5.4X10^{-2}$	1.2	$3.2X10^{1}$
Gd-153		$1.0X10^{1}$	$2.7X10^{2}$	9.0	$2.4X10^{2}$	$1.3X10^{2}$	$3.5X10^{3}$
Gd-159		3.0	$8.1X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.9X10^{4}$	$1.1X10^{6}$
Ge-68 (a)	Germanium (32)	$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$2.6X10^{2}$	$7.1X10^{3}$
Ge-71		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$5.8X10^{3}$	$1.6X10^{5}$
Ge-77		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.3X10^{5}$	$3.6X10^{6}$
Hf-172 (a)	Hafnium (72)	$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$4.1X10^{1}$	$1.1X10^{3}$
Hf-175		3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$3.9X10^{2}$	$1.1X10^{4}$
Hf-181		2.0	$5.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$6.3X10^{2}$	$1.7X10^{4}$
Hf-182		Unlimited	Unlimited	Unlimited	Unlimited	$8.1X10^{-6}$	$2.2X10^{-4}$
Hg-194 (a)	Mercury (80)	1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$1.3X10^{-1}$	3.5
Hg-195m (a)		3.0	$8.1X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$1.5X10^{4}$	$4.0X10^{5}$
Hg-197		$2.0X10^{1}$	$5.4X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$9.2X10^{3}$	$2.5X10^{5}$
Hg-197m		$1.0X10^{1}$	$2.7X10^{2}$	$4.0X10^{-1}$	$1.1X10^{1}$	$2.5X10^{4}$	$6.7X10^{5}$
Hg-203		5.0	$1.4X10^{2}$	1.0	$2.7X10^{1}$	$5.1X10^{2}$	$1.4X10^{4}$
Ho-166	Holmium (67)	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$2.6X10^{4}$	$7.0X10^{5}$
Ho-166m		$6.0X10^{-1}$	$1.6X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$6.6X10^{-2}$	1.8
I-123	Iodine (53)	6.0	$1.6X10^{2}$	3.0	$8.1X10^{1}$	$7.1X10^{4}$	$1.9X10^{6}$
I-124		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$9.3X10^{3}$	$2.5X10^{5}$
I-125		$2.0X10^{1}$	$5.4X10^{2}$	3.0	$8.1X10^{1}$	$6.4X10^{2}$	$1.7X10^{4}$
I-126		2.0	$5.4X10^{1}$	1.0	$2.7X10^{1}$	$2.9X10^{3}$	$8.0X10^{4}$
I-129		Unlimited	Unlimited	Unlimited	Unlimited	$6.5X10^{-6}$	$1.8X10^{-4}$
I-131		3.0	$8.1X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$4.6X10^{3}$	$1.2X10^{5}$
I-132		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$3.8X10^{5}$	$1.0X10^{7}$
I-133		$7.0X10^{-1}$	$1.9X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$4.2X10^{4}$	$1.1X10^{6}$
I-134		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$9.9X10^{5}$	$2.7X10^{7}$
I-135 (a)		$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.3X10^{5}$	$3.5X10^{6}$

Table D-1. A_1 and A_2 Values for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
In-111	Indium (49)	3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$1.5X10^{4}$	$4.2X10^{5}$
In-113m		4.0	$1.1X10^{2}$	2.0	$5.4X10^{1}$	$6.2X10^{5}$	$1.7X10^{7}$
In-114m (a)		$1.0X10^{1}$	$2.7X10^{2}$	$5.0X10^{-1}$	$1.4X10^{1}$	$8.6X10^{2}$	$2.3X10^{4}$
In-115m		7.0	$1.9X10^{2}$	1.0	$2.7X10^{1}$	$2.2X10^{5}$	$6.1X10^{6}$
Ir-189 (a)	Iridium (77)	$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$1.9X10^{3}$	$5.2X10^{4}$
Ir-190		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$2.3X10^{3}$	$6.2X10^{4}$
Ir-192		1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.4X10^{2}$	$9.2X10^{3}$
Ir-194		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$3.1X10^{4}$	$8.4X10^{5}$
K-40	Potassium (19)	$9.0X10^{-1}$	$2.4X10^{1}$	$9.0X10^{-1}$	$2.4X10^{1}$	$2.4X10^{-7}$	$6.4X10^{-6}$
K-42		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$2.2X10^{5}$	$6.0X10^{6}$
K-43		$7.0X10^{-1}$	$1.9X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.2X10^{5}$	$3.3X10^{6}$
Kr-81	Krypton (36)	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$7.8X10^{-4}$	$2.1X10^{-2}$
Kr-85		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$1.5X10^{1}$	$3.9X10^{2}$
Kr-85m		8.0	$2.2X10^{2}$	3.0	$8.1X10^{1}$	$3.0X10^{5}$	$8.2X10^{6}$
Kr-87		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$1.0X10^{6}$	$2.8X10^{7}$
La-137	Lanthanum (57)	$3.0X10^{1}$	$8.1X10^{2}$	6.0	$1.6X10^{2}$	$1.6X10^{-3}$	$4.4X10^{-2}$
La-140		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$2.1X10^{4}$	$5.6X10^{5}$
Lu-172	Lutetium (71)	$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$4.2X10^{3}$	$1.1X10^{5}$
Lu-173		8.0	$2.2X10^{2}$	8.0	$2.2X10^{2}$	$5.6X10^{1}$	$1.5X10^{3}$
Lu-174		9.0	$2.4X10^{2}$	9.0	$2.4X10^{2}$	$2.3X10^{1}$	$6.2X10^{2}$
Lu-174m		$2.0X10^{1}$	$5.4X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$2.0X10^{2}$	$5.3X10^{3}$
Lu-177		$3.0X10^{1}$	$8.1X10^{2}$	$7.0X10^{-1}$	$1.9X10^{1}$	$4.1X10^{3}$	$1.1X10^{5}$
Mg-28 (a)	Magnesium (12)	$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$2.0X10^{5}$	$5.4X10^{6}$
Mn-52	Manganese (25)	$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.6X10^{4}$	$4.4X10^{5}$
Mn-53		Unlimited	Unlimited	Unlimited	Unlimited	$6.8X10^{-5}$	$1.8X10^{-3}$
Mn-54		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$2.9X10^{2}$	$7.7X10^{3}$
Mn-56		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$8.0X10^{5}$	$2.2X10^{7}$
Mo-93	Molybdenum (42)	$4.0X10^{1}$	$1.1X10^{3}$	$2.0X10^{1}$	$5.4X10^{2}$	$4.1X10^{-2}$	1.1
Mo-99 (a)		1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.8X10^{4}$	$4.8X10^{5}$
N-13	Nitrogen (7)	$9.0X10^{-1}$	$2.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$5.4X10^{7}$	$1.5X10^{9}$
Na-22	Sodium (11)	$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$2.3X10^{2}$	$6.3X10^{3}$
Na-24		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$3.2X10^{5}$	$8.7X10^{6}$
Nb-93m	Niobium (41)	$4.0X10^{1}$	$1.1X10^{3}$	$3.0X10^{1}$	$8.1X10^{2}$	8.8	$2.4X10^{2}$
Nb-94		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$6.9X10^{-3}$	$1.9X10^{-1}$
Nb-95		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$1.5X10^{3}$	$3.9X10^{4}$
Nb-97		$9.0X10^{-1}$	$2.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$9.9X10^{5}$	$2.7X10^{7}$
Nd-147	Neodymium (60)	6.0	$1.6X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.0X10^{3}$	$8.1X10^{4}$

