NNSA Package Certification Engineer

Qualification Standard

DOE-STD-1026-2016

June 2016 | Reference Guide

The Functional Area Qualification Standard References Guides are developed to assist operators, maintenance personnel, and the technical staff in the acquisition of technical competence and qualification within the Technical Qualification Program (TQP).

Please direct your questions or comments related to this document to Learning and Career Management, TQP Manager, NNSA Albuquerque Complex.

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ACRONYMS

AC	administrative control
ADAST	Assistant Deputy Administrator for Secure Transportation
AEA	Atomic Energy Act
AEGL	acute exposure guideline level
AI	anodic index
ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AWS	American Welding Society
BPVC	boiler and pressure vessel code
Bq	becquerel
CAS	contractor assurance system
CEDE	committed effective dose equivalent
Ci	curie
cm	centimeter
CO	certifying official
CoC	certificate of compliance
CRAD	criteria review and approach document
CSI	criticality safety index
CV	containment vessel
DID	defense-in-depth
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
DSA	documented safety analysis
ERPG	emergency response planning guideline
FAE	first article evaluation
FAQS	functional area qualification standard
FN	ferrite number
ft	feet
gs	inertial load factors
h	hour
HA	hazard analysis
HAC	hypothetical accident condition
HAR	hazards analysis report
HAZ	Heat-affected zone
HCO	headquarters certifying official
HMR	hazardous material regulation
HS-40	HSS Office of Enforcement and Oversight
HSS	Office of Health, Safety, and Security
IAEA	International Atomic Energy Agency
in.	inch
ISM	integrated safety management
KSA	knowledge, skill, and ability

lb	pound
lbf	pound force
LDM	low dispersible material
MAWP	maximum allowable working pressure
MC&A	material control and accountability
MNOP	maximum normal operating pressure
MNSI	materials of national security interest
MOD	(United Kingdom) Ministry of Defense
mrem	millirem
MSLD	mass spectrometer leak detector
NCT	normal condition of transport
NNSA	National Nuclear Security Administration
Np	neptunium
NRC	Nuclear Regulatory Commission
NTS	noncompliance tracking system
OEM	Office of Emergency Management
OST	Office of Secure Transportation
OTA	offsite transportation authorization
OTC	offsite transportation certificate
OTD	offsite transportation direction
PAAA	Price Anderson Amendment Act
PCD	Packaging Certification Division
PRA	probabilistic risk assessment
psig	pounds per square inch gauge
PSO	program secretarial officer
Pu	plutonium
QA	quality assurance
QAP	quality assurance program
ref-cm^3/s	reference cubic centimeter per second
SAC	specific administrative control
SARP	safety analysis report for packaging
SCAPA	Subcommittee on Consequence Assessment and Protective Actions
SER	safety evaluation report
SFRP	shipment forecast and request procedure
SG	safety guide
SNM	special nuclear material
SS	safety-significant
SSC	structures, systems, and components
STAAB	Secure Transportation Asset Advisory Board
STD	standard
STPSC	Secure Transportation and Packaging Steering Committee
TBq	terabecquerel
TEEL	temporary emergency exposure limit
TI	transport index
TQP	Technical Qualification Program
TSD	transportation safety document

TSR	technical safety requirement
TSRA	transportation system risk assessment
TSRP	Transportation Safety Review Panel
TSS	Transportation Safeguards System
U	uranium
UK	United Kingdom
U.S.	United States
USL	upper subcritical limit
USQ	unreviewed safety question
USQD	unreviewed safety question determination

PURPOSE

The purpose of this reference guide is to provide a document that contains the information required for a Department of Energy (DOE)/National Nuclear Security Administration (NNSA) technical employee to successfully complete the NNSA Package Certification Engineer Functional Area Qualification Standard (FAQS). Information essential to meeting the qualification requirements is provided; however, some competency statements require extensive knowledge or skill development. Reproducing all the required information for those statements in this document is not practical. In those instances, references are included to guide the candidate to additional resources.

SCOPE

This reference guide has been developed to address the competency statements in the April 2016 edition of DOE-Standard (STD)-1126-2016, *NNSA Package Certification Engineer Functional Area Qualification Standard*. The qualification standard for NNSA Package Certification Engineer contains 20 competency statements.

PREFACE

Competency statements and supporting knowledge and/or skill statements from the qualification standard are shown in contrasting bold type, while the corresponding information associated with each statement is provided below it.

A comprehensive list of acronyms, abbreviations, and symbols is provided at the beginning of this document. It is recommended that the candidate review the list prior to proceeding with the competencies, as the acronyms, abbreviations, and symbols may not be further defined within the text unless special emphasis is required.

The competencies and supporting knowledge, skill, and ability (KSA) statements are taken directly from the FAQS. Most corrections to spelling, punctuation, and grammar have been made without remark. Only significant corrections to errors in the technical content of the discussion text source material are identified. Editorial changes that do not affect the technical content (e.g., grammatical or spelling corrections, and changes to style) appear without remark. When they are needed for clarification, explanations are enclosed in brackets.

Every effort has been made to provide the most current information and references available as of June 2016. However, the candidate is advised to verify the applicability of the information provided. It is recognized that some personnel may oversee facilities that utilize predecessor documents to those identified. In those cases, such documents should be included in local qualification standards via the TQP.

In the cases where information about an FAQS topic in a competency or KSA statement is not available in the newest edition of a standard (consensus or industry), an older version is referenced. These references are noted in the text and in the bibliography.

This reference guide includes streaming videos to help bring the learning experience alive. To activate the video, click on any hyperlink under the video title. Note: Hyperlinks to videos are shown in entirety, due to current limitations of eReaders and must be accessed through Google Chrome.

TECHNICAL COMPETENCIES

1. NNSA package certification engineers must demonstrate a working level knowledge of DOE O 461.1B or latest revision, *Packaging and Transfer or Transportation of Materials of National Security Interest*.

The information for the KSAs in this competency statement is taken from DOE O 461.1B, unless stated otherwise.

a. Describe the purpose of the SARP, the HAR, and the TSRA.

Safety Analysis Report for Packaging (SARP)
A SARP is a document that conforms to the Nuclear Regulatory Commission (NRC) Regulatory Guide 7.9, *Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material*, and provides a comprehensive technical evaluation of a package. The SARP consists of sections containing general information; structural, thermal, containment, shielding, and criticality evaluations; operating procedures; acceptance tests; and maintenance and quality assurance programs (QAPs). The purpose of the SARP is to demonstrate conformity with the applicable sections of 10 CFR 71, "Packaging and Transportation of Radioactive Material," and 49 CFR 171–180.

Hazards Analysis Report (HAR)
The HAR is a document that is submitted to the NNSA certifying official (CO) to support an applicant's request for an authorization for offsite transport of noncompliant packages and/or shipping configurations that contain less than a type B quantity of radioactive material or no radioactive materials, but which contain regulated hazardous materials as defined in 49 CFR 171–180. The HAR identifies the type and quantity of hazardous material, proposed packaging/handling gear, mode of transportation, shipment destinations, tie-down procedures, tests that will be performed on the unit, the post-test status of hazardous components, and procedures to verify that the shipment can be conducted safely (if applicable).

Transportation System Risk Assessment (TSRA)
The TSRA is a document that is submitted to the Assistant Deputy Administrator for Nuclear Safety and Operations to support an applicant's request for an authorization for offsite transport of noncompliant packages and/or shipping configurations that contain a type A(F) or type B quantity of radioactive material and may contain regulated hazardous materials as defined in 49 CFR 171-180.

b. Discuss the significance of the information required in a HAR.

Contractor requests for nonfissile less than type B offsite transportation authorizations (OTAs) for noncompliant packages and/or shipping configurations containing hazardous materials and/or no more than type A quantities of radioactive material must be supported by a HAR.

c. Discuss the information required in a TSRA.

The TSRA records the hazards, the assessment of the hazards, the analysis methods, and analysis and results used to determine the frequency and consequences of events that could pose risks to the workers, public, and/or the environment during the proposed offsite transportation of an uncertified package or a special assembly that contains a type A(F) or type B quantity of radioactive material in a prescribed transportation system and routes.

d. Discuss what function the TSRP serves, how it is organized and staffed, and the basic steps in the review process.

The Transportation Safety Review Panel (TSRP) is a committee chaired by a Federal employee and composed of persons with appropriate expertise that performs technical reviews to verify compliance with DOE O 461.1B and makes recommendations for offsite transportation certification or authorization.

e. Discuss the purposes of the OTA, OTC, OTD and how they differ. Describe the information required in an OTA, OTC and OTD.

Offsite Transportation Authorization (OTA)

The OTA is an NNSA CO approval that details the transportation configuration, authorized contents, regulatory and emergency response hazards, and transportation restrictions. The OTA is issued for nonfissile less than type B quantities of material. An OTC authorizes packages for shipment of radioactive materials within the TSS and for commercial carriers. An OTA may detail required positive measures, ACs, a declared maximum number of shipments per calendar year and a maximum number of units per trailer. An OTA may be issued for a one-time shipment or for a transportation campaign over a period of time not to exceed five years, at which time it must be re-authorized.

Offsite Transportation Certificate (OTC)

The OTC is an NNSA CO prepared document, analogous to an NRC or DOE headquarters CO certificate of compliance (CoC) that describes the compliant package configuration, authorized contents, and transportation restrictions. An OTC authorizes packages for shipment of radioactive materials within the transportation safeguards system (TSS) and for commercial carriers. An OTC can declare essential positive measures, administrative controls (ACs), and a maximum number of specified packages per transporter. Issuance of an OTC is demonstration of compliance with 10 CFR 71. It is issued for either a one-time use or multiple uses up to five years, at which point it must be renewed.

Offsite Transportation Direction (OTD)

The OTD is an NNSA approval that details the transportation configuration, authorized contents, regulatory and emergency response hazards, and transportation restrictions. The OTD is issued for type A(F) or type B quantities of material. An OTD always stipulates specific conditions of operations, including required transporter and shipment configuration. An OTD may detail required positive measures, ACs, a declared maximum number of shipments per calendar year and a maximum number of units per trailer. The OTD must state the national security purpose, the reason(s) why a compliant shipment cannot be made, and the reasons for determining that adequate safety has been achieved. An OTD may be issued for a one-time shipment or for a transportation campaign over a period of time not to exceed five years, at which time it must be re-authorized.

f. Explain the purpose of the SER and discuss the information required in the SER for HARs, SARPs, and TSRAs.

The safety evaluation report (SER) is a document that provides the results of the TSPA's safety evaluation, including its independent review of the HAR, SARP, and/or TSRA.

The SER highlights whether or not the shipment of the packages and/or shipping configurations proposed in the HAR presents an undue risk to the health and safety of the public, workers, and/or the environment.

The NNSA CO must document the TSRA review in a SER.

The SER documents the analysis of a TSRA for consideration in granting an OTD.

g. Discuss what is required for a contractor to be an authorized user of an NNSA type B package.

Contractors must conduct type B and other fissile radioactive materials packaging and transportation activities in accordance with a quality assurance (QA) plan that follows the requirements of 10 CFR 71, Subpart H. The QA plan must be submitted through the contractor's responsible DOE field organization to the NNSA CO for review and approval. The contractor is not authorized to begin packaging or transportation operations until it has been designated an authorized user of that package by the NNSA CO, and the NNSA CO will not designate the contractor an authorized user of the package until the contractor's packaging procedures and QA plan have been approved.

2. NNSA package certification engineers must demonstrate a working level knowledge of DOE O 460.1C, *Packaging and Transportation Safety*.

The following is taken from DOE O 460.1C.

a. Discuss the significance of the offsite safety requirements.

Each entity subject to DOE O 460.1C must perform packaging and transportation activities in accordance with the Department of Transportation (DOT) requirements of the Hazardous Materials Regulations (49 CFR 171–180).

b. Describe the application process for NRC or DOT certified packagings.

For a new NRC or DOT packaging certificate, each entity must file a request for a new certificate with the headquarters certifying official (HCO) or NNSA CO, as appropriate.

When DOE or NNSA is the holder of a packaging certificate issued by the NRC or DOT, each entity must file a request for revisions to or renewal of the existing NRC or DOT certificate with the HCO or NNSA CO, as appropriate.

In all cases the HCO or NNSA CO will review and forward, if appropriate, the request to the NRC or DOT.

c. Describe the application process for other type B or fissile materials certified packagings.

For a new DOE or NNSA type B or fissile material packaging, each entity must submit an application to the HCO or NNSA CO, as appropriate, that includes a SARP and any other supporting documentation to demonstrate that the packaging meets the requirements of 10 CFR 71, Subparts E, F, G, and H, and any other applicable standards for certification prior to use.

d. Discuss the QA requirements of this Order.

Each entity that participates in the design, fabrication, procurement, use, or maintenance of a hazardous materials packaging must

- have a QAP approved and audited by
  - the HCO or NNSA CO, as appropriate, for certified type B and fissile radioactive materials packagings satisfying the requirements of 10 CFR 71, Subpart H, "Quality Assurance," or
  - the head of operations office or field office/site office manager, as appropriate, for all other radioactive and hazardous materials packagings, satisfying the requirements of DOE O 414.1D, *Quality Assurance*;
- report deviations from the applicable requirements in compliance with DOE O 231.1B, *Environment, Safety, and Health Reporting*; and
- additionally, report deviations to the HCO or the NNSA CO within 30 days in which there is
  - any instance in which there is significant reduction in the effectiveness of any approved type B or fissile packaging during use;
  - any discovery of any defects with safety significance in type B or fissile packaging after first use, with details of the means employed to repair the defects and prevent their recurrence; or
  - any instances in which the conditions of approval in the CoC were not observed in making a shipment.

e. Discuss the significance of the onsite safety requirements.

Onsite transfer of hazardous materials, substances, and wastes must be conducted in accordance with one of the following:

- 49 CFR 171–180 and the Federal Motor Carrier Safety Regulations (49 CFR 350–399)
- A transportation safety document (TSD) approved by the head of operations office or field office/site office manager, as appropriate:
  - The TSD must describe the methodology and compliance process to meet equivalent safety for any deviation from 49 CFR 171–180 and 49 CFR 350–399.
  - For onsite transfers subject to 10 CFR 830, "Nuclear Safety Management," the TSD must comply with the safety basis requirements of 10 CFR 830, Appendix A to Subpart B, to identify the conditions, safe boundaries, and hazard controls necessary to protect workers, the public, and the environment from adverse consequences.
  - For multiple-tenant DOE/NNSA sites, safety documents for several contractor organizations may be combined into a single document.

- For onsite transfers not subject to 10 CFR 830, the TSDs must be approved and in effect no later than one year from incorporation of the contract requirements document of DOE O 460.1C into contracts.

f. Discuss the DOT HMR in 49 CFR Parts 171–180.

The contractor must perform offsite packaging and transportation activities in accordance with the DOT requirements of the hazardous materials regulations (HMR) (49 CFR 171–180).

For specific radioactive material packagings for offsite shipments, the following apply:

- Each contractor that offers for transportation or transports radioactive material in a type B or fissile material packaging, as appropriate, certified by the HCO, NNSA CO or the NRC, must meet the conditions specified in the CoC or OTC, as appropriate, for the package issued by the HCO, NNSA CO or NRC and register in writing with the HCO or the NNSA CO prior to use.
- For an import or an export shipment pursuant to 49 CFR 173.471, 173.472 or 173.473, each contractor must use a packaging certified by the U.S. competent authority (DOT) where the DOE or NNSA and the contractor have been registered with the DOT as a user, and the contractor has the required documentation for the use and maintenance of the packaging and makes the shipments in accordance with the terms of the certificate issued by the DOT.
- For a new NRC or DOT packaging certificate, the contractor must file a request for a new certificate with the responsible head of the operations office or the field office/site office manager for processing through the HCO or NNSA CO, as appropriate. When DOE or NNSA is the holder of a packaging certificate issued by the NRC or DOT, the contractor must file a request for revisions to or renewal of existing NRC or DOT certificate with the responsible head of the operations office or the field office/site office manager for processing through the HCO or NNSA CO.
- For a new DOE or NNSA type B or fissile material packaging, each contractor must submit an application to the responsible head of the operations office or the field office/site office manager for processing through the HCO or NNSA CO. This application must include a SARP and any other supporting documentation to demonstrate the packaging meets the requirements of 10 CFR 71, Subparts E, F, G, and H, and any other applicable standards for certification prior to use.

g. Describe the DOT special permit application process in 49 CFR 107 Subpart B.

Any offsite hazardous materials packaging or shipment that is regulated by DOT and is not prepared in accordance with the HMR must be prepared in accordance with a valid DOT special permit.

DOE applications for a DOT special permit must be submitted to the HCO to review, process, and forward to DOT. NNSA applications for a DOT special permit must be submitted to the NNSA CO to review, process, and forward to DOT. Applications must be prepared in accordance with the procedures in 49 CFR 107.105, "Application for Special Permit."

3. **NNSA package certification engineers must demonstrate a working level knowledge of DOE O 461.2, *Onsite Packaging and Transfer of Materials of National Security Interest* and DOE M 441.1-1, *Nuclear Material Packaging Manual.***

   a. **Discuss the purpose and scope of DOE O 461.2. Describe the following:**
      - **Packaging and transfer procedures**
      - **Transfer authorization**
      - **Review and approval process**
      - **Quality assurance**
      - **Transfer operations**
      - **Scheduling TSS transfers**

The following is taken from DOE O 461.2.

Packaging and Transfer Procedures

Each site must maintain a set of packaging and transfer procedures, approved by the appropriate authority. The responsible field organization managers must provide the oversight to ensure the implementation of written procedures and compliance with the packaging and transfer authorization basis.

NNSA type A(F) or type B packages issued an OTC require packaging procedures to be approved by the NNSA Service Center.

Non-NNSA approved type A(F) or type B packaging procedures shall be approved by the appropriate authority.

Transfer procedures must be approved by the contractor appropriate authority.

Non-type A or B packaging (e.g., IP-1, IP-II) procedures must be approved by the contractor appropriate authority.

Transfer Authorization

Compliant transfers of materials of national security interest (MNSI) do not require additional authorization. Noncompliant transfers must be authorized through a DOE-issued TSD SER.

Compliant transfers of radioactive material—Compliant transfers of radioactive MNSI do not require special authorization:

- Compliant packages must be used. Type A(F) or type B quantities of material must be packaged in a DOE, NNSA, or NRC certified package in accordance with the CoC or OTC.
- Transfer motor vehicles must follow DOT regulations.
- Personnel performing transfer operations must follow DOT regulations.

Noncompliant transfers of radioactive material—Noncompliant transfers for all MNSI must be authorized by an approved safety basis:

- The TSD establishes the approved safety envelope for packaging and transfer operations for MNSI. The TSD, when approved by a SER, satisfies the requirements of Attachment 3 of DOE O 461.2 and/or 10 CFR 830, Appendix A to Subpart B, Section F, "Documented Safety Analysis." The SER presents the results of the DOE review team and provides the framework for approval of the TSD giving the contractor authority to transfer these materials.

- Noncompliant transfers that were evaluated as part of DOE-approved safety basis per 10 CFR 830, Appendix A to Subpart B, Section F, do not require a TSD.

Review and Approval Process

A DOE team reviews the TSD and makes recommendations to the responsible field organization manager for approving the application/documentation for noncompliant transfers.

Quality Assurance

Packaging and transfer activities for MNSI must comply with DOE O 414.1D. Each site must maintain a compliant QAP approved by the appropriate field organization manager, for the packaging and transfer of MNSI. Compliant transfers of type A(F) or type B quantities of radioactive and fissile material must be conducted in accordance with a QA plan that meets the requirements of 10 CFR 71, Subpart H.

Transfer Operations

Each compliant transfer must be prepared and transported in accordance with the applicable hazardous materials regulations 49 CFR 171–180.

All transfer activities performed under the TSS must be conducted according to 10 CFR 830.

Government or contractor vehicles must be operated in compliance with the applicable Federal motor carrier safety regulations 49 CFR 350–399.

All noncompliant transfers must be performed under the purview of an approved TSD.

Transfer of nuclear explosives must meet the requirements of DOE O 452.1E, *Nuclear Explosive and Weapon Surety Program*, and DOE O 452.2E, *Nuclear Explosive Safety*.

Scheduling TSS Transfers

If TSS resources are required for onsite transfers, secure transportation shipping requirement forecasts must be developed for the Assistant Deputy Administrator for Secure Transportation (ADAST). Refer to DOE O 461.1C for the requirements and process to request and schedule TSS resources.

b. Discuss the purpose and scope of DOE M 441.1-1. Describe the following:

- **Overview and responsibilities**
- **Scope of materials**
- **Nuclear material packaging requirements**

The following is taken from DOE M 441.1-1.

Overview and Responsibilities

DOE M 441.1-1, *Nuclear Material Packaging Manual*, provides detailed packaging requirements for protecting workers from internal exposure to nuclear materials stored outside of an approved engineered contamination barrier. The nuclear materials of concern in DOE M 441.1-1 are those whose composition and quantity create the potential for an airborne contamination hazard that could result in a facility worker receiving an internal radiation dose in excess of five rem committed effective dose equivalent (CEDE). DOE M 441.1-1 does not address the shielding of nuclear materials that generate significant external direct radiation fields because those situations are addressed in other directives and rules.

FIELD ELEMENT MANAGERS' RESPONSIBILITIES

Review and approve the following contractor products:

- Technical basis for nuclear material packaging systems including the evaluation of the chemical, radiological, and physical characteristics of the stored nuclear material and the nuclear material storage packaging designs
- Nuclear materials packaging surveillance programs
- Process for documenting its nuclear materials storage program

Ensure oversight of contractor's implementation of the requirements of DOE M 441.1-1.

Ensure a process exists for proposing, reviewing, approving, and implementing site corrective actions when necessary to ensure the requirements of DOE M 441.1-1 are met and to address conditions that are not protective of the public, workers, or the environment.

Establish a risk-based prioritized schedule for implementing the nuclear material packaging requirements for material stored prior to the issuance of DOE M 441.1-1.

Maintain records according to National Archives and Records Administration-approved DOE record schedules.

Scope of Materials

Field element managers must ensure that nuclear materials that are stored outside engineered contamination barriers are packaged in accordance with the requirements described in Chapter III of DOE M 441.1-1. The nuclear materials of concern are those whose composition and quantity create the potential for an airborne contamination hazard that could result in a facility worker receiving an internal radiation dose in excess of five rem CEDE. Nuclear materials of this type with a total quantity in a storage package exceeding the A2 thresholds established in 49 CFR 173.435, "Table of A1 and A2 Values for Radionuclides," are subject to the specific container requirements of DOE M 441.1-1. A detailed description of the methodology for determining the type and quantities of materials is contained in Attachment 3 of DOE M 441.1-1.

Nuclear Material Packaging Requirements

STORED MATERIAL CHARACTERISTICS

Contractors must ensure the chemical, radiological, and physical characteristics of the stored material are evaluated for the lifetime of the storage of the material and are appropriate for the material package including the following:

- Explosion sensitive and/or flammable materials must be evaluated to determine if safe storage can be achieved or stabilization is necessary.
- Gas generation rates and gas composition must be evaluated and measures must be taken to minimize the formation or accumulation of gases inside the storage package with particular attention paid to limiting flammable gases. Minimization measures may

include placing limitations on contents of containers, stabilizing materials, or venting and filtering containers.

- Incompatible materials whose interaction could lead to failure of a containment barrier must not be packaged together.
- Physical and chemical form of material must be evaluated, including its corrosivity and potential for oxidative expansion to ensure proper packaging.
- Moisture content must be evaluated to determine if safe storage can be achieved or if stabilization is necessary. Stabilization must be considered for materials that can absorb/adsorb significant quantities of moisture when in contact with air. Moisture content amount must not violate the requirements by interacting with materials being stored to induce corrosion or to increase the potential for flammable mixtures or overpressure.
- Pyrophoricity must be evaluated to determine if safe storage can be achieved or if stabilization is necessary.
- Radioactive decay heat must be evaluated to ensure no unacceptable thermal degradation occurs. Package designers must include a maximum heat load in the design specifications.
- Radiation fields must be evaluated to ensure they will not cause failure of a containment barrier or other containers included in the storage package. Plastic should not be in direct contact with alpha-emitting materials. Package designers must include radiation fields resulting from radioactive decay of the stored material including daughter products that might have worse radiation fields than the originally stored material. Consideration should also be given to the radiation fields of contiguous containers in a storage array.
- Solution composition and its effect on the selection of packaging type, storage duration, and surveillance must be evaluated. Special consideration must be given to gas generation and venting during storage life.

PACKAGING DESIGN CRITERIA AND PERFORMANCE OBJECTIVES

Contactors must ensure concerns with corrosion, radiolytic and thermal degradation, oxidative expansion, pressurization, incompatible materials, and usage (handling) that may result in container failure are addressed as part of the package design. Contactors must ensure nuclear material storage packages meet the following design criteria and performance objectives:

- Corrosion resistance—The container, including filter vent and seals, must be constructed entirely of materials that are resistant to corrosion and chemical degradation from the materials being stored and generated gases as well as the ambient storage environment.
- Thermal degradation resistance—The container, including filter vent and seals, must be designed so that heat generation of the contents does not challenge the integrity of the container for the design life of the container.
- Radiation resistance—The container, including filter vent and seals, must be designed so that radiation from stored material and the storage environment does not challenge the integrity of the container for the container’s design life.
- Oxidative expansion accommodation—The container must be designed to accommodate the maximum volume increase and potential effects on the container from the oxidative expansion of stored material that may occur for the container’s design life.
- Pressurization—Containers must be designed to withstand pressure due to gas generation, material oxidation, volume expansion, or release of volatiles during the design storage period. Measures should be taken to minimize the formation or accumulation of gases

inside the storage package. Such measures include placing limitations on the contents of containers and permitted length of storage time, or providing mechanisms in the storage package design, such as venting.

- Incompatible and pyrophoric materials—Incompatible materials must not be stored together. Furthermore, pyrophoric materials must be containerized to prevent interaction with oxidative environments using techniques such as inert environments, double hermetic seals, encapsulation, etc. to prevent a pyrophoric reaction. Sites are encouraged to stabilize pyrophoric materials prior to storage.
- Filter performance—Filters must be capable of venting the maximum credible gas flow rate. Filters must be compatible with the material being stored, including any product of reaction between the environment and the material.
- Design life—The design life of the materials used in the container assembly as fabricated must be calculated or tested and the container must be examined under a regular documented maintenance and surveillance program. The impact of organic or other materials, such as silicone gaskets, that are subject to degradation must be accounted for in these calculations and in maintenance and surveillance plans. The container must meet design objectives such as the design release rate and design qualification release rate for the design life of the container. Container designs incorporating filters require useful-life determinations and performance verification measurements at the rated filtration efficiency.
- Container closure—Storage containers must be securely closed in a way that precludes accidental opening or breaching during normal operational conditions and/or postulated drops.
- Design release rate performance objective—The package must have a design release rate that will maintain the potential internal exposure to workers as low as reasonably achievable (ALARA) during normal storage or handling of the package.
- Design qualification release rate performance objective—The package must have a post-drop design qualification release rate that will prevent the exposure of workers to greater than five rem CEDE. The drop test must be from the maximum working or storage height but not less than four feet.
- Nondestructive examination—The package configuration should allow for nondestructive contents verification, inspection, and surveillance such as by radiography and weighing.
- Quality assurance—Packages must be designed, tested, and procured in accordance with the QA requirements of 10 CFR 830 and DOE O 414.1D.
- Labeling—Each storage container or package of radioactive material must have a unique permanent identification marking or bear a durable, clearly visible label for identification and documentation purposes. Labeling must meet the requirements of 10 CFR 835.605, “Labeling Items and Containers.” Where practical and appropriate, combine radiological, criticality safety, and material control and accountability (MC&A) data into a single label.

PACKAGING SURVEILLANCE PROGRAM

Field elements must ensure a surveillance program is established and implemented to ensure the nuclear material storage package continues to meet its design criteria:

- General—The surveillance program must evaluate appropriate attributes of stored packages to determine whether the packages can continue to be stored safely. Surveillance must also validate the package design life.

- ALARA—The surveillance program must be structured to minimize the overall risk to the facility workers and maintain radiation doses ALARA during the performance of surveillance activities. At a minimum, the following concepts should be considered:
  - Coordinating surveillance activities with MC&A measurements and inspections
  - Coordinating surveillance activities with package movements
  - Using MC&A and routine radiological survey data in the surveillance program
  - Performing surveillance on packages as they are being opened for material use
  - Conducting remote surveillance (e.g., camera, load cell, etc.) for items generating high radiation fields

- Objectives and techniques—The objective of the surveillance program is to identify indications of package degradation early to remediate the degraded package and to identify similar packages and materials that may need remediation. Surveillance techniques must be specified to provide early indications of package degradation which includes seal failure or loss of venting capability (if present). Any signs of degradation should be used for comparison with similar packages and containers. At a minimum, the following techniques must be considered:
  - Visual inspection of the container for indications of corrosion or pressurization, including examining the container for signs of degradation prior to routine handling of the container or opening for material use
  - Weight measurement for indications of mass change of the storage package that could indicate loss of package seal resulting in
    - ✓ possible oxidation and expansion of metal contents
    - ✓ absorption of moisture which could increase corrosion of the package
  - Contamination surveys which could indicate loss of container seal
  - Testing of vent filters for plugging
  - Radiography that could indicate pressurization or degradation of inner containers
  - Opening package for examination of interior contents and seals

- Frequency—Surveillance frequencies must be specified for each package design and content combination. At a minimum, the frequency should be based on the potential failure mechanism, failure consequence, and package design life. Statistical sampling of packages for surveillance may be used for large, uniform populations of packages and/or contents.
- Procedures—Surveillance procedures must specify the surveillance techniques to be used and must have defined acceptance criteria. Procedures must be established to address initial response to packages that fail the acceptance criteria.
- Evaluation—Surveillance data must be evaluated at least annually. Results of the surveillance program must be used to improve packaging design and modify the surveillance program, when appropriate. Sites should consider sharing data with other sites.

DOCUMENTATION

Field elements must ensure documentation of the nuclear material storage package design and surveillance is maintained:

- Records retention—Packaging and surveillance records must be maintained in accordance with DOE O 243.1B, *Records Management Program*.

- Content elements:
  - Material Information—Records must include, at a minimum, available information on the following material characteristics:
    - ✓ Description of the chemical and physical form
    - ✓ Best available isotopic content
    - ✓ Weight
  - Package information—At a minimum, the records must include identification of the following package characteristics:
    - ✓ Date of packaging for each container.
    - ✓ Baseline package gross weight.
    - ✓ Unique identification number associated with each package.
    - ✓ Configuration—quantity and type of containers in a package. This is not required for packages that existed before the manual was issued; however, it is a recommended good practice.
  - Surveillance information—At a minimum, the records must include the following surveillance and inspection data:
    - ✓ Unique identification number associated with each package
    - ✓ Surveillance and radioactive survey results and dates
    - ✓ Dates, location, and results of inspections
    - ✓ Name and site qualifications of individuals performing inspections
  - Technical basis information—At a minimum, records must include information pertaining to the technical basis for packaging and storing applicable nuclear materials, including a description of how the applicable stored material characteristics and packaging design criteria are satisfied, supported by test results and other evidence, as appropriate; threshold calculations for the packaged material; and surveillance and monitoring information.
- Database:
  - A database of relevant information about stored materials and packages including material information, packaging information, and surveillance information for each storage package must be maintained. An electronic database should be used to simplify scheduling of surveillance and repackaging of packages and evaluation of surveillance data. The database may consist of several files, which in themselves may be databases, some of which may be classified. For completeness, the database should be coordinated and generally compatible with the MC&A databases.

4. NNSA package certification engineers must demonstrate a working level knowledge of the information required in chapter 1 of a SARP.

a. Define and discuss the following items related to a packaging:

- **Containment features/boundaries**
- **Neutron and gamma shielding features**
- **Lifting and tie-down devices, impact limiters, and other structural features**
- **Heat transfer features**
- **Packaging tags, markings, and labels**

Containment Features/Boundaries

The following is taken from NNSA Safety Guide (SG)-100, Chapter 4.

The design of an NNSA package normally starts with the containment system. The containment system is defined as an assemblage of all the components required to retain the contents. In general, this includes the containment vessel (CV), seals, leak test and vent/fill port components, and closure bolts. The containment boundary is an assemblage of all the components required to retain the contents and is in direct communication with the internal cavity of the CV. Another way to interpret this is that any component whose failure could directly result in leakage of the contents is part of the containment boundary. Examples of containment boundary components are the CV, CV primary seal, and vent/fill port seal and plug.

The exact containment boundary should be defined, including the CV, welds, drain or fill ports, valves, seals, test ports, pressure relief devices, lids, cover plates, and other closure devices. Penetrations such as fill ports, valves, and test ports may be needed for operating purposes, such as backfilling for leak testing or for heat transfer enhancement.

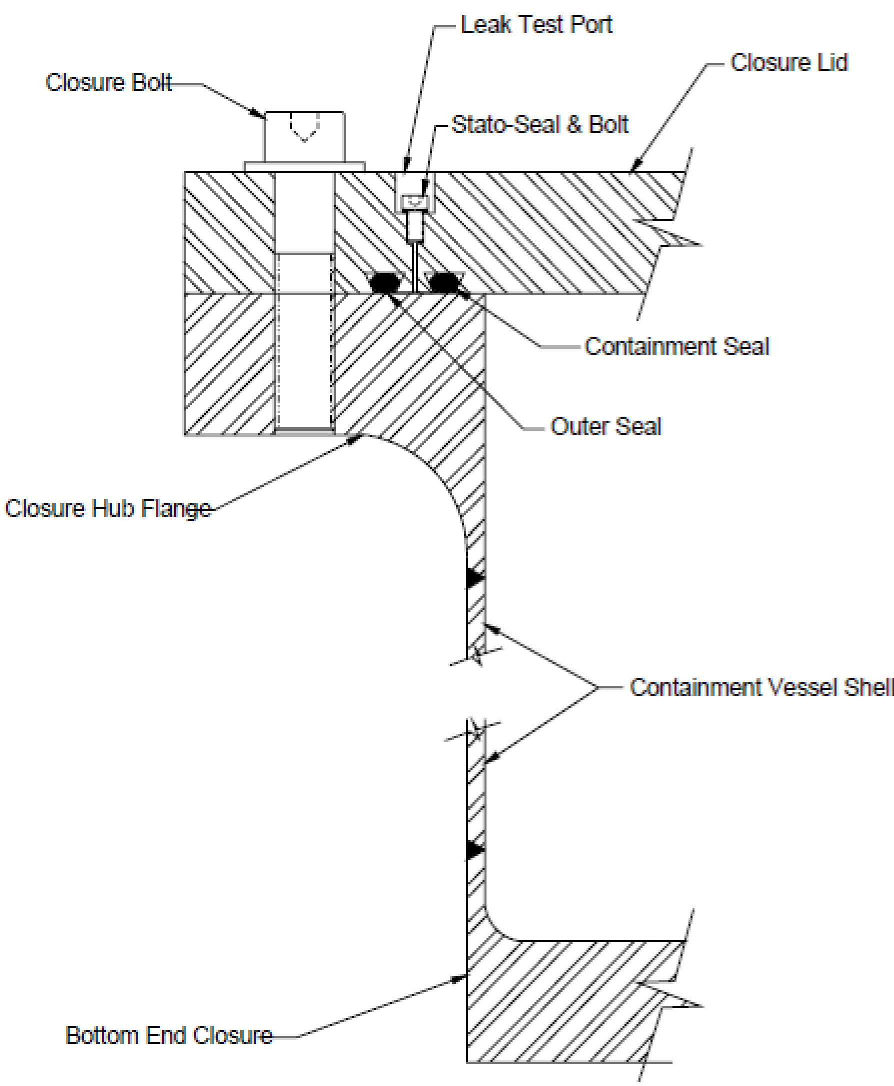
A CV usually has a bolted or other type of threaded closure to accommodate the loading and unloading of contents. The closure contains a seal or seals that minimize leakage from the CV to the environment. If there are multiple seals used for a single closure, the seal that is identified as the containment boundary seal should be clearly defined.

