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Criteria for Onsite Transfers of Radioactive Material

by

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DISCLAIMER

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LIST OF TABLES

1.0 INTRODUCTION

1.1 Scope

This document contains the following 11 chapters:

1. Introduction
2. Requirements for Onsite Transfers
3. Package Design Requirements
4. Structural Criteria
5. Thermal Criteria
6. Containment Criteria
7. Shielding Criteria
8. Criticality Safety Criteria
9. Operating Procedure Criteria
10. Acceptance Testing and Maintenance Program Criteria
11. Package Quality Assurance (QA) Criteria
12. Radiation Protection Criteria
13. Hazards Information Criteria

Background information is provided in Chapter 1, which also includes a description of the scope of application of these criteria and a discussion of the general approach.

A general description of the requirements for making onsite transfers of radioactive material is provided in Chapter 2. Also identified in this chapter is the required sequence of activities for determining whether the preferred method for the transfer is one of the following:

- use of authorized offsite packaging
- use of alternate onsite packaging
- unpackaged
- use of a Radiation Work Permit.

The various criteria for package use are identified in Chapters 3 through 13. These criteria, together with associated controls, provide protection against undue radiation exposure. Package shielding, containment, and surface contamination requirements are established. Criteria for providing criticality safety are enumerated in Chapter 6. Criteria for providing hazards information are established in Chapter 13.

Terms used in this document are defined in a glossary, which follows Chapter 13.

The requirements and format for the safety assessment document, which addresses alternate onsite packages, are in WSRC-TR-92-580, *Safety Assessment Requirements for Onsite Transfers of Radioactive Material*.

1.2 Applicability

These criteria are applicable to all onsite transfers of radioactive material at SRS. Whenever the requirements of the Environmental Protection Agency (EPA) for radioactive waste packages (i.e., package marking and labeling) conflict with these criteria, EPA requirements have precedence.

1.3 Approach

The basic approach for onsite transfers is to provide protection that is the same, or equivalent to, that provided during offsite shipments. This requires providing control of radiation exposure, criticality safety, and hazards information.

The approach to controlling radiation exposure is to ensure that the exposure is as low as reasonably achievable (ALARA) and below established limits. The established limits are no higher than those established for offsite shipments or the guidelines and limits in the Westinghouse Savannah River Company (WSRC) radiation protection program.

The approach to criticality safety is to comply with those U.S. Department of Energy (DOE) requirements for offsite shipments that are applicable to criticality safety or to comply with the DOE requirements for nuclear facilities applicable to criticality safety.

The approach to providing hazards information is to comply with the DOE requirements for offsite shipments that are applicable to placards, package marking, package labeling, shipping papers, and emergency response information or to provide the equivalent information by using locally approved written procedures.

2.0 REQUIREMENTS FOR ONSITE TRANSFERS

2.1 General Protection Requirements

The general requirement for making an onsite transfer of radioactive material is to provide the same or equivalent protection as is provided during offsite shipments of radioactive material. This involves providing control of radiation exposure, criticality safety, and hazards information.

The material to be transferred shall be adequately described. This description includes the following:

- the thermal, chemical, physical and nuclear descriptions
- usage of the package
- annual quantities transferred
- quantities per package
- transport origins and destinations.

An onsite package is approved for a specific authorized set of contents and specific usage.

2.2 Packaging Selection Requirements

The basic tenet of packaging selection is that the radiation exposure should be ALARA. This tenet is the primary consideration in determining whether the radioactive material is to be transferred packaged or unpackaged. If the material is to be transferred in a package, ALARA considerations primarily determine whether an authorized offsite packaging or an alternate packaging is preferred.

The procedure for making these choices shall follow the sequence specified in the following sections.

2.2.1 Preliminary Exposure Assessment

Unless an ALARA study shows that the unpackaged mode is preferable, radioactive material shall be packaged for onsite transfers. An ALARA study for the selection process is not required unless the circumstances suggest that using the unpackaged mode of transport for the onsite transfer will provide the lowest collective and lowest maximum individual radiation exposures.

If, due to the unusual size, shape, or characteristic of radioactive material, unpackaged transfer of the radioactive material is preferable, that unpackaged transfer is permitted provided the following requirements are met:

- a documented ALARA study that compares expected radiation exposure from the unpackaged transfer with the packaged transfer clearly shows reduced collective and reduced maximum individual radiation exposure when the unpackaged mode is used
- the cognizant WSRC division managers' approvals are obtained.

The ALARA study for the selection process shall estimate the expected collective and maximum individual radiation exposure from the transfer-related operational activities and the transfer activities for the packaged mode. These exposures shall be compared with estimated exposures for the comparable set of activities that are applicable to the unpackaged mode.

The cognizant WSRC division managers shall include division managers who:

- originally have custody of the radioactive material
- receive custody of the radioactive material at the conclusion of the transfer
- are responsible for site-wide radiation protection
- are responsible for emergency response during the transfer
- have a potential responsibility for some phase of the transfer.

2.2.2 Authorized Offsite Package Requirements

Authorized offsite packaging shall be the first packaging considered for onsite transfers of radioactive material. Authorized offsite packaging shall be used if it meets the following conditions:

- is available
- provides radiation exposure that is ALARA
- is authorized for all subsidiary hazards
- provides collective and maximum individual radiation exposure that is no higher than that from using alternate packaging.

When these criteria are met, authorized offsite packaging is the first choice for onsite transfers of radioactive material because of its large base of operating experience.

Authorized offsite packaging is approved for onsite transfers of radioactive material when the packaging, including its contents and use, complies with the following sections of the *Code of Federal Regulations* (CFR):

- 49 CFR 172.101, "Hazards Material Table"
- 49 CFR 173.24, "General Requirements for Packagings"
- 49 CFR 173.415, "Authorized Type A Packages"
- 49 CFR 173.416, "Authorized Type B Packages"
- 49 CFR 173.417, "Authorized Packagings—Fissile Material"
- 49 CFR 173.418, "Authorized Packagings—Pyrophoric Radioactive Materials"
- 49 CFR 173.419, "Authorized Packagings—Oxidizing Radioactive Material"
- 49 CFR 173.420, "Uranium Hexafluoride (fissile and low specific activity)"
- 49 CFR 173.421, "Limited Quantity of Radioactive Material"
- 49 CFR 173.441, "Radiation Limits"
- 49 CFR 173.442, "Thermal Limits"
- 49 CFR 173.443, "Contamination Limits".

2.2.3 Alternate Package Requirements

Alternate packaging may be used if authorized offsite packaging is not readily available.

Alternate packaging shall be used if authorized offsite packaging results in the following:

- requires any additional material preparation that causes more individual or collective personnel radiation exposure than using alternate packaging
- for any reason causes radiation exposure that is not ALARA.

Alternate packaging shall be used under the following circumstances:

- only for contents authorized in the safety assessment document
- in compliance with any limits and requirements specified in the safety assessment document
- in compliance with any specified operating procedures and acceptance and maintenance procedures in the safety assessment document.

The safety assessment for alternate packages shall show that the packaging and its contents, with associated controls and limits will provide collective and individual radiation exposure that is below established limits.

Separate detailed ALARA studies may be performed during the design process to verify that design features of the package minimize radiation exposure to workers handling the package. The details and requirements of these studies are outside the scope of the safety assessment.

2.3 Authorization of Alternate Package Use

Alternate onsite packagings shall be used only for the following:

- transportation to and from the authorized destinations
- authorized contents
- compliance with the limits and requirements established on the package authorization form.

The package authorization form is completed as part of the review and approval of the package safety assessment. The standards for these reviews are graded with the most stringent standards used for the most hazardous package contents. The approval process shall result in a definition of the following:

- authorized package contents
- authorized usage conditions
- any limits, restrictions, or requirements for the packaging, contents, or transfer related activities.

2.3.1 Alternate Type B Packages

Each safety assessment for an alternate onsite package for greater than A_2 quantity of radioactive material shall be reviewed by one or more technical specialists who are organizationally independent from the author of the safety assessment. To qualify as a technical specialist, the reviewer shall have competence equivalent to that of a licensed professional engineer in the packaging and transport of radioactive material.

In addition, the manager or designees of any affected WSRC departments shall review the following chapters of each safety assessment for an alternate Type B onsite package:

- Operating Procedures
- Acceptance Testing and Maintenance Program
- Hazards Information Assessment
- Assessment Results and Requirements.

The affected departments are those responsible for the following activities:

- originating the onsite transfer
- receiving the transferred radioactive material
- providing emergency response, if needed, during the transfer
- providing radiation exposure protection during the transfer
- operating the transport vehicle
- any other activity involved in the onsite transfer.

All chapters of the safety assessment documents for alternate onsite packages containing greater than A_2 quantities of liquid or gaseous radioactive material shall also be reviewed by the managers of affected WSRC Departments or their designees.

The review process shall be documented in a safety evaluation report, which describes the degree to which the requirements in the criteria document were met. The safety evaluation report shall include a draft copy of the onsite package authorization form. The onsite package authorization form shall identify the following:

- authorized contents for the package
- authorized usage for the contents

- all limits, restrictions, or requirements for the package or transfer activities.

Approval of the safety assessment document for Alternate Type B onsite packages shall be by WSRC senior management. Senior management approval shall be by those WSRC Level 2 managers responsible for the following:

- originating the onsite transfer
- receiving the transferred radioactive material
- providing emergency response, if needed, during the transfer
- providing radiation exposure protection during the transfer
- operating the transport vehicle
- any other activity involved in the onsite transfer.

Approval of the safety assessment shall be indicated by the signatures of WSRC senior management upon the onsite package authorization form. A signed authorization form provides authorization for package use in accordance with the terms inscribed on the authorization form.

2.3.2 Other Alternate Onsite Packages

Each safety assessment for alternate onsite packages for radioactive material that does not exceed an A_2 quantity per package shall be reviewed by one or more technical specialists who are organizationally independent from the author of the safety assessment. To qualify as a technical specialist, the reviewer shall have competence equivalent to that of a licensed professional engineer in the packaging and transport of radioactive material.

The review process shall be documented in a safety evaluation report, which describes how the requirements in the criteria document were met.

Approval of the safety assessment for an alternate onsite package that does not exceed an A_2 quantity per package shall be by the cognizant technical specialist. The signature on the completed onsite package authorization form shall indicate authorization for use of the package in accordance with the terms inscribed on the authorization form.

2.4 Criticality Control Requirements

Criticality safety for onsite packages during loading of the transfer vehicle, vehicle operation, and vehicle unloading shall be provided by compliance with the one or more of the following:

- applicable DOT regulations
- applicable NRC regulations
- applicable DOE directives.

Criticality safety during transport-related operational activities shall be ensured by compliance with the requirements in the applicable facility criticality safety requirements. Specific criteria for criticality safety for onsite transfers are provided in Chapter 8.

2.5 Hazard Information Requirements

The basic requirement is to provide information in compliance with the regulations for offsite shipments regarding vehicle placarding, package marking and labeling, shipping papers, and the emergency response information or to provide equivalent information by using locally approved procedures. Any locally approved procedures shall meet the following:

- operational requirements of the users
- radiation protection requirements
- needs of emergency responders.

Thus, locally approved procedures for providing hazards information must be approved by organizations representing the following:

- shipper
- carrier
- receiver
- radiation protection
- emergency response for the site or facility involved.