Table D-1. A_1 and A_2 Values for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Nd-149		$6.0X10^{-1}$	$1.6X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$4.5X10^{5}$	$1.2X10^{7}$
Ni-59	Nickel (28)	Unlimited	Unlimited	Unlimited	Unlimited	$3.0X10^{-3}$	$8.0X10^{-2}$
Ni-63		$4.0X10^{1}$	$1.1X10^{3}$	$3.0X10^{1}$	$8.1X10^{2}$	2.1	$5.7X10^{1}$
Ni-65		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$7.1X10^{5}$	$1.9X10^{7}$
Np-235	Neptunium (93)	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$5.2X10^{1}$	$1.4X10^{3}$
Np-236 (short-lived)		$2.0X10^{1}$	$5.4X10^{2}$	2.0	$5.4X10^{1}$	$4.7X10^{-4}$	$1.3X10^{-2}$
Np-236 (long-lived)		$2.0X10^{1}$	$5.4X10^{2}$	2.0	$5.4X10^{1}$	$4.7X10^{-4}$	$1.3X10^{-2}$
Np-237		$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{-3}$	$5.4X10^{-2}$	$2.6X10^{-5}$	$7.1X10^{-4}$
Np-239		7.0	$1.9X10^{2}$	$4.0X10^{-1}$	$1.1X10^{1}$	$8.6X10^{3}$	$2.3X10^{5}$
Os-185	Osmium (76)	1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$2.8X10^{2}$	$7.5X10^{3}$
Os-191		$1.0X10^{1}$	$2.7X10^{2}$	2.0	$5.4X10^{1}$	$1.6X10^{3}$	$4.4X10^{4}$
Os-191m		$4.0X10^{1}$	$1.1X10^{3}$	$3.0X10^{1}$	$8.1X10^{2}$	$4.6X10^{4}$	$1.3X10^{6}$
Os-193		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.0X10^{4}$	$5.3X10^{5}$
Os-194 (a)		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.1X10^{1}$	$3.1X10^{2}$
P-32	Phosphorus (15)	$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$1.1X10^{4}$	$2.9X10^{5}$
P-33		$4.0X10^{1}$	$1.1X10^{3}$	1.0	$2.7X10^{1}$	$5.8X10^{3}$	$1.6X10^{5}$
Pa-230 (a)	Protactinium (91)	2.0	$5.4X10^{1}$	$7.0X10^{-2}$	1.9	$1.2X10^{3}$	$3.3X10^{4}$
Pa-231		4.0	$1.1X10^{2}$	$4.0X10^{-4}$	$1.1X10^{-2}$	$1.7X10^{-3}$	$4.7X10^{-2}$
Pa-233		5.0	$1.4X10^{2}$	$7.0X10^{-1}$	$1.9X10^{1}$	$7.7X10^{2}$	$2.1X10^{4}$
Pb-201	Lead (82)	1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$6.2X10^{4}$	$1.7X10^{6}$
Pb-202		$4.0X10^{1}$	$1.1X10^{3}$	$2.0X10^{1}$	$5.4X10^{2}$	$1.2X10^{-4}$	$3.4X10^{-3}$
Pb-203		4.0	$1.1X10^{2}$	3.0	$8.1X10^{1}$	$1.1X10^{4}$	$3.0X10^{5}$
Pb-205		Unlimited	Unlimited	Unlimited	Unlimited	$4.5X10^{-6}$	$1.2X10^{-4}$
Pb-210 (a)		1.0	$2.7X10^{1}$	$5.0X10^{-2}$	1.4	2.8	$7.6X10^{1}$
Pb-212 (a)		$7.0X10^{-1}$	$1.9X10^{1}$	$2.0X10^{-1}$	5.4	$5.1X10^{4}$	$1.4X10^{6}$
Pd-103 (a)	Palladium (46)	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$2.8X10^{3}$	$7.5X10^{4}$
Pd-107		Unlimited	Unlimited	Unlimited	Unlimited	$1.9X10^{-5}$	$5.1X10^{-4}$
Pd-109		2.0	$5.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$7.9X10^{4}$	$2.1X10^{6}$

Table D-1. A_1 and A_2 Values for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Pm-143	Promethium (61)	3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$1.3X10^{2}$	$3.4X10^{3}$
Pm-144		$7.0X10^{-1}$	$1.9X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$9.2X10^{1}$	$2.5X10^{3}$
Pm-145		$3.0X10^{1}$	$8.1X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	5.2	$1.4X10^{2}$
Pm-147		$4.0X10^{1}$	$1.1X10^{3}$	2.0	$5.4X10^{1}$	$3.4X10^{1}$	$9.3X10^{2}$
Pm-148m (a)		$8.0X10^{-1}$	$2.2X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$7.9X10^{2}$	$2.1X10^{4}$
Pm-149		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.5X10^{4}$	$4.0X10^{5}$
Pm-151		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.7X10^{4}$	$7.3X10^{5}$
Po-210	Polonium (84)	$4.0X10^{1}$	$1.1X10^{3}$	$2.0X10^{-2}$	$5.4X10^{-1}$	$1.7X10^{2}$	$4.5X10^{3}$
Pr-142	Praseodymium (59)	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$4.3X10^{4}$	$1.2X10^{6}$
Pr-143		3.0	$8.1X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.5X10^{3}$	$6.7X10^{4}$
Pt-188 (a)	Platinum (78)	1.0	$2.7X10^{1}$	$8.0X10^{-1}$	$2.2X10^{1}$	$2.5X10^{3}$	$6.8X10^{4}$
Pt-191		4.0	$1.1X10^{2}$	3.0	$8.1X10^{1}$	$8.7X10^{3}$	$2.4X10^{5}$
Pt-193		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	1.4	$3.7X10^{1}$
Pt-193m		$4.0X10^{1}$	$1.1X10^{3}$	$5.0X10^{-1}$	$1.4X10^{1}$	$5.8X10^{3}$	$1.6X10^{5}$
Pt-195m		$1.0X10^{1}$	$2.7X10^{2}$	$5.0X10^{-1}$	$1.4X10^{1}$	$6.2X10^{3}$	$1.7X10^{5}$
Pt-197		$2.0X10^{1}$	$5.4X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.2X10^{4}$	$8.7X10^{5}$
Pt-197m		$1.0X10^{1}$	$2.7X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.7X10^{5}$	$1.0X10^{7}$
Pu-236	Plutonium (94)	$3.0X10^{1}$	$8.1X10^{2}$	$3.0X10^{-3}$	$8.1X10^{-2}$	$2.0X10^{1}$	$5.3X10^{2}$
Pu-237		$2.0X10^{1}$	$5.4X10^{2}$	$2.0X10^{1}$	$5.4X10^{2}$	$4.5X10^{2}$	$1.2X10^{4}$
Pu-238		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$6.3X10^{-1}$	$1.7X10^{1}$
Pu-239		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$2.3X10^{-3}$	$6.2X10^{-2}$
Pu-240		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$8.4X10^{-3}$	$2.3X10^{-1}$
Pu-241 (a)		$4.0X10^{1}$	$1.1X10^{3}$	$6.0X10^{-2}$	1.6	3.8	$1.0X10^{2}$
Pu-242		$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$1.5X10^{-4}$	$3.9X10^{-3}$
Pu-244 (a)		$4.0X10^{-1}$	$1.1X10^{1}$	$1.0X10^{-3}$	$2.7X10^{-2}$	$6.7X10^{-7}$	$1.8X10^{-5}$
Ra-223 (a)	Radium (88)	$4.0X10^{-1}$	$1.1X10^{1}$	$7.0X10^{-3}$	$1.9X10^{-1}$	$1.9X10^{3}$	$5.1X10^{4}$
Ra-224 (a)		$4.0X10^{-1}$	$1.1X10^{1}$	$2.0X10^{-2}$	$5.4X10^{-1}$	$5.9X10^{3}$	$1.6X10^{5}$
Ra-225 (a)		$2.0X10^{-1}$	5.4	$4.0X10^{-3}$	$1.1X10^{-1}$	$1.5X10^{3}$	$3.9X10^{4}$
Ra-226 (a)		$2.0X10^{-1}$	5.4	$3.0X10^{-3}$	$8.1X10^{-2}$	$3.7X10^{-2}$	1.0
Ra-228 (a)		$6.0X10^{-1}$	$1.6X10^{1}$	$2.0X10^{-2}$	$5.4X10^{-1}$	$1.0X10^{1}$	$2.7X10^{2}$
Rb-81	Rubidium (37)	2.0	$5.4X10^{1}$	$8.0X10^{-1}$	$2.2X10^{1}$	$3.1X10^{5}$	$8.4X10^{6}$
Rb-83 (a)		2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$6.8X10^{2}$	$1.8X10^{4}$
Rb-84		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$1.8X10^{3}$	$4.7X10^{4}$
Rb-86		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$3.0X10^{3}$	$8.1X10^{4}$
Rb-87		Unlimited	Unlimited	Unlimited	Unlimited	$3.2X10^{-9}$	$8.6X10^{-8}$
Rb(nat)		Unlimited	Unlimited	Unlimited	Unlimited	$6.7X10^{6}$	$1.8X10^{8}$
Re-184	Rhenium (75)	1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$6.9X10^{2}$	$1.9X10^{4}$