CONTAINMENT VESSEL

The function of all the CV and closure components is to maintain the containment boundary so that all normal condition of transport (NCT) and hypothetical accident conditions (HAC) containment requirements are met. The general standards for all packages listed in 10 CFR 71.43, “General Standard for all Packages,” also contain requirements that influence the structural design of the package containment.

For most NNSA packages, the CV is designed as an American Society of Mechanical Engineers (ASME) boiler and pressure vessel code (BPVC) pressure vessel. In other words, the CV is designed to be capable of holding pressure. The CVs of NNSA packages are usually configured as a hollow, thin-walled cylindrical shell with one welded end and a bolted flange closure at the other end for loading and unloading the contents. (See figure 1.)

The welded end closure of the CV is usually ellipsoidal, torispherical, hemispherical, or a flat head and welded to the CV shell by an ASME BPVC permissible weld joint. The bolted flange closure is usually a hub flange welded to the CV shell by an ASME BPVC permissible weld joint and bolted to a blind flange, which acts as the closure lid. The material of construction of an NNSA package CV component is usually stainless steel manufactured to ASME BPVC standards for pressure-retaining components or American Society for Testing and Materials (ASTM) standards that are essentially identical to ASME BPVC standards.



Source: NNSA SG-100

Figure 1. Typical containment system components

The closures of a containment system of NNSA packages are usually bolted closures and are the weakest component of the package. The structural integrity and containment performance of the bolted closure depend upon the number, strength, and tightness of the closure bolts and are vital in maintaining containment and adequate seal compression.

CLOSURE SYSTEM SEALS

The closure system of an NNSA package CV usually consists of a closure lid bolted to a hub flange. Closure system sealing is normally provided by two seals installed in dovetail or modified dovetail grooves on the underside of the closure lid. At assembly, the two seals are compressed between the polished machined surfaces on the hub flange and the closure lid grooves by installing and tightening the closure bolts to form the seal. In some designs, the closure lid is equipped with a vent/fill port which leads directly to the vessel cavity and a leak test port which leads to a small annulus between the two seals. The double-seal arrangement, coupled with a vent/fill and leak test ports, makes it possible to easily perform leakage tests of the inner seal. This is accomplished using various methods via a leak test port located between the two seals.

CLOSURE BOLTS

The weakest link in the CV closure system is typically the closure bolt joint. The integrity of the containment system depends on the number, strength, and tightness of the closure bolts. However, the structural behavior of bolted joints is very complex due to the interactions with the closure lid, vessel wall, and, at times, the seals. In addition, the behavior of bolted joints varies depending upon the design and application. Existing national standards, such as the ASME BPVC for vessels, have significantly different design parameters and loads from those of an NNSA CV. The closure bolts of an NNSA package CV must be able to experience significant fire and impact loads, as specified in normal conditions of transport and hypothetical transport condition requirements that are not experienced by a pressure vessel, and still perform their intended function.

Neutron and Gamma Shielding Features

The following is taken from NNSA SG-100.

For the majority of NNSA packages, structural parts of the CV are sufficient to provide shielding of short-range radiation such as alpha and beta, and usually are sufficient to reduce gamma and neutron radiation to acceptable levels. In cases where this is not true, shielding against gamma radiation can be achieved by using relatively dense materials such as lead, steel, or depleted uranium. At times, materials such as polyethylene or boron carbide are added to provide neutron shielding.

In all cases, the design should ensure the mechanical configuration of any shielding materials is maintained. In addition, when lead is used as a shield material, lead slump and changes in the shielding configuration should be considered. This can happen at relatively low temperatures. To prevent decomposition, it is important to protect any plastic or other hydrogenous neutron shielding from high temperatures or from intense gamma radiation.

Loads that can result in rupture or severe distortion of the CV also pose a threat to the shielding system. HAC stresses and concentrated loads may reduce the effectiveness of the shielding, even if the CV maintains containment.

Vibration and differential thermal expansion may also reduce the effectiveness of the shielding if the mechanisms holding the shielding components in place are subject to fatigue.

The package design should be evaluated to ensure the materials selected for the shielding are compatible under all conditions of transport. The materials selected should be such that there will be no significant chemical, galvanic, or other reaction among the package components and contents. The primary concern during NCT and HAC is thermal degradation or decomposition, particularly when plastics or other low-temperature materials are used. While not normally encountered in NNSA packages, radiolysis of hydrocarbons is a concern at higher radiation levels. Decomposition can affect the shielding properties of the material and can also cause pressurization of the CV if gases are generated. The most significant concern is a case in which decomposition of materials in the shielding results in a loss of mechanical properties or a physical configuration change that may lead to a loss of shielding effectiveness.

Lifting and Tie-Down Devices, Impact Limiters, and Other Structural Features

The following is taken from NNSA SG-100.

TIE-DOWN DEVICES

This section discusses the requirements for securing the packaging to a transport vehicle, including those for devices, inspection, and certification. Instructions may be included in the generic operating procedure or may be contained within a separate attachment or appendix. The following items may be included under this category:

- Tie-down devices (including tie-down straps, chains, etc.) that secure the package to its transport vehicle or shipping skid or pallet
- Inspection and certification requirements for the tie-down devices or shipping skids
- Alternatively acceptable means to provide proper package tie-down

Tie-down procedures used in the TSS conveyances are approved by the Office of Secure Transportation (OST).

IMPACT LIMITERS AND OTHER STRUCTURAL FEATURES

A number of empirical and classical calculational methods have been developed for scoping or preliminary determination of satisfactory or optimum performance of impact limiter or energy absorbing devices. In the past, the most common method used was a combination of testing, empirical, and classical methods. More recently, computer programs capable of modeling dynamic loadings on packages are being used to determine satisfactory or optimum performance of impact limiting materials and energy absorbing devices. In any method used, the most important parameter is the mechanical properties of the impact energy absorbing material under dynamic loading.

The main mechanical property of interest is the force deflection or stress-strain behavior of the material under dynamic load conditions. In the past, such materials as Celotex and/or plywood, polyurethane foam, wood, and Kaolite have been used for NNSA package impact limiters. In all cases, impact energy absorption is based on the crushing of these materials. The mechanical property information developed for these materials ranges from static force deflection curves to dynamic stress strain data. In this section, some classical and empirical methods are discussed that have been used in design and material selection of package impact limiting devices. They are based on selecting the materials and design of impact limiters by calculating the loads transmitted to critical components. Also discussed are the key parameters for assessing impact limiter performance using computer programs, which numerically simulate impacts of the package design.

In the past, several classical and empirical methods have been used for designing crushable impact limiting devices. Most are based on idealizing the containment system as a rigid body surrounded by a crushable material that deforms on impact. These methods are based on the assumption that the natural frequency of the containment system is substantially higher than that of the impact absorbing system and dampening can be ignored. The deceleration inertial factor on the containment system is then conservatively determined by balancing the impact energy to work done or by using the impulse momentum theory combined with empirical test data on the behavior of the energy absorbing material.

One empirical method, based on the impulse momentum theory for estimating maximum deceleration inertial load factors (gs) for free drops of packages protected by impact absorbing materials onto d surface, is provided in the *Shock and Vibration Handbook*. Based on field observations, knowing the drop height and assuming a sinusoidal deceleration pulse, the rise time (time to reach peak deceleration) and the peak deceleration can be determined and are dependent on the impact energy absorbing (or cushioning) material. The peak deceleration (G) in gs is determined based on the following equation:

$$G = \sqrt{\frac{0.0128 \times h}{t_r^2}}$$

where:
h = the drop height in inches
t_r = the rise time in seconds

The rise times given in table 41.1 of the *Shock and Vibration Handbook* are the approximated rise times from an assumed sinusoidal pulse for various packaging impact energy absorbing materials being dropped onto a rigid floor. The rise times provided are for metal containers, wooden boxes, cartons, and packages cushioned by 1-inch and 3-inch latex hairs. The reference also provides information and data on the material properties for impact absorbing materials. Unfortunately, the material data provided in this reference is limited by both type and the drop heights from which this data was obtained. For design purposes, this method provides the means for quickly and conservatively estimating impact limiting material performance if the rise time or deceleration time history data from a similar package impact limiter design using similar materials is known.

The most common classical method is the energy balance method. This method essentially equates the kinetic energy of the falling object to the work done in crushing the impact limiting material. The work done on the impact limiter can be represented as the stress applied multiplied by the volume crushed by the material. Assuming a uniform crush strength of the material, the volume of deformation is determined. Based on the geometry and from the volume, the area of crush is calculated and multiplied by the assumed uniform crush strength of the material. The product is then divided by the gross weight of the package to determine the average deceleration inertial factor. In using this method, the strain on the material crush is usually limited to 60 percent to account for material lock-up. The limitation of this method is that the crush strength of the impact limiting material must be relatively constant and known, the lock-up strain must be either known or estimated, and the determined deceleration initial load factors is only the average deceleration of the system.

A variation on this energy balance method is used in specifying the impact limiting properties of polyurethane foam. This method is presented in the General Plastics product information for LAST-A-FOAM® FR-3700 impact limiting and insulating materials. In this iterative method, a theoretical deceleration distance must be determined to reduce the forces on the containment system components to some desired level. This is simply done by dividing the free drop height by the desired inertial deceleration factor to determine the theoretical deceleration distance. A theoretically perfect impact limiting material would decelerate a payload uniformly through 100 percent of its thickness. It should be noted that impact limiting materials typically have efficiencies of 25 to 50 percent of theoretically perfect foams. As a result, the actual deceleration levels would be 2 to 4 times the theoretically determined values. If this deceleration distance is adequate, the next step is to determine if the energy levels are within the energy absorbing range of the foam. This is done by determining the strain energy density required by the foam, which is determined by dividing the kinetic energy of the object free dropped by the volume of the foam. From this strain energy density, the required density range of the foam is determined by integrating the stress-strain curve for various foam densities and equating this to the required strain energy density that needs to be absorbed. In the case of the foam manufactured by General Plastics, the foam dynamic properties have been well characterized by testing. The data and methods for determining desired foam properties are available in General Plastics publication *General Plastics LAST-A-FOAM® FR-3700 for Crash and Fire Protection of Nuclear Material Shipping Containers*. This method can also be applied to other impact limiting materials as long as the dynamic stress-strain or force-deflection properties and the impact energy absorbing efficiencies of the material are known. As with the previous methods, the limitation of this method is that the crush strength of the impact limiting material must be relatively constant and known, lock-up strain must be either known or estimated, and the determined deceleration initial load factors are only the average deceleration of the system.

There are also several empirical methods for design of impact limiters using steel structures. These are usually in the form of fins, tubular members, and steel cage type construction. Also in some cases, the spent nuclear fuel casks have been constructed of concentric cylinders with lead shielding poured into the annulus. In this case, the deformation of the outer shell and lead can be used as impact limiters as long as regulatory shielding limits are maintained. Although impact limiting devices made from steel structures are used in many commercial and DOE environmental management applications, they are not typically used in NNSA packages.

Heat Transfer Features

The following is taken from NNSA SG-100.

Specific heat is a measure of the amount of energy it takes to heat a specified quantity of a material a specified number of degrees. The units for this property can be expressed in kJ/(kg-K). The heat capacity of most materials increases with temperature.

Density is the mass per volume of a material and units are kg/m^3. For most materials, density decreases as temperature rises, although some materials may contract. Thermal conductivity, heat capacity, and density are needed to calculate non-steady state (transient) conduction heat transfer rates.

The only material property needed to calculate heat transfer rates due to radiation is emissivity (or absorbtivity). Emissivity is a measure of how efficiently a surface of a material emits radiant energy compared with an ideal radiator (blackbody). The emissivity of a material surface can vary greatly depending on the finish of the surface.

For instance, the emissivity of stainless steel can vary from 0.17 (highly polished) to 0.90 (oxidized) depending on the finish of the surface. While selecting a material with either a low or high emissivity due to its surface finish may help protect a package, attaining, keeping, certifying, and documenting the specified surface finish on a package part may be difficult. Most applications of radiative heat transfer dealing with a package are for the package surroundings to the package exterior (or vice versa). However, if gaps exist within the package, heat transfer across these gaps will be by radiation and convection (unless the space is a vacuum, in which case heat transfer is only by radiation).

Several properties must be known for the calculation of convection heat transfer coefficients, and they will always be fluid properties (i.e., either gas or liquid). The fluid properties needed to calculate the natural convection heat transfer coefficient are dependent on the correlations used, but generally the properties are fluid viscosity, density, volumetric thermal expansion coefficient, thermal conductivity, and heat capacity. For the vast majority of cases, the fluid will be air, although some other liquids, such as nitrogen and helium, are used in package design.

Packaging Tags, Markings, and Labels

The following is taken from the United States (U.S.) Nuclear Regulatory Commission, *Labeling.*

Labeling

Labels are used to visually indicate the type of hazard and the level of hazard contained in a package. Labels rely principally on symbols to indicate the hazard.

Although the package required for transporting radioactive material is based on the activity inside the package, the label required on the package is based on the radiation hazard outside the package. Radioactive material is the only hazardous material which has three possible labels, depending on the relative radiation levels external to the package. Also, labels for radioactive material are the only ones that require the shipper to write some information on the label. The information is a number called the transport index (TI), which, in reality, is the highest radiation level at 1 meter from the surface of the package.

The three labels are commonly called white I, yellow II, and yellow III, referring to the color of the label and the Roman numeral prominently displayed. A specific label is required if the surface radiation limit and the limit at 1 meter satisfy the requirements in table 1.

Table 1. Types of labels

Label	Surface Radiation Level		Radiation Level at 1 Meter
White I	Does not exceed 0.5 millirem (mrem) 1 hour (h)		Not applicable
Yellow II	Does not exceed 50 mrem/h	and	Does not exceed 1 mrem/h
Yellow III	Exceeds 50 mrem/h	or	Exceeds 1 mrem/h

Source: U.S. Nuclear Regulatory Commission, Labeling

Because the TI is the radiation level at 1 meter, it is clear that a white I label has no TI. A yellow II must have a TI no greater than 1, and a yellow III may have a TI greater than 1.

b. Define and discuss the following items related to radioactive material contents:

- **Maximum activity**
- **Fissile material spatial separation and quantity limits**
- **Neutron absorbers or moderators**
- **Materials subject to chemical or galvanic reactions**
- **Decay heat**

Maximum Activity

According to The American Nuclear Society *Tool Kit Glossary*, maximum activity is the maximum rate of disintegration (transformation) or decay of radioactive material. The units of activity are the curie (Ci) and the becquerel (Bq).

Fissile Material Spatial Separation and Quantity Limits

The following is taken from 49 CFR 174.700.

The number of packages of class 7 (radioactive) materials that may be transported by rail car or stored at any single location is limited to a total TI and a total criticality safety index (CSI) of not more than 50 each.

Each package of class 7 (radioactive) material bearing radioactive yellow II or radioactive yellow III labels may not be placed closer than 3 feet to an area that may be continuously occupied by any passenger, rail employee, or shipment of one or more animals, nor closer than 15 feet to any package containing undeveloped film (if so marked). If more than one package of class 7 materials is present, the distance must be computed from table 2 on the basis of the total TI number (determined by adding together the TI numbers on the labels of the individual packages) of packages in the rail car or storage area.

Table 2. Separation of class 7 packages by rail

Total Transport Index	Minimum separation distance to nearest undeveloped film		Minimum distance to area of persons or minimum distance from dividing partition of a combination car	
	Meters	Feet	Meters	Feet
None	0	0	0	0
0.1 to 10.0	4.5	15	0.9	3
10.1 to 20.0	6.7	22	1.2	4
20.1 to 30.0	7.7	29	1.5	5
30.1 to 40.0	10	33	1.8	6
40.1 to 50.0	10.9	36	2.1	7

Note: The distance in this table must be measured from the nearest point on the nearest packages of class 7 materials.
Source: 49 CFR 174.700

The following is taken from 49 CFR 177.842.

The number of packages of class 7 (radioactive) materials in any transport vehicle or in any single group in any storage location must be limited so that the total TI number does not exceed 50. The total TI of a group of packages and overpacks is determined by adding together the TI number on the labels on the individual packages and overpacks in the group.

Packages of class 7 (radioactive) material bearing radioactive yellow II or radioactive yellow III labels may not be placed in a transport vehicle, storage location, or in any other place closer than the distances shown in table 3 to any area which may be continuously occupied by any passenger, employee, or animal, nor closer than the distances shown in the table to any package containing undeveloped film (if so marked), and must conform to the following conditions:

- If more than one of these packages is present, the distance must be computed from table 3 on the basis of the total TI number determined by adding together the TI number on the labels on the individual packages and overpacks in the vehicle or storeroom.
- Where more than one group of packages is present in any single storage location, a single group may not have a total TI greater than 50. Each group of packages must be handled and stowed not closer than 20 feet (measured edge to edge) to any other group.

Table 3. Separation between class 7 packages by highway

Total Transport Index	Minimum separation distance in meters (feet) to nearest undeveloped film in various times of transit					Minimum distance in meters (feet) to area of persons, or minimum distance in meters (feet) from dividing partition of cargo compartments
	Up to 2 hours	2–4 hours	4–8 hours	8–12 hours	Over 12 hours	
None	0.0 (0)	0.0 (0)	0.0 (0)	0.0 (0)	0.0 (0)	0.0 (0)
0.1 to 1.0	0.3 (1)	0.6 (2)	0.9 (3)	1.2 (4)	1.5 (5)	0.3 (1)
1.1 to 5.0	0.9 (3)	1.2 (4)	1.8 (6)	2.4 (8)	3.4 (11)	0.6 (2)
5.1 to 10.0	1.2 (4)	1.8 (6)	2.7 (9)	3.4 (11)	4.6 (15)	0.9 (3)
10.1 to 20.0	1.5 (5)	2.4 (8)	3.7 (12)	4.9 (16)	6.7 (22)	1.2 (4)
20.1 to 30	2.1 (7)	3.0 (10)	4.6 (15)	6.1 (20)	8.8 (29)	1.5 (5)
30.1 to 40	2.4 (8)	3.4 (11)	5.2 (17)	6.7 (22)	10.1 (33)	1.8 (6)
40.1 to 50.0	2.7 (9)	3.7 (12)	5.8 (19)	7.3 (24)	11.0 (36)	2.1 (7)

Source: 49 CFR 177.842

QUANTITY LIMITS
The following is taken from DOE-HDBK-1129-2008.

Once an item has met the DOE and DOT requirements for shipment (e.g., properly packaged, radioactively surveyed, properly marked, properly completed shipping papers), the item inside the approved shipping package can be shipped to a new location.

During shipment, the item inside the approved shipping package is expected to be subject to the normal activities associated with its movement from one location to another; for example, loading and unloading the package from vehicles, transport to a shipping area, storage in the shipping area prior to transport, loading and unloading the package onto trucks/trains/airplanes, and storage in the receiving area after arrival at the new destination.

The packaging required by DOT and NRC regulations is designed to protect the workers, the public, and the environment from the radioactive material during normal package handling, transport, and shipping/receiving storage.

The applicable requirements for various quantities of tritium for transportation and storage are roughly as follows:

- Limited quantity—Limited quantities of tritium can be packaged and shipped in strong, tight containers (paper boxes, paint cans) with proper markings.
- Type A quantity (21.6 to < 1,100 Ci)—Type A quantities of tritium use DOT specification 7A containers, properly marked and surveyed prior to shipment. A number of different packages are available from small cans, 55- and 85-gallon drums, 4 x 4 x 7-foot steel boxes, up to and including oversized, specially designed containers. These containers are relatively inexpensive.

- Type B quantity (> 1,100 Ci)—Type B quantities of tritium must be shipped in a certified DOT type B package. There is only a limited number of these expensive type B packages available for tritium shipment, and special routing (i.e., prescribed routes employing highway route control such as using major highways or bypassing cities) is required.
- Type B quantity, low-level radioactive waste—For the purposes of storage at the waste site, type B quantity solid waste can be stored in type A containers. At the waste generation location, items containing greater than 1,100 Ci of tritium are normally stored in type A containers. These type A containers containing over 1,100 Ci must then be placed into type B containers for shipping to DOE waste sites. At the DOE waste site, the type A containers can be removed from the type B package and stored in the type A package. This allows the expensive type B package to be returned to the shipper and reused for another type B shipment.

Table 4. Allowable quantities of tritium

Form	Shipping Package Type	Maximum Quantity Specific Activity of Tritium per Package	Comments
All physical forms	None	• Activity concentration for exempt material of 10^{-6} TBq • Activity limit for exempt consignment of 10^{-3} TBq (0.027 Ci)	Both conditions must be satisfied.
Solid	Limited quantity		
	LSA-I	0.04 TBq (1.1 Ci)	
	LSA-II	4×10^{-5} TBq/g (0.001 Ci/g)	The conveyance limit for combustible solids is 4,000 TBq (1.1×10^5 Ci).
	LSA-III	0.08 TBq/g (2.16 Ci/g)	
	SCO-I	Limit based on surface contamination	The maximum nonfixed contamination on accessible surfaces is 4 Bq/cm^2. The maximum fixed contamination on accessible surfaces is 4×10^4 Bq/cm^2. The total surface contamination on the inaccessible surfaces is limited to 10^4 Bq/cm^2. The conveyance limit is 4,000 TBq.
	SCO-II	Limit based on surface contamination	The maximum nonfixed contamination on accessible surfaces is 400 Bq/cm^2. The maximum fixed contamination on accessible surfaces is 8×10^8 Bq/cm^2. The total surface contamination on the inaccessible surfaces is limited to 8×10^5 Bq/cm^2. The conveyance limit is 4,000 TBq.
	Type A	40 TBq (1,100 Ci)	
	Type B	Limited by CoC for the package	The type B UC-609 package is limited to 150 g of tritium.

Form	Shipping Package Type	Maximum Quantity Specific Activity of Tritium per Package	Comments
Liquid	Limited quantity (tritiated water)	37 TBq (1,000 Ci) 3.7 TBq (100 Ci) 0.037 TBq (1 Ci)	<0.0037 TBq (0.1Ci/L) 0.0037 to 0.037 TBq (0.1 to 1.0 Ci/L) >0.037 TBQ/L (1.0 Ci/L).
	Limited quantity (other liquids)	0.004 TBq (0.108 Ci)	
	LSA-II	4×10^{-4} TBq/g (0.001 Ci/g)	The conveyance limit is 4,000 TBq (1.1×10^5 Ci).
	Type A	40 TBq (1,100 Ci)	
	Type B	Limited by CoC for the package	The type B UC-609 package is limited to 150 g of tritium.
Gas	Limited quantity	0.8 TBq (21.6 Ci/g)	
	LSA-II	0.004 TBq/g (0.1 Ci/g)	The conveyance limit is 4,000 TBq (1.1×10^5 Ci).
	Type A	40 TBq (1,100 Ci)	
	Type B	Limited by CoC for the package	The type B UC-609 package is limited to 150 g of tritium.

Note: LSA (low specific activity) SCO (surface contaminated object)
Source: DOE-HDBK-1129-2008

Neutron Absorbers or Moderators

The following is taken from Wikipedia, *Neutron Poisons*.

A neutron poison (also called a neutron absorber or a nuclear poison) is a substance with a large neutron absorption cross section, in applications such as nuclear reactors. In such applications, absorbing neutrons is normally an undesirable effect. However, neutron absorbing materials, also called poisons, are intentionally inserted into some types of reactors to lower the high reactivity of their initial fresh fuel load. Some of these poisons deplete as they absorb neutrons during reactor operation, while others remain relatively constant.

The following is taken from NNSA SG-100.

The presence of a neutron absorber or neutron absorbing material in the packaging will be revealed by the fabrication records. A test should be performed on the package to ensure the neutron absorbing material is functional. This may be accomplished by loading the packaging with a measured neutron emitting source and taking direct measurements. If no neutrons are detected, the absorber is still functional and the packaging may be used for shipment.

A moderator is a material, typically light nuclei, used to reduce the kinetic energy of neutrons by scattering collisions without appreciable neutron capture.

Materials Subject to Chemical or Galvanic Reactions

CHEMICAL REACTIONS

The following is taken from eHow, *Types of Chemical Reactions of Metals.*

A chemical reaction is a process by which one type of chemical substance is converted to another type of chemical substance. All chemical reactions are characterized by a change in energy. Reactions either release or absorb energy from their surroundings as they proceed. In metals, there are four types of chemical reactions: synthesis reactions, decomposition reactions, single displacement reactions, and combustion reactions

When two elements combine to form a compound, it is called a synthesis reaction. Basically, two reactants combine to form a product. Types of synthesis reaction include: combining a metal and a nonmetal to form a compound; reaction between a metallic oxide and water to form a hydroxide; and a metal reacting with oxygen to form metal oxide. Examples include sodium reacting with chlorine to form sodium chloride and aluminum reacting with fluorine to form aluminum fluoride.

When a compound breaks into its elements or into simpler compounds, it is called a decomposition reaction. In this reaction, a reactant breaks into two or more products. Types of decomposition reaction include: metallic carbonates, when heated, produce metallic oxide and carbon dioxide; metallic chlorides break into a metallic element and chlorine; and metallic chlorates break into metallic chlorides and oxygen.

GALVANIC REACTIONS
The following is taken from eHow, *Aluminum Galvanic Reaction.*

A galvanic reaction is the movement of ions from one type of metal to another, which results in the corrosion and consumption of the metal losing ions and the fortification against corrosion of the metal gaining ions.

When two metals come into contact, a galvanic reaction begins if three conditions are in place: the metals are electrochemically dissimilar; there is an electrically conductive path between the two metals; and there is a path by which metal ions can move. Water that contains salt, such as seawater, provides electrical conductivity as well as a path for metal ion migration.

Metals with high anodic index (AI) values are more prone to galvanic corrosion than those with low AI values. Aluminum is considered a relatively reactive metal, with an AI value between 0.75 V and 0.95 V, depending on the composition of the aluminum alloy.

To prevent galvanic corrosion of aluminum when in proximity with a lower AI metal, such as corrosion-resistant steel (AI 0.50 V), either the metals must be treated with a nonreactive coating, such as paint, or a sacrificial metal such as magnesium (AI 1.75 V) must be attached to the aluminum.

Decay Heat
The following is taken from Massachusetts Institute of Technology, Nuclear Science and Engineering, *Explanation of Nuclear Reactor Decay Heat.*

Nuclear reactors produce electricity in a similar way to conventional coal plants in that they heat steam to drive a turbine that spins an electric generator. However, they differ on how that heat is produced. Coal plants burn coal to heat a boiler that produces the steam while nuclear reactors use nuclear fission to create the heat.

The heat in an operating reactor is produced mainly by the fission of fissile isotopes such as uranium (U)-235 and plutonium (Pu)-239. When a neutron causes one of these isotopes to split, a large amount of energy is released, which is then deposited in the fuel, cladding, coolant, and structures. On average, approximately eighty percent of the energy released in a fission reaction is imparted to the two or more fission products and these deposit their energy in the fuel, since they have a very short range. The rest of the energy is released in the form of neutrons, and other forms of radiation.

When there is a SCRAM (an emergency shutdown of a nuclear reactor), where all the control rods are inserted and the reactor is shutdown, the fission reactions essentially stop and the power drops drastically to about seven percent of full power in one second. The power does not drop to zero because of the radioactive isotopes that remain from the prior fissioning of the fuel. These radioactive isotopes, also called fission products, continue to produce various types of radiation as they decay, such as gamma rays, beta particles, and alpha particles. The decay radiation then deposits most of its energy in the fuel, and this is what is referred to as decay heat. As these radioactive isotopes continue to decay, more and more of them reach a stable state and stop emitting radiation, and thus no longer contribute to the decay heat.

The decay heat must be removed at the same rate it is produced or the reactor core will begin to heat up. The removal of this heat is the function of the various reactor core cooling systems that provide water flow through the reactor core and then reject the heat elsewhere.

Video 1. Decay heat
https://www.youtube.com/watch?v=Wb1JFzBgSl0

5. **NNSA package certification engineers must demonstrate a working level knowledge of the structural evaluation information required in chapter 2 of a SARP.**

   a. **Discuss the following:**
      - **NNSA SG-100 as it pertains to structural design and development: 1) mechanical design, 2) material selection, 3) fabrication (including welding), 4) examination, 5) testing, 6) SARP preparation, and 7) certification.**
      - **Determination of the design criteria for package design per NRC Regulatory Guide 7.11, *Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1 m)***
      - **NUREG/CR-3854*, Fabrication Criteria for Shipping Containers***
      - **The purpose and scope of NRC Regulatory Guide 7.6, *Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels***
      - **The purpose and scope of NRC Regulatory Guide 7.8, *Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material***
      - **Material properties and specifications**
      - **The application of austenitic stainless steel as opposed to ferritic steels in the design and construction of small type B packages**
      - **Chemical, galvanic, and other reactions, and the application of corrosion-inhibiting coatings**
      - **General requirements for all packages per 10 CFR 71.43**
      - **10 CFR 71 requirements for lifting and tie-down devices**

NNSA Safety Guide 100

STRUCTURAL DESIGN

NRC Regulatory Guide 7.6 was developed in the late 1970s specifically for spent nuclear fuel casks. It is based on the 1977 version of the ASME BPVC, Section III, Division 1, Subsection NB for Class 1 components. Therefore, much of the guidance in NRC Regulatory Guide 7.6 is based on the "design by analysis" and stress categorization methods outlined in Article NB-3000 for Class 1 components. At that time, the predominate method of analysis was classical linear elastic methods.

NRC Regulatory Guide 7.6 has not changed since its issuance in 1978, while the ASME BPVC and analytical methods have evolved over time. As a result, the NNSA guidance in Section 2.5 suggests that the current version of the ASME BPVC, Section III, Division 1, Subsection NB be used, along with the appropriate requirements from NRC Regulatory Guide 7.6 that are specific to packaging. The basis for this is that the packaging performance tests and the failure modes addressed by NRC Regulatory Guide 7.6 have not changed substantially. The appropriate criteria that should be considered are those outlined in NRC Regulatory Guide 7.6 regulatory positions 3, 4, 6, and 7. It should be recognized that these criteria are based on linear elastic methods and should be applied only to the containment system. Other appropriate nonlinear methods can be applied to other components, such as the impact limiters. Also, as the ASME BPVC design by analysis method is based on the maximum shear stress theory of failure, the stress intensity, which is the difference between the maximum and minimum principal stresses on the section being evaluated, must be evaluated. In using the guidance provided in Section 2.5 and NRC Regulatory Guide 7.6, it should be recognized that HAC performance test evaluations may use the criteria specified in ASME BPVC, Section III, Division 1, Appendices, Appendix F for Category I content package containment systems. Appendix F provides acceptance criteria for elastic system analysis and plastic system analysis in Section F-1330 and Section F-1340, respectively. Both sections define acceptance criteria for level D service limits for components designed in accordance with Subsection NB. Section F-1341 allows the use of any one of the following structural acceptance criteria for demonstrating the acceptability of components:

- Elastic analysis (F-1341.1)
- Plastic analysis (F-1341.2)
- Collapse load analysis (F-1341.3)
- Plastic instability analysis (F-1341.4)
- Interaction analysis (F-1341.5)

MECHANICAL DESIGN

NRC Regulatory Guide 7.6 describes design criteria acceptable to the NRC staff for use in the structural analysis of spent nuclear fuel cask CVs. These criteria are used to demonstrate the CV meets the performance tests of 10 CFR 71.71, "Normal Conditions of Transport," and 10 CFR 71.73, "Hypothetical Accident Conditions." NRC Regulatory Guide 7.6 was developed because there were no standards for demonstrating satisfactory performance of shipping cask CVs at the time. Although originally written for spent nuclear fuel cask, NRC Regulatory Guide 7.6 has traditionally been applied as the acceptable performance criteria for structural analysis of all type B package CVs with category I contents. The design criteria established in NRC Regulatory Guide 7.6 are only applicable to the structural evaluation of the CV of a package.

NRC Regulatory Guide 7.6 has adopted portions of the ASME BPVC that use the design by analysis approach for class 1 reactor components to form acceptable design criteria for shipping cask CVs. As with the 1977 ASME BPVC, NRC Regulatory Guide 7.6 is based on linear elastic methods, stress categorization, and the principle of superposition for determining the effect of combined loads on CVs. The authors of NRC Regulatory Guide 7.6 recognized that ASME BPVC, Section III contains requirements for the design of nuclear power plant components, so many of its failure modes are not considered applicable to shipping cask CV design. However, NRC Regulatory Guide 7.6 provides design criteria extracted from the 1977 ASME BPVC that are considered acceptable to the NRC for assessing the adequacy of CV designs. These requirements are stated as regulatory positions in NRC Regulatory Guide 7.6. Because NRC Regulatory Guide 7.6 is based on ASME BPVC design by analysis methods, the definitions and classification of stresses are essentially the same as the 1977 ASME BPVC, Section III, Division 1, Subsection NB.

In assessing the performance of a package under the performance tests specified in 10 CFR 71.71 for NCT and 10 CFR 71.73 for HAC, NRC Regulatory Guide 7.6 applies the ASME BPVC design by analysis approach. NRC Regulatory Guide 7.6 has adapted the service loading limits for reactor components to packaging performance test conditions. In NRC Regulatory Guide 7.6, NCT service loading limits are defined as level A service limits and HAC service loading limits are defined as level D service limits, as amended by NRC Regulatory Guide 7.6 for shipping cask CVs. In applying these limits, NRC Regulatory Guide 7.6 applies the ASME BPVC methods for determination of satisfactory performance of the CV. The ASME BPVC method applies the Tresca-Guest or maximum shear stress theory of failure by defining the term “stress intensity” as twice the maximum shear stress and equal to the difference between the algebraically largest and smallest principal stresses at any given point. These stress intensities are then categorized into three stress categories: primary, secondary, and peak. In addition, the primary stress intensity has three sub-categories: general membrane, local membrane, and bending. The stress is divided into various categories because the limit analysis method used in the ASME BPVC indicates that some stresses may be permitted to go to higher values.

Packaging materials are selected based on the package content material and the package safety functions required to transport this material. Materials typically packaged in NNSA packages are given in table 5.

Table 5. Typical NNSA package content materials

Material	Form	Source of Nuclear and Physical Properties
^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U	Solid (parts, powder, chips, etc.): pure metal, oxides, nitrides, other compounds	*SCALE: A Modular Code for Performing Standardized Computer Analyses for Licensing Evaluation,* NUREG/CR-0200, Rev.7, 2004
^{238}Pu, ^{239}Pu, ^{240}Pu	Solid (parts, powder, chips, etc.): pure metal, oxides, nitrides, other compounds	*SCALE: A Modular Code for Performing Standardized computer Analyses for Licensing Evaluation,* NUREG/CR-0200,Rev.7, 2004
Tritium	Solid, liquid, or gas (mixtures and compounds)	*Health Physics Manual of Good Practices for Tritium Facilities, MLM-3719, Draft EG&G Mound Applied Technologies*, December 1991

Source: NNSA SG-100

The materials used in NNSA packages must enable the package to satisfy the following safety functions: contain the radioactive material; provide shielding from radiation; and maintain subcriticality. The packaging must also keep the contents in place (confinement). These functions, in part, are accomplished by designing structural integrity into the packaging as well as designing the package to accommodate the thermal loads which may be experienced by the packaging. The following paragraphs discuss the material characteristics needed to accomplish each of these functions.