Specific hazard information requirements are provided in Chapter 13.

2.6 Other Requirements

The safety assessment document shall establish requirements for procedures or limits to manage the following:

- decay heat
- chemical compatibility of packaging and contents
- packaging tie downs
- packaging lifting and hoisting
- parameters associated with the transfer such as route, speed, access control, and number of transfers.

3.0 PACKAGE DESIGN REQUIREMENTS

3.1 Operational Requirements

Requirements pertaining to the package contents and package usage are necessary for the safety assessment. These requirements will affect the design of new packagings and evaluations of existing packagings.

3.1.1 Contents Characterization

The material to be packaged and transferred shall be adequately described to properly identify the shielding, containment, criticality, thermal and other packaging requirements in the safety assessment. Any characteristics, such as corrosivity, flammability, or toxicity shall be identified to assure these subsidiary hazards are also protected against during onsite transfers.

3.1.2 Package Usage

To perform a safety assessment, the intended package usage must be identified. The following information is required:

- routes
- transfer modes and frequency
- package size
- weight limits
- other facility interface requirements.

3.1.3 Data Sheets

The contents characterization and package usage data sheets must be completed by each organization responsible for filling an onsite packaging. The data sheets to be used are provided in Appendix A.

3.2 Design Conditions

The design conditions of interest are the conditions that a package may be subjected to during onsite transfer activities. These design conditions shall be established for each onsite package and shall be identified in the safety assessment. Design conditions are the ambient and maximum pressures, temperatures, and external loads on the package experienced during the transfer activities. The design conditions are also developed from the design basis events selected for onsite transfer (Section 3.2.2).

Note: Transfer-related operational activities and storage are outside the scope of the safety assessment. The design conditions unique to these activities and the credible accident events associated with these activities need not be considered in the safety assessment. However, to identify design conditions the package is required to withstand during storage or transfer-related operational activities, the preparers of the safety assessment may find it useful to review the functional design requirements for the package. These design conditions may be more severe than those encountered from the transfer activities, and credit could be taken for this in the safety assessment if a previous evaluation (i.e., facility safety assessment) has adequately demonstrated satisfactory performance of the package.

3.2.1 Credible Events

All credible events that might occur during transfer activities shall be identified and considered in the onsite package safety assessment. Table 3.1 lists examples of credible events that could occur in each event category.

Table 3.1. Possible Credible Events during Transfer Activities

Event Category	Examples of Credible Events
Normal	• Loading the package onto the transfer vehicle • Blocking, bracing, and tying down the package • Operating the transfer vehicle • Unloading the transfer vehicle
Anticipated	• Package falling from typical transport vehicle • Package hit and dented by fork truck tine • Package stacked 5 tiers high • Package subjected to rain storm
Off-Normal	• Package dropped from maximum height lifted during vehicle loading/unloading • Fire occurs from forklift or vehicle fuel leak, from combustible material around loading/unloading area, or from other flammable cargo • Package is restrained and hit by tine of forklift travelling at the physical or administrative speed limit • Truck accident occurs at the fender bender level
Emergency	• Transport vehicle collision with a fixed object near the road • Transport vehicle fuel spill and ignition due to an accident • Collision with typical vehicle sharing the roadway with the transport vehicle

Only normal and anticipated credible events are considered for alternate packages containing no more than an A_1 or A_2 quantity. The list of normal and anticipated credible events may be developed based upon the package usage information provided in the contents characterization and package usage data sheets. Alternatively, the performance tests listed in 49 CFR 173.465 and 49 CFR 173.466 may be used to represent the credible events for the normal and anticipated categories.

For packages containing more than an A_1 or A_2 quantity of material, credible events in the off-normal and emergency categories are also considered. The list of these credible events may be developed based upon the package usage information provided in the contests characterization and package usage data sheets.

Alternatively, the performance tests listed on 10 CFR 71.73 for hypothetical accidents may be used to represent the credible off-normal and emergency events associated with the transport phase of the onsite transfer. If the package is to be transferred by truck, weighs at least 3,000 pounds, and will experience no more than 1,000 loaded truck miles per year, the generic performance tests listed in Tables 3.2 and 3.3 may be used to represent the off-normal and emergency credible events associated with the transport phase of the transfer.

These generic performance tests were developed from the application of probabilistic models of package performance under transportation accident conditions.

The study that developed these generic accident conditions used information for truck accidents involving large packages.[1] Therefore, the use of these generic accident conditions is limited to onsite transfers using trucks. It is important to understand that these generic accident conditions are for truck accidents during the transfer. Other onsite transfer activities (i.e., loading and unloading) may have associated accidents that result in more severe accident conditions. For example, a drop from the maximum height that a package is raised during a loading operation may be more damaging to the package than the free drops specified in the generic sets of accident conditions. If this is the case, then the loading operation drop event should be used in the package evaluation.

Two sets of generic accident conditions were developed for use: one for off-normal events and one for emergency events. These generic sets of accident conditions can be used without further justification of probabilities if the package considered weighs at least 3000 lb and the shipment miles are less than 1000/yr.

Tables 3.2 and 3.3 of this section contain the generic set of accident conditions presented in a test/analysis sequence analogous to the sequence in the regulations for Type B packages. Table 3.2 contains the set of accident conditions for accidents that fall into the off-normal event category (accidents with a frequency between 10^{-1}/yr and 10^{-4}/yr). Table 3.3 contains the set of accident conditions for accidents that fall into the emergency event category (accidents with a frequency between 10^{-4}/yr and 10^{-6}/yr).

Table 3.2. Recommended Test/Analysis Sequence Representing Off-Normal Events: Accidents With Frequencies Between 10^{-1}/yr and 10^{-4}/yr in General Onsite Transportation

1.	A free drop of the container through a distance of 1.5 m (5 ft) onto a flat, rigid, horizontal surface; the container strikes the surface in a position for which maximum damage to the container is expected.
2.	A free drop of the container through a distance of 15 cm (6 in.) in a position for which maximum damage is expected onto a 15-cm (6-in.)-diameter bar mounted on a rigid, horizontal surface.
3.	Exposure to a 1.8-m (6-ft)-diameter pool fire for not less than 5 min. The fire shall be encompassing for container dimensions less than 1.8 m (6 ft).

Table 3.3. Recommended Test/Analysis Sequence Representing Emergency Events: Accidents With Frequencies Between 10^{-4}/yr and 10^{-6}/yr in General Onsite Transportation

1.	A free drop of the container through a distance of 4.6 m (15 ft) onto a flat, rigid, horizontal surface; the container strikes the surface in a position for which maximum damage to the container is expected.
2.	A free drop of the container through a distance of 51 cm (20 in.) in a position for which maximum damage is expected onto a 15-m (6-in.)-diameter bar mounted on a rigid, horizontal surface.
3.	Exposure to a 1.8-m (6-ft)-diameter pool fire for not less than 15 min. The fire shall be encompassing for container dimensions less than 1.8 m (6 ft).

A list of onsite transfer-specific credible events can be developed from the package usage information provided in the contents characterization and package usage data sheets. Credible events in each of the aforementioned categories shall be identified. Once a set of credible events has been established, a recognized methodical process shall be used to assign probable event frequencies to each event. The probable event frequencies for each credible event may be affected by the following:

- onsite administrative controls
- transport mode
- route

- road conditions, obstacles, topography
- trip frequency
- distance traveled
- any special features of the package contents or vehicle.

The frequency ranges for credible normal, anticipated, off-normal, and emergency events are defined below.

Credible Event Type	Probability Range
Normal	1/transfer
Anticipated	10^{-1}/year ≤ probability < 1/transfer
Off-Normal	10^{-4}/year ≤ probability < 10^{-1}/yr
Emergency	10^{-6}/year ≤ probability < 10^{-4}/yr

Events with a probability of less that 10^{-6}/yr are defined as incredible and are not considered in the safety assessment for onsite transfers of radioactive material.

3.2.2 Design Basis Events

Design basis events shall be selected from the list of credible events for each of the event categories. Each design basis event shall be the combination of the worst-case credible event for that event category and the worst-case combination of radionuclides for that failure mode.

For example, the design basis emergency event for containment is described below:

- the credible emergency event causing the most damage affecting containment, combined with
- that combination of radionuclides within the ranges permitted by the authorized contents that cause the largest health effect if released into the air or water.

Similarly, the design basis emergency event for shielding is described below:

- the credible emergency event causing the most damage affecting shielding, combined with
- that combination of radionuclides within the ranges permitted by the authorized contents that cause the largest health effect from penetrating radiation, (i.e., gammas and neutrons).

All normal events (collectively) and the applicable design conditions are the design basis for normal conditions. The following non-normal design basis events shall be identified:

- Design basis anticipated event for containment shielding
- Design basis anticipated event for shielding
- Design basis anticipated event for criticality
- Design basis off-normal event for containment
- Design basis off-normal event for shielding
- Design basis off-normal event for criticality
- Design basis emergency event for containment
- Design basis emergency event for shielding
- Design basis emergency event for criticality.

The collective set of design conditions for normal events and the identified design basis events are the required conditions under which the package must be shown to adequately perform.

3.3 General Package Requirements

This section provides general package requirements for onsite transfers. If the package is also used for storage, additional storage requirements may apply. Alternate packaging for onsite transfer of radioactive material shall meet the following requirements:

1. The impact resistance, strength, confinement and containment capability, thermal capacity, and other features important to safety provide required protection to workers, members of the public, and the environment from radiation exposure during normal operations and all other design basis events for containment.[2]

2. Radiation shielding components provide the required protection to workers, members of the public, and the environment from radiation exposure during normal operations and all other design basis events for shielding.[3]

3. The packaging maintains the contents in subcritical condition during normal operations and under all design basis events for criticality.[4]

4. Excessive stress from melting of solid contents or from expansion of liquid contents from temperature increases is precluded. Excessive overpressure from water vapor expansion from temperature increases during normal operations and all other design basis events is precluded.

5. Failure of any tie-down attachment from excessive load will not impair the function of any packaging safety system.

6. Incompatibility (including corrosivity, permeability, softening, premature aging, embrittlement, and chemical or galvanic reaction) between the packaging components and between the components and the package contents are precluded.[5]

7. Protection is provided for all subsidiary hazards.

8. Any gas created inside a package, during use or while the package was in storage prior to use, does not cause a fire or explosive hazard.

9. Penetrations through containment boundaries are protected from damage during use.

10. Containers involving any significant gas buildup, automatic pressure relief, or other venting cause no personnel radiation exposure or spreading of contamination from any release during normal operations and cause exposure or spread of contamination less than established limits during other design basis events.[6]

11. Decontamination is facilitated to the extent practical.[7]

12. Inner packagings of combination packagings are packed and secured to ensure the following:

    a) closures are upright
    b) breakage or leakage is prevented
    c) movement in the outer packaging is controlled.

13. The authorized gross weight is legibly marked on each alternate Type B packaging.

4.0 STRUCTURAL CRITERIA

4.1 General Criteria

Packaging structural integrity shall be established by analysis, testing, or comparison with existing authorized packagings or by a combination of analysis, testing, and comparison. The analysis or the combination of analysis and testing shall consider the containment, shielding, thermal, criticality, lifting, and tie-down functions and the stress, strain or deformation, structural stability, brittle fracture, and fatigue failure modes during normal operations and under the design conditions resulting from the selected design basis events. All loads identified using the criteria of Section 3.2 shall be considered.