Table D-1. A_1 and A_2 Values for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Re-184m		3.0	$8.1X10^{1}$	1.0	$2.7X10^{1}$	$1.6X10^{2}$	$4.3X10^{3}$
Re-186		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$6.9X10^{3}$	$1.9X10^{5}$
Re-187		Unlimited	Unlimited	Unlimited	Unlimited	$1.4X10^{-9}$	$3.8X10^{-8}$
Re-188		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$3.6X10^{4}$	$9.8X10^{5}$
Re-189 (a)		3.0	$8.1X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.5X10^{4}$	$6.8X10^{5}$
Re(nat)		Unlimited	Unlimited	Unlimited	Unlimited	0.0	$2.4X10^{-8}$
Rh-99	Rhodium (45)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$3.0X10^{3}$	$8.2X10^{4}$
Rh-101		4.0	$1.1X10^{2}$	3.0	$8.1X10^{1}$	$4.1X10^{1}$	$1.1X10^{3}$
Rh-102		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$4.5X10^{1}$	$1.2X10^{3}$
Rh-102m		2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$2.3X10^{2}$	$6.2X10^{3}$
Rh-103m		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$1.2X10^{6}$	$3.3X10^{7}$
Rh-105		$1.0X10^{1}$	$2.7X10^{2}$	$8.0X10^{-1}$	$2.2X10^{1}$	$3.1X10^{4}$	$8.4X10^{5}$
Rn-222 (a)	Radon (86)	$3.0X10^{-1}$	8.1	$4.0X10^{-3}$	$1.1X10^{-1}$	$5.7X10^{3}$	$1.5X10^{5}$
Ru-97	Ruthenium (44)	5.0	$1.4X10^{2}$	5.0	$1.4X10^{2}$	$1.7X10^{4}$	$4.6X10^{5}$
Ru-103 (a)		2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$1.2X10^{3}$	$3.2X10^{4}$
Ru-105		1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.5X10^{5}$	$6.7X10^{6}$
Ru-106 (a)		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$1.2X10^{2}$	$3.3X10^{3}$
S-35	Sulphur (16)	$4.0X10^{1}$	$1.1X10^{3}$	3.0	$8.1X10^{1}$	$1.6X10^{3}$	$4.3X10^{4}$
Sb-122	Antimony (51)	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.5X10^{4}$	$4.0X10^{5}$
Sb-124		$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$6.5X10^{2}$	$1.7X10^{4}$
Sb-125		2.0	$5.4X10^{1}$	1.0	$2.7X10^{1}$	$3.9X10^{1}$	$1.0X10^{3}$
Sb-126		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$3.1X10^{3}$	$8.4X10^{4}$
Sc-44	Scandium (21)	$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$6.7X10^{5}$	$1.8X10^{7}$
Sc-46		$5.0X10^{-1}$	$1.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$1.3X10^{3}$	$3.4X10^{4}$
Sc-47		$1.0X10^{1}$	$2.7X10^{2}$	$7.0X10^{-1}$	$1.9X10^{1}$	$3.1X10^{4}$	$8.3X10^{5}$
Sc-48		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$5.5X10^{4}$	$1.5X10^{6}$
Se-75	Selenium (34)	3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$5.4X10^{2}$	$1.5X10^{4}$
Se-79		$4.0X10^{1}$	$1.1X10^{3}$	2.0	$5.4X10^{1}$	$2.6X10^{-3}$	$7.0X10^{-2}$
Si-31	Silicon (14)	$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.4X10^{6}$	$3.9X10^{7}$
Si-32		$4.0X10^{1}$	$1.1X10^{3}$	$5.0X10^{-1}$	$1.4X10^{1}$	3.9	$1.1X10^{2}$
Sm-145	Samarium (62)	$1.0X10^{1}$	$2.7X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$9.8X10^{1}$	$2.6X10^{3}$
Sm-147		Unlimited	Unlimited	Unlimited	Unlimited	$8.5X10^{-1}$	$2.3X10^{-8}$
Sm-151		$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{1}$	$2.7X10^{2}$	$9.7X10^{-1}$	$2.6X10^{1}$
Sm-153		9.0	$2.4X10^{2}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.6X10^{4}$	$4.4X10^{5}$