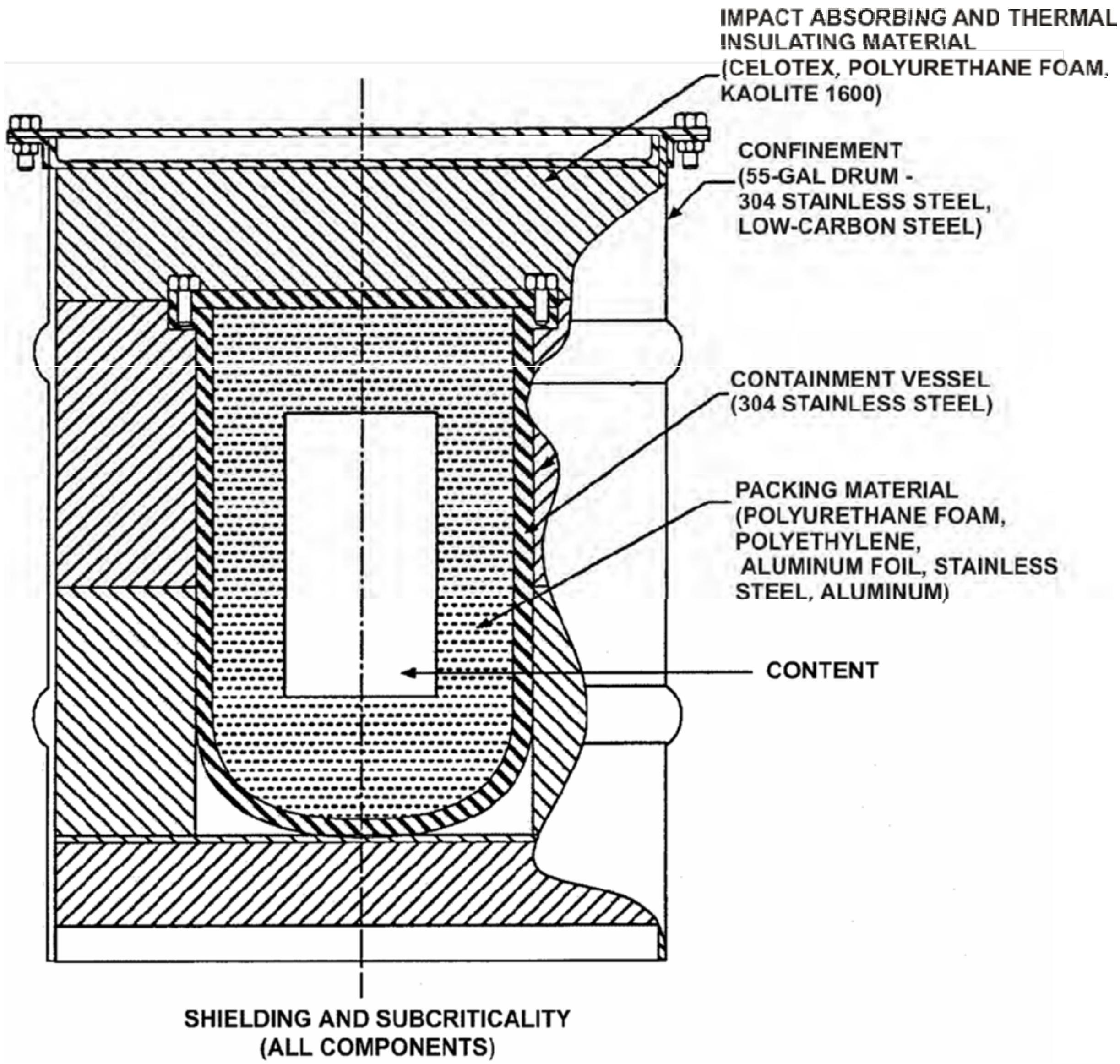
A sample packaging showing the function(s) of each of the components and possible materials for those components is shown in figure 2.

Containment

The containment boundary must prevent the release of radioactive materials during NCT and HACs as stipulated in the regulations. The walls, heads, welds, fasteners, and seals that make up the containment boundary must be capable of withstanding the mechanical, thermal, and physical conditions to which they may be subjected.

Shielding

The radiation dose rate must not exceed regulatory limits during use. For NCT, the regulations stipulate dose rates at the surface of the package and at 1m from the package surface; whereas for HAC only dose rates at 1 meter from the package surface are specified. If the packaging is deformed during HAC, consideration must be given to the possibility that the radiation source may be nearer to the surface of the packaging and, consequently, the radiation dose rate at 1 meter may be higher after deformation. In some cases, packaging design features that satisfy functions other than shielding also provide some shielding. In other cases, the materials used or the thicknesses of the materials used will have to be changed to provide adequate shielding. In addition to transportation, the packaging may be required to meet more stringent facility dose rates. These lower criteria must be considered during the material and fabrication selection process.



Source: NNSA SG-100

Figure 2. Sample package

Subcriticality

Subcriticality must be maintained during all NCT and HAC. In many cases, subcriticality is dependent on keeping moderating material out of the containment boundary, and on maintaining adequate spacing between multiple containers. With the advent of the crush test, possible reduced spacing of post-HAC units must also be considered.

Structural Integrity

Structural integrity is often supplied by impact limiters. The components of the packaging that act as an impact limiter are usually those components at or near the outer surface of the packaging. The impact limiters absorb energy from the HAC drops and crush tests such that the other functions of the packaging are preserved.

Thermal
The characteristics of the materials used must accommodate the thermal loads that may be applied to the package. Thermal loads include radioactive decay heat from the contents, heating from the sun (insolation), and heat from a HAC thermal event. The environment can also affect the thermal load on the package. Various combinations of thermal loading can occur during the shipment of a package. The thermal properties of the materials used and the configuration of those materials must be such that containment, shielding, and subcriticality are maintained.

Confinement
The components of the packaging that provide confinement must hold the package together during HAC to the extent that the other functions of the packaging are maintained. Usually, the confinement components are those that make up the outer surface of the packaging. In the sample packaging (figure 2), the confinement boundary consists of the stainless steel drum and drum lid. After the drop test and crush test portions of the HAC, the extent of the deformation and opening of the drum must be such that allowable radiation doses are not exceeded and containment is maintained during the fire test.

Packing Material
Packing material may be used inside the CV and holds the contents in place. It may also enhance the structural integrity and thermal capabilities of the package, as well as provide confinement. No single material can accomplish all of these functions; however, some materials contribute to more than one function. Materials should also be durable, easy to work with, and cost-effective. Materials that have been successfully used for various functions and components are listed in table 6. Materials other than these may be used in NNSA packagings; however, justification for their use must be documented.

Table 6. Packaging materials and their functions

Material	Function						
	Containment	Shielding	Subcriticality	Structural Integrity	Thermal	Confinement	Packing Material
Austenitic stainless steel	X	x	x	x		x	X
Carbon steel	X	X		X		X	
High strength steel	X					X	X
Aluminum alloy 6061	X					X	X
Cellulosic insulating board				X	X		
Inorganic refractory				X	X		
Ceramic fiber materials				X	X		
Fir plywood				X	X		
Redwood				X	X		
Urethane foam				X	X		
Silicone foam							X
Ethylene propylene	X						

Source: NNSA SG-100

FABRICATION

Fabrication generally addresses forming, fitting, and aligning; welding and brazing; heat treatment; and mechanical joints. If the fabrication specifications are prescribed by an acceptable code or standard, the code or standard should be identified on the engineering drawings. Unless otherwise indicated, specifications of the same code or standard used for design should also be used for fabrication. For components that have no fabrication codes or standards that are applicable, the specifications on which the evaluation depends should be described, and the method of control used to assure that the specifications are met should be identified. This information should be indicated on the engineering drawings.

EXAMINATION

Examination addresses the methods and criteria by which fabrication is determined to be acceptable. Unless it is justified otherwise, specifications of the same code or standard used for fabrication should also be used for examination. For components that have no fabrication codes or standards that are applicable, the examination method and acceptance criteria used should be described.

TESTING

In general, the requirements for pressure testing of components in Article 6000 are the same in Subsection NB and ND. Essentially when the construction is complete the component is closed and tested hydrostatically at a pressure of 1.25 times the design pressure. In the past this test was considered a structural integrity test even though it did not subject the vessel to conditions as severe as it might be expected to experience in service. The test is normally at atmospheric temperature and the only loads applied are the pressure loading and the weight of the test fluid. In cases where hydrostatic testing is not appropriate, pneumatic testing may be used. However, pneumatic pressure tests are specified to be 1.1 times the design pressure. No pressure testing is required for components constructed to Subsections NF and NG.

In the case of vessels constructed to the requirements, the requirements for pressure testing of components is a hydrostatic test at 1.3 times the maximum allowable working pressure (MAWP). For packaging purposes, MAWP is the same as the design pressure. As with Section III, Division 1, pneumatic testing is permitted where hydrostatic testing is not appropriate. However, pneumatic pressure tests are specified to be 1.1 times the MAWP.

SARP PREPARATION

The culmination of the design and verification phase will lead to the preparation of the SARP. The SARP should clearly demonstrate that all the requirements have been met with a clear margin of safety to protect the public, the workers, and the environment.

The information in the SARP is unique to the packaging and content. A suggested format for a SARP has been provided by the NRC in Regulatory Guide 7.9.

Revision 2 of NRC Regulatory Guide 7.9, was issued in March of 2005. NNSA SG-500 suggests the use of NRC Regulatory Guide 7.9 for SARP development. Appendix B of NNSA SG-500 provides additional guidance on the SARP development process.

NRC Regulatory Guide 7.9 has been developed from years of experience to standardize the SARP preparation and review process. While adherence to NRC Regulatory Guide 7.9 is not a requirement, it is highly recommended that it be followed closely. By using NRC Regulatory Guide 7.9, the SARP will be more easily and quickly reviewed. If applicants deviate from the suggested format and/or content, it is recommended that such deviations be well-documented in the SARP to aid the reviewer.

SARPs submitted to the DOE and NNSA, unlike those submitted to the NRC, must contain a section on QA. This is a deviation from the format outlined in NRC Regulatory Guide 7.9.

CERTIFICATION

Packages are usually certified for five years, and applications must be made to recertify packages when the certificate is due to expire. The application will include a revised SARP, supplemental report, and completed NNSA SG-200 checklist, and any other supporting documents that should be submitted no later than 26 work weeks prior to the expiration of the certificate. Additional time may be required for the recertification process if significant comments are generated during the review cycle or there are delays in transmitting sensitive or classified documents.

Numerous issues may impact the amount of time and effort needed for the recertification effort. Packages requiring design changes will require more effort and time than packages that simply need recertification. A listing of all packaging serial numbers that were used under the existing or previous certificate and their refurbishment history should be available for review by the CO. If the refurbishment history indicates an excessive amount of repair/refurbishment of packaging components, replacement of these components with more robust components may be warranted. In addition, the level of the change to the safety class will also affect the level of change to the SARP. If new or revised rules and regulations become effective, between the time the package was certified/recertified and this recertification, the changes to the rules and/or regulations must be properly considered, as the packaging may not meet the new requirements. The extent of the SARP revision will depend on how the new regulations are to be implemented, the extent of the rule changes, and how they will impact the packaging to be recertified.

Determination of the Design Criteria

The following is taken from U.S. Nuclear Regulatory Commission, NRC Regulatory Guide 7.11.

Table 7. Design categories

Levels of safety	Category I	Category II	Category III
Low specific activity		Greater than 30,000 Ci or greater than 3,000 A_1 or greater than 3,000 A_2	Less than 30,000 Ci and less than 3,000A_1 and less than 3,000 A_2
Special form	Greater than 3,000 A_1 or greater than 30,000 Ci	Between 3,000 A_1 and 30 A_1 and not greater than 3,000 Ci	Less than 30 A_1 and less than 30,000 Ci
Normal form	Greater than 3,000 A_2 or greater than 30,000 Ci	Between 3,000 A_2 and 30 A_2 and not greater than 30,000 Ci	Less than 30 A_2 and less than 30,000 Ci

Source: U.S. Nuclear Regulatory Commission, Regulatory Guide 7.11

NUREG/CR-1815, *Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers up to Four Inches Thick*, contains background and other information pertinent to the development of the criteria in this guide. These criteria are divided into three categories that are associated with the levels of safety appropriate for the radioactive contents being transported. Table 7 identifies the radioactivity limits for each of the three categories. The criteria contained in Section 5 of NUREG/CR-1815, other than for full-scale destructive testing and qualifying procedures for reduced stress levels, are acceptable to the NRC staff for assessing the fracture toughness of thin-wall base material ferritic steel CVs for the categories identified in table 7.

A category I container qualified in accordance with NRC Regulatory Guide 7.11 is acceptable for transporting either category II or category III radioactive materials. Similarly, a category II container qualified in accordance with the guide is acceptable for transporting category III materials.

Except in those cases in which an applicant or licensee proposes an acceptable alternative method for complying with specified portions of the NRC's regulations, the methods described in NRC Regulatory Guide 7.11 (which reflects public comments) will be used by the NRC staff in evaluating base material for all applications for new package designs and all requests that existing package designs be designated as type B(U) or type B(M) packages.

NUREG/CR-3854, Fabrication Criteria for Shipping Containers

The following is taken from U.S. Nuclear Regulatory Commission, NUREG/CR-3854.

NUREG/CR-3854 provides fabrication criteria for the metal components of shipping containers used for transporting radioactive materials. The criteria are divided into three categories that are associated with the levels of safety for the types and quantities of radioactive materials being transported. For each category, the fabrication criteria are subdivided into three component safety groups that are formed according to their safety function. The categories and component safety group designations are the same as those used in developing the welding criteria.

The fabrication criteria are based on the ASME BPVC as summarized in table 8 for each of the categories and component safety groups. Section 4.0 of NUREG/CR-3854 provides the detailed fabrication criteria, including any exceptions or modifications to the ASME code. The selected ASME code criteria provide levels of confidence in controlling fabrication processes consistent with the categories and component safety groups. The criteria should be used with the welding criteria when fabricating shipping containers for transporting radioactive materials.

Table 8. Summary of the fabrication criteria based on the ASME code[a]

Component safety group[b]	**Container contents**		
	Category I	Category II	Category III
Containment	Section III Subsection NB	Section III Subsection ND	Section VIII Division 1
Criticality	Section III, Subsection NG		
Other safety	Section VIII, Division 1 or Section III, Subsection NF		

[a] See Section 4.0 of NUREG/CR-3854 for detailed criteria for each category, component safety group, and fabrication process.
[b] DOT specifications 17C or 17H are acceptable for the fabrication of drums and pails used as shipping containers.
Source: NUREG/CR-3854

The criteria for fabricating metal component of shipping containers used for transporting radioactive materials are based on the ASME code as summarized in table 8. An acceptable method of assuring compliance with the criteria is to have the construction of a shipping container carried out by a fabricator having a valid certificate of authorization for the use of the ASME code stamp for the appropriate section. It is not intended that the ASME code stamp be applied to the shipping container. A fabricator having a Section III subsection NB, ND, or NF certificate of authorization is considered to be qualified for Section VIII fabrication and would not require a Section VIII certificate of authorization.

Purpose and Scope of NRC Regulatory Guide 7.6

10 CFR 71.35, "Package Evaluation," requires that packages used to transport radioactive materials meet the normal and HACs of Appendices A and B, respectively, to 10 CFR 71. This guide describes design criteria acceptable to the NRC staff for use in the structural analysis of the CVs of type B packages used to transport irradiated nuclear fuel. Alternative design criteria may be used if judged acceptable by the NRC staff in meeting the structural requirements of 10 CFR 71.35.

Purpose and Scope of NRC Regulatory Guide 7.8

10 CFR 71.71 and 10 CFR 71.73 describe normal conditions of transport and HACs that produce thermal and mechanical loads that serve as the structural design bases for the packaging of radioactive material for transport.

Initial conditions must be assumed before analyses can be performed to evaluate the response of structural systems to prescribed loads. This regulatory guide presents the initial conditions that are considered acceptable by the NRC staff for use in the structural analysis of type B packages used to transport radioactive material in the contiguous United States.

Any information collection activities mentioned in NRC Regulatory Guide 7.8 are contained as requirements in 10 CFR 71, which provides the regulatory basis for this guide.

Material Properties and Specifications

The following is taken from NNSA SG-100.

The materials selected for the fabrication of a packaging designed to transport nuclear materials are often chosen for their ability to shield and contain the radioactive contents.

However, if fissile material is being transported, the materials of construction can also be used to maintain the spatial distribution and subcriticality within the package. The materials of construction for the packaging containment system should ensure structural containment of the hazardous material. These materials of construction, supported by the material certifications and structural analysis, must be shown by design and testing to adequately contain the radioactive material.

10 CFR 71.33, "Package Description," requires applications to include a description of the proposed package in sufficient detail to identify the package accurately and provide a sufficient basis for evaluation of the package. The description must include specific information regarding the materials of construction, weights, dimensions, and fabrication methods.

10 CFR 71.31, "Contents of Application," requires the applicant to include the identification of any established codes and standards proposed for use in the package design, fabrication, assembly, testing, maintenance, and use. In the absence of any codes and standards, it requires the applicant to describe and justify the basis and rationale used to formulate the package's QAP. Although materials of construction are not specifically mentioned, their selection is an integral part of the design process.

Therefore, the codes and standards that are used to determine the properties for the materials of construction need to be identified.

10 CFR 71.31(c) provides the applicant the option of describing and justifying the basis and rationale used to formulate the package's QAP in the absence of any material codes or standards. However, the current regulatory practice has been to limit the choice of materials for vital package safety components to those that have existing codes and/or standards established by nationally recognized organizations or long-standing commercial manufacturers of products used in similar applications.

The regulatory requirement in combination with the regulatory practice compel the applicant to use materials whose characteristics and properties are well-known and documented by testing, and verified by operational experience in similar applications. This ensures that the materials selected have material properties, which serve as a basis for the various safety analyses that have been extensively tested and proven.

Any materials proposed for vital package safety components whose material properties are not described by such codes or standards must be adequately characterized and well-documented. The application should include a characterization of the materials with respect to the values of their mechanical, thermal, and physical properties; the means by which the quality of the materials is ensured; and any effects that the fabrication processes may have on the materials.

The material properties are external measures of how the materials behave in the presence of applied loads and environmental conditions such as internal or external pressure, dynamic and vibration loads, interactions with fluids, hot and cold temperatures, and so on. The objective is to use these properties to fully describe how the packaging materials will react to any combination of the applied loads and environmental conditions.

The Application of Austenitic Stainless Steel

The following is taken from the American Welding Society (AWS), *Classification of Stainless Steel.*

AUSTENITIC

Austenitic stainless steels are the most weldable of the stainless steels and can be divided rather loosely into three groups: common chromium-nickel; manganese-chromium-nickel-nitrogen; and specialty alloys. Austenitic is the most popular stainless steel group and is used for numerous industrial and consumer applications, such as in chemical plants, power plants, food processing and dairy equipment. Austenitic stainless steels have a face-centered cubic structure. Though generally very weldable, some grades can be prone to sensitization of the weld heat-affected zone (HAZ) and weld metal hot cracking.

Video 2. Austenitic stainless steel

https://www.youtube.com/watch?v=frwLBFoRj-0

FERRITIC

Ferritic stainless steel consists of iron-chromium alloys with body-centered cubic crystal structures. They can have good ductility and formability, but high-temperature strengths are relatively poor when compared to austenitic grades. Some ferritic stainless steels used, for example, in mufflers, exhaust systems, kitchen counters and sinks, cost less than other stainless steels. Other more highly alloyed steels low in chromium and nickel are more costly, but are highly resistant to chlorides.

SELECTING STAINLESS STEEL
The selection of a particular type of stainless steel will depend on what requirements a particular application poses. Environment, expected part life and extent of acceptable corrosion all help determine what type of stainless to use. In most cases, the primary factor is corrosion resistance, followed by tarnish and oxidation resistance. Other factors include the ability to withstand pitting, crevice corrosion, and intergranular attack. The austenitic/higher chromium stainless steels, usually required in very high or very low temperatures, are generally more corrosion-resistant than the lower chromium ferritic or martensitic stainless steels.

Most stainless steels are considered to have good weldability. It is important to make sure joint surfaces and any filler metal be kept free from oxide, organic material, or other contamination.

A principal concern in selecting welding filler metals for stainless steels is to match the important properties of the base metal. In addition, for nominally austenitic and duplex stainless steels, one should have some control over the weld metal's ferrite content. Specification of ferrite in nominally austenitic and duplex stainless steel welds are based upon ferrite numbers (FN) defined in AWS A4.2M/A4.2:2006, *Standard Procedures for Calibrating Magnetic Instruments to Measure the Delta Ferrite Content of Austenitic and Duplex Ferritic-Austenitic Stainless Steel Weld Metal*. Recommended by the ASME code, the magnetically determined FN is much simpler to obtain and is more reproducible than metallographically determined percent ferrite.

When selecting stainless steels, a welder must also consider something called sensitization. Ferritic stainless steels and some austenitic stainless steels, which contain appreciable free carbon can be rendered sensitive to intergranular corrosion in the HAZ of a weld. This sensitization occurs where a peak temperature of about 900 to 1,600°F is reached in the HAZ. Chromium carbides precipitate on grain boundaries, and in the process of doing so, chromium as an alloy element is depleted in the metal adjacent to the grain boundaries. Then, in corrosive service, this chromium-depleted metal is selectively attacked. Low welding heat input can limit, but not eliminate, sensitization. The best methods of preventing sensitization are selection of very low carbon base metal or selection of a grade stabilized with titanium or niobium, such as types 321 or 347. Note also that sensitization is almost never a weld metal problem—it is largely a HAZ problem.

Chemical, Galvanic, and Other Reactions
The following is taken from NNSA SG-100.

The materials selected should be such that there will be no significant chemical, galvanic, or other reaction among the package components and contents. The primary concern during NCT and HAC is thermal degradation or decomposition, particularly when plastics or other low temperature materials are used. While not normally encountered in NNSA packages, radiolysis of hydrocarbons is a concern at higher radiation levels. Decomposition can affect the shielding properties of the material and can also cause pressurization of the CV if gases are generated. The most significant concern is a case in which decomposition of materials in the shielding results in a loss of mechanical properties or a physical configuration change that may lead to a loss of shielding effectiveness.

General Requirements for all Packages per 10 CFR 71.43

The following is taken from 10 CFR 71.43.

The smallest overall dimension of a package may not be less than four inches.

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

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

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

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

A package must be designed, constructed, and prepared for shipment so that under the tests specified in 10 CFR 71.71, there would be no loss or dispersal of radioactive contents, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging.

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

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

10 CFR 71 Requirements for Lifting and Tie-Down Devices

The following is taken from 10 CFR 71.45.

LIFTING DEVICES

Any lifting attachment that is a structural part of a package must be designed with a minimum safety factor of three against yielding when used to lift the package in the intended manner, and it must be designed so that failure of any lifting device under excessive load would not impair the ability of the package to meet other requirements of 10 CFR 71.45, “Lifting and Tie-Down Standards for All Packages.” Any other structural part of the package that could be used to lift the package must be capable of being rendered inoperable for lifting the package during transport, or must be designed with strength equivalent to that required for lifting attachments.

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

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

Each tie-down device that is a structural part of a package must be designed so that failure of the device under excessive load would not impair the ability of the package to meet other requirements of 10 CFR 71.45.

b. Discuss the following related to normal conditions of transport:
- **Evaluation methods**
- **Most limiting initial test conditions**
- **Most damaging orientations**

Evaluation Methods
The following is taken from 10 CFR 71.71.

EVALUATION
Evaluation of each package design under normal conditions of transport must include a determination of the effect on that design of the conditions and tests specified in 10 CFR 71.71. Separate specimens may be used for the compression test, and the penetration test, if each specimen is subjected to the water spray test before being subjected to any of the other tests.

INITIAL CONDITIONS
With respect to the initial conditions for the tests in 10 CFR 71.71, the demonstration of compliance with the requirements of 10 CFR 71.71 must be based on the ambient temperature preceding and following the tests remaining constant at that value between -20ºF and +100ºF which is most unfavorable for the feature under consideration. The initial internal pressure within the containment system must be considered to be the maximum normal operating pressure (MNOP), unless a lower internal pressure consistent with the ambient temperature considered to precede and follow the tests is more unfavorable.

CONDITIONS AND TESTS
Heat—An ambient temperature of 100ºF in still air, and insolation according to table 9.

Table 9. Insolation data

Form and location of surface	Total insolation for a 12-hour period (g cal/cm$^{2)}$
Flat surfaces transported horizontally:	
Base	None
Other surfaces	800
Flat surfaces not transported horizontally	200
Curved surfaces	400

Source: 10 CFR 71.71

Cold—An ambient temperature of -40ºF in still air and shade.

Reduced external pressure—An external pressure of 3.5 pound force (lbf)/in.^{2} absolute.

Increased external pressure—An external pressure of 20 lbf/in.^{2} absolute.

Vibration—Vibration normally incident to transport.

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

Free drop—Between 1.5 and 2.5 hours after the conclusion of the water spray test, a free drop through the distance specified in table 10 onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected.

Table 10. Criteria for free drop test (weight distance)

Package weight		Free drop distance	
Kilograms	Pounds	Meters	Feet
Less than 5,000	Less than 11,000	1.2	4
5,000 to 10,000	11,000 to 22,000	0.9	3
10,000 to 15,000	22,000 to 33,100	0.6	2
More than 15,000	More than 33,100	0.3	1

Source: 10 CFR 71.71

CORNER DROP

A free drop onto each corner of the package in succession, or in the case of a cylindrical package onto each quarter of each rim, from a height of 0.1 foot (ft) onto a flat, essentially unyielding, horizontal surface. This test applies only to fiberboard, wood, or fissile material rectangular packages not exceeding 110 pounds (lbs) and fiberboard, wood, or fissile material cylindrical packages not exceeding 220 lbs.

COMPRESSION

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

- The equivalent of 5 times the weight of the package
- The equivalent of 2 lbf/in.^{2} multiplied by the vertically projected area of the package

PENETRATION

Impact of the hemispherical end of a vertical steel cylinder of 1.25 inch diameter and 13 lbs mass, dropped from a height of 40 inches onto the exposed surface of the package that is expected to be most vulnerable to puncture. The long axis of the cylinder must be perpendicular to the package surface.

Most Limiting Initial Test Conditions

The following is taken from NNSA SG-100.

NRC Regulatory Guide 7.9 identifies the tests that must be performed prior to the first use of the packaging: visual inspection for physical defects, structural and pressure tests, leak tests, and individual component tests. NRC Regulatory Guide 7.9 states that “components, such as gaskets, should be tested under conditions that simulate the most severe service conditions under which they are to perform.” Such testing should be performed on all components whose failure would impair the effectiveness of the packaging, not just gaskets. Clearly it is not required that all specimens of a specific component be tested, just that enough units must be tested to provide a high level of confidence that any specimen of a component specified for such service can be expected to perform properly. This is especially true for components that can only be tested in a destructive manner, such as rupture disks. It is also acceptable to test the component to criteria that exceed the most severe condition, assuming that exceeding the most severe condition will not have a detrimental effect on the performance of the package. These components include, but are not limited to, valves, rupture disks, fluid transport devices, as well as gaskets. All components must be procured under an approved QAP.

If any of the packaging components are required for shielding integrity or thermal protection, tests must be performed to ensure the shielding requirements for the packaging are met and the thermal protection maintains a temperature that is consistent with the thermal analyses and/or tests conducted for NCT.

Acceptance criteria for each test and inspection must be specified, and the packaging must not be used until all acceptance criteria are met. The acceptance and/or verification testing, inspection, and maintenance procedures must describe actions to be taken if acceptance criteria are not met. Tests and inspections performed and their acceptance criteria must be determined on the basis of the quality category of the component, or of the packaging as a whole.

Most Damaging Orientations

According to International Atomic Energy Agency (IAEA) TS-G.1-1 (ST-2), *Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material*, packages must be tested in whichever orientation results in the most damage.

According to 10 CFR 71.71, the impact of the hemispherical end of a vertical steel cylinder of 1.25 inches diameter and 13 lbs mass, dropped from a height of 40 inches onto the exposed surface of the package that is expected to be most vulnerable to puncture. The long axis of the cylinder must be perpendicular to the package surface.

According to NNSA SG-100, Rev 2, *Safety Guide: Design and Development Guide for NNSA Type B Packages*, if applicable, the effects of the dynamic crush test on a package should be described. The most unfavorable orientation should be justified.

c. Discuss the criteria that determine whether the dynamic crush test applies and describe how this test affects DOE's ability to add new contents to packages certified prior to October 2004.

The following is taken from NNSA SG-100.

10 CFR 71.73, Subsection c2 was modified by the addition of the following sentence: "For packages containing fissile material, the radioactive contents greater than 1000 A_2 criterion does not apply." (The pre-October 1, 2004, 10 CFR 71.73(c)(2) did not require this test for packages with radioactive material content less than or equal to 1,000 A_2.) 10 CFR 71.73(c)(2) now reads as follows:

> Subjection of the specimen to a dynamic crush test by positioning the specimen on a flat, essentially unyielding horizontal surface so as to suffer maximum damage by the drop of a 1,100-lb mass from 30 ft onto the specimen. The mass must consist of a solid mild steel plate 40 inches by 40 inches and must fall in a horizontal attitude. The crush test is required only when the specimen has a mass not greater than 1,100 lbs, an overall density not greater than 62.4 lb/ft^3 based on external dimension, and radioactive contents greater than 1,000 A_2 not as special form radioactive material. For packages containing fissile material, the radioactive contents greater than 1,000 A_2 criterion does not apply.

d. Describe the 10 CFR 71 tests required for hypothetical accident conditions and discuss the requirements that pertain to the sequence of HAC tests and the allowable and typical methods of demonstrating compliance.

The following is taken from 10 CFR 71.73.

Test Procedures
Evaluation for HAC is to be based on sequential application of the tests specified in 10 CFR 71.73, in the order indicated, to determine their cumulative effect on a package or array of packages.

Test Conditions
With respect to the initial conditions for the tests, except for the water immersion tests, to demonstrate compliance with the requirements of 10 CFR 71.73 during testing, the ambient air temperature before and after the tests must remain constant at that value between -29^{o}C (20^{o}F) and +38^{o}C (+100^o) which is most unfavorable for the feature under consideration. The initial internal pressure within the containment system must be the MNOP, unless a lower internal

pressure, consistent with the ambient temperature assumed to precede and follow the tests, is more unfavorable.

Tests

Tests for HACs must be conducted as follows:

- Free drop—A free drop of the specimen through a distance of 30 ft onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected.
- Crush—Subjection of the specimen to a dynamic crush test by positioning the specimen on a flat, essentially unyielding horizontal surface so as to suffer maximum damage by the drop of a 1,100-lb mass from 30 ft onto the specimen. The mass must consist of a solid mild steel plate 40 inches by 40 inches and must fall in a horizontal attitude. The crush test is required only when the specimen has a mass not greater than 1,100 lbs, an overall density not greater than 62.4 lb/ft^3 based on external dimension, and radioactive contents greater than 1,000 A_2 not as special form radioactive material. For packages containing fissile material, the radioactive contents greater than 1,000 A_2 criterion does not apply.
- Puncture—A free drop of the specimen through a distance of 40 inches in a position for which maximum damage is expected, onto the upper end of a solid, vertical, cylindrical, mild steel bar mounted on an essentially unyielding, horizontal surface. The bar must be 6 inches in diameter, with the top horizontal and its edge rounded to a radius of not more than 0.25 inch, and of a length as to cause maximum damage to the package, but not less than 8 inches long. The long axis of the bar must be vertical.
- Thermal—Exposure of the specimen fully engulfed, except for a simple support system, in a hydrocarbon fuel/air fire of sufficient extent, and in sufficiently quiescent ambient conditions, to provide an average emissivity coefficient of at least 0.9, with an average flame temperature of at least 1,475ºF for a period of 30 minutes, or any other thermal test that provides the equivalent total heat input to the package and which provides a time averaged environmental temperature of 800ºC. The fuel source must extend horizontally at least 40 inches, but may not extend more than 10 ft, beyond any external surface of the specimen, and the specimen must be positioned 40 inches above the surface of the fuel source. For purposes of calculation, the surface absorptivity coefficient must be either that value which the package may be expected to possess if exposed to the fire specified or 0.8, whichever is greater; and the convective coefficient must be that value which may be demonstrated to exist if the package were exposed to the fire specified. Artificial cooling may not be applied after cessation of external heat input, and any combustion of materials of construction must be allowed to proceed until it terminates naturally.
- Immersion, fissile material—For fissile material subject to 10 CFR 71.55, “General Requirements for Fissile Material Packages,” in those cases where water in-leakage has not been assumed for criticality analysis, immersion under a head of water of at least 3 ft in the attitude for which maximum leakage is expected.
- Immersion, all packages—A separate, undamaged specimen must be subjected to water pressure equivalent to immersion under a head of water of at least 50 ft. For test purposes, an external pressure of water of 21.7 $lbf/in.^2$ gauge is considered to meet these conditions.

e. Discuss the NNSA SG-600, *Regulatory Compliance Testing of NNSA Type B Packages*.

NNSA Service Center SG-600 is a comprehensive collection of information covering the many different subject areas that must be addressed to complete a successful package testing sequence. The purpose of NNSA SG-600 is to provide the necessary information to be used in conjunction with all applicable regulations to test packages for the transport of SNM components. The guide can also be used to test packages for other unirradiated nuclear materials. These specific radioactive materials can be shipped only in certified type B(F) containers that meet all applicable 10 CFR 71 and 49 CFR 100–178 requirements. Although much of this document has universal application, the guide is specifically focused on drum-type, type B, fissile packages used by the DOE/NNSA for transporting SNM components.

The primary goal of NNSA SG-600 is to document successful testing practices that have resulted in certified type B packages and to make the information on these practices readily available to NNSA organizations and/or individuals who can benefit from its use.

6. NNSA package certification engineers must demonstrate a working level knowledge of the thermal evaluation information required in chapter 3 of a SARP.

a. Discuss the NNSA SG-100 Section 3.2.3.2 (Physical Systems) through Section 3.2.5.2 (Successful Designs) as they pertain to thermal evaluation.

Packages used in the shipment of radioactive material must be capable of withstanding intense thermal environments while preventing the release of contents and maintaining shielding and nuclear subcriticality. Achieving this capability requires the use of construction materials that enable the package to withstand serious thermal insult.

Insulating materials are used to slow the transfer of heat toward the package content during a thermal accident condition. The material used to provide this thermal protection may also be used to increase structural stability. For packages designed to carry contents with significantly large heat sources, the issue of thermal protection becomes much more complicated. In these cases, the insulating material must work effectively to reduce the heat added to the package during an upset condition while also allowing internally generated heat to escape under regular operating conditions. These are conflicting requirements that must be carefully balanced during the package design process.

Other forms of thermal protection can be employed. Passive systems, such as heat pipes, can be used, but their viability in the event of an accident must be demonstrated before package certification. Fins can also be used to increase the rate of heat transfer from the package content to the package’s surroundings. However, fin cooling systems can also increase the rate of heating to the interior portions of the package during an accident scenario. One possibility is to make the fins of a material that is stable at lower temperatures but experiences a phase change at elevated temperatures. The problem with this design theory is that, while the fins may melt when exposed to the accident scenario outlined in 10 CFR 71, they may not melt if they were near but not fully engulfed in such an accident. In this case, it is possible for the fins to stay intact and actually increase the quantity of heat absorbed by the package.