The structural analysis in the safety assessment shall assess the following:

- Internal pressure resulting from gas buildup and temperature increases
- Radiation shielding capability of the package after design basis shielding events
- Confinement capability of the package after design basis containment events
- Damage affecting criticality safety
- Ability of inner packagings in combination packagings to retain their orientation, cushioning, and position within the outer packaging
- Stress effects of the melting of solid contents and the expansion of liquids subjected to an increased temperature
- Fatigue, embrittlement, corrosion, and buckling and other structural instability
- Excessive deformations in portions of the packaging that are not part of the safety system
- Vibration of the packaging components
- Stresses on any packaging feature that could be used inadvertently as a tie-down attachment.

4.2 New Alternate Type B Onsite Package Criteria

The structural criteria for new alternate Type B packages requires that stresses in the containment boundary be less than the limits in Table 4.1 and that the stresses in other structures be less than the limits in Table 4.2.

Table 4.1. Structural Design Criteria for the Containment Boundary Structures of Packagings With More Than an A_2 Quantity of Radioactive Material[a]

Stress Category	Normal and Anticipated Events	Off-Normal and Emergency Event
Primary Membrane, General P_m Local P_l[b]	S_m 1.5 S_m	Lesser of: 2.4 S_m or 0.7 S_u 3.6 S_m or S_u[c]
Primary Membrane + Bending P_m or $P_l + P_b$	1.5 S_m[b]	Lesser of: 3.6 S_m S_u[c]
Range of Primary + Secondary P_m or $P_l + P_b + Q$	3.0 S_m[d]	2 x S_a for 10 cycles (Code Sect III App.) (Including F)
Bearing Stress	S_y	S_y for seal surfaces S_u elsewhere
Pure Shear Stress	0.6 S_m	0.42 S_u
Fatigue	Cum. fac = 1	Cumulative factor = 1, if applicable
Buckling[e] (Suitable Factors Based Upon ASME Code)	F.S. = 2	F.S. = 1.34 (Code case N-284)
Average Stress in Fasteners	2/3 S_y	2/3 S_y
Maximum Combined Stress in Fasteners	S_y	S_y

[a] These limits are consistent with the ASME code.
[b] From Code article NB-3200 and Appendix F, paragraph F-1331.1.
[c] When using nonlinear elastic plastic analyses for end drop accidents, the general primary membrane stress intensity, P_m, shall not exceed 0.7 S_u and the maximum primary stress intensity at any location (P_l or $P_l + P_b$) shall not exceed 0.90 S_u (in accordance with Appendix F of Section III)
[d] Regulatory guide 7.6, position C4 provides an alternate criteria.
[e] Code design formulas to preclude buckling are acceptable.

P_m = Primary Membrane Stress
P_l = Local Primary Membrane Stress
S_m = Design Stress Intensity as given in appendix I, tables I-1.0 of Section III of the ASME Code for Class 1 Components
S_u = Minimum Ultimate Tensile Strength
P_b = Bending Stress
Q = Secondary Stress
S_a = Allowable Amplitude of the Alternating Stress Intensity
S_y = Minimum Yield Stress
F.S. = Factor of Safety

Table 4.2. Structural Design Criteria For Non-Containment Structures of Packagings With More Than A_2 Quantity of Radioactive Material

Stress Category	Normal and Anticipated Events	Off-Normal and Emergency Event
Primary Membrane, General P_m Local P_1[a]	S_m $1.5\ S_m$	$0.7\ S_u$[b] S_u[b]
Primary Membrane + Bending P_m or $P_1 + P_b$	$1.5\ S_m$[c]	S_u[b]
Range of Primary + Secondary P_m or $P_1 + P_b + Q$	$3.0\ S_m$[d]	Not Applicable
Bearing Stress	S_y	S_u
Pure Shear Stress	$0.6\ S_m$	$0.5\ S_u$
Buckling	Buckling to be precluded	
Average Stress in Fasteners	$2/3\ S_y$	Greater of: S_y or $0.7\ S_u$
Maximum Combined Stress in Fasteners	S_y	Greater of: $1.5\ S_y$ or S_u

[a] From Code article NB-3200 and Appendix F, paragraph F-1331.1.

[b] When using nonlinear elastic plastic analyses for end drop accidents, the general primary membrane Regulatory guide 7.6, Position C4 provides an alternate criteria stress intensity, P_m, shall not exceed $0.7\ S_u$, and the maximum primary stress intensity at any location (P_1 or $P_1 + P_b$) shall not exceed $0.90\ S_u$ (in accordance with Appendix F of Section III).

[c] $1.5\ S_m$ may be exceeded for non-containment structure if deflection is acceptable.

[d] Regulatory guide 7.6, Position C4 provides an alternate criteria.

P_m = Primary Membrane Stress
P_1 = Local Primary Membrane Stress
S_m = Design Stress Intensity as given in Appendix I, table I-1.0, Section III of the ASME Code for Class 1 Components
S_u = Minimum Ultimate Tensile Strength
P_b = Bending Stress
Q = Secondary Stress
S_y = Minimum Yield Stress

4.3 Existing Alternate Type B Onsite Package Criteria

The structural criteria for existing alternate Type B onsite packages requires that stresses in the containment boundary be less than the limits in Table 4.1 and that the stresses in other structures be less than the limits in Table 4.2.

An existing alternate Type B onsite package with solid, non-dispersible contents can be considered to have met the structural criteria if the following conditions have been met:

- Acceptable functional capability through usage has been demonstrated.
- All predicted stresses are below their limits except stresses in the closure components.
- The maximum permanent strain in the closure components is not more than 0.75 of their rupture strain from the design basis containment event, or other justification clearly demonstrates the package will retain its contents after the design basis containment event.
- Loss of function does not occur through excessive distortion.
- The rupture strain is determined by test, taking proper account of the actual stress and strain distribution.
- The amount of shielding degradation that would occur from structural loads resulting from the design basis shielding events is predicted.

An existing alternate Type B onsite package with liquid or dispersible solid contents can be considered to have met the structural criteria if the following conditions have been met:

- Acceptable functional capability through usage has been demonstrated.
- All predicted stresses are below their limits except stresses in the closure components.
- The pre-load in the closure bolts and other closure components properly maintains the closure seal after the design basis containment event, or other justification clearly demonstrates the leakage from the package would be a single spurt after the design basis containment event.
- The potential for higher-than-normal contamination on the package from a spurt of the contents is marked on the package.

- The amount of contents released in the spurt is used in predicting the radiation exposure to transport workers or emergency responders.

- The amount of shielding degradation from the design basis shielding events is predicted.

4.4 Alternate Type A, IP-2, and IP-3 Package Criteria

The structural criteria for Type A, IP-2, and IP-3 onsite packagings are as follows:

- The structural limits shown in Table 4.3 are used in conjunction with the defined credible events for that package or other structural criteria, if justified.

- Other justified structural criteria are used in conjunction with the defined credible events.

4.5 Excepted and IP-1 Package Criteria

The structural criteria for excepted quantity and article (49 CFR 173.423) and IP-1 packagings may be justifiable limits used in conjunction with the normal events.

Table 4.3. Structural Design Criteria for Packagings for Radioactive Material Contents of Less Than an A_2 Quantity, IP-2, IP-3 Packages.

Stress Category	Normal and Anticipated Events
Primary Membrane, General P_m Local P_1[a]	S_m $1.5\ S_m$
Primary Membrane + Bending P_m or $P_1 + P_b$	$1.5\ S_m$[b]
Range of Primary + Secondary P_m or $P_1 + P_b + Q$	$3.0\ S_m$[c]
Bearing Stress	S_y
Pure Shear Stress	$0.6\ S_m$
Buckling	Buckling to be precluded
Average Stress in Fasteners	$0.9\ S_y$
Maximum Combined Stress in Fasteners	S_y

[a] From Code article NB-3200 and Appendix F paragraph F-1331.1.
[b] 1.5 S_m may be exceeded for non-containment structure if deflection is acceptable.
[c] Regulatory guide 7.6, Position C4 provides an alternate criteria.

P_m = Primary Membrane Stress
P_1 = Local Primary Membrane Stress
S_m = Design Stress Intensity as given in appendix I, tables I-1.0 of Section III of the ASME Code for Class 1 Components
S_u = Minimum Ultimate Tensile Strength
P_b = Bending Stress
Q = Secondary Stress
S_y = Minimum Yield Stress

5.0 THERMAL CRITERIA

All contact-handled, alternate onsite packagings for radioactive materials shall be designed to provide adequate heat removal capability without using an active heat removal system. The maximum temperature on any exposed exterior surface shall not exceed 122°F in shade in 100°F still air ambient during all normal and anticipated transfer events. The maximum temperature on the exposed exterior surface can be extended to 180°F if appropriate controls equivalent to those used for exclusive use are in place.

Temperature calculation shall be based upon information in the contents characterization and package usage data sheets. The information used shall include the ambient temperatures and the isotopic composition of the contents identified as providing the highest decay heat load. The effect of solar insolation on the packaging and its contents shall be considered.

Any heat generated by friction between packaging components shall be considered in the temperature calculations.

6.0 CONTAINMENT CRITERIA

The containment criterion for all packages used for onsite transfer is to reduce leakage (and the resultant package contamination) to levels that are ALARA and that are below established limits.

The containment criterion for authorized offsite packages, when used for onsite transfers, is compliance with the appropriate leak rate and contamination limits imposed on offsite shipments.

The criterion for alternate packages is to meet the established environmental, contamination, and personnel radiation exposure limits found in Chapter 12.

6.1 Authorized Offsite Type B Packages

The containment criterion for an authorized offsite Type B packaging, when used for an onsite transfer, is compliance with:

- 10 CFR 71.51 "Additional Requirements for Type B Packages"

6.2 Authorized Offsite Type A, IP-2, and IP-3 Packages

The containment criterion for an authorized offsite Type A, IP-2, or P-3 packaging when used for an onsite transfer is compliance with:

- 49 CFR 173.412 "Additional Requirements for Type A Packages"
- IAEA Safety Series 6, Para 519 and 520, "Additional Requirements for Industrial Package Type 2 (IP-2). . .(IP-3)."

6.3 Authorized Offsite Excepted and IP-1 Packages

The containment criterion for excepted and IP-1 packagings is compliance with the requirements of:

- 49 CFR 173.421, "Limited Quantity of Radioactive Material"
- IAEA Safety Series 6, Para 518, "Requirements for Industrial Packages Type 1 (IP-1)."

6.4 Alternate Packages

Alternate packaging, together with the controls, limits, and requirements identified in the safety assessment; shall control radiation exposure to be ALARA and below established limits. The radiation protection criteria state that the predicted radiation exposure is below the environmental, contamination, and personnel exposure limits.

The containment criteria shall be established by the designer such that the combined exposure from released material (i.e., containment), from penetrating radiation (i.e., shielding), and from package surface contamination is ALARA and is predicted to be below the limits established in the radiation protection chapter of the criteria document.

7.0 SHIELDING CRITERIA

The shielding criterion for all packages used for onsite transfer is to reduce penetrating radiation to levels which are ALARA.

The shielding criterion for authorized offsite packages, when used for onsite transfers, is compliance with the package radiation limits imposed on offsite shipments.