Table D-1. A_1 and A_2 Values for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Sn-113 (a)	Tin (50)	4.0	$1.1X10^{2}$	2.0	$5.4X10^{1}$	$3.7X10^{2}$	$1.0X10^{4}$
Sn-117m		7.0	$1.9X10^{2}$	$4.0X10^{-1}$	$1.1X10^{1}$	$3.0X10^{3}$	$8.2X10^{4}$
Sn-119m		$4.0X10^{1}$	$1.1X10^{3}$	$3.0X10^{1}$	$8.1X10^{2}$	$1.4X10^{2}$	$3.7X10^{3}$
Sn-121m (a)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-1}$	$2.4X10^{1}$	2.0	$5.4X10^{1}$
Sn-123		$8.0X10^{-1}$	$2.2X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$3.0X10^{2}$	$8.2X10^{3}$
Sn-125		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{3}$	$1.1X10^{5}$
Sn-126 (a)		$6.0X10^{-1}$	$1.6X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.0X10^{-3}$	$2.8X10^{-2}$
Sr-82 (a)	Strontium (38)	$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$2.3X10^{3}$	$6.2X10^{4}$
Sr-85		2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$8.8X10^{2}$	$2.4X10^{4}$
Sr-85m		5.0	$1.4X10^{2}$	5.0	$1.4X10^{2}$	$1.2X10^{6}$	$3.3X10^{7}$
Sr-87m		3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$4.8X10^{5}$	$1.3X10^{7}$
Sr-89		$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.1X10^{3}$	$2.9X10^{4}$
Sr-90 (a)		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	5.1	$1.4X10^{2}$
Sr-91 (a)		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.3X10^{5}$	$3.6X10^{6}$
Sr-92 (a)		1.0	$2.7X10^{1}$	$3.0X10^{-1}$	8.1	$4.7X10^{5}$	$1.3X10^{7}$
T(H-3)	Tritium (1)	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$3.6X10^{2}$	$9.7X10^{3}$
Ta-178 (long-lived)	Tantalum (73)	1.0	$2.7X10^{1}$	$8.0X10^{-1}$	$2.2X10^{1}$	$4.2X10^{6}$	$1.1X10^{8}$
Ta-179		$3.0X10^{1}$	$8.1X10^{2}$	$3.0X10^{1}$	$8.1X10^{2}$	$4.1X10^{1}$	$1.1X10^{3}$
Ta-182		$9.0X10^{-1}$	$2.4X10^{1}$	$5.0X10^{-1}$	$1.4X10^{1}$	$2.3X10^{2}$	$6.2X10^{3}$
Tb-157	Terbium (65)	$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$5.6X10^{-1}$	$1.5X10^{1}$
Tb-158		1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$5.6X10^{-1}$	$1.5X10^{1}$
Tb-160		1.0	$2.7X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$4.2X10^{2}$	$1.1X10^{4}$
Tc-95m (a)	Technetium (43)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$8.3X10^{2}$	$2.2X10^{4}$
Tc-96		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.2X10^{4}$	$3.2X10^{5}$
Tc-96m (a)		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$1.4X10^{6}$	$3.8X10^{7}$
Tc-97		Unlimited	Unlimited	Unlimited	Unlimited	$5.2X10^{-5}$	$1.4X10^{-3}$
Tc-97m		$4.0X10^{1}$	$1.1X10^{3}$	1.0	$2.7X10^{1}$	$5.6X10^{2}$	$1.5X10^{4}$
Tc-98		$8.0X10^{-1}$	$2.2X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$3.2X10^{-5}$	$8.7X10^{-4}$
Tc-99		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-1}$	$2.4X10^{1}$	$6.3X10^{-4}$	$1.7X10^{-2}$
Tc-99m		$1.0X10^{1}$	$2.7X10^{2}$	4.0	$1.1X10^{2}$	$1.9X10^{5}$	$5.3X10^{6}$

Table D-1. A_1 and A_2 Values for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Te-121	Tellurium (52)	2.0	$5.4\text{X}10^{1}$	2.0	$5.4\text{X}10^{1}$	$2.4\text{X}10^{3}$	$6.4\text{X}10^{4}$
Te-121m		5.0	$1.4\text{X}10^{2}$	3.0	$8.1\text{X}10^{1}$	$2.6\text{X}10^{2}$	$7.0\text{X}10^{3}$
Te-123m		8.0	$2.2\text{X}10^{2}$	1.0	$2.7\text{X}10^{1}$	$3.3\text{X}10^{2}$	$8.9\text{X}10^{3}$
Te-125m		$2.0\text{X}10^{1}$	$5.4\text{X}10^{2}$	$9.0\text{X}10^{-1}$	$2.4\text{X}10^{1}$	$6.7\text{X}10^{2}$	$1.8\text{X}10^{4}$
Te-127		$2.0\text{X}10^{1}$	$5.4\text{X}10^{2}$	$7.0\text{X}10^{-1}$	$1.9\text{X}10^{1}$	$9.8\text{X}10^{4}$	$2.6\text{X}10^{6}$
Te-127m (a)		$2.0\text{X}10^{1}$	$5.4\text{X}10^{2}$	$5.0\text{X}10^{-1}$	$1.4\text{X}10^{1}$	$3.5\text{X}10^{2}$	$9.4\text{X}10^{3}$
Te-129		$7.0\text{X}10^{-1}$	$1.9\text{X}10^{1}$	$6.0\text{X}10^{-1}$	$1.6\text{X}10^{1}$	$7.7\text{X}10^{5}$	$2.1\text{X}10^{7}$
Te-129m (a)		$8.0\text{X}10^{-1}$	$2.2\text{X}10^{1}$	$4.0\text{X}10^{-1}$	$1.1\text{X}10^{1}$	$1.1\text{X}10^{3}$	$3.0\text{X}10^{4}$
Te-131m (a)		$7.0\text{X}10^{-1}$	$1.9\text{X}10^{1}$	$5.0\text{X}10^{-1}$	$1.4\text{X}10^{1}$	$3.0\text{X}10^{4}$	$8.0\text{X}10^{5}$
Te-132 (a)		$5.0\text{X}10^{-1}$	$1.4\text{X}10^{1}$	$4.0\text{X}10^{-1}$	$1.1\text{X}10^{1}$	$1.1\text{X}10^{4}$	$8.0\text{X}10^{5}$
Th-227	Thorium (90)	$1.0\text{X}10^{1}$	$2.7\text{X}10^{2}$	$5.0\text{X}10^{-3}$	$1.4\text{X}10^{-1}$	$1.1\text{X}10^{3}$	$3.1\text{X}10^{4}$
Th-228 (a)		$5.0\text{X}10^{-1}$	$1.4\text{X}10^{1}$	$1.0\text{X}10^{-3}$	$2.7\text{X}10^{-2}$	$3.0\text{X}10^{1}$	$8.2\text{X}10^{2}$
Th-229		5.0	$1.4\text{X}10^{2}$	$5.0\text{X}10^{-4}$	$1.4\text{X}10^{-2}$	$7.9\text{X}10^{-3}$	$2.1\text{X}10^{-1}$
Th-230		$1.0\text{X}10^{1}$	$2.7\text{X}10^{2}$	$1.0\text{X}10^{-3}$	$2.7\text{X}10^{-2}$	$7.6\text{X}10^{-4}$	$2.1\text{X}10^{-2}$
Th-231	Thorium (90)	$4.0\text{X}10^{1}$	$1.1\text{X}10^{3}$	$2.0\text{X}10^{-2}$	$5.4\text{X}10^{-1}$	$2.0\text{X}10^{4}$	$5.3\text{X}10^{5}$
Th-232		Unlimited	Unlimited	Unlimited	Unlimited	$4.0\text{X}10^{-9}$	$1.1\text{X}10^{-7}$
Th-234 (a)		$3.0\text{X}10^{-1}$	8.1	$3.0\text{X}10^{-1}$	8.1	$8.6\text{X}10^{2}$	$2.3\text{X}10^{4}$
Th(nat)		Unlimited	Unlimited	Unlimited	Unlimited	$8.1\text{X}10^{-9}$	$2.2\text{X}10^{-7}$
Ti-44 (a)	Titanium (22)	$5.0\text{X}10^{-1}$	$1.4\text{X}10^{1}$	$4.0\text{X}10^{-1}$	$1.1\text{X}10^{1}$	6.4	$1.7\text{X}10^{2}$
Tl-200	Thallium (81)	$9.0\text{X}10^{-1}$	$2.4\text{X}10^{1}$	$9.0\text{X}10^{-1}$	$2.4\text{X}10^{1}$	$2.2\text{X}10^{4}$	$6.0\text{X}10^{5}$
Tl-201		$1.0\text{X}10^{1}$	$2.7\text{X}10^{2}$	4.0	$1.1\text{X}10^{2}$	$7.9\text{X}10^{3}$	$2.1\text{X}10^{5}$
Tl-202		2.0	$5.4\text{X}10^{1}$	2.0	$5.4\text{X}10^{1}$	$2.0\text{X}10^{3}$	$5.3\text{X}10^{4}$
Tl-204		$1.0\text{X}10^{1}$	$2.7\text{X}10^{2}$	$7.0\text{X}10^{-1}$	$1.9\text{X}10^{1}$	$1.7\text{X}10^{1}$	$4.6\text{X}10^{2}$
Tm-167	Thulium (69)	7.0	$1.9\text{X}10^{2}$	$8.0\text{X}10^{-1}$	$2.2\text{X}10^{1}$	$3.1\text{X}10^{3}$	$8.5\text{X}10^{4}$
Tm-170		3.0	$8.1\text{X}10^{1}$	$6.0\text{X}10^{-1}$	$1.6\text{X}10^{1}$	$2.2\text{X}10^{2}$	$6.0\text{X}10^{3}$
Tm-171		$4.0\text{X}10^{1}$	$1.1\text{X}10^{3}$	$4.0\text{X}10^{1}$	$1.1\text{X}10^{3}$	$4.0\text{X}10^{1}$	$1.1\text{X}10^{3}$