Conduction pathways can be used to help remove heat from the CV on packages with a significant internal heat generation. Any type of high thermal conductivity material can be used as long as it does not interfere with any of the package's other functions. Special care must be taken to ensure such pathways do not allow heat an easy access to sensitive portions of the packaging during a thermal accident situation.

In general, the use of pressure vents and valves is not encouraged on packages in the NNSA. Pressure vents and valves should be used only if they are absolutely necessary and must comply with 10 CFR 71.43(e). If they are used, the concept of a fail-safe mode must be applied, and the applicant must be able to convince the regulators that a fail-safe situation does exist.

Thermophysical Properties

Several properties affect how a material or group of materials will respond to a thermal input. The discussion of these properties provided in this guide is brief, and more information can be found in most standard heat transfer texts.

Thermal conductivity is a measure of a material's ability to conduct heat. Units for this quantity are expressed in W/m-K, or more explicitly W-m/m2-K. Metals tend to be the most conductive class of materials, but conductivity within this group varies greatly. Of the common materials used for construction of packaging parts, aluminum is the most highly conductive with a thermal conductivity of about 240W/m-K. Carbon steel has a much lower conductivity of about 60 W/m-K, and stainless steels have an even lower conductivity of about 15 W/m-K. Insulating materials, as their name suggests, have a much lower thermal conductivity, typically on the order of 0.05 W/m-K, but this number is somewhat misleading. Most insulating materials employ air pockets or gaps. Through this type of material, heat transfer occurs by natural convection and radiation, not conduction. In most cases, these other heat transfer mechanisms are less efficient than conduction; thus, the overall or apparent thermal conductivity of the material is very low. In general, solids are more conductive than liquids, which are more conductive than gases. However, insulating materials (which are usually considered solids) often display thermal conductivities equal to or less than that of some liquids and gases as these materials are actually a mixture of solids and gases. Thermal conductivities of common materials can be found in most heat transfer texts. For less common materials only the thermal conductivity of a material is needed to calculate steady-state conduction heat transfer rates.

Specific heat is a measure of the amount of energy it takes to heat a specified quantity of a material a specified number of degrees. The units for this property can be expressed in kJ/(kg-K). The heat capacity of most materials increases with temperature. Density is the mass per volume of a material and units are kg/m3. For most materials, density decreases as temperature rises, although some materials may contract. All three of the material properties that have been discussed so far (thermal conductivity, heat capacity, and density) are needed to calculate non-steady state (transient) conduction heat transfer rates.

The only material property needed to calculate heat transfer rates due to radiation is emissivity (or absorbtivity). Emissivity is a measure of how efficiently a surface of a material emits radiant energy compared with an ideal radiator (blackbody). The emissivity of a material surface can vary greatly depending on the finish of the surface. While selecting a material with either a low or high emissivity due to its surface finish may help protect a package, attaining, keeping, certifying, and documenting the specified surface finish on a package part may be difficult. Most applications of radiative heat transfer dealing with a package are for the package surroundings to the package exterior (or vice versa). However, if gaps exist within the package, heat transfer across these gaps will be by radiation and convection.

Several properties must be known for the calculation of convection heat transfer coefficients, and they will always be fluid properties. The fluid properties needed to calculate the natural convection heat transfer coefficient are dependent on the correlations used, but generally the properties are fluid viscosity, density, volumetric thermal expansion coefficient, thermal conductivity, and heat capacity. For the vast majority of cases, the fluid will be air, although some other liquids, such as nitrogen and helium, are used in package design. Properties for these fluids can be found in most heat transfer texts.

One of the most important physical attributes involving thermal aspects of package design is the decomposition of materials of construction at elevated temperatures. Many forms of decomposition are possible and include dehydration, melting, pyrolysis, and combustion. Not all of these processes need to be avoided; some can serve as energy absorption mechanisms, while others tend to have degrading effects on package performance.

Dehydration is the act of vaporizing free water. The phase change of the H_2O from liquid to gas (steam) requires a considerable quantity of energy. Thus, during a thermal excursion, this process can absorb a considerable quantity of energy thereby allowing interior package temperatures to remain lower then would otherwise occur. Generally, the bulk of the steam produced is driven from the package, due to a buoyancy gradient, and escapes through vent holes or other openings possibly caused by previous HAC structural insults. Materials used for structural impact absorption and thermal insulation, such as Kaolite 1600TM and Celotex exhibit this trait. In fact, for the inorganic Kaolite 1600TM material, the only form of decomposition during thermal excursions up to about 2,300°F is the vaporization of free water from within the material matrix. Although the vaporization is usually good for a package’s thermal characteristics, there can be negative effects on the shielding and subcriticality properties of a shipping package due to the dehydration of materials of construction. This should be carefully considered when both shielding and criticality calculations are made. It is important to understand the materials of construction regarding the possibility of dehydration and to take these processes into account in subsequent calculations.

Many organic materials, such as Celotex and polyurethane foams, undergo chemical decomposition at the elevated temperatures associated with the HAC thermal event. If the decomposition occurs with oxygen present, the result is typically combustion; if oxygen is not present, the result is typically pyrolysis. Each of these processes can be either exothermic (heat releasing) or endothermic (heat absorbing), and should be examined for each material that is considered for use. Typically, the decomposition process—whether it be combustion, pyrolysis or a combination of both—is very complicated, especially for organic materials.

For most cases, the decomposition processes are not fully predictable or understood; thus, conservatism is important when dealing with this aspect of package design.

Finally, melting is typically associated with elastomeric O-rings that form the CV seal. Clearly, this type of behavior is to be avoided in package designs as melting of a containment boundary O-ring leads immediately to loss of containment. Typically, the manufacturers of the elastomeric O-rings used in drum-type packages specify a maximum continuous use temperature as well as specified durations at higher temperatures that the O-ring will survive. It may be necessary to determine, by test or analysis, the duration that a seal may be above the continuous use temperature, during the design or certification process. It should be noted that some plastic behavior has been seen in O-rings that continue to provide a leak-tight seal. Squared-off O-rings have been found in drum-type package design test units, even though prior to removal of the CV lid the O-rings provided a leak-tight containment seal.

Successful Designs

Many package designs ranging from very simple to quite complex have been successful. Packages with very small heat sources tend to have the more simplistic designs, whereas those with sizable heat sources tend to be more extravagant.

The simplest and most common package design now used in the NNSA is a stainless steel CV surrounded by a thick layer of material that acts as a thermal insulation and an impact limiter. This layer is then surrounded by the confinement drum, which is generally constructed of either mild or stainless steel.

This very basic design is usually used for packages with contents that have a reasonable heat output. If some very basic design strategies are followed, this package design can be effective and relatively inexpensive. The main concerns with this type of design are the use of the organic materials and their propensity to offgas when exposed to high temperatures. This property makes analytical modeling of the hypothetical thermal accident almost impossible unless an extremely conservative approach is taken. Offgassing notwithstanding, this design has repeatedly been able to withstand the rigors of the hypothetical thermal accident environment without loss of function.

A more sophisticated version of this design uses two layers of thermal insulation/impact limiting material. These two layers are sealed off from one another by creating compartments for the two layers. The outermost compartment contains the insulating material. In the event of a accident involving a fire, the outside wall of the insulation compartment would be exposed to very high temperatures. Heat from this exposure would be transferred into the insulating material adjacent to this wall. As this insulation is heated, decomposition processes will occur and hot gasses will be formed. As these gasses are formed, the insulation compartment will pressurize. Most of the gases will escape through vent holes provided for such an event; however, this internal pressurization will force hot gasses to the inside wall of the compartment. Due to compartmentalization, offgas from the outer most layer cannot reach the CV, but rather can reach only the outside of the inner wall of the insulation compartment. The insulation/impact limiting material used for the inner layer is different from that used on the outer layer. During physical thermal accident testing, some slight degradation of the inner thermal insulation does occur. But, in general, CVs withstand this type of test with a much smaller accumulation of offgas condensate on their outer surfaces then would the simpler design described earlier.

For some packages with significant heat sources, it is very important to spread out the heat quickly such that the content does not overheat. One method that has been employed is to use an aluminum packing material inside the CV to help carry off the heat coming from the source. Aluminum is an excellent conductor of heat and thus works very well for this purpose. Initial designs for this package called for the use of aluminum shot to be used. The package is quite small, and the weight of the shot was significant when compared with the rest of the package. As an alternative, an aluminum packing very similar to aluminum straws was tried. This packing was just as effective at removing heat at only about one-third the weight of the aluminum shot. The ability of aluminum to transfer heat so rapidly simulates a heat source the volume of the CV interior rather than just the size of the content. That is, heat generated by the content is transferred to the aluminum surrounding it. The aluminum is so efficient at conducting heat there is little temperature gradient throughout this material.

Because the entire CV is filled with either the content or the aluminum, the temperature profile throughout the entire CV cavity is nearly constant. In this way, the temperature of the content is minimized, and the heat transfer from the CV is maximized, to the extent possible for the package design being used. When employing this concept, the content is not constrained within the CV, and some shifting of the content may occur.

Another strategy used to keep heat down inside the CV is the use of aluminum conduction bars that attach directly to the content and to the lid of the CV. For this container, it was necessary to hold the content in place, and a polypropylene material was selected for this duty.

Placement of the content is critical to ensure there is good physical contact between the content and the conduction bars; therefore, the conduction bars are built into the CV lid such that good contact there is assured. This design is effective, and there is no reason why other similar designs could not be constructed. One possibility is to build an aluminum holder into the CV with the holder attached to the CV sidewall. Hence, both objectives of helping to remove heat while securing the content in a specific location are met.

b. Discuss NNSA SG-140.1, *Combination Test/Analysis Method Used to Demonstrate Compliance to DOE Type B Packaging Thermal Test Requirements*.

The following is taken from NNSA SG-140.1.

Applicants requesting certification of packaging for transport of defense programs radioactive material must, in general, show that the packaging conforms to the DOE performance requirements adopted from 10 CFR 71. The subject of DOE-SG-140.1 is the DOE interpretation of the hypothetical accident scenario described in 10 CFR 71.73, as follows:

> Thermal. Exposure of the specimen fully engulfed, except for a simple support system, in a hydrocarbon fuel/air fire of sufficient extent, and in sufficiently quiescent ambient conditions, to provide an average emissivity coefficient of at

least 0.9, with an average flame temperature of at least 800°C (1,475°F) for a period of 30 minutes, or any other thermal test that provides the equivalent total heat input to the package and which provides a time averaged environmental temperature of 800°C. The fuel source must extend horizontally at least 1 m (40 in.), but may not extend more than 3 m (10 ft), beyond any external surface of the specimen, and the specimen must be positioned 1 m (40 in.) above the surface of the fuel source. For purposes of calculation, the surface absorptivity coefficient must be either that value which the package may be expected to possess if exposed to the fire specified or 0.8, whichever is greater; and the convective coefficient must be that value which may be demonstrated to exist if the package were exposed to the fire specified. Artificial cooling may not be applied after cessation of external heat input, and any combustion of materials of construction, must be allowed to proceed until it terminates naturally.

NNSA SG-140.1 has been prepared as a guideline to ensure applicants are aware of DOE's interpretation of the requirement and to provide consistent review policy. It should be recognized by DOE regulatory certifying personnel and applicants using this guideline that other viable means of demonstrating conformance may exist and that it is not the intent of NNSA SG-140.1 to imply that any such methods may or may not be acceptable. Specifically, NNSA SG-140.1 is applicable to a combination test/analysis methodology that involves heating a package in a furnace or other enclosed radiative environment where the package and the furnace have similar geometric shapes or where dimensional considerations allow the assumption of similar geometric shape.

c. Discuss the purpose and scope of ASTM E 2230, *Standard Practice for Thermal Qualification of Type B Packages for Radioactive Material*.

The major objective of ASTM E 2230 is to provide a common reference document for applicants and certification authorities on the accepted practices for accomplishing package thermal qualification. Details and methods for accomplishing qualification are described in this document in more specific detail than available in the regulations. Methods that have been shown by experience to lead to successful qualification are emphasized. Possible problems and pitfalls that lead to unsatisfactory results are also described.

ASTM E 2230 defines detailed methods for thermal qualification of type B radioactive materials packages under 10 CFR 71 in the United States or, under IAEA regulation TS-R-1, *Regulations for the Safe Transport of Radioactive Material*. Under these regulations, packages transporting what are designated to be type B quantities of radioactive material shall be demonstrated to be capable of withstanding a sequence of hypothetical accidents without significant release of contents.

ASTM E 2230 is used to measure and describe the response of materials, products, or assemblies to heat and flame under controlled conditions, but does not by itself incorporate all factors required for fire hazard or fire risk assessment of the materials, products, or assemblies under actual fire conditions.

d. Discuss the NCT thermal analyses/tests that are evaluated in the SARP.

The following is taken from NNSA SG-100.

The following example demonstrates an actual thermal acceptance test used to evaluate new drum-type packagings. The test was developed to ensure that the heat load capabilities of the package are acceptable for NCT. A representative sample packaging is tested as follows:

- Assemble a packaging. In the CV, include a device that generates the maximum permissible heat load of the package being emulated.
- During assembly, attach thermocouples to the outside top and side of the CV and to the outside top and side of the drum surfaces.
- Place the packaging in a normal condition environment of 100°F until it reaches steady-state temperatures. Record the temperatures.
- The temperatures should not exceed 250°F on the CV and 136°F on the drum surface. If the temperatures are exceeded, notify the design agency for the packaging.

e. Discuss the HAC thermal analyses/tests that are evaluated in the SARP.

This topic is covered in KSA 3d.

f. Discuss the thermal test methods used to demonstrate compliance with 10 CFR 71 thermal requirements.

The following is taken from NNSA SG-600.

The thermal testing of a package represents a challenge to the packaging designer and test engineer. Along with this testing challenge may come some details that cannot be foreseen.

The thermal test procedure is covered in detail by ASTM Standard Practice E 2230. The ASTM E 2230 discusses the use of pool fires, furnaces, radiant heat facilities, and analysis for performing the thermal test described in 10 CFR 71.73. For pool fires and furnaces, the material covered in ASTM E 2230 is relatively conclusive. However, for radiant testing and analysis, some additional discussion is warranted.

It is paramount that applicants and testing authorities fully discuss their planned thermal testing methodology with NNSA Packaging Certification Division (PCD) early in the test planning process to allow for changes to their plans based on PCD input, if needed.

Thermal Test by Analysis

Because the vast majority of packages certified by NNSA are relatively small, NNSA expects the applicants to perform actual thermal testing, rather than a thermal analysis, for package certification.

Large spent fuel casks typically use metallic seals that do not fail even when exposed to the severe environment of the thermal test for very long periods of time. In contrast, most packages certified by NNSA use elastomeric O-rings that could fail at relatively low temperatures compared with temperatures reached in the thermal test environment. Furthermore, full-scale test units for spent fuel casks can cost millions of dollars.

Because the cost of fabricating the test units for the package designs that NNSA certifies is typically much less, NNSA expects its applicants to build prototype units and test them to all of the requirements. Therefore, it is strongly recommended that some form of physical testing, instead of computer analysis, be used to demonstrate compliance with the 10 CFR 71.73 thermal test requirements. However, computer analysis or other types of analysis may be used to make adjustments to results obtained from actual physical thermal tests performed.

Thermal Test by Radiant Heat

A radiant heat testing methodology has been developed at Sandia National Laboratories. The methodology uses a large array of very bright radiant heat lamps in proximity to the test unit to create an environment similar to that specified in 10 CFR 71.73. For a typical setup, a cylindrical package is oriented upright usually about two feet off the ground. The radiant heat lamp arrays are then placed so as to fully encircle the sides of the test unit. The heat lamps usually extend about one foot beyond the top and bottom of the test unit. The test unit surface temperature can be monitored and the radiant heat system can be controlled based on these surface temperature measurements.

However, in most cases, the open space between the test unit and the heat lamps, and the ambient air above and below the test unit, allow considerable airflow from the area external to the test rig. That is, as the air inside is heated, it rises toward the top of the test rig and out the top. This lost heated air is replaced with ambient-temperature air entering at the bottom of the test rig. This creates a convection current that carries off much of the heat imparted from the heat lamps to the package. For any test method using the radiant heat technology, consideration of the convective cooling must be taken into account, and test methods that effectively deal with the issue of airflow while still meeting all the requirements of 10 CFR 71.73 must be used and demonstrated to the satisfaction of the competent authority.

The materials of construction, especially any that are combustible, must be considered in determining the orientation in which the test unit will be placed for thermal testing, regardless of the type of thermal test method used. The possibility of chimney effects, especially in packages that have a high height-to-width ratio, should be considered. If previous testing created any breaches into the test unit's insulation, and the insulation is known to be combustible or will degrade significantly during a thermal event, the test unit should be oriented so the atmospheric gases within the thermal test can flow into the breach and out through any other available area. The orienting of the test unit is subjective; however, someone with considerable knowledge of heat transfer should be consulted to help determine the test unit's "worst case" orientation. This use of heat transfer expertise should be documented and provided as part of either the test report or the discussion in the associated SARP.

Applicants who are thinking of conducting the thermal test by the use of radiant heating should discuss their plan with PCD and get PCD's concurrence before finalizing their plan.

g. Discuss the application of fire resistant and consumable thermal insulation materials, and describe the benefits and disadvantages of each.

The following is taken from the American Society of Mechanical Engineers, *Operational Readiness of the ES-3100 Type B Shipping Container for Fissile Materials.*

The DOT specification 6M containers had been the workhorse for bulk highly enriched uranium shipping containers for DOE and many other shippers for over twenty years. This DOT-specification container was terminated for shipment of radioactive material on September 30, 2008. The anticipation of this action prompted DOE to develop and implement the ES-3100 shipping container as a replacement for the 6M. The ES-3100 was first licensed in April 2006. Since then, the license has been revised nine times. The ES-3100 was operationally ready for use at several sites by September 2007, and is now being used on a regular basis for materials that had been shipped in the DOT 6M. The ES-3100 has also been certified for air transport, in support of foreign research reactor fuel supply and international nonproliferation efforts. This container has a CoC from the NRC and a competent authority certificate from the DOT. The utility of the ES-3100 continues to grow. The ES-3100 CoC allows many forms of fissile material to be shipped, and continues to be amended to authorize additional contents for a variety of shippers.

Video 3. Thermal insulation material
https://www.youtube.com/watch?v=T24o3ZXmJDA

7. NNSA package certification engineers must demonstrate a working level knowledge of the containment information required in chapter 4 of a SARP.

a. Discuss the NNSA SG-100 Sections 4.2 through 4.8 as they pertain to containment.

Chapter 4 addresses containment for type B, radioactive material packaging systems. Requirements for type B packaging systems are contained in the applicable sections of 10 CFR 71. Containment for packaging is demonstrated through leakage rate testing. Standards for leakage rate testing of type B packages are presented in American National Standards Institute (ANSI) N14.5,*Radioactive Materials—Leakage Tests on Packages for Shipment*. This chapter provides guidance on the requirements for development and design of containment systems and on preparation of the containment section of the SARP.

This chapter identifies the regulatory requirements for containment, and describes, with examples, how those requirements may be satisfied in a packaging system.

b. Discuss the containment criteria for normal form radioactive material under NCT and HAC.

The following is taken from NNSA SG-100.

Regulatory requirements for packaging and transportation of radioactive materials, components, and special assemblies are governed by DOE Orders and various Federal regulations. This section describes the primary regulatory documents that contain the requirements for containment, including DOE O 461.1B, 49 CFR 100–185, and 10 CFR 71.

DOE O 461.1B

DOE O 461.1B, *Packaging and Transfer or Transportation of Materials of National Security Interest*, establishes requirements and responsibilities for offsite shipments and onsite transfers of naval nuclear fuel elements, category I and category II special nuclear material (SNM), nuclear components, special assemblies and other MNSI, and the certification of packages for category I and II SNM, nuclear components and other MNSI. DOE O 461.1B is applicable to all DOE elements, including the NNSA.

DOE O 461.1B requires that shipments of radioactive materials be prepared and transported in accordance with the applicable DOT hazardous materials regulations in 49 CFR 100–185.

49 CFR 100–185

49 CFR 173, "Shippers—General Requirements for Shipments and Packagings," lists general requirements for all hazardous materials and specific requirements for various classes of hazardous materials. General requirements are located in 49 CFR 173, Subpart A. 49 CFR 173.7(d) grants the DOE the authority to evaluate, approve, and certify packagings made by or under the direction of the DOE for the transportation of class 7 materials provided the DOE uses standards that are equivalent to those specified in 10 CFR 71.

Specific regulatory requirements for transportation of radioactive materials are located in 49 CFR 173, Subpart I, "Class 7 (Radioactive) Materials." In 49 CFR 173.410, "General Design Requirements," DOT specifies general design requirements for radioactive material packages. DOT also requires that type B packages be designed and constructed to meet the applicable requirements specified in 10 CFR 71.

A type B package is one designed to contain a type B quantity of radioactive material. A type B quantity is defined as any quantity of special form material having an activity of greater than A_1 or any quantity of normal form material having an activity of greater than A_2.

10 CFR 71

10 CFR 71, "Packaging and Transportation of Radioactive Material," specifies the design, construction, and testing requirements for type B packages. Containment requirements are specified in 10 CFR 71.43, which states:

> (f) A package must be designed, constructed, and prepared for shipment so that under the tests specified in 10 CFR 71 there would be no loss or dispersal of radioactive contents…
>
> (h) A package may not incorporate a feature intended to allow continuous venting during transport.

The phrase "no loss or dispersal of radioactive contents" is clarified in 10 CFR 71.51, "Additional Requirements for Type B Packages," which states:

> (a) A Type B package,…, must be designed, constructed, and prepared for shipment so that under the tests specified in: (1) Section 71.71, "Normal Conditions of Transport," there would be no loss or dispersal of radioactive contents—as demonstrated to a sensitivity of 10^{-6} A_2 per hour, no significant

increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging; and (2) Section 71.73, "Hypothetical Accident Conditions," there would be no escape of krypton-85 exceeding 10 A_2 in 1 week, no escape of other radioactive material exceeding a total amount A_2 in 1 week…

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

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

(d) For packages which contain radioactive contents with activity greater than 10^5 A_2, the requirements of 10 CFR 71.61, "Special Requirements for Type B Packages Containing More than $10^5 A_2$," must be met.

It must be demonstrated that containment, defined as a release rate of less than 10^{-6} A_2 per hour, is maintained for packages subjected to the normal conditions of transport tests specified in 10 CFR 71.71. The tests specified in 10 CFR 71.71 consist of vibration, water spray, free drop, corner drop, compression, and penetration. The 10 CFR 71.71 tests are to be conducted at the most unfavorable conditions of external temperature and pressure.

Packages must also demonstrate that containment, defined as a release of less than 1 A_2/wk (except krypton-85 may be released at up to 10 A_2/wk), is maintained for packages subjected to the HAC tests specified in 10 CFR 71.73. The tests defined in 10 CFR 71.73 must all be applied to a single package in the specified sequence. The test sequence is a 30-ft free drop, followed by crush, puncture, and fire. The tests are to be performed at the most unfavorable temperature, and the initial pressure within the containment system must be the MNOP unless a lower internal pressure is more unfavorable. An immersion test is also required, but may be performed on an undamaged test specimen.

There are important changes to the containment requirements included in the 10 CFR 71 version effective October 1, 2004. Each of the primary changes affecting containment requirements is discussed below.

10 CFR 71.61 was renamed and revised to specify requirements for the packages identified in its title.

Earlier versions of 10 CFR 71 applied this requirement only to irradiated nuclear fuel shipments. 10 CFR 71.61 now reads as follows:

A type B package containing more than 10^5 A_2 must be designed so that its undamaged containment system can withstand an external water pressure of 290 psi for a period of not less than 1 hour without collapse, buckling, or in-leakage of water.

10 CFR 71.63, "Special Requirements for Plutonium Shipments," was modified to eliminate

the double containment requirement for plutonium shipments. The section now reads as follows:

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

10 CFR 71.73(c)(2), "Crush," was modified by the addition of the following sentence: "For packages containing fissile material, the radioactive contents greater than 1,000 A_2 criterion does not apply." (The pre-October 1, 2004, 10 CFR 71.73 did not require this test for packages with radioactive material content less than or equal to 1,000 A_2.) 10 CFR 71.73(c)(2) now reads as follows:

> Subjection of the specimen to a dynamic crush test by positioning the specimen on a flat, essentially unyielding horizontal surface so as to suffer maximum damage by the drop of a 1,100-lb mass from 30 ft onto the specimen. The mass must consist of a solid mild steel plate 40 inches by 40 inches and must fall in a horizontal attitude. The crush test is required only when the specimen has a mass not greater than 1,100 lbs, an overall density not greater than 62.4 lb/ft^3 based on external dimension, and radioactive contents greater than 1,000 A_2 not as special form radioactive material. For packages containing fissile material, the radioactive contents greater than 1,000 A_2 criterion does not apply.

c. Discuss the following related to ANSI N14.5:

- **Different methods for leak testing a small drum-type package**
- **Standard(s) utilized by DOE for leak tests of type B radioactive materials**
- **Packages and the following containment boundary test requirements for: 1) design verification, 2) fabrication, 3) pre-shipment, 4) periodic, and 5) maintenance**

Different Methods for Leak Testing

The following is taken from NNSA SG-100.

ANSI N14.5-1997, Annex A lists eleven accepted leakage test methods with corresponding test sensitivities. Table A.2 of Annex A lists the accepted tests and nominal sensitivities. The table is repeated here as table 11. Summaries of these descriptions are provided after the table.

It is important to design the containment system closures, valves, seals, etc. with the required leakage testing in mind. It should be understood that the reference leakage rate, LR, will likely limit the available test method. For example, if the reference leakage rate is determined to be leak-tight (10^{-7} reference cubic centimeter per second [ref-cm^3/s]), the only leakage rate tests with acceptable sensitivities are gas-filled envelope and evacuated envelope (both techniques use a tracer gas [generally helium] placed on one side of a seal and a vacuum on the other side). In the gas-filled envelope and evacuated envelope techniques, mass spectrometry is used to detect the leakage of the tracer gas. It is also important to note that the current sensitivity of helium mass spectrometry, depending on the method and test conditions, can be orders of magnitude greater than listed in table 11.

The other tests described here may be used for packages where less sensitivity is acceptable and for pre-shipment leakage testing, which only requires detection of a leakage rate of 10^{-3} ref-cm^3/s.

Table 11. Accepted leakage test and sensitivities

Test method	Nominal test sensitivity[a]	
	ref-cm^3/s	Pa-m^3/s
Gas pressure drop[b,c]	$10^{-1} – 10^{-5}$	$10^{-2} – 10^{-6}$
Gas pressure rise[b,c]	$10^{-1} – 10^{-5}$	$10^{-2} – 10^{-6}$
Gas filled envelope (gas detector)	$10^{-3} – 10^{-8}$	$10^{-4} – 10^{-9}$
Evacuated envelope (gas detector)	$10^{-3} – 10^{-8}$	$10^{-4} – 10^{-9}$
Evacuated envelope (with back pressurization)	$10^{-3} – 10^{-8}$	$10^{-4} – 10^{-9}$
Gas bubble techniques		
Hot water bubble[d,e]	10^{-3}	10^{-4}
Vacuum bubble[d,e]	10^{-3}	10^{-4}
Pressurized cavity bubble[d,e]	10^{-3}	10^{-4}
Soap bubble[d,e]	10^{-3}	10^{-4}
Tracer gas (sniffer technique)	$10^{-3} – 10^{-6}$	$10^{-4} – 10^{-7}$
Tracer gas (spray method)	$10^{-3} – 10^{-6}$	$10^{-4} – 10^{-7}$

[a]For comparison purposes only. The listed values are referred to a common standard of dry air at 25ºC and a pressure of 1 atm. Bubble test sensitivities are for 1 atm differential pressure except for the hot water bubble, which is for 0.25 atm differential.
[b]Sensitivity depends upon volume tested, test time, and instrument sensitivity to pressure differentials.
[c]The listed sensitivity is typical. The actual sensitivity when applied to a containment system may be calculated using ANSI N14.5-1997, Annex B.
[d]The listed sensitivity applies to test methods under normal field conditions. Under favorable, well-controlled conditions, this sensitivity could be increased by a factor of 10.
[e]For tests with a liquid-gas interface, account should be taken of surface tension and the hydrostatic head of the liquid bath (see ANSI N14.5-1997, Annex B).
Source: NNSA SG-100

GAS PRESSURE DROP

This method involves pressurizing the test item cavity or interspace and measuring the pressure drop. This method is particularly useful for testing double O-ring seals, where the small interspace volume between the seals makes the method most sensitive and the primary seal of the cavity does not have to be broken. The sensitivity of this method is inversely proportional to the test volume. To have a sensitive test using this method, it is critical to accurately measure the volume of the test cavity. Measured leakage rates in pressure drop and pressure rise tests using double O-rings may be actually higher than the actual leakage rate from the containment boundary because the measured leakage rate is across both of the O-ring seals as opposed to only across the containment boundary seal.

GAS PRESSURE RISE

This method involves evacuating the test cavity and measuring a pressure rise during a specified time period. This method applies to test items with pressure tap connections, but can also be used for testing double O-ring seals. Again, test sensitivity is inversely proportional to test volume, and it is critical for a sensitive test to accurately measure the volume of the test cavity. Measured leakage rates in pressure drop and pressure rise tests using double O-rings may be actually higher than the actual leakage rate from the containment boundary because the measured leakage rate is across both of the O-ring seals as opposed to only across the containment boundary seal.

GAS-FILLED ENVELOPE

This method involves evacuating the test item connected to a gas detector and surrounding the item in an envelope filled with a tracer gas. This method is suitable for large test items that have a replaceable seal. The detector typically used for this test is a mass spectrometer that has the ability to detect the tracer gas at low concentrations. This test method can have sufficient sensitivity to verify a leak-tight seal. Because the containment void volume is typically evacuated in this test method, the connection point for the vacuum must also be tested.

EVACUATED ENVELOPE (GAS DETECTOR)

This method involves pressurizing the test item with a tracer gas while the item is placed in a vacuum chamber connected to a gas detector. This method is ideal for small test items that have a replaceable seal. This test method can have sufficient sensitivity to verify a leak-tight seal. This test method is often applied to packages with two concentric seals; the inner seal forms the containment boundary, and the outer seal establishes an interspace for testing. It is important to ensure both seals are not evaluated simultaneously. Additionally, the designer should ensure the two seal grooves communicate in this type configuration.

EVACUATED ENVELOPE (WITH BACK PRESSURIZATION)

This method involves externally pressurizing the test item in an envelope of the tracer gas for a period of time and subsequently transferring the test item to an evacuated envelope connected to a gas detector. This method is ideal for welded capsules from very small sizes up to the sizes limited by the pressurizing chamber. This method can have sufficient sensitivity to verify a leak-tight seal.

HOT WATER BUBBLE

This method involves submerging a room-temperature test item in hot water, which raises the internal pressure. A leak is indicated by a stream of bubbles. This method applies to welded capsules and small test items, usually without pressure tap connections. It can be used in the field without sophisticated equipment. This method provides sufficient sensitivity for pre-shipment leakage rate testing where a sensitivity of 1×10^{-3} ref-cm^3/s is acceptable.

VACUUM BUBBLE

This method involves immersing the test item in a liquid and then producing a vacuum above the liquid in which the test item is submerged. A leak is indicated by a stream of bubbles. The method is suitable for welded capsules and small resealable items. This method provides sufficient sensitivity for pre-shipment leakage rate testing where a sensitivity of 1×10^{-3} ref-cm^3/s is acceptable.

PRESSURIZED CAVITY BUBBLE
This method involves pressurizing the test item and immersing it in a liquid. A leak is indicated by a stream of bubbles. This method applies to welded capsules, containers with pressure tap connections, or where the pressure required within the cavity can be obtained by the vaporization of solid carbon dioxide. This method provides sufficient sensitivity for pre-shipment leakage rate testing where a sensitivity of 1 x 10^{-3} ref cm^3/s is acceptable.

SOAP BUBBLE
This method involves pressurizing the test item and coating the surface with a soap film or other viscous material. A leak is indicated by a soap bubble on the surface. This method applies to containers with pressure tap connections. The required pressure within the cavity can be obtained by the vaporization of solid carbon dioxide. Care must be taken to not exceed the test item design pressure. This method provides sufficient sensitivity for pre-shipment leakage rate testing where a sensitivity of 1 x 10^{-3} ref cm^3/s is acceptable.

TRACER GAS (SNIFFER TECHNIQUE)
This method involves pressurizing the test item with the tracer gas. A leak is detected by moving a probe across the areas that are likely to leak. This method is best used on large items where the area of potential leak is clearly visible. There must be some facility for pressurizing the inside of the weld or seal.

TRACER GAS (SPRAY METHOD)
This method involves evacuating a test item connected to a gas and spraying the tracer gas over the surface. This method can be used for testing partially finished vessels provided that one side of a potential leak can be evacuated and the other side is easily accessible with a supply of tracer gas.

Standard(s) Utilized by DOE for Leak Tests of Type B Radioactive Material Packages

10 CFR 34.27, “Leak Testing and Replacement of Sealed Sources.”

NUREG 1609, *Standard Review Plan for Transportation Packages for Radioactive Material*, provides guidance for the review and approval of applications for packages used to transport radioactive material under 10 CFR 71.

NRC Regulatory Guide 7.4, *Leakage Tests on Packages for Shipment of Radioactive Material*, endorses the methods and procedures developed by the ANSI Standards Committee on Packaging and Transportation of Radioactive and Nonnuclear Hazardous Materials in ANSI N14.5-1997 issued in 1997 and reaffirmed in 2008, as a process that the NRC staff considers acceptable for meeting the regulatory requirements.

The IAEA SSR-6, *Regulations for the Safe Transport of Radioactive Material* specify permitted release of radioactivity under normal and accidental conditions of transport, in terms of activity per unit of time, for type B packaging used to transport radioactive materials. Generally, it is not practical to measure activity release directly. The usual method used is to relate activity release to nonradioactive fluid leakage, for which several leakage test procedures are available. The appropriate procedure will depend on its sensitivity and its application to a specific package.

ISO 12807, *Safe Transport of Radioactive Materials—Leakage Testing on Packages*, specifies gas leakage test criteria and test methods for demonstrating that packages used to transport radioactive materials comply with the package containment requirements defined in reference of annex F for

- design verification
- fabrication verification
- pre-shipment verification
- periodic verification

The regulations specify permissible activity release for normal and accidental conditions of transport. These activity release limits can be expressed in maximum permissible activity release rates for the radioactive material carried within a containment system.

In general, it is not feasible to demonstrate that the activity release limits are not exceeded by direct measurement of activity release. In practice, the most common method to prove that a containment system provides adequate containment is to carry out an equivalent gas leakage rate test.

Packages, and the Following Containment Boundary Test Requirements for 1) Design Verification. 2) Fabrication, 3) Pre-shipment, 4) Periodic, and 5) Maintenance

DESIGN VERIFICATION

The following is taken from NNSA SG-100.

The selected method for verification may include testing, analysis, or a combination of the two. A detailed verification plan should be developed that addresses the methods of achieving the packaging requirements. The verification plan should include specific tests or analysis plan requirements. The test or analysis plan should then be followed by detailed test or analysis plan procedures. The test or analysis acceptance criteria must be clearly stated in the plans and procedures. The development of the verification plan, and the test or analysis plan and the associated procedures fulfill the requirements of 10 CFR 71.123, “Test Control,” by establishing measures to ensure applicable test programs are accomplished in accordance with written procedures. The verification plan and test or analysis plan identify the appropriate test prerequisites that are in turn properly translated into test procedures.