The criterion for alternate packages is to meet the established environmental, contamination, and personnel radiation exposure limits found in Chapter 12.

7.1 Authorized Offsite Type B Packages

The shielding criterion for an authorized offsite Type B packaging is compliance with:

- 49 CFR 173.441, "Radiation Level Limits"
- 10 CFR 71.51, "Additional Requirements for Type B Packages"

7.2 Authorized Offsite Type A, IP-2, and IP-3 Packages

The shielding criterion for an authorized offsite Type A, IP-2, or IP-3 packaging is compliance with:

- 49 CFR 173.441, "Radiation Level Limits"
- 49 CFR 173.412, "Additional Requirements for Type A Packages"
- IAEA Safety Series 6, Para 519 and 520, "Additional Requirements for Industrial Package Type 2 (IP-2). . .(IP-3)."

7.3 Authorized Offsite Excepted and IP-1 Packages

The shielding criterion for excepted and IP-1 packagings is compliance with the contents, radiation, and contamination requirements of:

- 49 CFR 173.421, "Limited Quantity of Radioactive Material"
- IAEA Safety Series 6, Para 518, "Requirements for Industrial Package Type 1 (IP-1)", 1985 ed.

7.4 Alternate Packages

Alternate packaging, together with the controls, limits, and requirements identified in the safety assessment, shall control radiation exposure to be ALARA and below established limits. The radiation protection criteria state that the predicted radiation exposure shall be below the environmental, contamination, and personnel exposure limits.

The shielding criteria shall be established by the designer such that the combined exposure from released material (i.e., containment), from penetrating radiation (i.e., shielding), and from package surface contamination is ALARA and is predicted to be below the limits established in the radiation protection chapter of the criteria document.

8.0 CRITICALITY SAFETY CRITERIA

Any package meeting the package descriptions in 49 CFR 173.453, "Fissile Materials—Exceptions" meets the criticality safety for onsite transfers.

Criticality safety for all other onsite packages containing fissile material shall be provided by compliance with the following applicable regulations and directives:

- DOE regulations
- NRC regulations
- DOE directives

8.1 Applicable DOT Regulations

DOT authorized packages identified in 49 CFR 173.417 and any alternate onsite packages, which are essentially identical to DOT authorized packages, may use the following criticality safety criteria for onsite transfers:

- The packages meet the associated package, vehicle, or package limits included in 49 CFR 173.417.
- The packages comply with the following regulations:
  - 49 CFR 173.451, "Fissile Materials—General Requirements"
  - 49 CFR 173.455, "Classification of Fissile Materials Packages"
  - 49 CFR 173.457, "Transportation of Fissile Class III Shipment Specific Requirements"
  - 49 CFR 173.459, "Mixing of Fissile Material Packages."
- The alternate packages do not differ from the DOT specification packaging in any way that might increase the criticality hazard.
- The alternate packages meet the associated package, vehicle, or package limits included in 49 CFR 173.417.

8.2 Applicable NRC Regulations

NRC regulations that apply to criticality safety of fissile packages appear in 10 CFR 71. The applicable requirements are listed below in a very slightly modified form. The modification is merely to remove references to NRC regulatory test conditions (i.e., hypothetical accident conditions tests) and substitute reference to the design basis events for criticality. The requirements listed in Sections 6.2.1 through 6.2.4 may be used as the criticality safety criteria for any onsite package.

8.2.1 General Requirements for All Fissile Material Packages

1. 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 maximum reactivity of the fissile material would be obtained under the following conditions:

    - The most reactive credible configuration consistent with the chemical and physical form of the material
    - Moderation by water to the most reactive credible extent
    - Close reflection by water on all sides.

2. Exceptions to requirement 1 can be proposed if the package incorporates special design features that ensure that no single packaging error would permit leakage and if appropriate measures are taken before each shipment to ensure the containment system does not leak.

3. A package used for fissile material shipment must be so designed and constructed and its contents so limited that under normal and anticipated design basis events the following occur:

    - The contents would be subcritical.
    - The geometric form of the package contents would not be substantially altered.
    - Water would not leak into the containment system unless, in the evaluation of the undamaged package arrays, it has been assumed that moderation is present to such an extent as to cause maximum reactivity consistent with the chemical and physical form of the material.
    - There will be no substantial reduction in the effectiveness of the packaging, including: (1) no more than five percent reduction in the effective volume of the packaging on which nuclear safety is assessed, (2) no more than five percent reduction in the effective spacing between the fissile contents and the outer surface of the packaging, and (3) no occurrence of an aperture in the outer surface of the packaging large enough to permit the entry of a 10-cm (4-in.) cube.

4. A package used for the fissile material shipment must be so designed and constructed and its contents so limited that under the most severe off-normal and emergency design basis events for criticality, the package would be subcritical. For this determination, the following must be assumed:

   - The fissile material is in the most reactive credible configuration consistent with the damaged condition of the package and the chemical and physical form of the contents.

   - Water moderation occurs to the most reactive credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents.

   - There is reflection by water on all sides, as close as is consistent with the damaged condition of the package.

8.2.2 Specific Standards for a Fissile Class I Package

A fissile Class I package must be so designed and constructed and its contents so limited that the following are true:

1. Any number of undamaged packages would be subcritical in any arrangement and with optimum interspersed hydrogenous moderation, unless there is a greater amount of interspersed moderation in the packaging, in which case the greater amount may be assumed for this determination.

2. Two hundred and fifty (250) packages, if each package were subjected to the most severe of the off-normal and emergency design basis events for criticality, would be subcritical if stacked together in any arrangement, closely reflected on all sides of the stack by water, and with optimum interspersed hydrogenous moderation.

8.2.3 Specific Standards for a Fissile Class II Package

1. A fissile Class II package must be controlled by the carrier during transport. To provide this control, the designer of a fissile Class II package must determine the allowable number of packages of that design that can be safely transported in a vehicle under the conditions specified in this subsection. This allowable number of packages determines the minimum transport index that the shipper of the package marks on the package label when the package is shipped. By limiting the total number of transport indexes in a vehicle or storage area to 50, the carrier provides adequate criticality control.

2. A fissile Class II package must be designed and constructed and its contents so limited, and the allowable number of these packages in a Fissile Class II shipment so determined, that the following are true:

- Five times the allowable number of undamaged packages would be subcritical if stacked together in any arrangement and closely reflected on all sides of the stack by water.

- Twice the allowable number of packages, if each package were subjected to the most severe of the off-normal and emergency design basis events for criticality, would be subcritical if stacked together in any arrangement, closely reflected on all sides of the stack by water, and with optimum interspersed hydrogenous moderation.

3. The transport index, with respect to criticality control for each fissile Class II package, must be calculated by dividing the number 50 by the allowable number of fissile Class II packages that may be transported together as determined under the limitations of criterion 1 of this subsection. The transport index so determined must not exceed 10 and must be rounded up to the first decimal place.

8.2.4 Specific Standards for a Fissile Class III Shipment

A package for fissile Class III shipment must be so designed and its contents so limited, and the number of packages in a fissile Class III shipment must be so limited, that the following are true:

- Twice this number of undamaged packages would be subcritical if stacked together in any arrangement, assuming close reflection on all sides of the stack by water.

- This number of packages would be subcritical if stacked together in any arrangement, closely reflected on all sides of the stack by water, and with optimum interspersed hydrogenous moderation. Each package must be considered to have been subjected to the most severe of the off-normal and emergency design basis events for criticality.

8.3 Applicable DOE Directives

The applicable DOE directive is 5480.5, "Safety of Nuclear Facilities," which provides alternate requirements for establishing and maintaining criticality safety during onsite transfers of fissionable material.

Subparagraph 12(f) requires the following for onsite transfers:

- For the onsite movement of fissionable materials that do not present a radiation hazard, the pertinent requirements set forth in this order shall be met.

- For the onsite movement of fissionable materials that presents a radiation hazard, as well as the possibility of an accidental chain reaction, the pertinent requirements of DOE Order 5480.3,[8] and DOE Order 5480.11[9], *Radiation Protection for Occupational Workers*, shall be met.

- In addition to the physical controls specified in this order, administrative controls, including traffic controls, shall be exercised as deemed necessary to minimize accident probabilities.

- Fire protection, security, health physics, and any other emergency personnel, when deemed appropriate shall be alerted and advised of movements and routings.

The aforementioned mandatory requirements are included in the various appropriate sections of this criteria document.

DOE Order 5480.5 contains the pertinent requirements for criticality safety in Paragraph 11, "Nuclear Criticality Safety Elements," and Paragraph 12, "Nuclear Criticality Safety Control Parameters." These requirements for criticality safety are acceptable criteria for any onsite package.

The second requirement from subparagraph 12(f) of DOE Order 5480.5 (second bullet of this section) implies that DOE Order 5480.3 applies to onsite transfers. DOE Order 5480.3 was not written or intended for use for onsite transfers of radioactive material. DOE Order 5480.3 requires that packages used for offsite transport be in compliance with all applicable safety regulations of the DOT and follow the applicable packaging standards of the NRC for offsite shipments. The criticality safety requirements from the applicable DOT and NRC regulations were previously provided in Sections 8.1 and 8.2 of this chapter.

DOE Order 5480.11 contains requirements for site-wide radiation protection programs. The pertinent requirements for radiation protection during onsite transfers are included in Chapter 2.

8.4 Calculation of K_{eff}

A limit of 0.95 for K_{eff} should be used for criticality calculations to provide an appropriate safety margin. This limit is often used in Special Nuclear Material (SNM) licenses and is recommended in the criticality evaluation section of the DOE Packaging Review Guide.[10] The review guide also describes how to determine and apply the calculational bias to the calculated K_{eff}.

9.0 OPERATING PROCEDURE CRITERIA

Written operating procedures are required to be followed for those activities involving packaging components or functions that are important to radiation safety, criticality safety, and providing hazard information to transfer workers or emergency responders.

Written procedures shall also be used whenever the following occur:

1. Filling, closing, leak testing, loading on transport vehicle, tying down, unloading, opening, and emptying activities for any packages containing greater than A_2 quantities of radioactive material

2. Performing routine determination activities related to package operations identified in the package quality assurance criteria that require reports or initialed check sheets

3. Conducting operations activities as specified in the chapter on operating procedures of the safety assessment document.

10.0 ACCEPTANCE TESTING AND MAINTENANCE PROGRAM CRITERIA

Written acceptance testing procedures are required to be followed for those activities involving packaging components or functions that are important to radiation safety, criticality safety, and providing hazard information to transfer workers or emergency responders.

Written acceptance testing procedures shall also be used whenever the following occur:

- Inspecting, acceptance testing, or reviewing and evaluating vendor inspection and testing activities prior to first use of any packages containing greater than A_2 quantities of radioactive material

- Performing preliminary determination activities related to package acceptance identified in the package quality assurance criteria that require reports or initialed check sheets

- Conducting acceptance testing as specified in the acceptance testing and maintenance chapter of the safety assessment document.

Written maintenance procedures are required to be followed for those activities involving packaging components or functions that are important to radiation safety, criticality safety, and providing hazard information to transfer workers or emergency responders.

Written maintenance procedures shall also be used whenever the following occur:

- Conducting periodic maintenance and inspection activities for any packages containing greater than A_2 quantities of radioactive material

- Performing periodic determination and maintenance activities identified in the package quality assurance criteria that require reports or initialed check sheets

- Conducting maintenance and inspection activities as specified in the acceptance testing and maintenance chapter of the safety assessment document.