Table D-1. A_1 and A_2 Values for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
U-230 (fast lung absorption) (a)(d)	Uranium (92)	$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-1}$	2.7	$1.0X10^{3}$	$2.7X10^{4}$
U-230 (medium lung absorption) (a)(e)		$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-1}$	2.7	$1.0X10^{3}$	$2.7X10^{4}$
U-230 (slow lung absorption) (a)(f)		$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-1}$	2.7	$1.0X10^{3}$	$2.7X10^{4}$
U-232 (fast lung absorption) (d)		$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-2}$	$2.7X10^{-1}$	$8.3X10^{-1}$	$2.2X10^{1}$
U-232 (medium lung absorption) (e)		$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-2}$	$2.7X10^{-1}$	$8.3X10^{-1}$	$2.2X10^{1}$
U-232 (slow lung absorption) (f)		$4.0X10^{1}$	$1.1X10^{3}$	$1.0X10^{-2}$	$2.7X10^{-1}$	$8.3X10^{-1}$	$2.2X10^{1}$
U-233 (fast lung absorption) (d)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-2}$	2.4	$3.6X10^{-4}$	$9.7X10^{-3}$
U-233 (medium lung absorption) (e)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-2}$	2.4	$3.6X10^{-4}$	$9.7X10^{-3}$
U-233 (slow lung absorption) (f)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-2}$	2.4	$3.6X10^{-4}$	$9.7X10^{-3}$
U-234 (fast lung absorption) (d)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-2}$	2.4	$2.3X10^{-4}$	$6.2X10^{-3}$
U-234 (medium lung absorption) (e)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-2}$	2.4	$2.3X10^{-4}$	$6.2X10^{-3}$
U-234 (slow lung absorption) (f)		$4.0X10^{1}$	$1.1X10^{3}$	$9.0X10^{-2}$	2.4	$2.3X10^{-4}$	$6.2X10^{-3}$
U-235 (all lung absorption types) (a),(d),(e),(f)		Unlimited	Unlimited	Unlimited	Unlimited	$8.0X10^{-8}$	$2.2X10^{-6}$
U-236 (fast lung absorption) (d)		Unlimited	Unlimited	Unlimited	Unlimited	$2.4X10^{-6}$	$6.5X10^{-5}$
U-236 (medium lung absorption) (e)		Unlimited	Unlimited	Unlimited	Unlimited	$2.4X10^{-6}$	$6.5X10^{-5}$

Table D-1. A_1 and A_2 Values for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
U-236 (slow lung absorption) (f)		Unlimited	Unlimited	Unlimited	Unlimited	$2.4X10^{-6}$	$6.5X10^{-5}$
U-238 (all lung absorption types) (d),(e),(f)		Unlimited	Unlimited	Unlimited	Unlimited	$1.2X10^{-8}$	$3.4X10^{-7}$
U (nat)		Unlimited	Unlimited	Unlimited	Unlimited	$2.6X10^{-8}$	$7.1X10^{-7}$
U (enriched to 20% or less)(g)		Unlimited	Unlimited	Unlimited	Unlimited	N/A	N/A
U (dep)		Unlimited	Unlimited	Unlimited	Unlimited	0.0	(See Table A-3)
V-48	Vanadium (23)	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$6.3X10^{3}$	$1.7X10^{5}$
V-49		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$3.0X10^{2}$	$8.1X10^{3}$
W-178 (a)	Tungsten (74)	9.0	$2.4X10^{2}$	5.0	$1.4X10^{2}$	$1.3X10^{3}$	$3.4X10^{4}$
W-181		$3.0X10^{1}$	$8.1X10^{2}$	$3.0X10^{1}$	$8.1X10^{2}$	$2.2X10^{2}$	$6.0X10^{3}$
W-185		$4.0X10^{1}$	$1.1X10^{3}$	$8.0X10^{-1}$	$2.2X10^{1}$	$3.5X10^{2}$	$9.4X10^{3}$
W-187		2.0	$5.4X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$2.6X10^{4}$	$7.0X10^{5}$
W-188 (a)		$4.0X10^{-1}$	$1.1X10^{1}$	$3.0X10^{-1}$	8.1	$3.7X10^{2}$	$1.0X10^{4}$
Xe-122 (a)	Xenon (54)	$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$4.8X10^{4}$	$1.3X10^{6}$
Xe-123		2.0	$5.4X10^{1}$	$7.0X10^{-1}$	$1.9X10^{1}$	$4.4X10^{5}$	$1.2X10^{7}$
Xe-127		4.0	$1.1X10^{2}$	2.0	$5.4X10^{1}$	$1.0X10^{3}$	$2.8X10^{4}$
Xe-131m		$4.0X10^{1}$	$1.1X10^{3}$	$4.0X10^{1}$	$1.1X10^{3}$	$3.1X10^{3}$	$8.4X10^{4}$
Xe-133		$2.0X10^{1}$	$5.4X10^{2}$	$1.0X10^{1}$	$2.7X10^{2}$	$6.9X10^{3}$	$1.9X10^{5}$
Xe-135		3.0	$8.1X10^{1}$	2.0	$5.4X10^{1}$	$9.5X10^{4}$	$2.6X10^{6}$
Y-87 (a)	Yttrium (39)	1.0	$2.7X10^{1}$	1.0	$2.7X10^{1}$	$1.7X10^{4}$	$4.5X10^{5}$
Y-88		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$5.2X10^{2}$	$1.4X10^{4}$
Y-90		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$2.0X10^{4}$	$5.4X10^{5}$
Y-91		$6.0X10^{-1}$	$1.6X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$9.1X10^{2}$	$2.5X10^{4}$
Y-91m		2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$1.5X10^{6}$	$4.2X10^{7}$
Y-92		$2.0X10^{-1}$	5.4	$2.0X10^{-1}$	5.4	$3.6X10^{5}$	$9.6X10^{6}$
Y-93		$3.0X10^{-1}$	8.1	$3.0X10^{-1}$	8.1	$1.2X10^{5}$	$3.3X10^{6}$
Yb-169	Ytterbium (79)	4.0	$1.1X10^{2}$	1.0	$2.7X10^{1}$	$8.9X10^{2}$	$2.4X10^{4}$
Yb-175		$3.0X10^{1}$	$8.1X10^{2}$	$9.0X10^{-1}$	$2.4X10^{1}$	$6.6X10^{3}$	$1.8X10^{5}$
Zn-65	Zinc (30)	2.0	$5.4X10^{1}$	2.0	$5.4X10^{1}$	$3.0X10^{2}$	$8.2X10^{3}$
Zn-69		3.0	$8.1X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.8X10^{6}$	$4.9X10^{7}$
Zn-69m (a)		3.0	$8.1X10^{1}$	$6.0X10^{-1}$	$1.6X10^{1}$	$1.2X10^{5}$	$3.3X10^{6}$
Zr-88	Zirconium (40)	3.0	$8.1X10^{1}$	3.0	$8.1X10^{1}$	$6.6X10^{2}$	$1.8X10^{4}$