Test results may be presented on calculation data sheets, test results data forms, photographs, videos, computer-generated graphics, plots, and charts. To meet the requirements of 10 CFR 71.123, measures must be established to ensure test results are documented, evaluated, and maintained as QA records. These records must be readily available should questions arise concerning operational aspects of the packages. The acceptability of the records should be determined by a qualified individual or group. Test results should be documented in a formal test report that will be used in the SARP either directly or by reference.

FABRICATION

The following is taken from NNSA SG-100.

Methods for fabricating the packaging will usually depend on the quantity of packages made, schedule for completion, and funding available. Complicated designs may require intricate fabrication techniques.

The applicant's QAP should detail the approach to the control of purchased items and services to fulfill the requirements of 10 CFR 71.115, "Control of Purchased Material, Equipment, and Services," and 10 CFR 71.109, "Procurement Document Control." Vendors should be carefully selected based on their capability to comply with applicable sections of 10 CFR 71, Subpart H, their facility and QAP, and their previous records and performance. Vendor evaluations are to be performed before the vendor is released for production. The evaluation of the QAP should be performed by qualified QA personnel from the applicant's organization prior to initiation of activities affected by the program. Depending on the critical nature of the packaging and the vendor's performance, the applicant should perform the following activities for each vendor to address the requirements of 10 CFR 71.115(a):

- Evaluate and select packaging component vendors
- Establish controls to be imposed on these vendors
- Perform audits at vendors' facilities (as required)
- Establish conditions for the receiving inspection.

Procurement documents provided to the vendor will supply the design basis technical requirements, including the applicable regulatory requirements; material and component identification requirements; drawings; specifications; applicable codes and standards; special process instructions; and test and inspection requirements for the components to be fabricated. Procurement documents should also identify the records to be retained, controlled, and maintained by the vendor and the records to be delivered to the applicant prior to installation of hardware. These records should include the pertinent documentation to be furnished with the procured materials or services. If the pertinent documentation is in an electronic format, the software system the documentation is to be delivered in should be specified. If the item requires inspection and/or certification by the applicant, then documentation of such inspections and/or certifications should also be entered into the QA records. If the packaging components are manufactured in-house, then similar steps should be conducted for those manufacturing elements.

In some cases, to ensure adequate quality, it may be necessary to require that, prior to fabrication, the vendor or in-house shop supply a sample of the component to be fabricated for a first article evaluation (FAE). This requirement should be specified in the procurement documents. The component will be subjected to verification tests and inspections that are specified in the procurement documents to ensure the article was fabricated in accordance with its design and that the manufacturing process was properly performed. If the FAE is acceptable, the vendor will be released to begin production.

PRE-SHIPMENT
The following is taken from NNSA SG-100.

The purpose of packaging pre-shipment leakage rate testing is to confirm the containment system is properly assembled for shipment. Pre-shipment leakage rate testing must be performed before each shipment, after the contents are loaded and the containment system is fully assembled.

Pre-shipment leakage rate testing is required only for the containment boundary areas that have been opened during the unload-loading cycle. (Note: NNSA has determined that CV seal (O-ring), CV washer and/or CV fastener replacement is not necessarily a maintenance activity for NNSA certified type B packages.)

The acceptance criterion for pre-shipment leakage rate testing shall be the greater of either 1) a leakage rate of not more than the reference air leakage rate, LR, or 2) no detected leakage when tested to a sensitivity of at least 10^{-3} ref-cm3/s. Pre-shipment leakage rate tests for packages that have non-reusable seals, or have reusable seals that have been replaced, shall demonstrate a leakage rate according to the guidance for maintenance leakage rate testing.

PERIODIC
The following is taken from NNSA SG-100.

The purpose of periodic leakage rate testing is to confirm that the containment capabilities of packagings built to an approved design have not deteriorated during a period of use. Periodic testing must be performed within 12 months prior to the initiation of each shipment. Periodic leakage rate testing need not be performed for out-of-service packagings, but must be performed prior to placing the packaging back in service. Periodic leakage rate testing must be performed for all containment boundary seals, closures, valves, rupture disks, etc. Periodic leakage rate testing does not need to include inaccessible surfaces. If the contribution of the individual component leakage rate to the packaging leakage rate is unknown, the periodic leakage rate testing must be performed for the entire containment boundary.

The acceptance criterion for packaging fabrication leakage rate testing is the reference air leakage rate.

MAINTENANCE
The following is taken from NNSA SG-100.

The purpose of the maintenance leakage rate test is to confirm that any maintenance, repair, or replacement of components has not degraded the containment system. Maintenance leakage rate testing must be performed prior to returning a package to service following maintenance, repair, or replacement of components of a containment system. Maintenance leakage rate testing need only be performed on the affected area of the package.

Reuse of components like removable lids, O-rings, and closure bolts is not considered replacement. The containment boundaries of some packaging systems have multiple port openings. The total leakage rate from the package is the sum of the leakage rates from the multiple openings. If the contribution of the individual component leakage rate to the packaging leakage rate is unknown, the maintenance leakage rate testing must be performed for the entire containment boundary.

The acceptance criterion for packaging fabrication leakage rate testing is the reference air leakage rate.

d. Discuss the following:

- **The definition of A_1 and A_2 and the units commonly used**
- **How to calculate A_2 for a mixture of radionuclides, and how to calculate the number of A_2s in the package contents using the guidance in 10 CFR 71**
- **The definition of a type A and type B package relative to the quantity of the content**
- **The definition of exclusive use and highway route control, and how these declarations alter package transport, whether by the DOE or for the DOE by a commercial carrier**
- **The definition of normal, depleted, and enriched uranium based on isotopic content**
- **The definition of weapons grade, reactor grade, and heat source grade plutonium**

The Definition of A_1 and A_2 and the Units Commonly Used

The following is taken from NNSA SG-100.

A_1 is the maximum activity of special form radioactive material permitted in a type A package. This value is either listed in 10 CFR 71, table A-1, or can be derived in accordance with the procedures prescribed in 10 CFR 71, Appendix A.

A_2 is the maximum activity of radioactive material, other than special form radioactive material, permitted in a type A package. This value is either listed in 10 CFR 71, table A-1, or can be derived in accordance with the procedures prescribed in 10 CFR 71, Appendix A.

The following is taken from 10 CFR 71, Appendix A.

The Ci values specified are obtained by converting from the TBq value. The TBq values are the regulatory standard. The Ci values are for information only and are not intended to be the regulatory standard. Where values of A_1 and A_2 are unlimited, it is for radiation control purposes only. For nuclear criticality safety, some materials are subject to controls placed on fissile material.

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

a. For special form radioactive material, the maximum quantity transported in a type A package is as follows:

$$\sum_{1} \frac{B(i)}{A_1(i)} \leq 1$$

b. For normal form radioactive material, the maximum quantity transported in a type A package is as follows:

$$\sum B(i)/A_2\,(i) \leq 1$$

where B(i) is the activity of radionuclide i, and A2(i) is the A_2 value for radionuclide i.

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

$$A_1 \text{ for mixture} = \frac{1}{\sum_1 \frac{f(i)}{A_1(i)}}$$

where f(i) is the fraction of activity for radionuclide I in the mixture, and A_2 (i) is the appropriate A_2 value for radionuclide I.

d. Alternatively, the A_2 value for mixtures of normal form material may be determined as follows:

$$A_2 \text{ for mixture} = \frac{1}{\sum_1 \frac{f(i)}{A_2(i)}}$$

where f(i) is the fraction of activity for radionuclide I in the mixture, and A_2(i) is the appropriate A_2 value for radionuclide I.

e. The exempt activity concentration for mixtures of nuclides may be determined as follows:

$$\text{Exempt activity concentration for mixture} = \frac{1}{\sum_1 \frac{f(i)}{[A](i)}}$$

where f(i) is the fraction of activity concentration of radionuclide I in the mixture, and [A] is the activity concentration for exempt material containing radionuclide I.

f. The activity limit for an exempt consignment for mixtures of radionuclides may be determined as follows:

$$\text{Exempt consignment activity limit for mixture} = \frac{1}{\sum_1 \frac{f(i)}{A(i)}}$$

where f(i) is the fraction of activity of radionuclide I in the mixture, and A is the activity limit for exempt consignments for radionuclide I.

When the identity of each radionuclide is known, but the individual activities of some of the radionuclides are not known, the radionuclides may be grouped, and the lowest A_1 or A_2 value, as appropriate, for the radionuclides in each group may be used in applying the formulas. Groups may be based on the total alpha activity and the total beta/gamma activity when these are known, using the lowest A_1 or A_2 values for the alpha emitters and beta/gamma emitters.

The Definition of a Type A and Type B Package Relative to the Quantity of the Content

The following definitions are taken from NNSA SG-100.

TYPE A

A shipping package used to ship radioactive material having an aggregate radioactivity not greater than that specified in table A-1 of 10 CFR 71 or determined by procedures described in Appendix A of 10 CFR 71.

TYPE B

A shipping package used to ship radioactive materials in excess of the type A definition.

The Definition of Exclusive Use and Highway Route Control, and How These Declarations Alter Package Transport, Whether by the DOE or for the DOE by a Commercial Carrier

The following is taken from NNSA SG-100.

EXCLUSIVE USE

Exclusive use is the sole use of a conveyance by a single consignor and for which all initial, intermediate, and final loading and unloading are carried out in accordance with the direction of the consignor or consignee.

HIGHWAY ROUTE CONTROL

The following is taken from 49 CFR 173.403.

Highway route controlled quantity means a quantity within a single package which exceeds

- 3,000 times the A_1 value of the radionuclides as specified in 49 CFR 173.435 for special form class 7 (radioactive) material;
- 3,000 times the A_2 value of the radionuclides as specified in 49 CFR 173.435 for normal form class 7 (radioactive) material; or
- 1,000 TBq (27,000 Ci), whichever is least.

The following is taken from 49 CFR 397.101.

A carrier or any person operating a motor vehicle containing a highway route controlled quantity of class 7 (radioactive) materials, as defined in 49 CFR 173.403, shall operate the motor vehicle only over preferred routes.

A preferred route is an interstate system highway for which an alternative route is not designated by a state routing agency; a state-designated route selected by a state routing agency pursuant to 49 CFR 397.103, “Requirements for State Routing Designations,” or both of the above.

The motor carrier or the person operating a motor vehicle containing a highway route controlled quantity of class 7 (radioactive) materials, as defined in 49 CFR 173.403, shall select routes to reduce time in transit over the preferred route segment of the trip.

An interstate system bypass or interstate system beltway around a city, when available, shall be used in place of a preferred route through a city, unless a state routing agency has designated an alternative route.

The Definition of Normal, Depleted, and Enriched Uranium Based on Isotopic Content

The following definitions are taken from the U.S. Nuclear Regulatory Commission, *Glossary*.

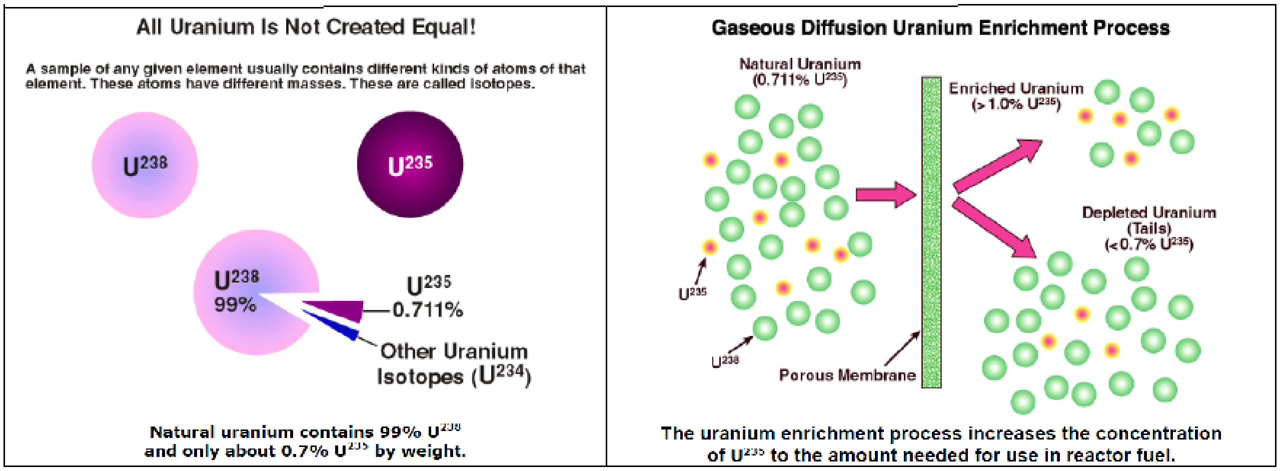
NORMAL (NATURAL) URANIUM
Uranium containing the relative concentrations of isotopes found in nature (0.7 percent U-235, 99.3 percent U-238, and a trace amount of U-234 by mass). In terms of radioactivity, however, natural uranium contains approximately 2.2 percent U-235, 48.6 percent U-238, and 49.2 percent U-234. Natural uranium can be used as fuel in nuclear reactors.

DEPLETED URANIUM
Uranium with a percentage of U-235 lower than the 0.7 percent (by mass) contained in natural uranium. (The normal residual U-235 content in depleted uranium is 0.2–0.3 percent, with U-238 comprising the remaining 98.7–98.8 percent.) Depleted uranium is produced during uranium isotope separation and is typically found in spent fuel elements or byproduct tailings or residues. Depleted uranium can be blended with highly enriched uranium, such as that from weapons, to make reactor fuel.

ENRICHED URANIUM
The following is taken from the U.S. Nuclear Regulatory Commission, *Uranium Enrichment*.

Enriching uranium increases the proportion of uranium atoms that can be split by fission to release energy (usually in the form of heat) that can be used to produce electricity. Not all uranium atoms are the same. When uranium is mined, it consists of about 99.3 percent U-238, 0.7 percent U-235, and < 0.01 percent U-234. These are the different isotopes of uranium, which means that while they all contain 92 protons in the atom's center, or nucleus (which is what makes it uranium), the U-238 atoms contain 146 neutrons, the U-235 atoms contain 143 neutrons, and the U-234 atoms contain only 142 neutrons. (The total number of protons plus neutrons gives the atomic mass of each isotope—that is, 238, 235, or 234, respectively.)

The fuel for nuclear reactors has to have a higher concentration of U-235 than exists in natural uranium ore. This is because U-235 is fissionable, meaning that it starts a nuclear reaction and keeps it going. Normally, the amount of the U-235 isotope is enriched from 0.7 percent of the uranium mass to about 5 percent, as illustrated in figure 3.



Source: U.S. Nuclear Regulatory Commission, Uranium Enrichment

Figure 3. Uranium enrichment

Video 4. Uranium enrichment
https://www.youtube.com/watch?v=pl_E3aIL7G0

The Definition of Weapons Grade, Reactor Grade, and Heat Source Grade Plutonium

WEAPONS GRADE

The following is taken from Wikipedia, *Weapons Grade*.

A weapons-grade substance is one that is pure enough to be used to make a weapon or has properties that make it particularly suitable for weapons use. Plutonium and uranium in grades normally used in nuclear weapons are the most common examples. (These nuclear materials have other categorizations based on their purity.)

Only fissile isotopes of certain elements have the potential for use in nuclear weapons. For such use, the concentration of fissile isotopes U-235 and Pu-239 in the element used must be sufficiently high. Uranium from natural sources is enriched by isotope separation, and plutonium is produced in a suitable nuclear reactor.

REACTOR GRADE

The following is taken from the Federation of American Scientists, *Plutonium Production.*

Reactor-grade plutonium is produced in the core of a reactor when U-238 is irradiated with neutrons. Unlike weapon grade plutonium (which is relatively pure Pu-239), reactor grade plutonium is a mixture of Pu-238, 239, 240, 241, and 242. It is this mixture of isotopes which renders reactor grade plutonium less suitable as a weapon-grade material.

The even numbered isotopes (Pu-238, 240, and 242) fission spontaneously producing high-energy neutrons and a lot of heat. In fact, the neutron and gamma dose from this material is significant and the heat generated in this way would melt the high-explosive material needed to compress the critical mass prior to initiation. The neutrons can also initiate a premature chain reaction thus reducing the explosive yield, typically to a few percent of the nominal yield, sometimes called the “fizzle yield.” Such physical characteristics make reactor-grade plutonium extremely difficult to manipulate and control and therefore explains its unsuitability as a bomb-making ingredient.

The odd numbered isotope, Pu-241, is also a highly undesirable isotope as it decays to americium-241, which is an intense emitter of alpha particles, X, and gamma rays. Pu-241 has a half-life of 13.2 years which means americium-241 accumulates quickly, causing serious handling problems.

HEAT SOURCE GRADE PLUTONIUM
The following is taken from DOE-HDBK-1145-2013.

Heat-source plutonium has the highest Pu-238 content and can be produced by exposing U-235 to neutron bombardment until U-237 is formed. U-237 has a short half-life (6.75 days) and decays to long-lived (2 million years) neptunium (Np)-237. Neutron activation of Np-237 produces Np-238, which then decays to Pu-238.

For a nuclide to be used for thermal (heat) energy, it has to have a half-life greater than 100 days, but less than 100 years. If the half-life is less than 100 days, the nuclide needs to be replenished often. If the half-life is greater than 100 years, the decay rate (activity) is not high enough to create enough heat to be considered a good heat source.

The half-life of Pu-238 is short enough (88 years) to create a high heat output and long enough to provide long-term power without replenishment. These characteristics make it an ideal heat source for thermoelectric generators. These generators have been used to power ocean buoys and space satellites where long-term, reliable power is essential.

Video 5. Plutonium
https://www.youtube.com/watch?v=89UNPdNtOoE

8. NNSA package certification engineers must demonstrate a working level knowledge of the shielding evaluation information required in chapter 5 of a SARP.

a. Discuss NNSA SG-100, Sections 5.2 through 5.8 as they pertain to shielding.

The radiation shielding aspects of the weapon components and special assembly packaging design guides are concerned with establishing that the radiation dose rate limits on the package exterior are not exceeded. DOE requires the application of relevant Federal regulations to ensure the protection of the public safety and health and the environment from the inherent risks of the public transportation of nuclear weapon components, special assemblies, and radioactive material.

The purpose of NNSA SG-100, Section 5, is to aid in the identification and efficient resolution of any radiation shielding issues arising from the public domain transportation of radioactive material associated with that portion of the U.S. nuclear weapons program under the control of DOE. NNSA SG-100 supports the shipment of type B quantities of dispersible forms of radioactive material and SNM.

The contents of NNSA SG-100 follow closely the packaging requirements for safe transport of SNM. The following list of recommendations for preparing a SARP are given with reference to the section of this chapter in which they are discussed.

- Fully describe the bounding source configuration, including the justification for why the source is bounding. Describe the parameters used to generate the source and indicate any sources that were omitted and justification for the omission. (Section 5.2)
- Provide a complete description of the packaging including physical dimensions, material compositions, and material densities. (Section 5.3)
- Provide a description, including sketches with dimensions and materials, of the calculational models. Note differences between calculational models and the actual package design, and discuss how these differences affect the results of the calculations. (Section 5.3)
- Clearly present summary dose rate information for both normal and accident conditions, indicating the limiting locations. (Sections 5.4 and 5.5)
- Provide a description of the code(s), cross-section data, and flux-to-dose conversion factors used in the analysis, together with references that provide complete information. Discuss software capabilities and limitations, and how the resultant calculation may be affected (see Section 4 of Reference 1 for more details). (Sections 5.6 and 5.7)
- Provide sufficient information in the application to support independent analyses. (Section 5.8)

b. Discuss the maximum allowable dose rates for NCT, and the impact of package measurement locations on dose rate.

The following is taken from NNSA SG-100.

Maximum Surface Dose Rate for NCT

A package containing radioactive material must be designed and prepared for shipment such that under NCT, the radiation dose rate does not exceed 200 mrem/h at any point on the external surface of the package, as specified by 10 CFR 71.47, “External Radiation Standards for All Packages,” 49 CFR 173.441, “Radiation Level Limitations and Exclusive Use Provisions,” and in IAEA Safety Standards Series TS-R-1, Section VI, “Requirements for Radioactive Materials and for Packagings and Packages.”

Maximum One-Meter Dose Rate for NCT

The maximum dose rate at one meter from any external surface position of the package under NCT must not exceed 10 mrem/h, as specified in 10 CFR 71.47, using the definition of TI in 10 CFR 71.4, “Definitions.”

Exclusive-Use Conveyance, Long-term Storage, and ALARA

The NCT dose rate limits apply to a shipping package without regard to the method of shipment. If the package is shipped as exclusive use, the NRC limits can be relaxed to take into account the material and geometric shielding properties of the conveyance vehicle. A maximum package external dose of 1,000 mrem/h for NCT is allowed in a closed vehicle if 200 mrem/h limit is met on the external surface of the vehicle. The details of the exclusive-use limits are given in 10 CFR 71.47.

All DOE MNSI Packages are Shipped in Exclusive-use DOE Conveyance

However, the DOE policy of ALARA for external package dose rates can be interpreted so as to not allow the higher NRC exclusive-use limit, and, generally, all DOE MNSI shipments comply with the limits for NCT and HAC. Most probably, the exclusive-use limit of 1,000 mrem/h would exceed the local plant radiation limits where the packages are assembled before shipment and unpacked after shipment.

The ALARA requirement can also be examined with respect to the nonexclusive use dose rate limits. The design of most packages for the shipment of weapon components and special assemblies is usually dictated by considerations from structural, thermal, containment, and criticality aspects.

Generally, no specific shielding materials are included in the package, but the external dose rates will in most cases be much lower than the nonexclusive-use dose rate limits. The dose rates could conceivably be reduced to or near background levels by including appropriate liners of lead, steel, or other shielding materials into the package designs. However, this package enhancement might be considered unreasonable considering the increased package weight and the increased cost of the package fabrication and shipping procedure.

The ALARA concept takes on greater importance if the shipping packages or the CVs are also to be used for storage and/or several packages are clustered in the same general location. Under these circumstances, the combined doses from all of the packages, each of which meets the nonexclusive-use dose rate limits individually, may pose a radiation hazard.

The 10 CFR 71.47 requirements state only that the dose rates on the conveyance exterior must not exceed 200 mrem/h; they do not address the combined package dose rates in the conveyance interior. The possible use of shipping packages for storage must be examined to determine if additional shielding is needed for the storage facility to be in compliance with 10 CFR 20, “Standards for Protection Against Radiation.”

c. Describe how the transport index is determined.

The following is taken from 10 CFR 71.4.

TI means the dimensionless number placed on the label of a package, to designate the degree of control to be exercised by the carrier during transportation. The TI is the number determined by multiplying the maximum radiation level in millisieverts per hour at 3.3 feet from the external surface of the package by 100.

d. Discuss typical modeling assumptions applied for DOE/NNSA type B packages.

The following is taken from NNSA SG-100.

Conservative modeling assumptions should be used with caution in nonlinear, dynamic models. For example, in static analysis, assuming a connection is fixed or free can result in conservative, or bounding response levels; however, this typically isn’t true for dynamic modeling. A typical shipping package dynamic model is nonlinear by nature and therefore path-dependent, so interpolation or extrapolation cannot be employed. Even if a component itself is not an item of concern, it may require a fairly detailed model due to its effects on an adjacent component that is of concern. Typically in dynamic models, the accurate modeling of a component is the only appropriate course of action.

e. Discuss how representative source terms are developed for shielding analysis.

The following is taken from NNSA SG-100.

Following the recommendations of the NRC Regulatory Guide 7.9, Section 5 of a SARP for a particular packaging and contents will contain a calculational analysis confirming that the external dose rates for the package are in compliance with the limits as specified in 10 CFR 71.47 and 10 CFR 71.51. The confirmation of compliance with the regulations is acceptable if the calculations follow from commonly accepted radiation shielding analysis practices.

The primary reason for the shielding analysis is to ensure beforehand that neither the NCT nor the HAC dose rate limits will be exceeded. The required pre-shipment measurements will record the NCT dose rates resulting from the radioactive source material at the time of loading. However, the analysis must account for any source buildup or decay that may occur during the period specified in the authorization certificate and compute dose rates based on the maximum possible source strength during that time. The use of the transportation package for long-term storage of its contents will increase the time period for source maximization.

The conservatisms built into the source calculation and other modeling items will, in general, yield calculated dose rates that exceed the measured values, sometimes by substantial amounts. If the measured values are greater than the calculated values, all measurements and calculations should be carefully examined to explain this difference, even when none of the regulatory dose rate limits are exceeded by either method. In some cases, for weak radioactive sources, the external package dose rates may be comparable to or less than the local background radiation, especially for the off-surface dose rates. For unirradiated uranium source material, the package surface neutron dose rates will ordinarily be comparable to background levels.

f. Discuss shielding models used in type B package shielding evaluations.

The following is taken from NNSA SG-100.

The most efficient shielding analysis for a SARP submittal is one that contains simple and conservative models. These conservatively calculated dose rates should always exceed measured values, sometimes by substantial amounts. Overly conservative unrealistic shielding models should be avoided, as they will most likely grossly overestimate the dose rates and could be misinterpreted as actual values.

The calculations and considerations characteristic of a conservative shielding analysis are discussed below:

- Determine the radioactive source strength at or very near the time of maximum strength. This may be done with a few calculations to determine the time within a year or a half-year.
- Include any high radiation-producing isotopes, such as trace amounts of Pu-236 in plutonium or U-232 in uranium, at the highest measured or theoretical values.

- Include any trace element impurities in the source material, such as boron, beryllium, fluorine, lithium, etc., at maximum measured values to maximize the (α, n) reaction neutron source.
- An even more conservative (α, n) neutron source is to use data in the source generation code for a uranium oxide matrix (the default for ORIGEN). However, this method may over-predict the neutron source by more than an order of magnitude as compared to a metal matrix.
- When calculating the total source strength or source normalization factor, use the maximum measured or theoretical density for the radioactive material.
- Account for subcritical multiplication. Subcritical multiplication can be modeled either in the calculation or afterward by multiplying the resulting neutron dose rates by $1/(1-K_{eff})$, where K_{eff} is the highest value determined by the HAC criticality calculation.
- In the calculation of the dose rates, after the source has been determined, use the minimum measured values of density and dimensions for shielding materials and the maximum values for materials that produce sources during the dose rate calculation. The concerns regarding the maximum and minimum values of material density and dimensions may lead to conflicting choices due to the production of secondary radiation, and use of nominal or average values may be necessary for dose rate calculations. The effect on the calculated dose rates from the use, or lack of use, of maximum or minimum measured densities and dimensions should be at most only a few percent. In contrast to the effect of such variations on the results of a criticality safety analysis, small percentage changes in the results of a shielding analysis are usually of little concern.
- The overall efficiency of a shielding calculation may be enhanced by conservatively omitting from the calculational model nonsource materials that present some geometric, compositional, or other difficulty. This difficulty may appear as a result of some complication in the contents or in the packaging material. Uranium containing none of the U-232 isotope may be conservatively omitted from a shielding model when the computed exterior package dose rates result primarily from gamma rays in other radioactive source material. The gamma ray shielding properties of uranium outweigh any contributions to the dose rate from its small neutron and gamma ray source when U-232 is not present.
- The overall efficiency of a shielding analysis will also be increased by conservatively omitting the outer packing/insulation material and the shipping drum. The NCT and HAC shielding models are identical, except for detector locations, and all dose rate calculations from both configurations can be made simultaneously. For most shipping packages containing MNSI, the omission of this outer packaging material will increase the calculated exterior dose for NCT by fifty percent at most.
- If a series of packages is to be shipped containing weapons components or special components that are similar in some way from package to package, the dose rates computed for the most conservative of the models may be used to represent all the packages. A situation often encountered is the necessity to ship many radioactive source material parts separately in many packages with identical packaging materials. If all these parts are identical in composition, and differ only in mass and shape, one dose-rate calculation for the most massive part in the most conservative position in the CV may be used for all packages. It may be convenient to change the part shape to some generic shape spread out over some portion of the top, bottom, and side of the CV, with no

internal packing material, for three separate calculations to give maximum possible dose rates. The procedure for using conservative, generic shapes, and masses may also reduce any security classification concerns in connection with the shielding analysis and presentation.

- The neutron-source dose rate calculation may be computer intensive, even when the calculated value is comparable to background levels as for uranium or thorium. Compliance with the required analysis and regulatory limits can be accomplished with a very conservative and approximate method. The neutron dose rate can be computed from the neutron-source strength and spectra using a point-source-in-void flux calculation. This method is not recommended for plutonium.
- Many conservative and approximate methods are available for package source strength and dose rate calculations that are specific to a particular computer code or group of computer codes.

9. NNSA package certification engineers must demonstrate a working level knowledge of the criticality evaluation information required in chapter 6 of a SARP.

a. Discuss NNSA SG-100, Sections 6.2 through 6.9 as they pertains to criticality.

Criticality safety is the practice of ensuring that adequate protection is provided against an accidental self-sustaining or divergent fission chain reaction. For packages that transport fissile material, this “adequate protection” is provided by using a design and safety-assessment philosophy that effectively eliminates the possibility of a criticality event occurring under any credible scenario. Thus, the package design and allowable loading specifications must be such that the safety evaluation can demonstrate, under all credible transport conditions, that more neutrons are lost from the system than are produced; that is, the system must always be subcritical.

Restriction of fissile mass, use of a favorable geometry (to provide enhanced neutron leakage from the package), incorporation of neutron poison materials, and moderator control are potential means of controlling the neutron balance. A detailed consideration of the many parameters that interact to influence the neutron behavior is needed to provide an adequately safe, yet efficient, package design. Neutron poisons added to package materials require special attention because their presence must be ensured under all conditions and because their incorporation may change the mechanical and/or thermal properties of host materials.

Whatever the control mechanism, an adequate margin of subcriticality must be demonstrated for both the single package in isolation and for arrays of packages. Undamaged (normal transport conditions) and damaged (subsequent to accident conditions) packages must be considered using the credible fissile material configuration and the moderator and reflector conditions that provide the maximum effective neutron multiplication factor, k_{eff}. From the criticality safety evaluation, one determines the CSI for a package design. The CSI is the operational parameter that determines the number of packages allowed on a transport vehicle.

Although criticality safety assessments can sometimes be developed using safe subcritical limits for mass or dimensions, the evaluation of complex package designs under the conditions prescribed by the regulations typically requires the use of sophisticated computer codes that incorporate either deterministic or statistical techniques to model neutron transport and predict k_{eff}. Proper validation of the computer code and nuclear data is necessary to establish calculational biases and uncertainties. The margin of subcriticality allowed for a package configuration should include the effect of these biases and uncertainties, together with design uncertainties, and an additional subcritical margin that would assure subcriticality even in the absence of all uncertainties.

Chapter 6 offers guidance that should be of value in the criticality safety aspects of design and safety evaluation for transportation packages containing fissile material. The guidance is built on earlier efforts to provide such information. Performance and documentation of a thorough criticality safety evaluation, consistent with the package design and specified contents, provides the demonstration of compliance called for in the domestic and international transportation regulations.

b. Discuss the effects and applications of the following factors relevant to criticality safety:

- **Mass**
- **Geometry**
- **Interaction and separation**
- **Moderation**
- **Reflection**
- **Concentration**
- **Volume**
- **Density**
- **Neutron absorbers**
- **Heterogeneity**
- **Enrichment**
- **Fissile material**

The following definitions are taken from DOE G 421.1-1 (Archived) unless stated otherwise.

Mass

Mass control may be used on its own or in combination with other nuclear parameter controls. In either case, administrative means of control are required.

Mass control often takes on aspects of nuclear materials accountability, particularly when used in laboratory situations. Mass control limits are frequently established for individual laboratory rooms, or groups of rooms, and detailed records are kept of mass transfers into and out of the room.

Extensive ACs are generally implemented involving the transfer of fissionable material, documentation of fissionable masses currently in the facility, posting of limits, and surveillance of the laboratories, records, and posted limits. Mass limits for adjoining rooms should also account for significant interaction. Alternatively, room walls could be designed to preclude potential interaction.

Mass control may be used to limit the quantity of fissionable material in processes such as casting of metal, disposal, storage, collection, or withdrawal, or in transportation containers. Sampling or nondestructive measurements are often required to verify masses. Establishment of mass limits for containers of fissionable material should involve consideration of potential moderation and reflection, geometry, enrichment, spacing, concentration, and neutron poisons. Safe mass varies considerably, depending on the other nuclear parameters involved. Controls should be implemented to ensure that unexpected changes in these other nuclear parameters will not cause a criticality accident.

Geometry

Geometry control is the preferred method of criticality safety control based on limiting one or more characteristic dimensions. Where practicable, reliance should be placed on the use of geometry control rather than the control of any other nuclear parameter. The practicality of using geometry control depends on the type of equipment needed (it may be impossible to incorporate a geometrically safe design for a large-scale pit), the process flow rates and volumes, and inherent complexity. Because each system and facility may be different, decisions should be made and approved on a case-by-case basis.

Geometry control is based on physical design limits, such as "geometrically safe" or "geometrically favorable" cylinder diameter, annulus inner and outer diameter, slab thickness, and spherical diameter for a given fissionable material. Geometrically safe is defined as the characteristic dimension of importance for a single unit of a specific geometrical shape such that nuclear criticality safety is not dependent on any other nuclear parameter. A geometrically safe dimension is determined assuming optimal moderation, thick reflection, and no control on concentration, enrichment, mass, or neutron poison.

Geometrically favorable is defined as the characteristic dimension of importance for a single unit of a specific geometrical shape such that nuclear criticality safety is maintained in conjunction with one or more other nuclear parameters such as concentration and limited reflection.

If geometrically favorable dimensions are used, care should be taken to guard against the possibility of losing control over the other nuclear parameters upon which favorable geometry depends. Geometrically favorable control may require active protective devices or ACs, or both, such as concentration- or moderation-control monitoring equipment, on-line enrichment, or mass-control monitors, and sampling. If there is no possibility that the geometrically favorable conditions will be violated, then it may be reasonable to consider the geometrically favorable dimensional parameters as passive-engineered control rather than as active-engineered or administrative control.

Geometry control limits preclude the possibility of criticality by virtue of neutron leakage from the system. This control method provides inherent criticality protection that 1) is not subject to random failures (as may occur with an "active" control device); 2) is not susceptible to the common types of human errors occurring during operating and maintenance activities that may act to defeat the control; and 3) provides inherent protection against unforeseen criticality scenarios. This control method requires a minimum of facility operational support to maintain effectiveness. Note that spacing between geometrically safe units shall be considered because of the potential for interaction.

Geometry control has many applications. Arrays of geometrically controlled cylindrical columns or slab tanks may be used to store or process fissionable material solutions. Geometrically controlled slab geometry may also be used for drip pans and for tables used for cleaning small pieces of contaminated equipment with various solutions. Process piping and drain lines often need to be designed to be geometrically controlled. Other equipment or portions thereof that normally do not process or contain fissionable material may also need to be controlled by volume or geometry. For example, if significant quantities of fissionable material can enter lubricating oil in a pump, the pump and its oil reservoir, if any, may need to be limited to a safe volume. Alternatively, it may be necessary to conservatively approximate the geometry of the interior of the pump and any oil reservoir and perform calculations to show that the geometry is safe for all credible cases.

Interaction and Separation

Spacing is a highly preferred method of control consisting of the use of passive devices or systems, or ACs, to ensure the maintenance of favorable spacing. Safe spacing maintains neutron leakage and reduces neutron interaction among units containing fissionable material.