11.0 PACKAGE QUALITY ASSURANCE CRITERIA

A graded level of quality assurance (QA) for radioactive material packagings is necessary. Packagings for the material with higher activity require a more comprehensive and stringent QA program than for those packagings containing material with lower activity. A QA program that incorporates a graded level of QA shall be established for each package category listed below:

1. NRC, DOE, or International Atomic Energy Association (IAEA) certified
2. DOT 6M, 6L, 20WC, or 21WC
3. Alternate onsite Type B
4. DOT 7A
5. Existing alternate onsite Type A
6. New alternate onsite Type A, IP-2, and IP-3
7. New onsite low-specific activity (LSA) packagings
8. Excepted packagings and IP-1
9. Empty packagings

11.1 NRC, DOE, or IAEA Certified

The performance objective of the QA program for certified packagings is that all packaging components and features that are important to safety are designed, fabricated, maintained, and used in a manner so they continue to perform as expected. To ensure meeting this objective for certified packagings used for onsite transfers, the QA program should implement the applicable sections of 10 CFR 71, Subpart H and follow the guidance of NRC Regulatory Guide 7.10. The QA program shall be designed to meet minimum criteria for preliminary and routine determinations (see Glossary), and periodic determinations (see Glossary) and maintenance.

11.1.1 Preliminary Determination

The performance objective for the preliminary determination of NRC, DOE, or IAEA certified packaging is that all design, fabrication, acceptance testing, receiving inspection, and other QA requirements, identified in either the Safety Analysis Report for Packaging (SARP) or certificate of compliance (CoC), are met.

The criteria that ensure the preliminary determination for a NRC, DOE, or IAEA certified packaging are met include the following:

- A current CoC is readily available for each package used.
- The current operating and maintenance instructions are readily available for each package used.
- Fabrication records are available.
- The packaging configuration being used is the same as that certified.
- Documentation is available showing compliance with all preliminary determination requirements in the CoC and the SARP.

11.1.2 Routine Determination

The performance objective for the routine determination of a NRC, DOE, or IAEA certified packaging is that all inspections, tests, and activities in the SARP or CoC important to containment, shielding, criticality, and thermal control are performed. The packaging meets all routine maintenance requirements in the CoC and the SARP.

The criteria that ensure the routine determination for a NRC, DOE, or IAEA certified packaging are met include the following:

- Either reports or initialed check sheets for required inspections are available.
- Reports with test results for the required tests are available.
- Evidence (e.g., memos, work sheets, work request forms, or transfer and shipping papers) is available showing the material shipped is authorized by the CoC.
- Evidence is available showing the material shipped does not exceed the authorized amount.
- Evidence is available showing the material shipped does not exceed the authorized decay heat limit.
- Evidence is available showing the material shipped is in the authorized chemical and physical form.
- Evidence is available showing the material meets any other restriction in the SARP or CoC.

11.1.3 Periodic Determination and Maintenance

The performance objective for the periodic determination and maintenance of a NRC, DOE, or IAEA certified packaging is that all periodic inspections, tests, and maintenance required by the SARP and CoC are performed.

The criteria that ensure the periodic determination and maintenance for a NRC, DOE, or IAEA certified packaging are met include the following:

- Reports or initialed check sheets for required inspections are available.
- Reports with test results for required tests are available.

11.2 DOT 6M, 6L, 20WC, or 21WC

The performance objective of the QA program for these DOT specification packagings is that all packaging components and features that are important to safety are designed, fabricated, maintained, and used in a manner so they continue to perform as expected. To ensure meeting this objective for these DOT specification packagings used for onsite transfers, the QA program shall meet the minimum criteria for preliminary and routine administrations, and periodic determination and maintenance.

11.2.1 Preliminary Determination

The performance objective for the preliminary determination of a DOT, 6M, 6L, 20WC, or 21WC packaging is that all dimensional, material, and configuration requirements, testing, and other design specifications identified in 49 CFR 178, subpart K were met.

The criteria that ensure the preliminary determination for a DOT, 6M, 6L, 20WC, or 21WC packaging are met include the following:

- A vendor certification of compliance with the appropriate DOT specification is available.
- A receiving inspection report confirming dimensional and configurational compliance with DOT regulations is available.

11.2.2 Routine Determination

The performance objective for the routine determination of a DOT, 6M, 6L, 20WC, or 21WC packaging is that all inspections, tests, and activities required by 49 CFR 178, Subpart K and other inspections, tests, and activities determined to be important to containment, shielding, criticality, and thermal control are performed.

The criteria that ensure the routine determination for a DOT, 6M, 6L, 20WC, or 21WC packaging are met include the following:

- Either reports or initialed check sheets are readily available for inspections or tests that are required by 49 CFR 178, subpart K.
- Reports with test results for required tests are available.
- Initialed check sheets for loading and package closing activities required by 49 CFR 178, subpart K or those activities important to containment, shielding, criticality, and thermal control are available.
- Evidence (e.g., memos, work sheets, work request forms, or shipping papers) is available showing the material shipped is authorized by 49 CFR 178, subpart K.
- Evidence is available showing the material shipped does not exceed authorized amount.
- Evidence is available showing the material shipped does not exceed authorized decay heat limit.
- Evidence is available showing the material shipped is in the authorized chemical and physical form.
- Evidence is available showing the material meets any other restriction in 49 CFR 178, subpart K.

11.3 Alternate Onsite Type B

The performance objective of the QA program for existing and new alternate onsite Type B packages is that all packaging components and features which are important to safety are designed, fabricated, maintained and used in a manner so they continue to perform as expected. To assure meeting this objective for existing and new alternate onsite Type B packages used for onsite transfers, the QA program should implement the applicable sections of 10 CFR 71, subpart H and follow the guidance of NRC Regulatory Guide 7.10. The QA program shall be designed to meet minimum criteria for preliminary and routine determinations, periodic determination, and maintenance.

11.3.1 Preliminary Determination

The performance objective for the preliminary determination of an existing or new alternate onsite Type B package is that all design, fabrication, acceptance testing, receiving inspection, and other QA requirements, identified in the approved safety assessment are met.

The criteria that ensure meeting the preliminary determination for an existing or new alternate onsite Type B package are as follows:

- A approved safety assessment is readily available for each package used.
- Packaging components and features important to safety (e.g., containment, criticality control, radiation protection, and thermal control) are identified in the approved safety assessment.
- Material specifications and fabrication activities necessary to ensure that components and features important to safety will perform as desired and are identified in the approved safety assessment.
- Acceptance testing specifications were met and other activities necessary to ensure that components and features important to safety will perform as desired and were satisfactorily completed.
- Review of the safety assessment confirms that correct packaging components and features important to safety are identified, correct material specifications and fabrication activities are identified, and correct acceptance testing specifications and other activities are identified.
- Current operating and maintenance instructions are readily available for each package used.
- Fabrication records are available.
- The packaging configuration being used is the same as that described in the safety assessment.
- Documentation is available showing compliance with all preliminary determination requirements in the approved safety assessment.

11.3.2 Routine Determination

The performance objective for the routine determination of an existing or new alternate onsite Type B package is that all inspections, tests, and activities in the approved safety assessment important to containment, shielding, criticality, and thermal control are performed and that packaging meets all routine maintenance requirements identified in the approved safety assessment.

The criteria that ensure the routine determination for an existing or new alternate onsite Type B package are as follows:

- Either reports or initialed check sheets are available for required inspections.

- Reports with test results for required tests are available.

- Evidence (e.g., memos, work sheets, work request forms, or other papers) is available showing the material shipped is authorized by the safety assessment.

- Evidence is available showing the material shipped does not exceed the authorized amount.

- Evidence is available showing the material shipped does not exceed the authorized decay heat limit.

- Evidence is available showing the material shipped is in the authorized chemical and physical form.

- Evidence is available showing the material meets any other restriction in the approved safety assessment.

11.3.3 Periodic Determination and Maintenance

The performance objective for the periodic determination and maintenance of an existing or new alternate onsite Type B package is that all special inspections, tests, and maintenance required by the approved safety assessment are performed.

The criteria that ensure meeting the periodic determination and maintenance for an existing or new alternate onsite Type B package are as follows:

- Reports or initialed check sheets are available for required inspections.

- Reports with test results are available for required tests.

11.4 DOT Specification 7A Type A

The performance objective of the QA program for these DOT Specification 7A Type A packagings is that all packaging components and features that are important to safety are designed, fabricated, maintained and used in a manner so they continue to perform as expected. To ensure meeting this objective for these DOT Specification 7A Type A packagings used for onsite transfers, the QA program shall meet minimum criteria for preliminary and routine determinations.

11.4.1 Preliminary Determination

The performance objective for the preliminary determination of a DOT Specification 7A Type A packaging is that all dimensional, material, configuration, testing, and other QA requirements identified in 49 CFR 178, subpart K are met, and performance tests that qualify this packaging as a 7A were performed in compliance with DOT regulations.

The criteria that ensure the preliminary determination for a DOT Specification 7A Type A packaging are met include the following:

- A vendor certification of compliance with the appropriate DOT specification is available.
- A receiving inspection report is available confirming regulatory compliance with those dimensional and configurational package features important to containment, shielding, and criticality control.
- A test report is available that describes the package configuration tested, provides the test conditions, and reports the test results.

11.4.2 Routine Determination

The performance objective for the routine determination of a DOT 7A packaging is that all inspections, tests, and activities required by 49 CFR or important to containment, shielding, and criticality were performed.

The criteria that ensure the routine determination for a DOT 7A packaging are met include the following:

- Evidence (e.g., special nuclear material (SNM), inventory control, laboratory test results, historical data including uncertainty limits, or expert knowledge of the custodian) is available to show the amount of material shipped is authorized for a 7A container.
- Documents are available that show the material shipped does not exceed the weight of the package tested.
- Written evidence is available showing the material shipped was similar to that in the package performance tests.
- Evidence is available showing the material shipped is in the chemical and physical form authorized by the package performance test.
- Evidence (e.g., laboratory measurements, historical data including uncertainty limits, or expert knowledge of the custodian) is readily available, which shows the material shipped does not exceed the authorized decay heat limit.
- Either reports are available that describe results of the required inspections, or operating procedures are used that require performing the required inspections.

11.5 Existing Alternate Onsite Type A

The performance objective of the QA program for existing alternate onsite Type A packagings is that all packaging components and features important to safety are designed, fabricated, maintained, and used in a manner so they continue to perform as expected. To ensure meeting this objective for existing alternate onsite Type A packagings used for onsite transfers, the QA program shall meet minimum criteria for preliminary and routine determinations.

The performance objective for the preliminary determination of an existing alternate onsite Type A packaging is that the package features important to containment, shielding, criticality, and thermal control perform as intended.

11.5.1 Preliminary Determination

The criteria that ensure meeting the preliminary determination for an existing alternate onsite Type A packaging are as follows:

- The user has an approved safety assessment and onsite authorization form.
- Drawing configuration control and fabrication shop practices were used for design and fabrication.
- A test report is available, which describes the package configuration tested, provides the test conditions, and reports the test results.

11.5.2 Routine Determination

The performance objective for the routine determination of an existing alternate onsite Type A packaging is that all inspections, and activities important to containment, shielding, and criticality control were performed.