Table D-1. A_1 and A_2 Values for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	A_1 (TBq)	A_1 (Ci)	A_2 (TBq)	A_2 (Ci)	Specific activity (TBq/g)	Specific activity (Ci/g)
Zr-93		Unlimited	Unlimited	Unlimited	Unlimited	$9.3X10^{-5}$	$2.5X10^{-3}$
Zr-95 (a)		2.0	$5.4X10^{1}$	$8.0X10^{-1}$	$2.2X10^{1}$	$7.9X10^{2}$	$2.1X10^{4}$
Zr-97 (a)		$4.0X10^{-1}$	$1.1X10^{1}$	$4.0X10^{-1}$	$1.1X10^{1}$	$7.1X10^{4}$	$1.9X10^{6}$

Table D-2. Exempt Material Activity Concentrations and Exempt Consignment Activity Limits for Radionuclides

Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Ac-225 (a)	Actinium (89)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Ac-227 (a)		$1.0X10^{-1}$	$2.7X10^{-12}$	$1.0X10^{3}$	$2.7X10^{-8}$
Ac-228		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ag-105	Silver (47)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ag-108m (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ag-110m (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ag-111		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Al-26	Aluminum (13)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Am-241	Americium (95)	1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Am-242m (a)		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Am-243 (a)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Ar-37	Argon (18)	$1.0X10^{6}$	$2.7X10^{-5}$	$1.0X10^{8}$	$2.7X10^{-3}$
Ar-39		$1.0X10^{7}$	$2.7X10^{-4}$	$1.0X10^{4}$	$2.7X10^{-7}$
Ar-41		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{9}$	$2.7X10^{-2}$
As-72	Arsenic (33)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
As-73		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
As-74		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
As-76		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
As-77		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
At-211 (a)	Astatine (85)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Au-193	Gold (79)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Au-194		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Au-195		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Au-198		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Au-199		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ba-131 (a)	Barium (56)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ba-133		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ba-133m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ba-140 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Be-7	Beryllium (4)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Be-10		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{6}$	$2.7X10^{-5}$
Bi-205	Bismuth (83)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Bi-206		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Bi-207		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Bi-210		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Bi-210m (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Bi-212 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$

Table D-2. Exempt Material Activity Concentrations and Exempt Consignment Activity Limits for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Bk-247	Berkelium (97)	1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Bk-249 (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Br-76	Bromine (35)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Br-77		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Br-82		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
C-11	Carbon (6)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
C-14		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ca-41	Calcium (20)	$1.0X10^{5}$	$2.7X10^{-6}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ca-45		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ca-47 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cd-109	Cadmium (48)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cd-113m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cd-115 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cd-115m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ce-139	Cerium (58)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ce-141		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ce-143		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ce-144 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cf-248	Californium (98)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cf-249		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Cf-250		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cf-251		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Cf-252		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cf-253 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cf-254		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Cl-36	Chlorine (17)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cl-38		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cm-240	Curium (96)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cm-241		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cm-242		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cm-243		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cm-244		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cm-245		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Cm-246		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Cm-247 (a)		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cm-248		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Co-55	Cobalt (27)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$

Table D-2. Exempt Material Activity Concentrations and Exempt Consignment Activity Limits for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Co-56		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Co-57		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Co-58		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Co-58m		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Co-60		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cr-51	Chromium (24)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Cs-129	Caesium (55)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cs-131		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cs-132		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cs-134		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cs-134m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cs-135		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Cs-136		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Cs-137 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Cu-64	Copper (29)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Cu-67		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Dy-159	Dysprosium (66)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Dy-165		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Dy-166 (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Er-169	Erbium (68)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Er-171		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-147	Europium (63)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-148		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-149		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Eu-150 (short lived)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-150 (long lived)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-152		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-152 m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-154		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Eu-155		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Eu-156		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
F-18	Fluorine (9)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Fe-52 (a)	Iron (26)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Fe-55		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{6}$	$2.7X10^{-5}$
Fe-59		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Fe-60 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$

Table D-2. Exempt Material Activity Concentrations and Exempt Consignment Activity Limits for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Ga-67	Gallium (31)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ga-68		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ga-72		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Gd-146 (a)	Gadolinium (64)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Gd-148		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Gd-153		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Gd-159		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ge-68 (a)	Germanium (32)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ge-71		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{8}$	$2.7X10^{-3}$
Ge-77		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Hf-172 (a)	Hafnium (72)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hf-175		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hf-181		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hf-182		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hg-194 (a)	Mercury (80)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hg-195m (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hg-197		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Hg-197m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Hg-203		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ho-166	Holmium (67)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ho-166m		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
I-123	Iodine (53)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
I-124		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
I-125		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
I-126		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
I-129		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
I-131		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
I-132		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
I-133		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
I-134		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
I-135 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
In-111	Indium (49)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
In-113m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
In-114m (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
In-115m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ir-189 (a)	Iridium (77)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ir-190		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$