Fixed (passive) spacing controls are used for the separation of fissionable material in operating activities and storage of many types of fissionable materials, including weapons components, wet or dry storage of reactor fuel, storage of oxides or nitrates, storage of fissionable material in shipping containers, and storage of fissile-containing solutions. Examples of such devices and systems include pool storage racks, floor storage racks of various types, dollies having a base of sufficient dimensions to provide favorable spacing, fissionable material birdcages, and safely spaced shelving. To be considered a strictly passive control, spacing should not be dependent on other nuclear parameters, or the other nuclear parameters should be fixed. For example, spacing that also incorporates fixed neutron poisons is considered passively favorable spacing; spacing that passively limits container size to those containers that may be safely spaced is considered favorable spacing; spacing that is designed for materials having a limited enrichment is considered passively favorable spacing as long as other materials having higher enrichments are either not available or cannot fit into the safely spaced positions; and spacing that is designed for containers having limited moderation is considered passively favorable if containers without such moderation control limits are not available or cannot fit into the safely spaced positions. In situations where spacing is established for a specific container and its contents based on specific mass, dimension, chemical composition, or fissile nuclide, features should be incorporated into the design to preclude the placement of other containers into the storage spaces. In addition, design features should eliminate the possibility of placing more than one container into a given storage space or placing additional containers into the regions between storage positions. When the containers of fissionable material are relatively light and have low radiation levels such that they can be handled hands-on, the potential exists for human error, particularly in moving containers into and out of storage. It is better to design spaced arrays such that they cannot be easily defeated by human error rather than to rely strictly upon procedures. When several types of containers of fissionable material are to be stored in the same spacing structure, the spacing should be established for the most reactive feasible combination of packages. This is normally the entire array filled with the most reactive package, and one position near the center double-batched if this is a possible loss of one control.

Passive control of spacing is highly reliable, not subject to random failures, and provides coverage against potential unforeseen accident scenarios. However, several problems with spacing are possible. Fixed (passive) spacing structures and devices may be susceptible to structural failures due to such conditions as exceeding the load limits, corrosion, ramming by forklifts, items falling from overhead cranes, and earthquakes. Spacing control may require the use of nondestructive gamma or neutron counting equipment or physical sampling and analysis to determine fissionable material package content before assigning a given spacing to a given package. In such situations, passive-engineered spacing control takes on aspects of active-engineered or administrative control.

If a favorable spacing arrangement can potentially be defeated by human error or equipment failure, then it may be necessary to consider the spacing control as active-engineered or administrative in nature. In fact, it is sometimes difficult to strictly distinguish between passive spacing control and active-engineered or administrative spacing control.

If spacing control is dependent on such things as signs, marks on the floor, procedures such as the spacing of packages one meter apart, or temporarily erected structures such as chained-off areas rather than fixed engineered storage structures, then it becomes administrative in nature and is less preferred than truly passive-engineered spacing design.

Moderation

Moderating and reflecting materials (such as water, heavy water, acids, oil, plastic, beryllium, concrete, heavy metals, and carbon) tend to substantially reduce the quantity of fissile nuclides that may be safely handled. For this reason, processes involving fissile nuclide compounds or metals are often designed to specifically exclude or control the use of moderators. Moderation control is the purposeful control of the quantity of moderating material mixed with or intermingled with fissile nuclides. Fissile nuclides may be safely handled using moderation control in combination with other control methods, such as mass control and geometry control. In this way, larger masses of fissile nuclides in larger geometries may be handled than by using mass or geometry control alone. Measurement of the ratio of moderating atoms to fissile atoms may be necessary to verify moderation control. Because of the need to verify moderation level, moderation control is generally implemented using active-engineered and ACs such as sampling, drying, or moisture detectors. When implementing moderation control, designers should take necessary steps to preclude potential sources of moderation from areas handling fissile nuclide powders and solids. Doubly containing water pipes and employing a leak detection system between the inner and outer pipes is an acceptable compromise in many situations where water piping must pass through or over moderation controlled areas. Processes involving fissile nuclide powders and metals may also make use of suitable enclosures to reduce or eliminate the potential for unwanted moderation. Otherwise, a design should take suitable precautions for potential moderation and reflection and rely on other control methods.

Moderation control is an important consideration in selecting a fire control system. Available options include water, borated water, carbon dioxide, inert gases, and foam.

When analyzing a fissionable material system consisting of fissile nuclides, it is important to understand the effect of added moderation to dry fissile nuclides. Since fissionable materials of fissile nuclides can only fission with intermediate or fast neutrons, the addition of moderators will make such materials less reactive to the extent that criticality may be precluded. On the other hand, when handling solutions of fissionable material constituted with fissile nuclides, consideration should be given to potential loss of moderation that would make such systems more reactive.

Mixtures of fissible and fissile materials require special attention as they have conflicting contributions to the system reactivity depending upon the neutron energy spectra of the system.

Reflection

Neutron-reflecting materials reduce the quantity of fissionable material that may be safely processed, stored, or transported. Generally, the degree of reflection evaluated for a given situation is taken to be the maximum credible available unless mitigating factors are, or reflection control is, ensured.

Reflection may be controlled to prevent unacceptable thicknesses of reflectors in contact with, or surrounding, process equipment or fissionable material units. One should be aware that controlling other parameters may increase reflection. For example, when controlling neutron interaction between units by adding material between them, the undesired and unintended effect could occur by which a single unit is made critical because of reflection.

The effectiveness of standard and credible composite reflectors incidental to normal or abnormal conditions of processing, storing, or transporting fissionable material should be considered and evaluated, as appropriate.

Concentration

Fissile concentration control is used in situations in which the concentration of fissile nuclides in solution must be controlled to maintain subcriticality in large tanks. At least two circumstances exist for which different emphases of concentration control are required. In the first circumstance, a physicochemical process of an operation may ensure the fissile nuclide concentration of a solution will be within safely subcritical values. In such cases, the monitoring and control of concentration becomes a secondary control in the event the physicochemical process is corrupted or bypassed. In the second circumstance, processes involving potentially concentrated solutions of fissile nuclides can also use concentration control as one of the process variables to include as part of implementing double-contingency. Examples include large volumes of dilute waste solutions, such as raffinates from chemical extraction processes, evaporator condensates, or laboratory sample solutions that must be processed or stored in large and unfavorable geometry tanks or process equipment; however, geometry control should be used, if practical. Active-engineered or administrative means of control are normally used to implement concentration control. Procedures such as sampling, automatic concentration, or density measurements with or without automatic shutoff valves, or prescribed dilution are always necessary as passive-engineered control of concentration generally is not feasible.

Volume

Geometry control is a very effective means of providing inherent criticality protection for equipment items not intended for fissionable material processing or storage (such as pumps, valves, and filters). This control may be achieved by limiting the maximum holdup volume in an equipment item to less than the minimum critical volume for the fissionable material being processed, as determined by calculations or applicable tables.

Additionally, in areas having overhead liquid fissionable material process lines or storage vessels, policies and procedures should be established to deter incidental or unintentional containers such as unrestricted volume toolboxes, mop buckets, sponges, mops, open-topped plastic-lined containers, or other containers that could collect liquids.

Density

Density control of solids is similar to concentration control for liquids, and areal density control may be applied to either solids or liquids. High density of solid fissionable material tends to reduce the volume or geometric dimensions (and sometimes the mass) that may be safely handled compared to lower densities of the same or similar material whether alone or in a mixture. Higher density of fissionable nuclides means that it is less likely for a neutron to escape without causing fission. Moderation and mass control are normally required as well when using density control for solids. Also, maximum density of the fissile nuclide as a solid or in a mixture of solids is normally an assumption in many evaluations, hence not a control. The difference is that assumptions are those factors thought to be immutable and not readily subject to measurement or control. When density is used as a control, it is often represented indirectly, that is, in terms of what can be directly observed and controlled. For example, storage containers in a moderation control area may have a lower mass limit for the fissile nuclide as metal than for the same nuclide in a compound. However, some processes such as super-compacting solid wastes are becoming more prevalent to minimize storage or repository space, and the effects of greatly increasing density of the fissionable material within a container should not be overlooked, especially if many such containers are to be stored in an array with minimum spacing between them. It may be necessary to establish control on the overall fissionable nuclide density of the array as well as on the fissionable nuclide density within units.

Areal density control, a related concept to overall array density, is defined by making a projection perpendicular to a planar surface, such as a floor or tank bottom, and limiting the mass of fissionable nuclides per unit area on this projection. Areal density control may be very beneficial when the area of the planar surface is large. In such cases, the mass of fissionable nuclides in an area or within a vessel may be safely increased by a large factor over the minimum critical mass, and it does not matter whether the fissionable material is in the solid or liquid form. Areal density control may also be applied to discrete items, equipment, or containers of either solids or liquids, and if so, is akin to spacing control. In all cases, care should be taken to ensure no localized region containing more than a minimum critical mass can credibly exceed the overall limit of mass per unit area.

Neutron Absorbers

A neutron poison is any material for which credit is intentionally taken for an operation or a piece of equipment to maintain subcriticality. Control using solid neutron poisons incorporated into passive-engineered controls such that the neutron poisons are protected from dissolution or dispersion is preferable to soluble neutron poisons controlled by active-engineered controls. That form of control is preferable to concentration of soluble neutron poisons controlled by ACs. When poisons are specified, use of other than solid neutron poisons incorporated into protected passive-engineered controls shall be justified, including a description of the need for a neutron poison, its distribution, concentration, and permanence.

Heterogeneity

The following is taken from the Los Alamos National Laboratory, *Nuclear Science Self-Study Guide*.

Heterogeneity is also known as intermixing. When working with uranium or plutonium suspended in a chemical solution, it is common for the chemicals to separate, similar to the way oil and water do. This can prove (and has proved) to be dangerous if there is enough fissile material in the suspension that could concentrate and reach critical mass. It is important to be aware that some chemicals separate from solution, unless the mixture is kept well-stirred. This information is vital because accidents can and have happened due to separation of solution.

Heterogeneity and enrichment are discussed together due to the resonance capture effect. Heterogeneous systems can be made critical with lower masses than similar homogeneous systems.

Enrichment

The following is taken from Los Alamos National Laboratory, *Nuclear Science Self-Study Guide* and the Bureau of Radiological Health and the Training Institute, *Radiological Health Handbook.*

This is the process]in which the relative amount of one or more isotopes of a constituent has been increased. For example, uranium that has been processed to raise the concentration of U-235 is referred to as enriched uranium. All other things being equal, increasing the relative amount of a fissile isotope such as U-235 above its normal abundance in a material such as naturally occurring uranium, will lower the mass necessary to achieve criticality compared with uranium with a natural abundance of U-235. Enrichment as a criticality control parameter must be managed with respect to other parameters such as concentration and density.

Fissile Material

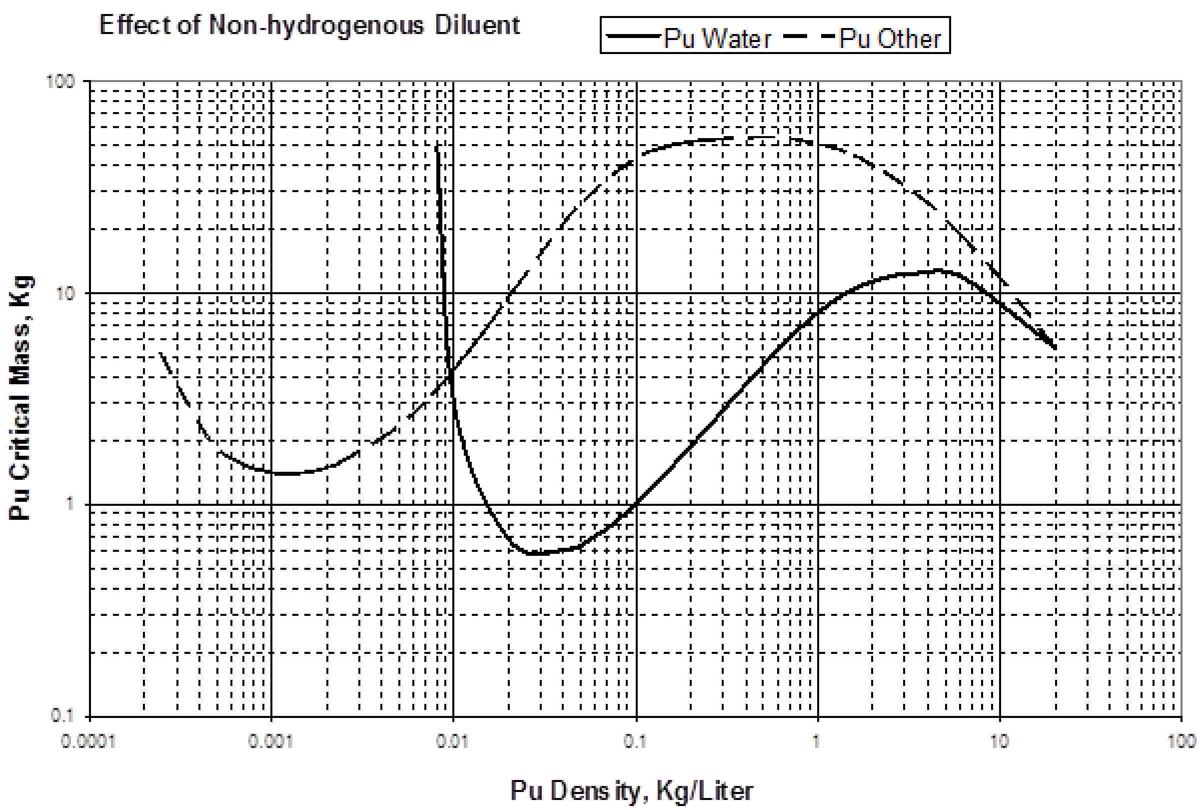
Fissile material in the presence of moderators more effective than water—that is, moderators with values of $\xi \sum_8 \sum_8$ greater than that of water, for example, heavy water, beryllium or beryllium compounds, carbon, and hydrogenous materials with hydrogen densities greater than that of water—can be expected to have minimum criticality masses, safe geometries, exempt quantities, etc. that are more restrictive than those determined for aqueous solutions or for flooding by water.

Fissile material in the presence of moderators other than water that are made more effective than water by the neutron energy spectrum of the fissile-moderator system can also be expected to have a minimum critical mass, safe geometry, exempt quantity, etc., each of which is more restrictive than that determined for aqueous solutions or for flooding by water.

c. **Discuss the influence of the presence of nonfissionable materials mixed with, or in contact with, fissionable material on nuclear criticality safety.**

The following is taken from DOE Nuclear Criticality Safety Program website, *Nuclear Criticality Safety Engineer Training*.

The addition of nonfissionable materials to a Pu or U metal system initially raises the minimum critical mass (more so with Pu than U), as the core density effect predominates. As more diluent is added, the moderation effect decreases the critical mass until the absorption properties of the diluent drive the system subcritical. The general effect of nonhydrogenous diluents is to raise the minimum critical mass and lower the minimum critical concentration as compared to hydrogenous diluents as shown in figure 4.



Source: DOE Nuclear Criticality Safety Program Website, Nuclear Criticality Safety Engineer Training
Figure 4. Effect of nonhydrogenous diluent

d. **Discuss the criticality requirements in 10 CFR 71.55. Describe the factors that would be considered in determining whether it is acceptable to grant the 10 CFR 71.55(c) exception.**

The following is taken from 10 CFR 71.55.

(a) A package used for the shipment of fissile material must be designed and constructed in accordance with 10 CFR 71.41 through 10 CFR 71.47. When required by the total amount of radioactive material, a package used for the shipment of fissile material must also be designed and constructed in accordance with 10 CFR 71.51.

(b) Except as provided in paragraph (c) or (g) of 10 CFR 71.55, a package used for the shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system, or liquid contents were to leak out of the containment system so that, under the following conditions, maximum reactivity of the fissile material would be attained:

- 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 full reflection of the containment system by water on all sides, or such greater reflection of the containment system as may additionally be provided by the surrounding material of the packaging

(c) The NRC may approve exceptions to the requirements of paragraph (b) of 10 CFR 71.55 if the package incorporates special design features that ensure that no single packaging error would permit leakage, and if appropriate measures are taken before each shipment to ensure that the containment system does not leak.

(d) A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in 10 CFR 71

- the contents would be subcritical;
- the geometric form of the package contents would not be substantially altered;
- there would be no leakage of water into the containment system unless, in the evaluation of undamaged packages under paragraph (a)(1) of 10 CFR 71.59, “Standards for Arrays of Fissile Material Packages,” it has been assumed that moderation is present to such an extent as to cause maximum reactivity consistent with the chemical and physical form of the material; and
- there will be no substantial reduction in the effectiveness of the packaging, including
    - no more than five percent reduction in the total effective volume of the packaging on which nuclear safety is assessed;
    - no more than five percent reduction in the effective spacing between the fissile contents and the outer surface of the packaging; and
    - no occurrence of an aperture in the outer surface of the packaging large enough to permit the entry of a four-inch cube.

(e) A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in 10 CFR 71.73, the package would be subcritical. For this determination, the following is 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 full reflection by water on all sides, as close as is consistent with the damaged condition of the package.

(f) The following applies for fissile material package designs to be transported by air:

- The package must be designed and constructed, and its contents limited so that it would be subcritical, assuming reflection by 7.9 inches of water but no water in-leakage, when subjected to sequential application of
  - the free drop test in 10 CFR 71.73(c)(1);
  - the crush test in 10 CFR 71.73(c)(2);
  - a puncture test, for packages of 250 kg or more, consisting of a free drop of the specimen through a distance of 120 inches in a position for which maximum damage is expected at the conclusion of the test sequence, onto the upper end of a solid, vertical, cylindrical, mild steel probe mounted on an essentially unyielding, horizontal surface. The probe must be 20 cm (7.9 inches) in diameter, with the striking end forming the frustum of a right circular cone with the dimensions of 30 cm height, 2.5 cm top diameter, and a top edge rounded to a radius of not more than 6 mm (0.25 inches). For packages less than 250 kg, the puncture test must be the same, except that a 250 kg probe must be dropped onto the specimen which must be placed on the surface; and
  - the thermal test in 10 CFR71.73(c)(4), except that the duration of the test must be 60 minutes.
- The package must be designed and constructed, and its contents limited, so that it would be subcritical, assuming reflection by 20 cm (7.9 inches) of water but no water in-leakage, when subjected to an impact on an unyielding surface at a velocity of 90 meters/second (m/s) normal to the surface, at such orientation so as to result in maximum damage. A separate, undamaged specimen can be used for this evaluation.
- Allowance may not be made for the special design features in paragraph (c) of 10 CFR 71.55, unless water leakage into or out of void spaces is prevented following application of the tests in paragraphs (f)(1) and (f)(2) of 10 CFR 71.55, and subsequent application of the immersion test in 10 CFR 71.73(c)(5).

(g) Packages containing uranium hexafluoride only are excepted from the requirements of paragraph (b) of 10 CFR 71.55 provided that

- following the tests specified in 10 CFR 71.73, there is no physical contact between the valve body and any other component of the packaging, other than at its original point of attachment, and the valve remains leak tight;
- there is an adequate quality control in the manufacture, maintenance, and repair of packagings;
- each package is tested to demonstrate closure before each shipment; and
- the uranium is enriched to not more than 5 weight percent U-235.

e. Discuss how the criticality safety index is determined.

The following is taken from 10 CFR 71.22.

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

$$\mathbf{CSI = 10}\left[\frac{\boldsymbol{grams\ ^{235}U}}{\boldsymbol{X}} + \frac{\boldsymbol{grams\ ^{233}U}}{\boldsymbol{Y}} + \frac{\boldsymbol{gramsPu}}{\boldsymbol{Z}}\right]$$

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

The values of X, Y, and Z used in the CSI equation must be taken from tables 71-1 or 71-2 in 10 CFR 71.22,"General License: Fissile Material," as appropriate.

If table 71-2 in 10 CFR 71.22 is used to obtain the value of X, then the values for the terms in the equation for U-233 and plutonium must be assumed to be zero and the values for X, Y, and Z in table 71-1 in 10 CFR 71.22 must be used to determine the CSI if

- U-233 is present in the package;
- the mass of plutonium exceeds 1 percent of the mass of U-235;
- the uranium is of unknown U-235 enrichment or greater than 24 weight percent enrichment; and
- substances having a moderating effectiveness are present in any form, except as polyethylene used for packing or wrapping.

f. Discuss the three characteristics of neutron balance that controls the criticality of a fissile system, and discuss how k_{eff} is determined.

The following is taken from DOE-HDBK-1122-2009.

The effective multiplication constant, or k_{eff}, is defined as the ratio of the number of neutrons in the reactor in one generation to the number of neutrons in the previous generation.

- Subcritical - $k_{eff} < 1$
- Critical - $k_{eff} = 1$
- Supercritical - $k_{eff} > 1$

g. Discuss criticality control devices

The following is taken from DOE-HDBK-1145-2013.

Many facility-specific engineered and administrative controls have been put in place in an effort to prevent an uncontrolled criticality. Examples of engineered controls are specific piping, container shape, and poisons (neutron absorbing material). Examples of administrative controls are procedures on container spacing and the amount of material in the container.

Only workers who are properly trained should handle fissile material. If you are not trained as a fissile material handler, do not, under any circumstances, handle fissile material.

Because liquid containing plutonium is a special criticality concern, care is required when handling plutonium-bearing liquids. Containers that could hold liquid may not be placed in or under gloveboxes or hoods unless they are criticality safety approved.

h. Discuss 10 CFR 71.59 standards for arrays of fissile material packages.

The following is taken from 10 CFR 71.59.

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

- Five times "N" undamaged packages with nothing between the packages would be subcritical.
- Two times "N" damaged packages, if each package were subjected to the tests specified in 10 CFR71.73, would be subcritical with optimum interspersed hydrogenous moderation.
- The value of "N" cannot be less than 0.5.

The CSI must be determined by dividing the number 50 by the value of "N" derived using the procedures specified in paragraph (a) of 10 CFR 71.59. The value of the CSI may be zero provided an unlimited number of packages are subcritical, such that the value of "N" is effectively equal to infinity under the procedures specified in paragraph (a) of 10 CFR 71.59. Any CSI greater than zero must be rounded up to the first decimal place.

For a fissile material package which is assigned a CSI value

- less than or equal to 50, that package may be shipped by a carrier in a nonexclusive use conveyance, provided the sum of the CSIs is limited to less than or equal to 50;
- less than or equal to 50, that package may be shipped by a carrier in an exclusive use conveyance, provided the sum of the CSIs is limited to less than or equal to 100; and
- greater than 50, that package must be shipped by a carrier in an exclusive use conveyance, provided the sum of the CSIs is limited to less than or equal to 100.

i. Discuss the acceptance criteria for subcriticality, and define over-moderation and why it is such a beneficial and achievable feature for safe transportation of arrays of small type B packages.

Acceptance Criteria for Subcriticality

The following is taken from Oak Ridge National Laboratory, *Application of Sensitivity and Uncertainty Analysis Methods to a Validation Study for Weapons-Grade Mixed-Oxide Fuel.*

A calculated k_{eff} for a fissile system is considered to be acceptably subcritical provided the calculated k_{eff} plus two standard deviations is less than a specified upper subcritical limit (USL). The following relationship can be used to establish the acceptance criterion for a calculated multiplication factor for a subcritical system, ks:

$$k_s + 2\sigma \leq 1.00 + \beta - \Delta\beta - \Delta k_m$$

where β and Δβ represent the calculational bias and uncertainty in the calculational bias, respectively.

The quantity Δk_m represents an administrative margin, and for transportation package applications, the minimum administrative margin of subcriticality is 5 percent. The bias, β, in the calculational method is the difference between the mean value of the calculated k_{eff} for the critical experiments, k_c, and 1.0.

A USL is a limit such that there is a specified level of confidence that a calculated k_{eff} is considered to be subcritical. Using the acceptance criteria for a subcritical system with an administrative margin of 0.05, the USL can be defined as follows:

$$k_s + 2\sigma \leq USL = 0.95 + \beta - \Delta\beta$$

A fissile system is considered to be acceptably subcritical provided the calculated k_{eff} plus two standard deviations is below the USL.

The calculational bias in the acceptance criteria can be positive if k_c is greater than one; however, a positive bias is not used in this study. Therefore, the bias is always less than or equal to zero.

Regarding the uncertainty in the validation, the sources of uncertainty include the calculational method, the experimental data or technique, and calculational models as well as the particular analyst.

The sources of uncertainty are cumulatively observed in the variability of the calculated k_{eff} results obtained for the modeled critical experiments. Furthermore, this variability includes the Monte Carlo standard deviation in each calculated k_{eff} for the critical experiment as well as changes in the calculated value due to consideration of the experimental uncertainties. Consequently, the noted uncertainties are included in the bias and uncertainty in the bias.

Over Moderation

The following is taken from NNSA SG-100.

Optimum moderation should be considered in each calculation unless it is demonstrated that there would be no leakage of water into void spaces under the appropriate test conditions. Optimum moderation is the condition that provides the maximum k_{eff} value. Partial and preferential flooding should be considered in determination of optimum moderation conditions. If there is no leakage of water into the system, then the actual internal moderation provided by the materials in the package can be assumed in the array model. Similarly, if the internal moderator provides more than optimum moderation and by its physical and chemical form cannot leak from the CV, then its moderating properties can be considered in the model. For example, a solid moderator that is shown to over-moderate the fissile material can be considered in the calculational model if its presence is verified. The criteria for internal moderation should be assessed and separately applied for NCT and HAC.

Each model for arrays of undamaged packages should assume a void between the packages consistent with the requirement of 10 CFR 71.59. For the assessment of arrays of damaged packages per 10 CFR 71.59, this optimum interspersed hydrogenous moderation condition should be determined. Optimum is considered the hydrogenous condition that provides the highest k_{eff} value. Interspersed moderation should be considered that moderation which separates one package in the array from another package. This interspersed moderation should not be taken to include the moderation within the package. If packages consistent with HAC and a void between packages provide an over-moderating condition, then it should be acceptable to take credit for the excess moderation provided by the packaging.

j. Discuss key factors and conservatisms used in criticality models.

The following is taken from NNSA SG-100.

When using a Monte Carlo code, the criticality safety analyst should always consider the imprecise nature of the values provided by the statistical technique. Every predicted value, k_{eff} or otherwise, should be reported with a standard deviation. Typical Monte Carlo codes provide an estimate of the standard deviation of the calculated k_{eff}. For some situations, the analyst may wish to obtain a better estimate for the standard deviation by repeating the calculation with different valid random numbers and using this set (five to ten calculations) of k_{eff} values to determine the standard deviation. Also, the statistical nature of Monte Carlo methods makes it difficult to use in determining small changes in k_{eff} due to problem parameter variations. The change in k_{eff} due to a parameter change should be greater than 3σ to indicate a significant trend in k_{eff}.

Monte Carlo codes give group-dependent fluxes and energy-integrated responses. The fluxes and responses obtained by a Monte Carlo code have a statistical uncertainty. Frequently, the uncertainties on individual group fluxes can be quite high; however, the uncertainties for the most important groups can easily be reduced to acceptable values. Empirically, the statistical uncertainty should be less than ten percent, except for point detectors, and less than five percent when using point detectors, to obtain reliable results.

The geometry model limitations of deterministic discrete-ordinates methods typically restrict their applicability to calculation of bounding, simplified models, and investigation of the sensitivity of k_{eff} to changes in system parameters. These sensitivity analyses can use a model of a specific region of the full problem to demonstrate changes in reactivity with small changes in model dimensions or material specification. Such analyses should be used when necessary to ensure or demonstrate that the full package model has used conservative assumptions relative to calculation of the system k_{eff} value.

10. NNSA package certification engineers must demonstrate a working level knowledge of the package operations information required in chapter 7 of a SARP.

a. Discuss the NNSA SG-100 as it pertains to package operations.

Chapter 7 discusses packaging design considerations based on the operation of the unique class of radioactive material transportation containers designed for weapons components, special assemblies, and other MNSI.

Radioactive materials considered for packaging under NNSA SG-100 include uranium metal parts, plutonium parts, and tritium bottles, as well as assemblies containing these materials. Packaging design is driven by regulatory requirements that are addressed elsewhere in this guide. Chapter 7 includes general discussions of packaging operating principles and features which, based on user experience, packaging developers should apply to their designs and should incorporate into operating and maintenance procedures or instructions. The chapter stresses the importance of addressing the user's needs and establishing the user-desired operational features early in the packaging design, and of avoiding oversights and operational problems that would require the packaging users or operators to improvise afterwards. The chapter also addresses the importance of keeping exposure to radiation ALARA standards, human factors, and system reliability within the design effort.

Chapter 7 of the guide focuses on the operational aspects of NNSA packages, specifically type B packages. NNSA SG-100 is written for the packaging-design team and discusses desirable features that are consistent with the regulatory requirements, and take into consideration human factors issues, the expected operational environment, and the level of expertise of the prospective users. In many cases, the users could be personnel who are not necessarily nuclear or packaging experts. It also addresses the needs of high-volume production-line users and their operations.

It is applicable to the specific packaging handling and operations required to use the package in compliance with applicable certification and regulatory requirements, including the following:

- Loading
- Closure and pre-shipment testing
- Preparation for shipment
- Unloading and empty shipment preparation
- Long-term storage
- Inspection, maintenance, and repairs
- Records and documentation

Chapter 7 establishes a framework to ensure that the needs of operators are given proper consideration during packaging design and development, and to establish a communication channel to provide the designer with feedback from lessons learned by the field operators (users) and fabricators. To accomplish this communication, a cooperative working relationship should be established early in the development process. Chapter 7 describes concepts and specific packaging features that a team should consider during the initial design phase.

A packaging design is a compromise that uses the available resources to best meet both the regulatory and operational requirements of the user(s). Each design feature represents an effort to balance the range of user needs and requirements; the designer looks at the range of options available and selects the ones that meet the regulatory requirements, protects the user, costs the least in the process, and provides the appropriate level of quality.

The designer must address user issues early in the design process. User requirements that are ignored at this stage often must be dealt with after the design is completed; and their resolution often has significant cost and schedule impacts. The package development team must consider ALARA whenever design decisions are made, and must give appropriate attention to component reliability, especially when a component failure can significantly affect achievement of ALARA objectives. The packaging user should be provided with a highly reliable, unambiguous, and easily verifiable packaging designed to be operated and maintained with a minimum of special tools, hardware, equipment, or instruments at a variety of different facilities and under a wide range of operating conditions. The packaging design team is responsible for the development of generic operating procedures and task or activity lists for proper use of the packaging.

The packaging user must develop, get NNSA approval of, maintain and use facility-specific operating procedures, for each package used at their facility, that provide the interface between the facility's equipment and capabilities, and complies with the SARP and OTC requirements.

The development team should interview packaging users who currently use packages similar to the type proposed, then incorporate their feedback to improve upon the final packaging design. The development team should document the process throughout the development effort to demonstrate its consideration of ALARA and human factors issues. For packages designed to be used at one or a limited number of specific facilities, the applicable characteristics of the user's facility should be factored into the design effort as practical.

Human factors should be an important consideration in packaging designs. The designer can address the main human factors issues by simplifying both the packaging design and the users' requirements. Current packaging operations should be assessed to determine their strengths and/or weaknesses, and empirically proven design features should be retained or refined. Any substantive changes to those proven designs or operational techniques should be justified. Packaging operations that are unambiguous, intuitively obvious, and provide ease of handling will reduce the risk of errant operations caused by mistakes during use, circumvention of approved procedures, etc. Addressing human factors in a packaging design effort will also help develop the documentation program needed to ensure conformance to the packaging certification requirements.

b. Discuss several key factors that must be addressed in chapter 7 related to package loading, unloading, and transport of empty packages.

The following is taken from NNSA SG-100.

Operational steps included in the SARP must meet the minimum requirements of NRC Regulatory Guide 7.9, Section 7. The SARP must list the steps necessary for loading, unloading, and preparing the packaging for transportation in a manner that is consistent with the analyses upon which the package approval is based. The designer is cautioned not to provide overly detailed procedures that could limit packaging-user discretion in performing the activity. The SARP must contain steps that will ensure the user meets the routine regulatory determination requirements. The steps should be presented in the expected order of completion and should contribute to maintaining personnel radiation exposure ALARA. Specific guidelines include the following:

- Operations procedures developed for inclusion in the SARP must address minimum operating requirements by listing only the functional steps necessary to ensure conformance to applicable design bases.
- Special tools or processes required must be clearly identified.

- Content limits must be reiterated within the procedure. Applicable controls must be identified to verify compliance with content limits.
- Procedures must detail unloading operations even if they do not differ from loading operations.
- Procedures must also detail preparation for shipping empty packaging.

c. Discuss issues related to package operations by multiple users, including equipment differences, facility differences, and internal procedure differences.

The following is taken from NNSA SG-100.

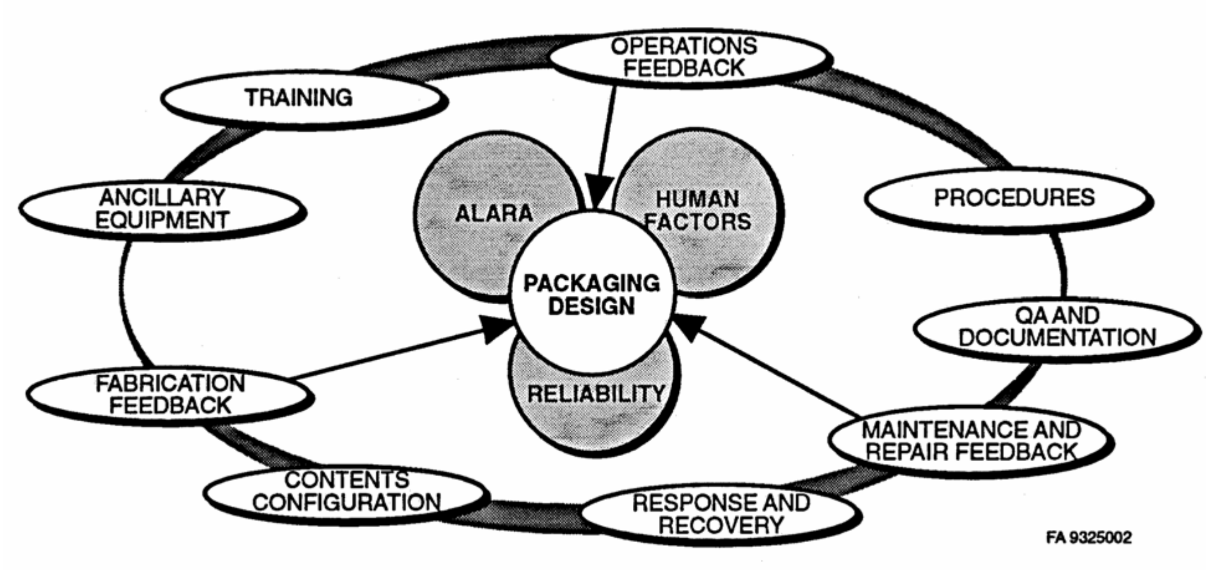
The requirements for the use of ancillary equipment supplied by the packaging user, owner, or OTC holder include the following instructions:

- Instructions for using owner-supplied ancillary equipment, including unpacking and packing instructions and instructions for shipping the ancillary equipment either with or separate from the package
- Inspection requirements and owner approval for ancillary equipment (primarily related to lifting fixtures or devices), which must be reviewed for compliance to facility-specific programs
- Utilities (such as electricity, air, water) or supplies required to operate ancillary equipment
- Alternately acceptable means of packaging handling or operations without using the ancillary equipment

The site-specific procedure should incorporate specific site information, including work rules, prerequisites, and personnel qualifications, into the appropriate locations in the generic operating procedure. This incorporation results in a single procedure for the site operator to use that conforms to the requirements of the SARP and the OTC. Checklists, which provide written and signed documentation of completion of an activity, can either be incorporated within the procedure or stand alone. Nevertheless, if a checklist is required for package use, each use of the package should be documented on a separate checklist. Each packaging user should establish a schedule for procedure review and revision, and should implement a program to permit periodic or temporary revisions in response to specific conditions.

d. Describe the linkage of chapters 1–6 evaluation results with package operations requirements provided in chapter 7.