The criteria that ensure meeting the routine determination for an existing alternate onsite Type A packaging are as follows:

- Either reports are available that describe results of the required each use inspections, or operating procedures are used that require performing the required inspections.
- Written evidence is available that shows the package contents transferred and the package contents in the performance tests have similar containment characteristics.
- Documents are available that show the material shipped does not exceed the content weight of the package tested.

- Evidence (e.g., laboratory measurements, historical data including uncertainty limits, or expert knowledge of the custodian) is available that shows the material shipped does not exceed the authorized decay heat limit.

- Evidence is available that shows the material shipped is in the chemical and physical form authorized by the package performance test.

- Evidence (e.g., SNM inventory control, laboratory test results, historical data including uncertainty limits, or expert knowledge of the custodian) is available that shows the amount of material shipped is less than Type B.

11.6 New Alternate Onsite Type A, IP-2, and IP-3

The performance objective of the QA program for new alternate onsite Type A, IP-2, and IP-3 packagings is that all packaging components and features important to safety are designed, fabricated, maintained, and used in a manner so they continue to perform as expected. To ensure meeting this objective for new alternate onsite Type A, IP-2, and IP-3 packagings used for onsite transfers, the QA program shall meet minimum criteria for preliminary and routine determinations.

11.6.1 Preliminary Determination

The performance objective for the preliminary determination of a new alternate onsite Type A, IP-2, and IP-3 packagings is that the packaging features that are important to containment, shielding, criticality, and thermal control perform as intended. Performance tests that qualify this packaging as an onsite Type A packaging comply with the onsite criteria document.

The criteria that ensure meeting the preliminary determination for a new alternate onsite Type A, IP-2, and IP-3 packagings include the following:

- The user has an approved safety assessment and onsite authorization form.

- Drawing configuration control and fabrication shop practices were used for design and fabrication.

- A test report is available to the packaging user, which describes the package configuration tested, provides the test conditions, and reports the test results.

11.6.2 Routine Determination

The performance objective for the routine determination of a new alternate onsite Type A, IP-2, and IP-3 packagings is that all inspections and activities important to containment, shielding, and criticality control were performed.

The criteria that ensure meeting the routine determination for a new alternate onsite Type A, IP-2, and IP-3 packagings include the following:

- Either reports are available that describe results of the required inspections or operating procedures are used that require performing the required inspections.
- Written evidence is available that shows the material transferred onsite had containment characteristics similar to the contents used in the package performance tests.
- Documents are available that show the material being transferred does not exceed the contents weight of the package tested.
- Evidence (e.g., laboratory measurements, historical data including uncertainty limits, or expert knowledge of the custodian) is available that shows the material being transferred onsite does not exceed the authorized decay heat.
- Evidence is available that shows the material being shipped is in the chemical and physical form authorized by the package performance test.
- Evidence (e.g., SNM inventory control records, laboratory test results, historical data including uncertainty limits, or expert knowledge of the custodian) is available that shows the amount of material being transferred onsite does not exceed an A_2 quantity of material.

11.7 New Onsite LSA Packagings

The performance objective of the QA program for new onsite LSA packagings is that all packaging components and features important to safety are designed, fabricated, maintained, and used in a manner so they continue to perform as expected. To ensure meeting this objective for new onsite LSA packagings used for onsite transfers, the QA program shall meet minimum criteria for preliminary and routine determinations.

11.7.1 Preliminary Determination

The performance objective for the preliminary determination of new onsite LSA packages is that the package used was properly represented by the package tested and the tests complied according to DOT regulations.

The criteria that ensure meeting the preliminary determination for a new onsite industrial package include the following:

- The user has an approved safety assessment and onsite authorization form.

- Drawing configuration control and fabrication shop practices were used for design and fabrication.

- A test report is available to the package user that describes the package configuration tested, provides the test conditions, and reports the test results.

11.7.2 Routine Determination

The performance objective for the routine determination of a new onsite LSA packagings is that all inspections and activities important to containment, shielding, thermal control, and criticality control were properly performed.

The criteria that ensure meeting the routine determination for a new onsite LSA packagings include the following:

- Either reports are available that describe results of the required inspections, or operating procedures are used, which require performing the required inspections.

- Written evidence is available that shows the material being shipped had similar containment characteristics to the material used in the package performance tests.

- Documents are available that show the material being shipped does not exceed the content weight of the package tested.

- Evidence (e.g., laboratory measurements, historical data including uncertainty limits, or expert knowledge of the custodian) is available that shows the material being shipped does not exceed the authorized decay heat.

- Evidence is available that shows the material being shipped is in the chemical and physical form authorized by the package performance test.

- Evidence (e.g., SNM inventory control, laboratory test results, historical data including uncertainty limits, or expert knowledge of the custodian) is available that shows that the material being shipped meets the appropriate definition of LSA for the package used.

11.8 Excepted Packaging and IP-1

The performance objective of the QA program for excepted and IP-1 packagings is that all packaging components and features important to safety are designed, fabricated, maintained, and used in a manner so they continue to perform as expected. To ensure meeting this objective for excepted and IP-1 packagings used for onsite transfers, the QA program shall meet minimum criteria for preliminary and routine determinations.

11.8.1 Preliminary Determination

The performance objective for the preliminary determination of excepted and IP-1 packaging for onsite transfers is that the packaging obtained will contain the material during transportation if unexpected incidents do not occur.

The criteria that ensure meeting the preliminary determination of excepted and IP-1 packaging for onsite transfers include the following:

- Experience with this package design and procurement process indicate that it has satisfactorily contained similar material during a similar transfer or shipment.

11.8.2 Routine Determination

The performance objective for the routine determination of excepted and IP-1 packaging for onsite transfers is that the packaging is appropriate for the radioactive, chemical, thermal, and physical characteristics of the contents.

The criteria that ensure meeting the routine determination of excepted and IP-1 packaging for onsite transfers include the following:

- Evidence (e.g., SNM inventory control records, laboratory test results, historical data including uncertainty limits, or expert knowledge of the custodian) shows the contents is a limited quantity.

- Evidence shows the amount shipped does not exceed the weights previously transferred or shipped in similar packaging.

- Evidence shows the contents is the same chemical and physical form previously transferred or shipped in this packaging.

11.9 Empty Radioactive Packagings

The performance objective in the packaging QA programs for transferring empty onsite packagings is that the empty packaging will pose a minimal radiation hazard.

11.9.1 Routine Determination

The criteria that ensure meeting the routine determination for empty onsite packagings include the following:

- Evidence is available that shows the radiation, external contamination, and internal contamination were ALARA and within applicable contamination control limits in the site specific radiation control manual.

- Evidence is available that shows the residual uranium is less than that allowed by 49 CFR 173.431
- Evidence is available (e.g., check sheets or operating procedures) that shows the packaging was unimpaired, securely closed, previous labels were removed, and information that the packaging was "empty" was provided to the transport workers.

12.0 RADIATION PROTECTION CRITERIA

All packagings for onsite transfers of radioactive material must effectively perform the following:

- reduce radiation exposure to ALARA levels
- reduce radiation exposure to levels that are below established limits
- retain radioactive material to meet environmental release limits.

12.1 Authorized Offsite Type B Packages

The radiation protection criteria for an authorized offsite Type B packaging, when used for onsite transfer, must comply with the following:

- 49 CFR 173.441, "Radiation Level Limits"
- 49 CFR 173.443, "Contamination Control"
- 10 CFR 71.51, "Additional Requirements for Type B Packages".

12.2 Authorized Offsite Type A, IP-2, and IP-3 Packages

The radiation protection criteria for an authorized offsite Type A, IP-2, or IP-3 packaging, when used for onsite transfer, must comply with the following:

- 49 CFR 173.441, "Radiation Level Limits"
- 49 CFR 173.443, "Contamination Control"
- 49 CFR 173.412, "Additional Requirements for Type A Packages"
- IAEA Safety Series 6, paragraphs 519 and 520, "Additional Requirements for Industrial Package Type 2 (IP-2). . .(IP-3)," 1985 ed.

12.3 Authorized Offsite Excepted and IP-1 Packages

The radiation protection criterion for excepted and IP-1 packagings, when used for onsite transfer, must comply with the contents, radiation, and contamination requirements of the following:

- 49 CFR 173.421, "Limited Quantity of Radioactive Material"
- 49 CFR 173.443, "Contamination Control"

- IAEA Safety Series 6, paragraph 518, "Requirements for Industrial Package Type 1 (IP-1)," 1985 ed.

12.4 Alternate Packages, Radiation Protection Criteria

Alternate packaging, including the controls, limits, and requirements identified in the safety assessment, shall control radiation exposure to be ALARA and below established limits. The predicted radiation exposure shall be below the limits described in the following subsections.

12.4.1 Environment Limits

Each alternate package must provide protection such that individual members of the public will not exceed effective dose equivalent radiation exposures of the following amounts:

- 100 mrem/yr from both external and internal radiation
- 10 mrem/yr from airborne radiation
- 4 mrem/yr from contaminated drinking water.

12.4.2 Contamination Limits

All alternate onsite packagings shall meet the same contamination limits as offsite shipments specified in 49 CFR 173.443.

12.4.3 Personnel Exposure Limits

Empty alternate onsite packages and those packages containing any of the following are excepted from the containment and shielding radiation exposure criteria limits.

- limited quantity
- LSA-1
- SCO-1.

Type A packages and packagings containing radioactive material meeting the definition of the following shall meet the radiation exposure criteria as shown in Table 10.1.

- instruments and articles
- manufactured articles

- LSA-2
- LSA-3
- SCO-2.

Table 12.1. Radiation Exposure Criteria for Type A Packages and Other Lower Hazard Packages

Credible Event Category	Limit[a]
Normal	200 mrem/yr goal
Design Basis Anticipated Event	500 mrem/event (maximum)

[a] The combined exposure limit for normal and anticipated design basis events is 500 mrem. The estimated exposure from all normal activities associated with onsite transfers has an established goal limit of 200 mrem. No single anticipated design basis event should result in the combined exposure for normal activities and the event exceeding 500 mrem.

Type B packages shall meet the radiation exposure criteria shown in Table 10.2.

Table 12.2. Radiation Exposure Criteria for Alternate Type B Packages

Credible Event Category	Limit[a]
Normal	200 mrem/yr goal
Design Basis Anticipated Event	500 mrem/event (maximum)
Design Basis Off-Normal Event	2000 mrem/event
Design Basis Emergency Event	5000 mrem/event

[a] The combined exposure limit for normal and anticipated design basis events is 500 mrem. The estimated exposure from all normal activities associated with onsite transfers has an established goal limit of 200 mrem. No single anticipated design basis event should result in the combined exposure for normal activities and the event exceeding 500 mrem.

12.5 Radiological Work Permit

If the predicted radiation exposure is higher than the limits in tables 10.1 or 10.2 and an ALARA study shows that the collective and maximum individual exposure is ALARA, the material transfers might be permitted by a radiological work permit.

13.0 HAZARDS INFORMATION CRITERIA

Hazards information criteria permit using locally approved procedures for providing hazards information. Locally approved procedures are acceptable if they meet the criteria listed in the sections following, and approval is obtained from the managers of all affected WSRC departments. The affected departments include those departments responsible for the following activities:

- originating the transfer
- receiving the transferred material
- transporting the material
- emergency response during the transfer
- radiation protection during the transfer
- any other activity required as part of the transfer.