Table D-2. Exempt Material Activity Concentrations and Exempt Consignment Activity Limits for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Ir-192		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Ir-194		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
K-40	Potassium (19)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
K-42		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
K-43		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Kr-81	Krypton (36)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Kr-85		$1.0X10^{5}$	$2.7X10^{-6}$	$1.0X10^{4}$	$2.7X10^{-7}$
Kr-85m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{10}$	$2.7X10^{-1}$
Kr-87		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{9}$	$2.7X10^{-2}$
La-137	Lanthanum (57)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
La-140		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Lu-172	Lutetium (71)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Lu-173		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Lu-174		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Lu-174m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Lu-177		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Mg-28 (a)	Magnesium (12)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Mn-52	Manganese (25)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Mn-53		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{9}$	$2.7X10^{-2}$
Mn-54		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Mn-56		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Mo-93	Molybdenum (42)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{8}$	$2.7X10^{-3}$
Mo-99 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
N-13	Nitrogen (7)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{9}$	$2.7X10^{-2}$
Na-22	Sodium (11)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Na-24		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$

Table D-2. Exempt Material Activity Concentrations and Exempt Consignment Activity Limits for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Nb-93m	Niobium (41)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Nb-94		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Nb-95		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Nb-97		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Nd-147	Neodymium (60)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Nd-149		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ni-59	Nickel (28)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{8}$	$2.7X10^{-3}$
Ni-63		$1.0X10^{5}$	$2.7X10^{-6}$	$1.0X10^{8}$	$2.7X10^{-3}$
Ni-65		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Np-235	Neptunium (93)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Np-236 (short-lived)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Np-236 (long-lived)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Np-237		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Np-239		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Os-185	Osmium (76)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Os-191		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Os-191m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Os-193		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Os-194 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
P-32	Phosphorus (15)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{5}$	$2.7X10^{-6}$
P-33		$1.0X10^{5}$	$2.7X10^{-6}$	$1.0X10^{8}$	$2.7X10^{-3}$
Pa-230 (a)	Protactinium (91)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pa-231		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Pa-233		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pb-201	Lead (82)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pb-202		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pb-203		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pb-205		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pb-210 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Pb-212 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Pd-103 (a)	Palladium (46)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{8}$	$2.7X10^{-3}$
Pd-107		$1.0X10^{5}$	$2.7X10^{-6}$	$1.0X10^{8}$	$2.7X10^{-3}$
Pd-109		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$

Table D-2. Exempt Material Activity Concentrations and Exempt Consignment Activity Limits for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Pm-143	Promethium (61)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pm-144		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pm-145		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pm-147		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pm-148m (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pm-149		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pm-151		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Po-210	Polonium (84)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Pr-142	Praseodymium (59)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Pr-143		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pt-188 (a)	Platinum (78)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pt-191		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pt-193		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pt-193m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pt-195m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pt-197		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pt-197m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Pu-236	Plutonium (94)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Pu-237		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Pu-238		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Pu-239		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Pu-240		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Pu-241 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Pu-242		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Pu-244 (a)		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Ra-223 (a)	Radium (88)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ra-224 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ra-225 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Ra-226 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Ra-228 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Rb-81	Rubidium (37)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Rb-83 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Rb-84		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Rb-86		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Rb-87		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Rb(nat)		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Re-184	Rhenium (75)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$

Table D-2. Exempt Material Activity Concentrations and Exempt Consignment Activity Limits for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Re-184m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Re-186		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Re-187		$1.0X10^{6}$	$2.7X10^{-5}$	$1.0X10^{9}$	$2.7X10^{-2}$
Re-188		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Re-189 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Re(nat)		$1.0X10^{6}$	$2.7X10^{-5}$	$1.0X10^{9}$	$2.7X10^{-2}$
Rh-99	Rhodium (45)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Rh-101		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Rh-102		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Rh-102m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Rh-103m		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{8}$	$2.7X10^{-3}$
Rh-105		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Rn-222 (a)	Radon (86)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{8}$	$2.7X10^{-3}$
Ru-97	Ruthenium (44)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ru-103 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ru-105		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ru-106 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
S-35	Sulphur (16)	$1.0X10^{5}$	$2.7X10^{-6}$	$1.0X10^{8}$	$2.7X10^{-3}$
Sb-122	Antimony (51)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{4}$	$2.7X10^{-7}$
Sb-124		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sb-125		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sb-126		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Sc-44	Scandium (21)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Sc-46		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sc-47		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sc-48		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Se-75	Selenium (34)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Se-79		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Si-31	Silicon (14)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Si-32		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sm-145	Samarium (62)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Sm-147		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Sm-151		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{8}$	$2.7X10^{-3}$
Sm-153		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sn-113 (a)	Tin (50)	$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Sn-117m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sn-119m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$

Table D-2. Exempt Material Activity Concentrations and Exempt Consignment Activity Limits for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Sn-121m (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Sn-123		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sn-125		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Sn-126 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Sr-82 (a)	Strontium (38)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Sr-85		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sr-85m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Sr-87m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sr-89		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Sr-90 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{4}$	$2.7X10^{-7}$
Sr-91 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Sr-92 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
T(H-3)	Tritium (1)	$1.0X10^{6}$	$2.7X10^{-5}$	$1.0X10^{9}$	$2.7X10^{-2}$
Ta-178 (long-lived)	Tantalum (73)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Ta-179		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Ta-182		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Tb-157	Terbium (65)	$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Tb-158		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tb-160		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tc-95m (a)	Technetium (43)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tc-96		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tc-96m (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Tc-97		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{8}$	$2.7X10^{-3}$
Tc-97m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Tc-98		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tc-99		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
Tc-99m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Te-121	Tellurium (52)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Te-121m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Te-123m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Te-125m		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Te-127		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Te-127m (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Te-129		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Te-129m (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Te-131m (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Te-132 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$

Table D-2. Exempt Material Activity Concentrations and Exempt Consignment Activity Limits for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
Th-227	Thorium (90)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Th-228 (a)		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Th-229		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Th-230		1.0	$2.7X10^{-11}$	$1.0X10^{4}$	$2.7X10^{-7}$
Th-231		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Th-232		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
Th-234 (a)		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{5}$	$2.7X10^{-6}$
Th (nat)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
Ti-44 (a)	Titanium (22)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
Tl-200	Thallium (81)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tl-201		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tl-202		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tl-204		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{4}$	$2.7X10^{-7}$
Tm-167	Thulium (69)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tm-170		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Tm-171		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{8}$	$2.7X10^{-3}$
U-230 (fast lung absorption) (a)(d)	Uranium (92)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
U-230 (medium lung absorption) (a)(e)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
U-230 (slow lung absorption) (a)(f)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
U-232 (fast lung absorption) (d)	Uranium (92)	1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
U-232 (medium lung absorption) (e)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
U-232 (slow lung absorption) (f)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
U-233 (fast lung absorption) (d)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-233 (medium lung absorption) (e)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-233 (slow lung absorption) (f)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-234 (fast lung absorption) (d)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$