Chapter 7 is written for the packaging design team and discusses desirable features that are consistent with the regulatory requirements, and takes into consideration human factor issues, the expected operational environment, and the level of expertise of the prospective users. In many cases, the users could be personnel who are not necessarily nuclear or packaging experts. It also addresses the needs of high-volume production-line users and their operations. Figure 5 shows how operational considerations should be integrated within any packaging design effort.



Source: NNSA SG-100

Figure 5. Operational considerations in packaging design

e. Discuss the routine determinations required by 10 CFR 71.87.

The following is taken from 10 CFR 71.87.

Before each shipment of licensed material, the licensee shall ensure that the package with its contents satisfies the applicable requirements of 10 CFR 71.87, "Routine Determinations," and of the license. The licensee shall determine that

- the package is proper for the contents to be shipped;
- the package is in unimpaired physical condition except for superficial defects such as marks or dents;
- each closure device of the packaging, including any required gaskets, is properly installed and secured and free of defects;
- every system for containing liquid is adequately sealed and has adequate space or other specified provision for expansion of the liquid;
- every pressure relief device is operable and set according to written procedures;
- the package has been loaded and closed according to written procedures;
- for fissile material, each moderator or neutron absorber, if required, is present and in proper condition;
- any structural part of the package that could be used to lift or tie down the package during transport is rendered inoperable for that purpose, unless it satisfies the design requirements of 10 CFR 71.45;
- the level of nonfixed (removable) radioactive contamination on the external surfaces of each package offered for shipment is ALARA, and within the limits specified in DOT regulations in 49 CFR 173.443, "Contamination Control";
- external radiation levels around the package and around the vehicle, if applicable, will not exceed the limits specified in 10 CFR 71.47 at any time during transportation; and
- accessible package surface temperatures will not exceed the limits specified in 10 CFR 71.43(g) at any time during transportation.

11. NNSA package certification engineers must demonstrate a working level knowledge of the acceptance tests and maintenance program information required in chapter 8 of a SARP.

a. Discuss the NNSA SG-100 as it pertains to package testing and maintenance.

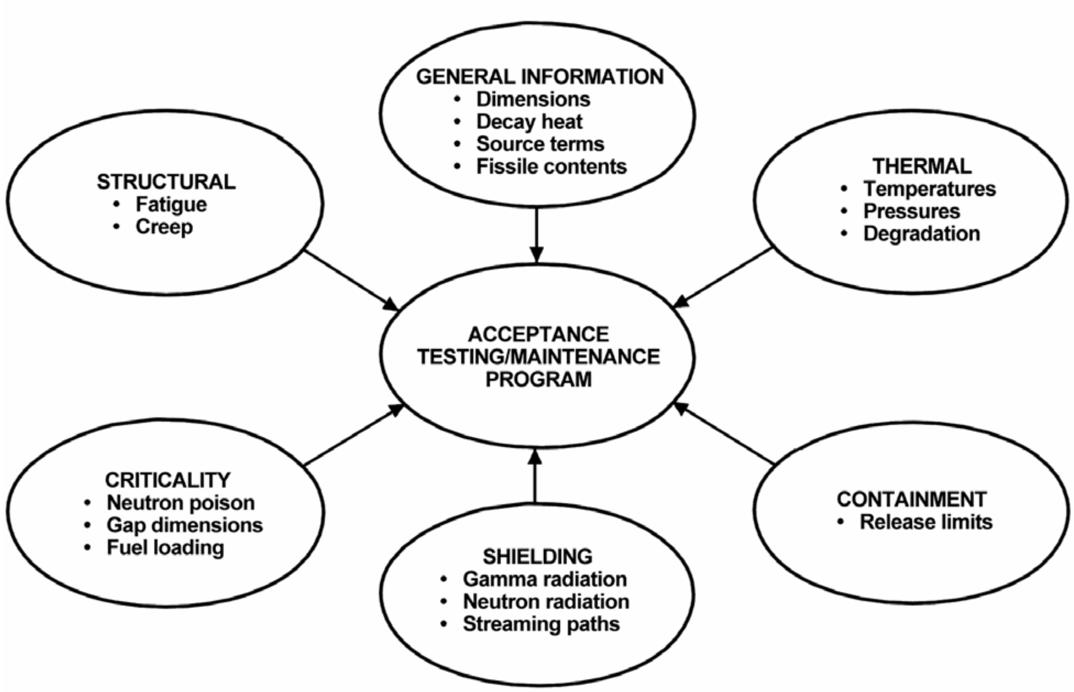
Chapter 8 describes the kinds of acceptance and verification tests and inspections required for a type B packaging and its components, appropriate criteria for determining the adequacy of the item(s) undergoing tests and inspections, and the constituents of an acceptable maintenance program. The goal of chapter 8 is to give the reader a basic understanding of the regulations concerning acceptance and verification testing and inspections, and maintenance guidelines as well as guidance on how to document that program in the SARP.

The requirements for acceptance and verification testing and inspections, and maintenance cover all packaging components, the assembled packaging, and the complete package.

The first step in developing a successful acceptance and/or verification, and maintenance program is to organize teams that understand all of the processes and operations involved in assembling, disassembling, inspecting, refurbishing, maintaining, leak testing, loading and unloading a type B shipping package. The acceptance and/or verification testing and inspection team should consist of qualified, experienced representatives from packaging design and safety, procurement, quality, and operations. The maintenance team should consist of representatives from packaging design and safety, quality, procurement, and operations.

Each team must be trained for the package under consideration, as package-specific training ensures that each team member fully understands all processes. Development of an acceptance and/or verification testing and inspection, and maintenance program depends highly on the packaging design and the quality categories of the packaging. Test requirements are specified in the regulations and DOE Orders. However, the exact type of test, the test procedure, and acceptance criteria vary depending on the specific components used in the packaging and their individual quality category.

The ability to inspect, test, and maintain a package impacts package design. Therefore, it is important to understand the interdependency among the individual development efforts for the overall package. The input needed from other design disciplines to develop a successful acceptance and/or verification and maintenance program is shown in figure 6.



Source: NNSA SG-100

Figure 6. Relationship of acceptance/maintenance programs to other disciplines

The driving force behind developing an acceptance, verification, and maintenance program is the need to comply with the regulations governing the transportation of radioactive materials. Therefore, each team member is responsible for having an adequate understanding of the applicable regulations.

b. Discuss the preliminary determinations required by 10 CFR 71.85.

The following is taken from 10 CFR 71.85.

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

- the licensee shall ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce the effectiveness of the packaging;
- where the MNOP will exceed 5 lbf/in^2 gauge, the licensee shall test the containment system at an internal pressure at least 50 percent higher than the MNOP, to verify the capability of that system to maintain its structural integrity at that pressure; and
- the licensee shall conspicuously and durably mark the packaging with its model number, serial number, gross weight, and a package identification number assigned by NRC. Before applying the model number, the licensee shall determine that the packaging has been fabricated in accordance with the design approved by the NRC.

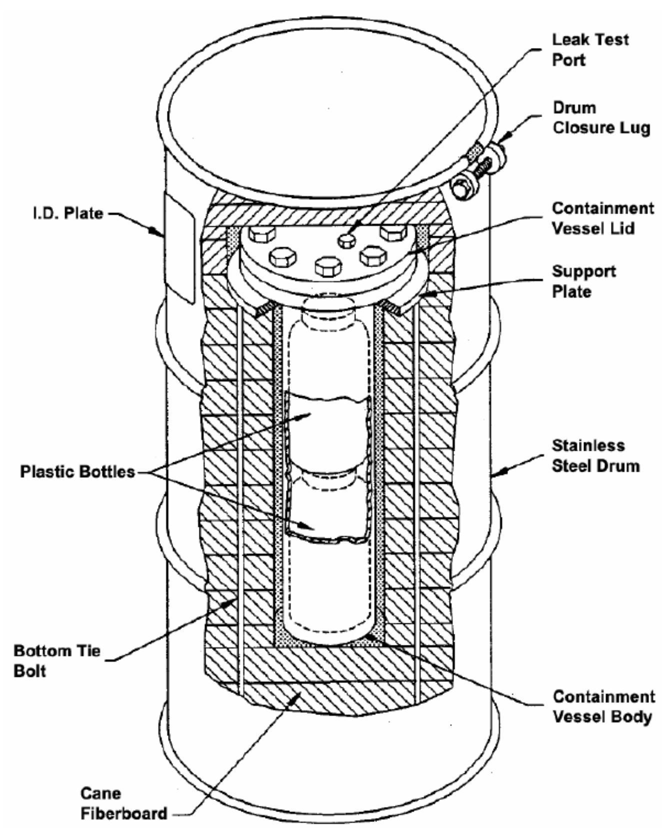
c. Discuss the following acceptance tests performed on type B packages:

- **Visual inspections and measurements**
- **Weld examinations**
- **Structural and pressure tests**
- **Leakage tests**
- **Component and material tests**
- **Shielding tests**
- **Thermal tests**

The following is taken from NNSA SG-100.

Visual Inspections and Measurements

All accessible components of the packaging are visually inspected. This inspection verifies that all items are clean, and free of nicks, gouges, and other damage that could affect package performance, and that all components are assembled in accordance with the design drawings. Each packaging component must be compared with the appropriate drawings to verify that the component is in the correct orientation or position, and that its design and dimensions conform to the design specifications. According to standard practice, a packaging is visually inspected each time it is loaded.



Source: NNSA SC SG-100

Figure 7. Typical drum-type package

Visual inspections typically fall into two categories based on the level of detail required. For overpacks such as drums, visual inspections are meant to evaluate the overall fit and finish, whereas the visual inspection of CV components requires more rigorous examination.

OVERPACK

Packagings should undergo acceptance and/or verification inspections on their arrival onsite. These inspections should be performed in accordance with approved site-specific inspection/product acceptance and/or verification procedures that are tailored to the specific packaging. The results should be recorded in forms (usually checklists) that are a subset of the inspection procedures. When completed, these forms should be maintained as inspection records.

When inspecting a drum-type packaging such as that shown in figure 7, the components critical to the structural integrity of the package, such as the drum body and lid and associated sealing surfaces should be examined. For example, visual inspections of the following items should be performed: the sealing surfaces to ensure that they are clean; the drum body to ensure that there are no scratches or gouges or other gross defects that could compromise the drum's performance; the gasket (when present) to ensure that it is in good condition, flexible, and makes a good seal; the weld seam to ensure that it has not split; the vent holes to ensure that they are present and plugged with the specified plug, if applicable; the bolts to ensure they are the specified bolts and that there is no damage that could reduce their effectiveness; the insulation, if accessible, to ensure that its thickness is adequate (a tape measure should be used to check the radial clearance between the insulation and the drum and the axial clearance between the lid and the insulation); the marking and labeling to ensure that they are appropriate for the intended use of the packaging, and the required paperwork, as defined in the applicable QAP, to ensure that the correct forms and information are provided.

Any nonconformances found should be documented on the inspection form. If the overpack is rejected, the required paperwork should be forwarded to the personnel responsible for packaging disposition, as defined in the applicable QAP, and the component should be tagged and sent to storage, where it should be stored in an area segregated from the other containers to await resolution by the personnel responsible for packaging disposition. If an acceptable way to resolve the discrepancy is found, the required paperwork should be forwarded to the personnel responsible for packaging disposition for their records. Packagings that successfully complete the acceptance tests/inspections and are accepted, should be sent to storage until they are needed.

Weld Examinations

Since welded CVs are used only once, the design must be able to accommodate a process by which the vessel's compliance with the requirements of 10 CFR 71.85, "Preliminary Determinations," can be confirmed. One solution is to develop a sampling program that tests a production vessel after a statistically defined interval. This test interval depends on how frequently CVs will be fabricated and how many vessels will be fabricated and shipped during a year. These qualification tests are also performed on a sampling of test specimens prior to the first production run to qualify the fabrication process. Furthermore, to establish a level of confidence, each welder is tested and qualified to perform the welding process.

Structural and Pressure Tests

Historically, structural and pressure tests have been used to check the structural integrity of a variety of packaging designs. The purpose of these tests is to ensure the packaging meets specified design criteria.

INTERNAL PRESSURE

This test verifies the structural integrity of the CV and ensures the vessel conforms to the design requirements. Pneumatic or hydrostatic pressure tests of CVs use air or water, as applicable, at an internal pressure as specified by the package design. The test pressure is commonly based on the design pressure times a factor of safety or the maximum possible calculated pressure. In cases where the vendor performs these tests, the vendor is required to provide reports on the results of the pressure tests for all CVs supplied. Individuals representing the package owner must examine the vendor-supplied data from these tests and should observe the tests when possible.

It should be noted that an internal pressure test will verify the quality of the welds of pressure vessels that are to be subjected to static pressure loads. However, it is not completely adequate for welds subject to dynamic loading or a combination of static and dynamic loading. Therefore, the design should allow for the CV welds to be radiographed.

EXTERNAL PRESSURE

This hydrostatic test, which is similar to the internal pressure test, is generally designed to satisfy the HAC 50-ft water immersion requirement. This test is designed to determine whether or not leakage of water into the CV will occur.

Leakage Tests

Acceptance leakage rate tests should be performed to ensure that a package meets the containment requirements specified by the package design. This section summarizes the leak test methods available. Some of these methods provide measurements of integrated (total) leakage and others are go/no-go tests that are designed to locate discrete leaks. The leak-test method should be selected on the basis of the containment system design and the sensitivity required.

GAS PRESSURE DROP

This procedure applies to test items with pressure tap connections, such as gas cylinders. The leakage rate is determined by pressurizing the test specimen to the specified test pressure and measuring the change of pressure and temperature within the test volume during a specified time period.

HOT WATER BUBBLE

This test applies to small, thermally responsive test items usually sealed without pressure tap connections. The test relies on an internal pressure increase, caused by heat transferred from the hot water, to form bubble streams that will reveal the location of individual leaks.

GAS BUBBLE

This procedure applies to test vessels with pressure tap connections. The formation of gas bubble streams in a liquid bath reveals the location of individual leaks. Various liquids, such as water, alcohol, mineral oil, or silicone oil can be used in conjunction with various tracer gases to vary the test sensitivity.

SOAP BUBBLE

A common method in pressure-vessel testing is the gas and bubble formation test. Leaks in a vessel are detected by applying a soapy solution at the prospective area(s). Gas passing through this solution will cause the creation of bubbles.

GAS PRESSURE RISE

This procedure, which is similar to the gas pressure drop test, applies to test items with pressure tap connections. The gas pressure rise procedure, although less affected by temperature changes than the pressure drop method, may require that all or part of the leakage flow be in the direction opposite to that typically encountered in normal operation. The total leakage rate may be derived from the rate of pressure rise, after correction for temperature change. Some leaks may be one-directional; therefore, the regulators may require additional justification for using this test since the leak flow may be in the opposite direction from that which would be expected in normal operations.

GAS BUBBLE, VACUUM, AND GLYCOL

This procedure is a variation of the gas bubble test, and usually applies to test items without pressure tap connections. The formation of gas bubble streams in an ethylene glycol bath reveals the location of individual leaks.

HALOGEN DETECTOR

This procedure applies to test items with or without pressure tap connections, and can be used to measure total leakage rates, or to locate individual leaks. Accumulation and shroud techniques can be used to increase accuracy and sensitivity, and to measure total leakage. Halogen vapor is detected by the rate of ionization either on a heated platinum anode or in an electron capture cell, or by a change in thermal conductivity.

MASS SPECTROMETER, SPRAY, OR SNIFFER

This procedure applies to test items with pressure tap connections. A mass spectrometer leak detector (MSLD) measures a tracer medium, usually helium gas. If a leak is present, the leak detector detects the helium gas and sends an electrical signal. For the mass spectrometer spray test, a small quantity of tracer gas is sprayed onto the test item. The MSLD is then used to check the response time for each portion of the test specimen. A change in the response time indicates a leak. In the mass spectrometer sniffer test, leaks of tracer gas from the pressurized side of the test item to the nonpressurized side are detected using an MSLD. The test can be performed under a vacuum, as a pressure test, or as a combination of both.

HELIUM BACK PRESSURIZING

This procedure applies to test items without pressure taps, and sealed sources that cannot be filled with helium during final closure. The items must be able to withstand the selected external pressure without damage. An MSLD measures leakage by detecting helium.

HELIUM MASS SPECTROMETER ENVELOPE

This procedure applies to test items with pressure taps. For this test, the test item is placed in an envelope, such as a well-sealed metal box, then purged and pressurized with helium at slightly greater than atmospheric pressure. The envelope is then evacuated with an MSLD. Conversely, a second method is to evacuate the item and place the item into an envelope that is slightly greater than atmospheric pressure.

Component and Material Tests

Component tests prove that individual packaging subsystems meet their design performance goals. Thorough testing is an essential part of the QAP. All components of the containment system (valves, rupture disks, and gaskets) should be procured under a QAP adequate to ensure that acceptance testing of a given component is equivalent to acceptance testing of all components supplied and identified by that manufacturer and identified as being a particular model.

VALVES AND RUPTURE DISKS

Acceptance tests of valves and rupture disks are required to show compliance with the package design requirements and to ensure the system will conform to regulatory requirements.

Individual acceptance tests for valves can be useful for screening borderline components before their inclusion in prototype or production hardware. These tests are simple and relatively inexpensive to perform. In general, borderline components should be tested under the most severe service conditions for which the package design assumes their acceptable performance. If adequate QA is evident, screening may be unnecessary or necessary only intermittently. The designer is responsible for determining the necessity of having to perform the valve acceptance tests.

Containment system verification requires more complicated test systems. Type B package containment systems should be leak- or release-tested per ANSI N14.5. The method of testing is determined by the degree of accuracy required and the package design, as noted in the standard. ANSI N14.5 also provides details for calculating the maximum permissible leak rates. Test models should be prepared as for shipment, subjected to NCT and HAC, and tested to show that they have leakages less than or equal to the maximum permissible rates. The radioactive contents may be simulated during the test, and only one specimen of a particular design need be tested to verify the integrity of a containment system. Note also that the integrity of a containment system can be verified by comparison, if a previously verified comparable design exists. Limited testing may be used to supplement the comparison.

Containment system fabrication tests are required before the first use of each reusable containment system. Where possible, all joints and seals on the containment system should be tested in the fully assembled state, but radioactive contents may be simulated. In some cases, joints and seals may have to be tested at the subassembly or component level. Single-trip containment systems must be tested to the same requirements as reusable systems, except that the sample size may be less than 100 percent and the testing must have been completed in the twelve months preceding shipment.

Periodic verification tests of the containment system are required for reusable type B packagings. In addition, before use for shipment, the packagings must have been tested according to the containment fabrication test within the preceding twelve months. Periodic verification of the containment system need not include the testing of inaccessible joints and seams, but should include all components such as closures, valves, pipe fittings, and rupture disks.

Containment system assembly verification tests are required prior to each shipment of type B packages. These tests are designed to verify proper assembly and to ensure functional containment.

Rupture disks, when used with a valve, provide additional protection against leakage of contents; when used without a valve, they are designed to relieve overpressure. The rupture disk burst pressure is certified by the manufacturer and should be marked on the disk. The user should verify that the rupture disk’s burst pressure falls within the specified design range.

For example, the ASME BPVC, suggests using one of the three following methods for acceptance testing of rupture disks:

1. Burst at least two sample rupture disks from each lot of rupture disks at the specified disk operating temperature. A lot is defined as a set of rupture disks manufactured of the same material at one time and of the same size, thickness, type, heat, and manufacturing

process, including heat treatment. If the specified disk temperature is not room temperature, burst at least one additional disk at room temperature. The rating at the specified disk temperature will be the average of the tests conducted at that temperature.

2. Burst at least four sample rupture disks, but not less than five percent, from each lot of rupture disks at four different temperatures. The test should be distributed over the applicable temperature range for which the disk would be expected to operate. These data should establish a smooth curve of burst pressure versus temperature for the lot of disks, with no point falling more than three percent from this curve. The rating at the specified disk temperature must be interpolated from this curve.
3. For tested disks whose data plots a smooth curve with no point outside of three percent of this data, a series of four or more burst tests can be made on one lot of disks at the specified disk temperature. The parameters from these tests can then be used to manufacture the disk.

GASKETS
Gaskets should be tested under conditions simulating the most severe service conditions under which the gaskets are to perform. If the actual system that the gasket will be part of is not used in the test, the simulation must adequately represent the actual system's consequential conditions. Many gasket physical characteristics such as hardness, tensile strength, elongation, and modulus are important to seal performance, but the specific application determines which of these characteristics are most crucial, and the developer is responsible for evaluating which parameters of those characteristics, if any, should be subjected to acceptance testing.

Shielding Tests

The user must be able to confirm that the shielding is present and functioning; that the external radiation levels on and around the package are below regulatory limits; and that the level of nonfixed radioactive contamination on the accessible surfaces of the package is ALARA.

Shielding integrity must be assured prior to each shipment of a package. Physical presence of the shielding can be confirmed by visual inspection. External radiation levels are usually determined by direct measurement with a known source simulating the contents.

It may not always be possible to ensure the package's shielding integrity by measuring external radiation levels from a known radiation source used to simulate the approved contents. Some packages may have to be loaded with the actual contents. In the case of one isotope container, no adequate neutron sources for shielding integrity tests were available at the time of fabrication. Therefore, shielding integrity was ensured by the imposition of rigid controls on the mixing and pouring of the package's concrete shield.

In this example, limonite concrete was used because it is a very effective neutron absorber. In accordance with the requirements of 10 CFR 71, direct measurements on the external surfaces of the package and at three feet are performed after loading, but prior to shipment, to verify the integrity of the shielding. Nonfixed contamination of the external surface of the package is determined by the swipe method. An area of the external surface is swiped with a cloth. A portable radiation detector is then used to measure the amount of contaminants on the cloth. The radiation level of contaminants found on the cloth is divided by the area swiped to determine the level of radioactive contamination per unit area. Storage and handling activities must not degrade the integrity of the shielding. Therefore, any special storage or handling requirements must be considered, recognized, and identified by the designer.

Thermal Tests

Containers designed to transport radioactive material with decay heat should be tested or analyzed to demonstrate their heat-load capabilities.

The acceptance criteria for acceptance and/or verifications tests should be determined with the aid of the packaging developers to ensure the manufactured packagings are the same as those whose design was analyzed and/or tested, and documented in the SARP as meeting all regulatory requirements. For instance, if a packaging design uses an insulating material, and the thermal analysis and/or tests documented in the SARP used the thermal properties of this insulating material, the acceptance criteria for the insulating material should be specified to ensure the insulating material in the manufactured packagings has the same properties as those used in the analysis and/or tests.

Packaging components—A typical packaging component essential to the thermal performance of a package is an insulating material. Insulating materials can vary by manufacturer and by lots produced at different times by one manufacturer. Therefore, demonstrating that the material is consistent with that specified by the package design is essential.

Frequently, manufacturers provide QA data on their insulating materials. This information can be used to verify that the insulating material meets the design requirements. However, when manufacturer QA data are not provided, measurements should be made to verify the acceptability of the material.

Whole packaging—The packaging as a whole should be evaluated to determine the acceptability of its thermal characteristics. The acceptance test should be designed to ensure that the actual packaging is consistent with the packaging design. For example, a new packaging can be loaded with a known heat source equivalent to the heat generated by the decay of the radioactive material being shipped. Temperature gradients across major materials and at material interfaces can then be measured. These temperature measurements can be made with the use of thermocouples or temperature indicator labels. The acceptance criteria should account for the sensitivity of the method for measuring temperature.

The following example demonstrates an actual thermal acceptance test used to evaluate new drum-type packagings. The test was developed to ensure the heat load capabilities of the package are acceptable for NCT. A representative sample packaging is tested as follows:

- Assemble a packaging. In the CV, include a device that generates the maximum permissible heat load of the package being emulated.
- During assembly, attach thermocouples to the outside top and side of the CV and to the outside top and side of the drum surfaces.
- Place the packaging in a normal condition environment of 100°F until it reaches steady-state temperatures. Record the temperatures.
- The temperatures should not exceed 250°F on the CV and 136°F on the drum surface. If the temperatures are exceeded, notify the design agency for the packaging.

d. Discuss the following maintenance program tests:

- **Structural and pressure tests**
- **Leakage tests**
- **Component and material tests**
- **Thermal tests**

The following is taken from NNSA SG-100.

Structural and Pressure Tests

Structural and pressure tests should be designed to verify the integrity of the packaging subjected to design loads. These tests should be based on a set of well-defined acceptance criteria that bound the design requirements of the package. Acceptance criteria and test frequency should be based on the quality level requirements of the component(s) being tested.

The hydrostatic test, which calls for immersion under a head of water of at least 50 feet, is required. For test purposes, an external pressure of water of 21.7 pounds per square inch gauge (psig) is considered to meet these conditions.

During each reuse inspection in one specific case, special attention was given to the O-ring sealing surfaces serving as a water barrier. If the sealing surfaces had been damaged, the vessel was hydrostatically tested after repairs had been performed and was subsequently approved for reuse before being placed back into operation.

In another instance, a CV was pressurized to its maximum allowable internal working pressure at 12-month intervals. Because of the geometry involved, it was possible to establish, as the acceptance criterion, that no visual deformation of the vessel occurred.

In a third instance, each CV was proof-tested at 200 psig with helium for a minimum of 4 hours before its first use, and every 2 years thereafter. Before each shipment, the container was proof tested at 1.5 times the maximum pressure possible during the planned shipment.

In a final example, a CV was pressure tested at 425 + 10 psi at 70 +10°F as ASME BPVC. The test was repeated every 2 years. The criterion for acceptance was no loss of pressure over a period of about 1 hour. The test sensitivity was dependent on the test gauges; at ± 1 percent accuracy for a full scale of 1,000 psi, this would be + 10 psi.

Note that there is a trade-off between pressurization testing and the cumulative stress that the CV is subjected to as a result of such testing. Nevertheless, when the MNOP of the CV exceeds 5 psig, the containment system must be tested at an internal pressure at least 50 percent higher than the MNOP. Reusable containers are fitted with pressure connections to enable periodic monitoring. The testing of welded CVs requires several special considerations.

The first consideration is how to test such vessels. One method is to fit a sample vessel with pressure fittings to provide a means of pressurizing the test vessel. A second consideration is how to test the vessel at the operating temperature. Two possible solutions have been proposed. The first is to use an internal heating unit that heats the vessel to the operating temperature during the test. A second, less invasive method is to perform the pressure test at ambient temperature conditions and raise the pressure at which the vessel is tested. Raising the pressure compensates for the difference in material properties at ambient and operating conditions. The amount of pressure applied depends upon the actual operating temperature.

A third consideration is how often to perform this test. Testing each vessel individually is impractical; one solution would be to develop a sampling plan.

The following example illustrates the use of a sampling plan. In this example, two sample vessels equipped with pressure fittings prepared by each certified welder are tested to qualify the fabrication process. If all test specimens satisfy the design requirements, production of the CVs may begin. However, if a test vessel fails to satisfy any requirement, the fabrication process must be modified, and the test series must be repeated until each certified welder sequentially produces two vessels that successfully pass the tests used to verify the vessels' compliance with the requirements. Thereafter, every 20th CV produced by each welder will be subjected to the overpressure test. If a vessel does not satisfy the test requirements, the sequential test described previously must be repeated and all of the untested vessels previously produced by this welder must be subjected to the pressure test.

Another approach to handling welded vessels is to present appropriate material certifications, together with documentation that the wall thickness of the container has been radiographed to show that it meets required minimum thickness everywhere. In conjunction, an analysis of the welds is performed in accordance with the ASME code. The case is then made that all of the vessels are acceptable.

Leakage Tests

The following sections discuss the instances in which leak testing is needed as well as appropriate testing methodologies to be employed under each condition. It should be noted that, when selecting a leak test method, the direction in which the seal is intended to block leakage should be taken into account in deciding how to set up the leak test.

ANNUAL VERIFICATION TEST

Prior to use, the CVs of all certified packages must have undergone a leakage rate test within the past twelve months. This test is often called an annual verification test and corresponds to the "periodic test" discussed in ANSI N14.5-1997.

For many CV systems, an annual helium leak test can be used to verify the CV's integrity. Sometimes a package design may have two concentric seals where the designer relies on only one of the seals for containment; the other seal is used to facilitate testing of the containment boundary. The second seal should not be evaluated simultaneously with the containment seal. The personnel involved in the testing shall be properly trained and qualified. The tests should be performed using established test procedures defined by the QAP. These test procedures will establish the requirements not only for the testing but also for the documentation of these tests. If the CV fails, it should be disassembled and the O-rings should be checked and replaced if necessary. The CV should be reassembled and the test should be performed again. If the CV continues to fail or if gross failure is observed, the package owner should be notified. This notification should initiate the process whereby the packaging owner informs others, such as the CO and the design engineer. The container should be tagged and stored in an official nonconforming segregation area until disposition is determined.

If the CV passes the tests, the packaging should then be reassembled and moved to refurbishment inspection until approval is received from the applicable QA organization. The refurbishment inspection and maintenance forms, as applicable, are reviewed by QA personnel, who issue the approval for the packaging. After approval, the packagings should be stored until needed, sent to another user, or used.

In some cases, the measured leak rate during annual qualification tests must be ≤ 1 × 10-7 std cm^3/s air. To accomplish this test, one packaging owner performs the annual leak tests with the aid of a leak-test adapter plate. Experience using the adapter plate for the leak test has shown that the leak rate will be well within the leak-tight range. This adapter plate is necessary as a means for loading helium into the CV cavity. Used O-rings should be discarded and new ones installed prior to performing the annual leak test.

In another case, the packaging owner leak-tested the CV at 1-year intervals using helium at 150 psig. In this case, the leak test, which was performed in accordance with ANSI N14.5, had to show that the leak rate was ≤ 1 × 10-7 std cm^3/s air. The sensitivity of the leak test was ≤ 5 × 10-8 std cm^3/s air. The leak test had to be performed during the 12 months preceding the vessel's use for actual shipment. Personnel performing the helium leak test were required to verify helium fill of the pressure vessel by valving-off the helium supply during the fill process and checking to see if the pressure dropped. A drop in pressure indicates that helium is flowing into the CV. If the CV side of the regulator immediately rises to regulator pressure, the flow passageway is blocked. The blockage must be cleared before a valid leak test can be performed. In another example, a CV was leak-tested at 1-year intervals. Helium gas at 100 ± 5 psig was introduced into the CV through the leak-test port in the vessel top. As per the ANSI code, the flange closure was leak-tested by using the helium mass spectrometer sniffer with sensitivity of ≤ 5 × 10-8 std cm^3/s air. In this case, the acceptable leakage was ≤ 1 × 10-7 std cm^3/s air.

At one site, two methods are used to leak-test the CV. One method is the pressure test, in which the CV is filled with helium and then sniffed with a helium mass spectrometer. The other method is the vacuum test in which helium is sprayed on the vessel's outsides. Some leaks are one-directional. Accordingly, performing both tests wherever possible is desirable. The spray test seems to work better. Helium sniffing is subject to the "cleanliness" of the surrounding atmosphere—that is, the freedom from helium, which can accumulate.

TESTS PERFORMED PRIOR TO SHIPMENT

After the contents are placed into the CV and the vessel lid is secured, a post-load leak test should be performed in accordance with ANSI N14.5-1997. This test measures the pressure rise in the annulus between the O-rings. The measured leak rate must be less than 1 × 10-3 std cm^3/s for test acceptance. The sensitivity should be documented with the test results. If leakage is determined to exceed the acceptable limit, the condition must be corrected prior to shipment. This correction may involve removal of the lid, examination and/or replacement of the O-rings, and examination of the seal contact surfaces. If the leakage criterion cannot be satisfied, the package must be set aside and not used until it is brought into compliance.

Component and Material Tests

Valves, if present, should be tested before each usage and replaced as necessary. If the package design employs any copper gaskets, these should be replaced prior to each shipment. Frequently, packages are purposely specified to be used without a drum gasket. The resulting strong closure offers better impact protection. Also, in a fire the gasket would burn and leave an open path for further combustion.

The CV O-rings should be visually inspected for surface defects and discontinuities, such as rough or porous conditions. Each time the package is used, a leak test to better than 1 × 10-3 std cm^3/s air must be performed to verify the sealing ability of the O-rings. O-rings must be replaced when they no longer satisfy the visual or leak tests. Often the package's maintenance procedure requires replacement of O-rings at the time of the annual leak test.

All O-rings must be individually wrapped to prevent damage in shipment and labeled to ensure traceability. Certification of materials and size shall be furnished by the vendor. All O-rings must be examined for defects before use. O-rings must be controlled spare parts. O-rings should be retained for use until an established shelf life limit is reached.

Thermal Tests

For packages that transport radioactive material with decay heat, a maintenance schedule should be developed to ensure heat is being dissipated from the package at the design rate. Therefore, the developer of the packagings' maintenance schedules should be aware of the packagings' components, whose heat transfer capabilities degrade with time and use. The frequencies of inspections and tests in the maintenance schedule should be sufficient to ensure all packaging components that degrade with time and use are functioning properly for each shipment.

Methods used to detect degradation of packaging components are visual inspection, measurements, and testing.

As stated in 10 CFR 71.87, "Routine Determinations," prior to each shipment of licensed material the licensee must ensure the temperature of the accessible surface of the package will not exceed the limits specified in 10 CFR 71.43(g) at any time during transportation. This test can be accomplished by placing a heat source equivalent to the heat generated by the decay of the actual authorized contents in the package for testing. As applicable, the thermal gradients should be recorded for each anticipated shipment configuration and correlated with the decay heat. This process is used to determine whether the heat that will be generated by the decay of the authorized radioactive material contents will be dissipated at the design rate.

All packaging components that are important to the thermal performance of the package should be inspected and measured. All maintenance repairs should be made according to an approved procedure, and degraded parts should be replaced.

At one site, a random sample of a particular type of packaging is selected at twelve-month intervals. The CV is instrumented by applying maximum temperature indicating labels or the equivalent to the outer surface. The labels selected indicate temperatures in the range of 200°F to 280°F in approximately 20°F increments. The package is then used for shipment and the temperature-indicating labels are surveyed upon disassembly. If the temperature criterion is exceeded, the information is reported to the shipping container design group for correction of the deficiency.

In the case of another packaging, each packaging is loaded with the maximum quantity of plutonium oxide desired to be shipped, allowed time to equilibrate, and the exterior temperatures, at predetermined locations on the package surface, are measured and recorded. The exterior surface temperature of the package and CVs is verified to remain within the range of values established by analysis and/or testing as safe limits. This evaluation is made on every new packaging and all packagings in use at two-year intervals.

12. NNSA package certification engineers must demonstrate a working level knowledge of the required QAP information required in chapter 9 of a SARP.

a. Discuss the purpose and scope of 10 CFR 71, Subpart H.

The following is taken from 10 CFR 71.101.