13.1 Vehicle Marking Criteria

Any vehicle transferring more than an A_2 quantity of radioactive material must comply with one of the following:

- The transfer vehicle shall be placarded in accordance with 49 CFR 172.500 through 49 CFR 172.560.
- Locally approved procedures providing equivalent information shall be used.

The information is equivalent to that required in the DOT regulations if the local approved procedures have accomplished one of the following:

- Provided information to trained personnel that the transfer vehicle is transporting radioactive material
- Permitted a trained individual to acquire the knowledge that the vehicle is carrying radioactive material while that individual is still 100 feet from the vehicle.

13.2 Package Marking Criteria

Any packaging for greater than an A_2 quantity of radioactive material shall be conspicuously and durably marked with a model number, a unique identification number, and the weight of the empty packaging.[1]

Any packaging for unirradiated fissile material used for storage of the material prior to onsite transfer shall be marked or coded to indicate the type or category of material, amount, degree of enrichment, and the radiation level at the outside surface of the vessel.[8]

Each package containing radioactive material shall be in compliance with the regulations in 49 CFR 172.200 through 49 CFR 172.450 for package marking and labeling, or locally approved procedures shall provide equivalent information.

In order for the procedures to provide equivalent information, those procedures must provide the following listed information to properly trained individuals or permit a properly trained individual who is within 10 ft of the radioactive materials being transferred to acquire the following information about each radioactive material package on that vehicle:

- The package contains radioactive material.
- The approximate radiation level is marked on the package surface (i.e., whether the radiation is less than 0.5, between 0.5 and 50, or greater than 50 mRem/hr).
- Whether the package contains limited quantity, instruments and articles, LSA, special form, less than or more than an A_2 quantity of material, or if the packaging is empty
- The technical name of the package contents (i.e., Uranium)
- The gross weight of package
- The package type (i.e., DOT 7A)
- The local origin or destination of the transfer.

13.3 Shipping Paper Criteria

The shipping paper criteria are that each onsite transfer of radioactive material shall be accompanied by paper work that complies with the regulations in 49 CFR 172.200 through 49 CFR 172.205 or equivalent information using locally approved procedures.

In order for the procedures to provide equivalent information, the following listed information must be provided to properly trained individuals or permit a properly trained individual who has access to the inside of the transfer vehicle to acquire the following information about each radioactive material transported on that vehicle:

- The material is radioactive

- Whether the radioactive material is limited quantity, a manufactured article, LSA, fissile, or special form
- Total quantity of material (i.e., 130 lb)
- Name of each radionuclide
- Description of the physical and chemical form of the material
- Activity of the material in the package in curies (Ci)
- Approximate radiation level on the package surface (i.e., whether the radiation is less than 0.5, between 0.5 and 50, or greater than 50 mRem/hr).
- Fissile class, if applicable
- Package limit per vehicle, if applicable.

13.4 Emergency Response Information

Each onsite transfer of radioactive material shall comply with one of the following:

- Be accompanied by paper work that complies with the regulations in 49 CFR 172.600 through 49 CFR 172.604
- Equivalent information shall be provided using locally approved procedures.

In order to provide equivalent information, the locally approved procedures shall either provide the following information to properly trained individuals or permit any properly trained individual having access to the inside of the transfer vehicle to acquire the following information about each radioactive material on that vehicle:

- The material is radioactive
- Whether the radioactive material is limited quantity, a manufactured article, LSA, fissile, or special form
- Total quantity of material (i.e., 130 lb)
- Identification of immediate hazards to health
- Identification of risks of fire or explosion
- Description of immediate precautions to be taken in the event of an accident or incident

- Descriptions of immediate methods for handling fires
- Description of initial methods for handling spills or leaks in the absence of fires
- Description of preliminary first aid measures
- Emergency contact phone number.

REFERENCES

1. W.R. Rhyne. *Accidents for Evaluating Onsite Transport Packages*. H&R Technical Associates, Inc. Report H&R 445-14, Oak Ridge, TN, 37831-4159 (May 1994).

2. *Safety of Nuclear Facilities*. USDOE order 5480.5, U.S. Department of Energy, Washington, D.C. (September 23, 1986).

3. 10 CFR 50.

4. See ref. 3.

5. *Safety of Nuclear Facilities*. USDOE Order 5480.5, 13.b.(13), U.S. Department of Energy, Washington, D.C. (September 23, 1986).

6. See ref. 5.

7. See ref. 3.

8. *Safety Requirements for the Packaging and Transportation of Hazardous Materials, Hazardous Substances, and Hazardous Wastes*. USDOE Order 5480.3, U.S. Department of Energy, Washington, D.C. (July 9, 1985).

9. *Radiation Protection for Occupational Workers*. USDOE Order 5480.11, U.S. Department of Energy, Washington, D.C.

10. *Packaging Review Guide for Reviewing Safety Analysis Reports for Packages*. USDOE Report UCID-21218, Lawrence Livermore National Laboratory, Livermore, CA, 94550 (October 1988).

GLOSSARY

For purposes of onsite transfer of radioactive materials, the following definitions are established.

ALARA is a process to ensure that personnel radiation exposures are as low as reasonably achievable.

An **ALARA study** is an assessment of the expected radiation exposure during both transfer activities and transfer-related operational activities.

An **alternate package** is any package that is not an authorized offsite package. Alternate packages are authorized for use by approval of a written safety assessment. An alternate package shall be used in accordance with all controls and limits identified in its safety assessment.

An **authorized offsite package** is packaging authorized by, and in total compliance with DOE Order 5480.3.

A **credible event** is any event that has a reasonable probability of occurring. The credible event categories considered in a safety assessment for an onsite package are defined here.

Event Type	Probability Range
Normal	1/transfer
Anticipated	10^{-1}/yr $\leq$ probability $<$ 1/transfer
Off-Normal	10^{-4}/yr $\leq$ probability $<$ 10^{-1}/yr
Emergency	10^{-6}/yr $\leq$ probability $<$ 10^{-4}/yr

Events with a probability of less than 10^{-6}/yr are defined as incredible and are not considered in the safety assessment for onsite transfers of radioactive material.

Criticality safety is the system of conservative assumptions, calculations, moderators, neutron absorbers, spacers, package material limits package labels, and other controls that ensures maintaining a subcritical fissionable mass during use, handling, and transfer of fissile material.

The **Design Basis** is set of environmental and loading conditions that the packaging might reasonably experience during its intended use. If the package will be used for storage, the storage conditions are part of the design basis. The design basis includes the following:

- Amount and isotopic composition ranges of package contents

- Operational restraints or special requirements including maximum and minimum temperatures, maximum pressures, and maximum structural loads experienced during normal, anticipated, off-normal, and emergency events, and during any use of the package for material storage

- Design criteria for structural, thermal, containment, criticality, and shielding analysis and the fabrication acceptance.

The design requirements for use of the package in storage are included in the overall design basis but are not considered in the safety assessment of the package for onsite transfer.

Design Basis Events are the credible scenarios upon which the package analysis is based. Each design basis event shall be the combination of the worst-case credible event for that event category and the worse-case combination of radionuclides for the failure mode of interest.

An **empty** package is one that has previously contained radioactive material and has been emptied of contents, insofar as practical, and has surface radiation below 0.5 mrem/hr, meets the exterior surface contamination limits established by DOT, and contains less than 15 g of residual uranium.

Equivalent protection is a level of protection including radiation protection, criticality safety, and hazards information provided during onsite transfers of radioactive material equivalent to that of offsite shipments. To be equivalent, the radiation protection must comply with the recommendations of the National Council on Radiation Protection and Measurements. The limits and requirements in offsite regulations are derived to provide this level of radiation protection. In addition, the criticality safety and hazards information provided must be equivalent to that provided for offsite shipments.

Excepted quantities and articles are small quantities of radioactive materials that meet the limits in 49 CFR 173.423.

Hazards information is the information provided to workers, emergency responders, and other persons about the hazards of material being transported. The information is provided by means of placards, package markings, package labels, shipping papers, emergency response information, or other means.

An **intrafacility movement** is movement of hazardous material within a facility. (Criteria for these movements is not within the scope of this document.)

Instruments and articles are instruments and manufactured articles containing small amounts of radioactive material. Each instrument or article shall contain no more than 10^{-2} A_2 of solid radioactive material or no more than 10^{-3} A_2 of gaseous radioactive material.

Limited quantity is radioactive material that does not exceed the following amount per package:

- 10^{-3} A_1 for special-form material
- 10^{-3} A_2 for non-special-form material
- 1000 Ci of tritiated water if the specific activity is less than 0.2 Ci/L
- 100 Ci of tritiated water if the specific activity is from 0.1 Ci/L to 1 Ci/L
- 1 Ci of tritiated water if the specific activity is equal to or greater than 1 Ci/L
- 10^{-4} A_2 for liquids other than tritiated water
- 20 Ci of tritium gas
- 10^{-3} A_2 for gas in special form
- 10^{-3} A_2 for gas in non-special form.

LSA is radioactive material that by its nature has a limited specific activity. External shielding materials surrounding the LSA material shall not be considered in determining the estimated average specific activity. LSA material shall be in one of three groups:

- LSA-1 is any one of the following materials:
    - Ores containing naturally occurring radionuclides (e.g., uranium, thorium), and uranium or thorium concentrates of such ores.
    - Solid unirradiated natural uranium or depleted uranium or natural thorium or their solid or liquid compounds or mixtures.
    - Radioactive material, other than fissile material, for which the A_2 value is unlimited.
- LSA-II is any one of the following materials:
    - Water with tritium concentration no greater than 20 Ci/L.
    - Other material in which the activity is distributed throughout and the estimated average specific activity does not exceed 10^{-4} A_2/g for solids and gases and 10^{-5} A_2/g for liquids.

- LSA-III is a solid (e.g., consolidated wastes, activated materials) for which each of the following is true:

  - 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.).

  - The radioactive material is relatively insoluble, or it is intrinsically contained in a relatively soluble matrix, 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.

  - The estimated average specific activity of the solid, excluding any shielding material does not exceed 2×10^{-3} A_2/g.

A **manufactured article** is an article in which the sole radioactive material content is natural or depleted uranium or natural thorium and the outer surface of the uranium or thorium is enclosed in an active sheath made of metal or other durable protective material.

Members of the public are persons who have no occupationally related reason for being onsite or who have not received appropriate radiation protection and training.

Normal events include all events that are expected to occur.

An **offsite shipment** of radioactive material is transported on a motorized vehicle to or from a location outside the boundary of SRS but is accessible to the public. The portion of an offsite shipment that might occur on roads within the boundary of SRS but not accessible to the public is part of the offsite shipment provided the offsite carrier continues the transport as part of the offsite shipment.

An **onsite transfer** of radioactive material is transport of radioactive material on a motorized vehicle over routes from which **members of the public** have been excluded. Intrafacility movements are not considered onsite transfers.

An **over-pack** is the container used to provide protection during transport, to add convenience in handling a package, or to consolidate two or more packages.

Periodic determinations are the activities performed at scheduled times during the useful life of the package to meet the specified maintenance and performance criteria.

Preliminary determinations are the activities performed prior to first use of a package to meet the specified acceptance criteria.