Table D-2. Exempt Material Activity Concentrations and Exempt Consignment Activity Limits for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
U-234 (medium lung absorption) (e)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-234 (slow lung absorption) (f)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-235 (all lung absorption types) (a),(d),(e),(f)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-236 (fast lung absorption) (d)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-236 (medium lung absorption) (e)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-236 (slow lung absorption) (f)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U-238 (all lung absorption types) (d),(e),(f)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{4}$	$2.7X10^{-7}$
U (nat)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
U (enriched to 20% or less)(g)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
U (dep)		1.0	$2.7X10^{-11}$	$1.0X10^{3}$	$2.7X10^{-8}$
V-48	Vanadium (23)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$
V-49		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$

Table D-2. Exempt Material Activity Concentrations and Exempt Consignment Activity Limits for Radionuclides (Continued)

Symbol of radionuclide	Element and atomic number	Activity concentration for exempt material (Bq/g)	Activity concentration for exempt material (Ci/g)	Activity limit for exempt consignment (Bq)	Activity limit for exempt consignment (Ci)
W-178 (a)	Tungsten (74)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
W-181		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
W-185		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{7}$	$2.7X10^{-4}$
W-187		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
W-188 (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Xe-122 (a)	Xenon (54)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{9}$	$2.7X10^{-2}$
Xe-123		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{9}$	$2.7X10^{-2}$
Xe-127		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{5}$	$2.7X10^{-6}$
Xe-131m		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{4}$	$2.7X10^{-7}$
Xe-133		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{4}$	$2.7X10^{-7}$
Xe-135		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{10}$	$2.7X10^{-1}$
Y-87 (a)	Yttrium (39)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Y-88		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Y-90		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{5}$	$2.7X10^{-6}$
Y-91		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{6}$	$2.7X10^{-5}$
Y-91m		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Y-92		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Y-93		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{5}$	$2.7X10^{-6}$
Yb-169	Ytterbium (79)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{7}$	$2.7X10^{-4}$
Yb-175		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Zn-65	Zinc (30)	$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Zn-69		$1.0X10^{4}$	$2.7X10^{-7}$	$1.0X10^{6}$	$2.7X10^{-5}$
Zn-69m (a)		$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Zr-88	Zirconium (40)	$1.0X10^{2}$	$2.7X10^{-9}$	$1.0X10^{6}$	$2.7X10^{-5}$
Zr-93		$1.0X10^{3}$	$2.7X10^{-8}$	$1.0X10^{7}$	$2.7X10^{-4}$
Zr-95 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{6}$	$2.7X10^{-5}$
Zr-97 (a)		$1.0X10^{1}$	$2.7X10^{-10}$	$1.0X10^{5}$	$2.7X10^{-6}$

NUREG/CR-6711

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 3.7×10^{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

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[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.

Harmonization of NRC's Transportation Regulations With Other Regulatory Agencies

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.

NRC-Initiated Changes

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.

[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

[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.

Option 1: No-Action Alternative

Under the No-Action alternative (Option 1), NRC would not modify Part 71 to incorporate the TS-R-1 UF_6 requirements.

Option 2: Proposed Action

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.

Option 1: No-Action Alternative

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.

Option 2: Proposed Action

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.

Option 1: No-Action Alternative

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.

Option 2: Proposed Action

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 and Type B tests with respect to release, shielding, and maintaining subcriticality (single package and 5xN array/normal and 2xN array/accident), and

  (B) comply with 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.

Option 1: No-Action Alternative

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.

Option 2: Proposed Action

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.

Option 1: No-Action Alternative

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.

Option 2: Proposed Action

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.

<u>Option 1: No-Action Alternative</u>

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.

<u>Option 2: Proposed Action</u>

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.

Option 1: No-Action Alternative

[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

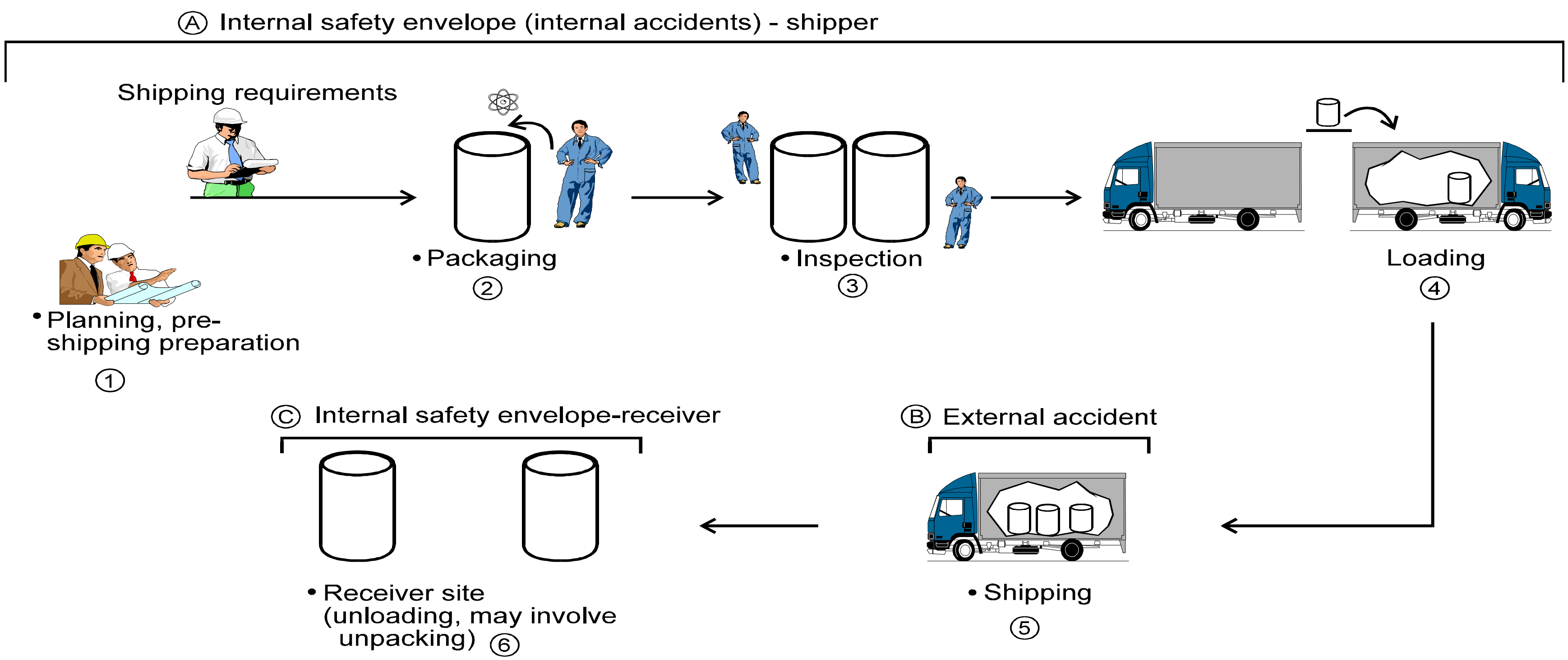
2. Packaging
    a. Changes in the number of loads
    b. Changes in length of time for packing a load
    c. Changes in worker exposure for packing a load

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

Impacts of the Proposed Action

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.

Impacts of the No-Action Alternative

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

Impacts of Proposed Action

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.

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[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 $k_{eff} \leq 0.95$	Critical $k_{eff} = 1.0$	Subcritical $k_{eff} \leq 0.95$	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

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[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%