Radiation exposure is exposure to ionizing radiation from internal or external sources. External radiation exposure comes from radiation emanating from the package, primarily gammas and neutrons, and from material released from the package. Internal radiation exposure may occur from inhalation or ingestion of radiation material released from the package. Leaking packages contribute to airborne radioactive material as a source for inhalation. Surface contamination of any package with nonfixed radioactive material is a source for an external dose to the skin or an internal dose from ingestion of the material. Immersion in an airborne cloud of some radioactive material may cause external radiation exposure or lead to internal radiation exposure by absorption through the skin in addition to inhalation.

Radiation exposure to transfer workers is exposure during transfer activities. (See also *transfer activities* and *transfer-related operational activities*.)

A **radiation protection** program is the system of requirements, controls, measurements, and limits that accomplishes the following:

- Ensures that activities are conducted only if the benefit is greater than the health effects from the radiation exposure
- Manages activities associated with radioactive materials to ensure the exposure is ALARA
- Predicts anticipated exposure and ensures that it is below established limits
- Measures and records personnel radiation exposure to confirm predictions
- Conducts medical exams, tests, and bioassays to confirm dosimeter and other measurements.

Radioactive material is material meeting the definition in 49 CFR 173.403 or material otherwise defined as radioactive by a cognizant federal agency.

Routine determinations are the activities performed prior to each onsite transfer of a package to meet the specified transfer criteria.

A **safety assessment** is a written assessment of the activities, hardware, and equipment involved in an onsite transfer to determine if the radiation protection, criticality safety, and hazards information provided meets the criteria established in this document. The criteria were derived to ensure that the protection provided is equivalent to that provided during offsite shipments. Assessment of radiation protection requires containment and shielding assessments. These assessments in turn require thermal and structural assessments of normal and accident conditions. The safety assessment identifies the authorized contents of the onsite package. A safety assessment document establishes, collects, and enumerates requirements for procedures such as loading and closing the packaging, inspections and tests, lifting and hoisting activities, and tie downs. The safety assessment document also establishes and enumerates limits such as contents weight or activity per package, isotopic concentrations, decay heat, transfer vehicle speed, and number of packages per vehicle.

SCO is a Surface Contaminated Object, which is a solid object that is not itself radioactive but has radioactive material distributed on its surfaces. SCO shall be in one of two groups:

- SCO-I is a solid object on the surface of which contamination is characterized as follows:
  - 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 10^{-4} $\mu Ci/cm^2$ for beta and gamma emitters or 10^{-5} $\mu Ci/cm^2$ for alpha emitters.
  - The fixed contamination on the accessible surface averaged over 300 cm^2 (or the area of the surface is less than 300 cm^2) does not exceed 1 $\mu Ci/cm^2$ for beta and gamma emitters or 0.1 $\mu Ci/cm^2$ for alpha emitters.
  - The nonfixed contamination plus 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 1 $\mu Ci/cm^2$ for beta and gamma emitters or 0.1 $\mu Ci/cm^2$ for alpha emitters.
- SCO-II is a solid object on the surface of which either the fixed or nonfixed contamination exceeds the applicable limits specified for SCO-I and on the surface of which contamination is characterized as follows:
  - The nonfixed contamination on the accessible surface averaged over 300 cm^2 (or the area of the surface is less than 300 cm^2) does not exceed 10^{-2} $\mu Ci/cm^2$ for beta and gamma emitters or 10^{-3} $\mu Ci/cm^2$ for alpha emitters.
  - The fixed contamination on the accessible surface averaged over 300 cm^2 (or the area of the surface is less than 300 cm^2) does not exceed 20 $\mu Ci/cm^2$ for beta and gamma emitters or 2 $\mu Ci/cm^2$ for alpha emitters.
  - The nonfixed contamination plus 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 20 $\mu Ci/cm^2$ for beta and gamma emitters or 2 $\mu Ci/cm^2$ for alpha emitters.

Special form is radioactive material which has the following characteristics:

- The material is a single solid piece or is contained in a sealed capsule that can be opened only by destroying the capsule.
- The piece or capsule has at least one dimension not less than 0.5 cm and passes through the tests prescribed by DOT for special-form material (49 CFR 173.469).

Subsidiary hazards of the radioactive material are characteristics such as flammability, toxicity, chemical instability, corrosivity, explosivity, or other hazardous characteristics that may exist or develop during transport.

Transfer activities are as follows:

- loading packages onto transfer vehicle
- blocking, bracing, and tie-down activities to secure packages to transfer vehicle
- operating transfer vehicle
- unloading transfer vehicle.

The radiation exposure expected during these activities is considered in determining whether the package meets its radiation exposure goals and limits.

Transfer-related operational activities are defined as follows for the purposes of developing the criteria in this document:

- preparing material for transfer
- packaging preparation
- putting material into the packaging
- closing the package
- any package inspection or testing
- temporary storage of filled package awaiting transfer
- long-term storage of filled package
- opening the package
- emptying the package.

The transfer-related operational activities are not included in the determination of radiation exposure chargeable to onsite transfers of radioactive material. The transfer-related operational activities are considered part of the facility operations. Therefore, these transfer-related operational activities and any credible events associated with them are not evaluated in the safety assessment for onsite transfers. However, these transfer-related operational activities are included in the package selection process when performing an ALARA study to assure that the packaging selected for a particular material and process stream does indeed reduce the radiation exposure to ALARA levels.

A **Type A** package is one authorized to carry greater than an A_1 quantity of special-form radioactive material or an A_2 quantity of non-special-form radioactive material.

A **Type B** package is one authorized to carry greater than an A_1 quantity of special-form-radioactive material or greater than an A_2 quantity of non-special-form radioactive material.

Appendix A.

Content Characterization and Package Usage Data Sheets

Contents Characterization and Package Usage Data Sheets — Sheet 1

A. Identification of Organization Originating the Transfer

Data Sheet Identification Number ____________________
Date ________ Material Name ____________________
Cognizant Person ____________________ Phone __________
Cognizant Organization ____________________

B. Materials Description

Package Contents ____________________
Physical Form (check one)Solid ______ Liquid______ Gas______ Other______
Basic material unit ____________ (lb., gal., piece, ft^3, ml., etc.)
If the material is a liquid list its specific gravity ____________
and its vapor pressure at 131°F (55°C)____________
Specific chemical name (technical name)____________
Generic chemical name ____________
Industrial or Commercial name____________
All known synonyms ____________

____

____

Is this waste material ____________

C. Radioactive Characteristics

List major radionuclides, and their design concentrations to consider for:

******Shielding******		***Containment***		***Criticality****	
Nuclide	Percent	Nuclide	Percent	Nuclide	Percent

*****Decay Heat******	
Nuclide	Percent

**Isotopic percent values are preferred. Indicate the basis of percent values, if they are different than isotopic.

Contents Characterization and Package Usage Data Sheets Sheet 2

Activity:
Most probable value: Ci/unit________________, Ci/preferred package________________
Maximum Design value: Ci/unit________________, Ci/preferred package________________
Decay heat:
Most probable value: Kw/unit________________, Kw/preferred package________________
Maximum Design value: Kw/unit________________, Kw/preferred package________________
**Identify the unit, e.g. gram, kg, gallon, lb. etc.
**Identify the preferred package size, (e.g., 55 gallon, rail car, 4 ft. x 4 ft x 8 ft B-25 Metal box, etc.)

D. Subsidiary Hazard Characteristics

Identify all of the following characteristics which the material may possess.
Flammability ____________, Toxicity____________, Chemical Instability________________,
Corrosivity____________, Explosivity____________, or other characteristics
which may make it hazardous during normal transport or accident________________________

Amount of hydrocarbon which may cause gas generation________________________
Provide details about any characteristics checked.

If needed use additional sheets. Identify additional sheets with the Contents Characterization Data Sheet Identification Number at the top of each additional sheet.

Contents Characterization and Package Usage Data Sheets　　　　Sheet 3

E. Chemical Compatibility

List all common packaging component materials which may be chemically incompatible with the material to be shipped.

____

____

____

____

(Provide additional supplemental information, as appropriate. Identify the supplemental information on the top of each sheet with the Data Sheet Identification Number listed on sheet of this form).

F. <u>Transfer Description</u>

List each destination where this material might be sent onsite.
1.________________, 2.________________, 3.______________, 4.____________
List estimated amount per year transferred to each destination.
1.________________, 2.________________, 3.______________, 4.____________
Package Size:
Preferred________________, Min. Practical______________, Max. Practical__________
For each destination list the preferred and any alternate transfer mode, e.g. road, rail.
1.______/______, 2.______/______, 3.______/______, 4.______/______

List any package handling requirements and limitation at the location originating these onsite transfers of radioactive material.
Gross package weight limit ________________
Package size limit, Length____________, Width__________, Height______________

Any unique hoisting requirements________________________________

____

Any unique interface or portal requirements__________________________

____

Any other unique requirements or limits to be considered in package design or selection_______

____

G. Describe any external mechanical loads imposed on the packaging during filling, closing opening, storage, empty, and cleaning or decontamination. Also list any thermal loads that are imposed during these activities____________________________

____

____

____

____

Contents Characterization and Package Usage Data Sheets — Sheet 4

H. List any package handling requirements and limitation at each destination.
Destination Number 1.
Gross package weight limit ________________
Package size limit, Length ________, Width ________, Height ________
Any unique hoisting requirements ________________

____

Any unique interface or portal requirements ________________

____

Any other unique requirements or limits to be considered in package design or selection ________

____

Destination Number 2
Gross package weight limit ________________
Package size limit, Length ________, Width ________, Height ________
Any unique hoisting requirements ________________

____

Any unique interface or portal requirements ________________

____

Any other unique requirements or limits to be considered in package design or selection ________

____

Destination Number 3
Gross package weight limit ________________
Package size limit, Length ________, Width ________, Height ________
Any unique hoisting requirements ________________

____

Any unique interface or portal requirements ________________

____

Any other unique requirements or limits to be considered in package design or selection ________

____

Contents Characterization and Package Usage Data Sheets Sheet 5

List conditions at any location where the package will be used to store its contents for more than 90 days.

Storage Location Number 1.
Storage location____________________ Max duration, days____________________
Ambient Conditions:
Temperature Range, F____________________, Estimated Average Temp. F__________
Description of relative humidity surrounding package during storage____________________
Mechanical Conditions:
How high are packages stacked during storage____________________
Describe any other external mechanical load on package during storage____________________

Storage Location Number 2.
Storage location____________________ Max duration, days____________________
Ambient Conditions:
Temperature Range, F____________________, Estimated Average Temp. F__________
Description of relative humidity surrounding package during storage____________________
Mechanical Conditions:
How high are packages stacked during storage____________________
Describe any other external mechanical load on package during storage____________________

Storage Location Number 3.
Storage location____________________ Max duration, days____________________
Ambient Conditions:
Temperature Range, F____________________, Estimated Average Temp. F__________
Description of relative humidity surrounding package during storage____________________
Mechanical Conditions:
How high are packages stacked during storage____________________
Describe any other external mechanical load on package during storage____________________

Storage Location Number 4.
Storage location____________________ Max duration, days____________________
Ambient Conditions:
Temperature Range, F____________________, Estimated Average Temp. F__________
Description of relative humidity surrounding package during storage____________________
Mechanical Conditions:
How high are packages stacked during storage____________________
Describe any other external mechanical load on package during storage____________________