10 CFR 71, Subpart H describes QA requirements applying to design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of components of packaging that are important to safety. As used in this subpart, QA comprises all those planned and systematic actions necessary to provide adequate confidence that a system or component will perform satisfactorily in service. QA includes quality control, which comprises those QA actions related to control of the physical characteristics and quality of the material or component to predetermined requirements. The licensee, certificate holder, and applicant for a CoC are responsible for the QA requirements as they apply to design, fabrication, testing, and modification of packaging. Each licensee is responsible for the QA provision that applies to its use of a packaging for the shipment of licensed material subject to 10 CFR 71.

b. Discuss, at a summary level, the required elements of 10 CFR 71, Subpart H.

10 CFR 71.101, Quality Assurance Requirements

This subpart describes quality assurance requirements applying to design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of components of packaging that are important to safety. As used in this subpart, "quality assurance" comprises all those planned and systematic actions necessary to provide adequate confidence that a system or component will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to control of the physical characteristics and quality of the material or component to predetermined requirements. Each certificate holder and applicant for a package approval is responsible for satisfying the quality assurance requirements that apply to design, fabrication, testing, and modification of packaging subject to this subpart. Each licensee is responsible for satisfying the quality assurance requirements that apply to its use of a packaging for the shipment of licensed material subject to this subpart.

10 CFR 71.103, Quality Assurance Organization

The licensee, certificate holder, and applicant for a CoC shall be responsible for the establishment and execution of the QAP. The licensee, certificate holder, and applicant for a CoC may delegate to others, such as contractors, agents, or consultants, the work of establishing and executing the QAP, or any part of the QAP, but shall retain responsibility for the program.

These activities include performing the functions associated with attaining quality objectives and QA functions.

10 CFR 71.105, Quality Assurance Program

The licensee, certificate holder, and applicant for a CoC shall establish, at the earliest practicable time consistent with the schedule for accomplishing the activities, a QAP that complies with the requirements of 10 CFR 71.101, “Quality Assurance Requirements,” through 10 CFR 71.137, “Audits.” The licensee, certificate holder, and applicant for a CoC shall document the QAP by written procedures or instructions and shall carry out the program in accordance with those procedures throughout the period during which the packaging is used. The licensee, certificate holder, and applicant for a CoC shall identify the material and components to be covered by the QAP, the major organizations participating in the program, and the designated functions of these organizations.

10 CFR 71.107, Package Design Control

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

10 CFR 71.109, Procurement Document Control

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

10 CFR 71.111, Instruction, Procedures, and Drawings

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

10 CFR 71.113, Document Control

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

10 CFR 71.115, Control of Purchased Material, Equipment, and Services

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

10 CFR 71.117, Identification and Control of Materials, Parts, and Components

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

10 CFR 71.119, Control of Special Processes

The licensee, certificate holder, and applicant for a CoC shall establish measures to ensure special processes, including welding, heat treating, and nondestructive testing are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

10 CFR 71.121, Internal Inspection

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

10 CFR 71.123, Test Control

The licensee, certificate holder, and applicant for a CoC shall establish a test program to ensure all testing required to demonstrate that the packaging components will perform satisfactorily in service is identified and performed in accordance with written test procedures that incorporate the requirements of 10 CFR 71.123 and the requirements and acceptance limits contained in the package approval. The test procedures must include provisions for ensuring all prerequisites for the given test are met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions. The licensee, certificate holder, and applicant for a CoC shall document and evaluate the test results to assure that test requirements have been satisfied.

10 CFR 71.125, Control of Measuring and Test Equipment

The licensee, certificate holder, and applicant for a CoC shall establish measures to ensure tools, gauges, instruments, and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated, and adjusted at specified times to maintain accuracy within necessary limits.

10 CFR 71.127, Handling, Storage, and Shipping Control

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

10 CFR 71.129, Inspection, Test and Operating Status

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

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

10 CFR 71.131, Nonconforming Materials, Parts, and Components

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

10 CFR 71.133, Corrective Action

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

10 CFR 71.135, Quality Assurance Records

The licensee, certificate holder, and applicant for a CoC shall maintain sufficient written records to describe the activities affecting quality. The records must include the instructions, procedures, and drawings required by 10 CFR 71.111 to prescribe QA activities and must include closely related specifications such as required qualifications of personnel, procedures, and equipment. The records must include the instructions or procedures that establish a records retention program consistent with applicable regulations and designates factors such as duration, location, and assigned responsibility. The licensee, certificate holder, and applicant for a CoC shall retain these records for three years beyond the date when the licensee, certificate holder, and applicant for a CoC last engage in the activity for which the QAP was developed. If any portion of the written procedures or instructions is superseded, the licensee, certificate holder, and applicant for a CoC shall retain the superseded material for three years after it is superseded.

10 CFR 71.137, Audits

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

c. For a DOE/NNSA drum-type package, discuss the quality levels typically assigned to packaging components.

The following is taken from NNSA SG-100.

When inspecting a drum-type packaging, the components critical to the structural integrity of the package, such as the drum body and lid and associated sealing surfaces should be examined. For example, visual inspections of the following items should be performed: the sealing surfaces to ensure they are clean; the drum body to ensure there are no scratches or gouges or other gross defects that could compromise the drum's performance; the gasket (when present) to ensure it is in good condition, flexible, and makes a good seal; the weld seam to ensure it has not split; the vent holes to ensure they are present and plugged with the specified plug, if applicable; the bolts to ensure they are the specified bolts and that there is no damage that could reduce their effectiveness; the insulation, if accessible, to ensure its thickness is adequate (a tape measure should be used to check the radial clearance between the insulation and the drum and the axial clearance between the lid and the insulation); the marking and labeling to ensure they are appropriate for the intended use of the packaging, and the required paperwork, as defined in the applicable QAP, to ensure the correct forms and information are provided.

d. Discuss the requirements for obtaining NNSA regulatory approval for packaging component changes.

The following is taken from 10 CFR 71.31.

An application for modification of a package design, whether for modification of the packaging or authorized contents, must include sufficient information to demonstrate that the proposed design satisfies the package standards in effect at the time the application is filed.

The following is taken from 10 CFR 71.19.

NRC will approve modifications to the design and authorized contents of a type B package, or a fissile material package, previously approved by NRC, provided

- the modifications of a type B package are not significant with respect to the design, operating characteristics, or safe performance of the containment system, when the package is subjected to the tests specified in 10 CFR 71.71 and 10 CFR 71.73;
- the modifications of a fissile material package are not significant, with respect to the prevention of criticality, when the package is subjected to the tests specified in 10 CFR 71.71 and 10 CFR 71.73; and
- the modifications to the package satisfy the requirements of 10 CFR 71.

e. Discuss QA requirements for being approved as an authorized user of a DOE/NNSA type B package.

The following is taken from the U.S. Department of Transportation, Pipeline and Hazardous Material Safety Administration, *Approvals Program Standard Operating Procedure.*

Upon receipt of an application for certification, three evaluations are conducted: sufficiency of application; safety evaluation of application; and fitness evaluation of the applicant and registered users of the certificate.

Sufficiency evaluation—The project/technical officer ensures the application contains the information required by the regulations. For package design certifications, the application must contain an application letter, a copy of the package design certificate from the NRC, DOE or foreign country, and a safety analysis report meeting the application guidelines of NRC Regulatory Guide 7.9. For material classification certifications, the application must include an application letter, description and specification of material design and technical data showing compliance with regulations and testing. When an application is incomplete, the project/technical officer has the option of requesting additional information from the applicant or rejecting the application as incomplete. For most cases, the project officer requests the data by phone or email. In some cases, letters requesting additional information are sent. Letters are sent for all rejected applications.

Safety evaluation—Once the application is complete, the project/technical officer determines if the design or material classification is safe and complies with the applicable domestic and international regulations. This evaluation is documented by a SER for package designs and a special form evaluation sheet for material classifications.

Fitness evaluation—The project/technical officer determines if the applicant and parties registered to use the certification are fit to conduct the activity authorized by the competent authority certification. Fitness evaluation entails review of the applicant's compliance and incident and event history.

f. Discuss the NRC Regulatory Guide 7.10, *Establishing Quality Assurance Programs for Packaging Used in Transport of Radioactive Material*.

This regulatory guide provides guidance for use in developing QAPs for packaging to be used in shipping type B and fissile radioactive materials. This guide also provides guidance for use in preparing and submitting QAP descriptions for review by the staff of the NRC. This guidance describes a method that is acceptable to the NRC staff for complying with the related regulatory requirements in 10 CFR 71. Specifically, 10 CFR 71.37(a), "Quality Assurance," requires that applicants requesting package design approval must describe, with respect to Subpart H of 10 CFR 71, the QAPs that they will apply in designing, fabricating, assembling, testing, maintaining, repairing, modifying, and using the proposed packaging. In addition 10 CFR 71.101 requires that licensees, certificate holders, and applicants for a CoC must implement and use a QAP that the NRC staff has previously approved as satisfying the provisions of Subpart H of 10 CFR 71. Specifically, Subpart H requires, in part, that QAPs of licensees, certificate holders, and applicants for a CoC must satisfy each of the applicable criteria specified in 10 CFR 71.101–71.137 to an extent that is consistent with their respective importance to safety.

g. Discuss the QA program information required in Chapter 9 of a SARP.

Chapter 9 of a SARP discusses the material quality of the following. Please refer to NNSA SG-100 for a complete description of this information:

- Austenitic stainless steel
- Carbon steel
- High-strength steel
- Aluminum alloy 6061
- Insulating board (cellulosic fiber)
- Inorganic refractory
- Ceramic fiber material
- Fir plywood
- Redwood
- Urethane foam
- Silicone foam
- Ethylene propylene elastomer

13. NNSA package certification engineers must demonstrate a working level knowledge of the required elements and their evaluation within the probabilistic risk assessment section of a TSRA.

a. Describe the different risks addressed in the probabilistic risk assessment:

- **Risk of a criticality event and its consequences**
- **Risk of special nuclear material release and its consequences**
- **Risk of hazardous material release and its consequences**
- **Consequences of a combined release event with two or three of the above risks**

The following is taken from NNSA SG-500.

The TSRA will contain a probabilistic risk assessment (PRA) for those requirements in 10 CFR 71 that cannot be met by the package. The PRA confirmatory analysis includes an independent assessment of the variables and basic data used to calculate accident frequencies and consequences, critical review of the accident progression models (i.e., combinations of events that lead to a release of radioactive materials), hand calculations that attempt to duplicate the results of the accident frequency calculations, validation of release fractions used in the analysis, application of alternative atmospheric dispersion models to calculate concentrations of released radioactive and hazardous materials at various receptor locations, and verification that risk-related conclusions are valid. If necessary, an independent PRA will be prepared to verify the results in the TSRA.

An assessment of the frequencies of accidents and the consequences of those accidents to members of the public is required. The PRA should include details on the shipping campaign, the number of units in a shipment, number of potential shipments, routes, and any other campaign information that would have potential impact on public safety or the environment.

The PRA should determine consequences and frequencies of accidents using input from structural, thermal, containment, and criticality analyses of the packaging system as appropriate. The significant potential hazards (flammable gas, explosives, toxic, hazardous when wet, etc.) in the package payload should be considered in the PRA with the dominant contributors to accident frequency and consequences identified. Other potential accidents that the 10 CFR 71 addresses may be considered given any unique transportation environments.

A composite summary of the risk to the public presented by the shipping campaign should be provided. The PRA should estimate the frequency and consequences of potential accidents. The PRA should include sensitivity/uncertainty information to support decision-making and development of ACs and operating procedures, where appropriate. Risk may be subdivided by accident type (for instance, criticality versus particulate dispersion) or hazard type (radiological versus toxicological). Transportation authorization will be based on the estimation of composite public risk and will be decided on a case-by-case basis. The overall risk consequence should be express in person-rem for a population or for the maximum exposed individual or radiation dose (mrem/h) at some distance from the source. The likely shipping configurations will have a direct impact on the assessment and the TSS may be included in the assessment for the HAC.

Video 6. Risk assessment and job hazard analysis
https://www.youtube.com/watch?v=dBKsdSDMIvc

b. Discuss the differences between direct dose and latent dose.

Direct Dose
The following is taken from DOE-HDBK-1121-2008 (Archived).

Direct dose is the measurements of radioactive material in the human body using instrumentation that detects radiation emitted from the radioactive material in the body.

Latent Dose
The following is taken from DOE-HDBK-1122-2009.

Latent dose is an asymptomatic period between the prodromal stage and the onset of symptoms of later stages. The higher the dose the shorter the latent phase. At sufficiently high doses the latent phase effectively disappears.

c. Describe the following aspects of dose reduction:
- **Time**
- **Distance**
- **Shielding**
- **Inverse square law**
- **ALARA**

The following descriptions are taken from DOE-HDBK-1130-2008.

Time

Reducing the time spent in a field of radiation will lower the dose received by the workers:

- Plan and discuss the task thoroughly prior to entering the area. Use only the number of workers actually required to do the job.
- Have all necessary tools present before entering the area.
- Use mock-ups and practice runs that duplicate work conditions.
- Take the most direct route to the job site if possible and practical.
- Never loiter in an area controlled for radiological purposes.
- Work efficiently and swiftly.
- Do the job right the first time.
- Perform as much work outside the area as possible. When practical, remove parts or components to areas with lower dose rates to perform work.
- Do not exceed stay times. In some cases, the radiological control organization may limit the amount of time a worker may stay in an area due to various reasons. This is known as "stay time," which should not be exceeded.

Distance

The worker should stay as far away as possible from the source of radiation:

- Stay as far away from radiation sources as practical given the task assignment. For point sources, the dose rate follows a principle called the inverse square law. Be familiar with radiological conditions in the area.
- During work delays, move to lower dose rate areas.
- Use remote handling devices when possible.

Shielding

Shielding reduces the amount of radiation dose to the worker. Different materials shield a worker from the different types of radiation:

- Take advantage of permanent shielding, such as nonradiological equipment/structures.
- Use shielded containments when available.
- Wear safety glasses/goggles to protect eyes from beta radiation, when applicable.
- Temporary shielding can only be installed when proper procedures are used.
- Shielding reduces the amount of radiation dose to the worker. Different materials shield a worker from the different types of radiation.
- Temporary shielding will be marked or labeled with wording such as "Temporary Shielding—Do Not Remove Without Permission from Radiological Control."
- Once temporary shielding is installed, it cannot be removed without proper authorization.
- When evaluating the use of shielding, the estimated dose saved is compared to the estimated dose incurred during shield installation and removal.

Inverse Square Law

This law states that at double the distance, the dose rate falls to 1/4 of the original dose rate. When tripling the distance, the dose rate falls to 1/9 of the original dose rate.

As Low As Reasonably Achievable (ALARA)

ALARA is an approach to radiation safety that strives to manage and control doses (both individual and collective) to the work force and the general public to as low as is reasonable, taking into account social, technical, economic, practical, and public policy considerations.

d. Discuss the following and their use in consequence assessment:

- **Emergency response planning guidelines (ERPG)**
- **Temporary emergency exposure limits (TEEL)**
- **Subcommittee on consequence assessment and protective actions (SCAPA)**

The following is taken from DOE-HDBK-1046-2008.

Emergency Response Planning Guidelines (ERPG)

ERPG-1 is the maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing other than mild transient adverse health effects or perceiving a clearly defined, objectionable odor.

ERPG-2 is the maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing irreversible or other serious health effects or symptoms that could impair an individual's ability to take protective action.

ERPG-3 is the maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing life-threatening health effects.

Temporary Emergency Exposure Limits (TEEL)

TEEL-0 is the threshold concentration below which most people will experience no appreciable risk of health effects.

TEEL-1 is the maximum concentration in air below which it is believed nearly all individuals could be exposed without experiencing other than mild transient adverse health effects or perceiving a clearly defined, objectionable odor.

TEEL-2 is the maximum concentration in air below which it is believed nearly all individuals could be exposed without experiencing or developing irreversible or other serious health effects or symptoms that could impair their abilities to take protective action.

TEEL-3 is the maximum concentration in air below which it is believed nearly all individuals could be exposed without experiencing or developing life-threatening health effects.

Subcommittee on Consequence Assessment and Protective Actions (SCAPA)

The following is taken from DOE-HDBK-1046-2008.

SCAPA is an outgrowth of the DOE Subcommittee on Dose Assessment in the Office of Emergency Management (OEM). Its mission, as formulated in its charter, is to "support the Office of Emergency Management by developing and disseminating, throughout the DOE/NNSA community, technical guidance, recommendations, and resources to improve emergency preparedness, consequence assessment capabilities, and the formulation of protective actions."

SCAPA is composed of Federal employees and contractors from a wide spectrum of DOE and NNSA facilities. The OEM appoints the chairperson.

Recognizing that acute exposure guideline levels (AEGLs) and ERPGs exist only for a limited number of chemicals, SCAPA developed TEELs so that DOE facilities can conduct appropriate hazard analyses and consequence assessments for chemicals lacking AEGLs or ERPGs.

SCAPA has a number of working groups, including the chemical exposures working group and the chemical mixtures working group. One of the principal tasks of the chemical exposures working group has been developing TEELs. It has provided peer review and oversight of TEELs since approximately 1997.

TEELs were first developed in the early 1990s. Since then, SCAPA has supported the development of TEELs, and updates are presented at most of its meetings. The TEEL methodology was first approved by DOE and was incorporated into its revised emergency management guidelines in 1999.

14. NNSA package certification engineers must demonstrate a working level knowledge of the required format and content of a HAR.

a. Discuss the basic information required to be included in a HAR.

The following is taken from DOE-STD-3009-94.

The initial analytical effort for all facilities is a hazard analysis (HA) that systematically identifies facility hazards and accident potentials through hazard identification and hazard evaluation. The focus of the HA is on thoroughness and requires evaluation of the complete spectrum of hazards and accidents. This largely qualitative effort forms the basis for the entire safety analysis effort, including specifically addressing defense-in-depth (DID) and protection of workers and the environment. Basic industrial methods for HA, its interface with more structured quantitative evaluations, and the basis for both have been described in references such as the American Institute of Chemical Engineers Guidelines for Hazard Evaluation Procedures. The Occupational Safety and Health Administration has accepted these guidelines as the standard for analytical adequacy in characterizing commercial chemical processes that perform the same type of unit operations conducted at DOE nonreactor nuclear facilities. Appropriately applied, they help fulfill the requirements of documented safety analyses (DSAs) for hazard category 2 and 3 facilities as specified in 10 CFR 830.

Chapter 3 of a DSA covers the topics of hazard identification, facility hazard categorization, hazard evaluation, and accident analysis. Expected products of this chapter, as applicable based on the graded approach, include the following:

- Description of the methodology for and approach to hazard and accident analyses.
- Identification of hazardous materials and energy sources present by type, quantity, form, and location.

- Facility hazard categorization, including segmentation in accordance with DOE-STD-1027, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*.
- Identification in the HA of the spectrum of potential accidents at the facility in terms of largely qualitative consequence and frequency estimates. The summary of this activity will also include
  - identification of planned design and operational safety improvements;
  - summary of DID, including identification of safety-significant (SS) structures, systems, and components (SSCs), specific administrative controls (SACs) and other items needing technical safety requirement (TSR) coverage in accordance with 10 CFR 830;
  - summary of the significant worker safety features, including identification of SS SSCs and any relevant programs to be covered under TSR and ACs, including those controls designated as SACs;
  - summary of design and operational features that reduce the potential for large material releases to the environment; and
  - identification of the limited set of unique and representative accidents to be assessed further in accident analysis.
- Accident analysis of design basis accidents identified in the HA. The summary of this activity will include for each accident analyzed
  - an estimation of source term and consequence;
  - documentation of the rationale for binning frequency of occurrence in a broad range in HA (detailed probability calculations not required); and
  - documentation of accident assumptions and identification of safety-class SSCs based on the evaluation guideline.

The HA section describes the hazard identification and evaluation performed for the facility. The purpose of this information is to present a comprehensive evaluation of potential process-related, natural events, and man-made external hazards that can affect the public, workers, and the environment due to single or multiple failures.

Consideration will be given to all modes of operation, including startup, shutdown, and abnormal testing or maintenance configurations. As is standard industrial practice, examination of all modes of operation considers the potential for equipment failure and human error.

Hazard identification and evaluation provide a thorough, predominantly qualitative evaluation of the spectrum of risks to the public, workers, and the environment due to accidents involving any of the hazards identified. The evaluation identifies preventive and mitigative features, including identification of expected operator response to incidents and provisions for operator protection in the accident.

b. Discuss the information required to be included in a hazardous materials table.

The following is taken from 49 CFR 172.101.

The hazardous materials table designates the materials listed therein as hazardous materials for the purpose of transportation of those materials. For each listed material, the table identifies the hazard class or specifies that the material is forbidden in transportation, and gives the proper shipping name or directs the user to the preferred proper shipping name. In addition, the table specifies or references requirements pertaining to labeling, packaging, quantity, limits aboard aircraft, and stowage of hazardous materials aboard vessels.

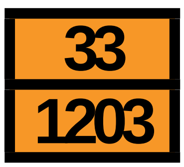
c. Describe the purpose and use of Technical Publication (TP) 20-11 hazardous material categories/designators and the Department of Transportation (DOT) *Emergency Response Guidebook* (ERG) numbers

Technical Publication (TP) 20-11

DOE Technical Publication TP 20-11, *General Firefighting Guidance*, provides line numbers and hazardous category cross reference lists. It is intended for use by firefighting personnel. The publication applies to emergencies where nuclear weapons, nuclear weapon components, or related nonnuclear weapon types are involved in a fire.

DOT Emergency Response Guidebook

Hazard identification numbers used under European and some South American regulations may be found in the top half of an orange panel on some intermodal bulk containers. The United Nations 4-digit identification number is in the bottom half of the orange panel.



The hazard identification number in the top half of the orange panel consists of two or three digits. In general, the digits indicate the following hazards:

- 2—Emission of gas due to pressure or chemical reaction
- 3—Flammability of liquids (vapors) and gases or self-heating liquid
- 4—Flammability of solids or self-heating solid
- 5—Oxidizing (fire-intensifying) effect
- 6—Toxicity or risk of infection
- 7—Radioactivity
- 8—Corrosivity
- 9—Risk of spontaneous violent reaction

NOTE: The risk of spontaneous violent reaction within the meaning of digit 9 include the possibility, due to the nature of a substance, of a risk of explosion, disintegration and polymerization reaction followed by the release of considerable heat or flammable and/or toxic gases.

Doubling of a digit indicates an intensification of that particular hazard (e.g., 33, 66, 88).

Where the hazard associated with a substance can be adequately indicated by a single digit, the digit is followed by a zero (e.g., 30, 40, 50).

A hazard identification number prefixed by the letter “X” indicates that the substance will react dangerously with water (e.g., X88).

d. Discuss the requirements for analyzing potential hazards and their associated consequences.

The following is taken from DOE-STD-1027-92.

The HA process consists of the identification of the relative and absolute hazards of the materials in a facility. The objective is to focus the safety assessment effort on those hazards that have the potential to present significant, nonroutine concerns to the worker, the public, and the environment.

HA is the initial step in the process of identifying and evaluating potential accidents in a facility. It is used to identify the hazardous chemical or radioactive material in a process or facility and the energy sources and initiating events that could lead to the potential consequences of an accident.

The objectives of HA are to 1) identify the hazards contained in a facility; 2) perform final hazard categorization according to Section 3 and Attachment 1 of DOE-STD-1027-92, based on the hazardous material quantity identified and the energy sources and initiating events identified; 3) provide an overall assessment of the importance of the various hazards; 4) identify occupational hazards and related DOE-prescribed standards; and 5) characterize and analyze the remaining nonroutine hazards that are unique and representative hazards to be analyzed in the DSA. To accomplish these objectives, each facility preparing a DSA must perform an HA.

HA consists of collecting and integrating four interrelated sets of information:

1. Hazardous material quantity, form, and location
2. Energy sources and potential initiating events
3. Preventive features
4. Mitigative features

Hazardous Materials Quantity, Form, and Location

HA identifies the hazardous chemical and radiological materials at risk in the facility. The quantity of material is assumed to be the maximum inventory permitted to be processed or present in specific locations in the facility. This quantity is generally determined from either process flow information or existing facility operating experience. Examples of material form are powder, metal, sludge, gas, solid waste, and liquid. Location indicates the part of the building, glovebox, or process line in which the hazardous material is present.

Occupational hazards, including common industrial hazards, should be identified, and the applicable DOE-prescribed occupational safety and health regulations, standards, and analyses should be referenced in the DSA.

Energy Sources and Potential Initiating Events

HA then identifies potential energy sources and potential initiating events that could affect the hazardous material and lead to a release of material or other occurrence. Such events include internally initiated events, process-initiated events, and externally initiated events. Inherent energies within the process should also be described.

Those accident initiators inappropriate for the facility or process under consideration should be eliminated, and the HA should include the rationale for doing so. For example, if the process does not include any liquid material and the potential for spill does not exist, this potential initiator should be eliminated at the HA level, with a brief discussion of the reasons for the elimination.

Preventive Features

HA identifies any SSC that serves to prevent the release of hazardous material in an accident scenario. Preventive features may include passive barriers such as piping, material containers, material cladding, and gloveboxes, or facility SSCs such as pressure relief valves, monitoring systems for material concentrations with automatic actions to stop or isolate the process, or dilution systems to control explosive or flammable mixtures. The discussion should begin with the preventive feature closest to the hazardous material or mixture, end with the preventive feature farthest from the hazardous material or mixture, and include all preventive features that may contribute to preventing the release of the hazardous chemical or radioactive material.

Mitigative Features

HA identifies any SSC that serves to mitigate the consequences of a release of hazardous materials in an accident scenario. Mitigative features may include passive barriers such as dikes, confinement systems, or containment systems, or active systems or components such as air cleanup systems, sump systems, dilution systems, and liquid cleanup system. The discussion should begin with the mitigative feature closest to the point of uncontrolled release, end with the mitigative feature farthest from the hazardous material or mixture, and include all mitigative features that may contribute to reducing the consequences of a release of the hazardous chemical or radioactive material to affected onsite and offsite populations.

e. Discuss the use of post-test inspection/evaluation procedures to ensure special assemblies are safe to ship.

The following is taken from NNSA SG-500.

DOE/NNSA PCD must be notified prior to offering the post-test unit for shipment anytime observations during the test or the post-test inspection indicate that the unit experienced an abnormal test. DOE/NNSA PCD shall perform a safety assessment of the unit and authorize its shipment before it can be offered for shipment. The DOE/NNSA PCD assessment may include the development and imposition of additional administrative measures that are designed to mitigate the hazards associated with the post-test unit.

15. NNSA package certification engineers must demonstrate a working level knowledge of the following guides and regulations:

- **NRC Regulatory Guide 7.9, *Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material***
- **NNSA SC Safety Guide 500, *Defense Programs Package Certification and Offsite Transportation Authorization Guide***

a. Discuss the purpose, scope, and applicability of each document.

NRC Regulatory Guide 7.9, Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material

NRC Regulatory Guide 7.9 provides guidance on preparing applications for approval of type B and fissile material transportation packages. This guidance describes a method that is acceptable to the staff of the NRC for complying with the agency's regulations in 10 CFR 71. This guidance is not intended as an interpretation of NRC regulations, within the context of 10 CFR 71.2. Nothing contained in this guide is to be construed as having the force or effect of NRC regulations, or as indicating that applications supported by safety analyses and prepared in accordance with the recommendations of this regulatory guide will necessarily be approved, or as relieving any licensee from the requirements of 10 CFR 71 or any other pertinent regulations.

The primary purpose of this regulatory guide is to assist applicants in preparing applications that thoroughly and completely demonstrate the ability of the given packages to meet the regulations. In addition to package approval, applicants must have an approved QAP in accordance with the provisions of 10 CFR 71.101–71.137. NRC Regulatory Guide 7.10, "Establishing Quality Assurance Programs for Packaging Used in Transport of Radioactive Material," provides guidance on the details of developing QAPs. The NRC may request additional information in support of an application, if such information is necessary to provide reasonable assurance of the ability of the package to meet the regulations. In preparing applications for package approval, applicants may find it useful to refer to other regulatory guides in Division 7, "Transportation," of the NRC's regulatory guide series.

NNSA Service Center Safety Guide 500, Defense Programs Package Certification and Offsite Transportation Authorization Guide

The purpose of NNSA SC SG-500 is to establish and disseminate standardized DOE/NNSA processes for establishing and maintaining packaging programs that ensure offsite shipments of nuclear weapon components, category I and II SNM, nuclear weapon program special assemblies, and other MNSI are safe, comply with the applicable Federal regulatory and DOE O 461.1B requirements, and present no undue risk to the public, worker safety and health, or the environment.

b. Discuss the role of a package certification engineer in utilizing/applying the above guides and regulations.

The role of the package certification engineer will vary depending on the site and the package configuration. The Qualifying Official will evaluate the completion of this KSA.

16. NNSA package certification engineers must demonstrate a working level knowledge of the process for certifying United Kingdom (UK) packages within the United States (U.S.).

a. Discuss the scope and content of the memorandum of arrangement between the U.S. and UK for the certification of packages and transport of radioactive materials.

Note: This is a classified agreement and is not available for reproduction in this reference guide.

b. Discuss the International Atomic Energy Agency (IAEA) *Regulations for the Safe Transport of Radioactive Material (SSR-6, 2012)* and IAEA *Safety Guide* (SSG-26, 2012) in regards to UK shipments.

IAEA SSR-6

IAEA SSR-6 establishes standards of safety that provide an acceptable level of control of the radiation, criticality, and thermal hazards to persons, property, and the environment that are associated with the transport of radioactive material. Compliance with these regulations is deemed to satisfy the principles of the basic safety standards in respect of transport. The prime responsibility for safety must rest with the person or organization responsible for facilities and activities that give rise to radiation risks.

IAEA Safety Guide SSG-26

In general, the transport regulations aim to provide a uniform and adequate level of safety that is commensurate with the inherent hazard presented by the radioactive material being transported. To the extent feasible, safety features are required to be built into the design of the package. By placing primary reliance on the package design and preparation, the need for any special actions during carriage (i.e., by the carrier) is minimized. Nevertheless, some operational controls are required for safety purposes.

Video 7. Safe transport of radioactive materials
https://www.youtube.com/watch?v=GW0IKvQ9saM

c. Discuss the 10 CFR air transport requirements for radioactive material, and describe any significant differences with IAEA requirements.

The following is taken from *Federal Register 3645.*

IAEA TS-R-1 has introduced two new concepts for the air transport of radioactive material: the type C package and low dispersible material (LDM). Type C packages are designed to withstand severe accident conditions associated with air transport without loss of containment or significant increase in external radiation levels. The LDM is a material exception to these new air transport standards that is granted based on a material's limited radiation hazard and low dispersibility.

If qualified as LDM, material in quantities that would otherwise require a type C package could continue to be transported by aircraft in a type B package. U.S. regulations do not contain a type C package or LDM category, but do have specific requirements for the air transport of plutonium (10 CFR 71.64 and 71.74). These specific NRC requirements for air transport of plutonium will continue to apply.

The type C requirements apply to all radionuclides packaged for air transport that contain a total activity value above 3,000 A_1 or 100,000 A_2, whichever is less, for special form material, or above 3,000 A_2 for all other radioactive material. Below these thresholds, type B packages may be used in air transport.

The type C package performance requirements are significantly more stringent than those for type B packages. For example, a 90 m/s impact test is required instead of the 9-meter drop test. A 60-minute fire test is required instead of the 30-minute for type B packages.

These stringent tests are expected to result in package designs that will survive more severe aircraft accidents than type B package designs.

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

d. Describe the process used to issue an OTC for a package certified by the UK.

The following is taken from NNSA SG-500.

The weapon system program manager shall submit a request to the PCD and the OST for authorization to transport type AF package United Kingdom Ministry of Defense (UK MOD) in the TSS. The PCD manager shall assign a certification engineer or TSRP to evaluate the proposed UK MOD type AF package's compliance with 10 CFR 71 and 49 CFR 173 requirements, and to assess the adequacy of the UK MOD's safety documentation and the package by conducting or managing an independent confirmatory technical assessment. This review shall consist of validation of the proper use of the packaging for the application intended. The PCD shall assure that the proposed configuration is within maintenance and serviceability dates, and can be delivered within prescribed deadlines. Review of DOE/NNSA users operating procedures are completed in accordance to work process defined for U.S. origin type A packaging designs. The certification engineer shall also work with OST to ensure the package is compatible with TSS equipment and the tie-down procedures are contained in DOE Technical Publication 45-51D, *Transportation of Nuclear Weapons Material (Supplement), Shipment by Safe-Secure-Trailer*, or approved in a separate correspondence from OST to PCD before an OTC/OTA is issued.

e. Discuss the National Security Exemption, 10 CFR 871.1, "National Security Exemption."

The following is taken from 10 CFR 871.1.

The following DOE air shipments of plutonium are considered as being made for the purposes of national security within the meaning of section 502(2) of Public Law 94-187:

- Shipments made in support of the development, production, testing, sampling, maintenance, repair, modification, or retirement of atomic weapons or devices
- Shipments made pursuant to international agreements for cooperation for mutual defense purposes
- Shipments necessary to respond to an emergency situation involving a possible threat to the national security

17. NNSA package certification engineers must demonstrate a working level knowledge of the transportation requirements and the interface with the Office of Secure Transportation (OST).

a. Discuss the DOE/NNSA process used to schedule TSS shipments.

The following is taken from DOE O 461.1B.

Secure transportation shipping requirement forecasts must be developed by each program secretarial officer (PSO) and NNSA Deputy Administrator for the ADAST for analyses. The results of the analyses must be provided to the Secure Transportation and Packaging Steering Committee (STPSC) and the Secure Transportation Asset Advisory Board (STAAB) as required.

The shipping forecast horizon, frequency, and information fields must be defined in the shipment forecast and request procedure (SFRP). The forecasts must be sufficient to meet planning and operational needs. PSOs and NNSA Deputy Administrators or their designees shall provide information to the ADAST and STPSC as soon as possible (between submittals) concerning any new campaigns to allow time to integrate the new requirements into the current schedule.

In the event of scheduling conflict, the STPSC may recommend priorities for shipments, subject to the review and approval of the STAAB.

Each site must confirm their shipment needs with a transportation shipping request, no less than 60 days prior to material availability date, with updates before the 30- and 7-day submittal requirements, as indicated on OST transportation shipping request form 1540.5. The transportation shipping request content, format, and mechanism must be defined by OST, as addressed in the SFRP. Requests for significant variances to the TSS schedule must be submitted per the SFRP.

b. Discuss TSS requirements regarding the use of OTAs, OTCs, and OTDs.

The following is taken from DOE O 461.1B.

The NNSA CO will issue a CoC or an OTC to certify a package based on the level of safeguards and security protection required for the package. Packages certified by an OTC must be transported via the TSS and packages certified by a CoC may be transported via an approved commercial carrier. The NNSA CO is responsible for defining the required content, format, and review procedures for the NNSA CoCs, OTCs, SARPs, and supporting documentation.

The NNSA CO documents the approval of these DOE-authorized shipments by issuance of an OTA or an OTD. The approval process requires the applicant to identify risks to public health and safety, workers, and the environment, and to identify and implement risk mitigation measures. These shipments should be evaluated for suitability for shipment by qualified commercial carriers provided the OST ensures that the commercial carrier proposed, approved, and used for the shipment complies with all additional requirements specified for the cargo. An OTA and OTD may be issued for a period not to exceed five years.

c. Discuss the applicable tie-down procedures for MNSI assemblies and type B containers in aircraft and safeguards transporter (SGT) modes of transport.

The information regarding tie-down procedures for special assemblies and type B containers is available in the following DOE technical publications:

- TP 35-51, *General Instructions Applicable to Nuclear Weapons*
- TP 45-51, *Transportation of Nuclear Weapons Materiel, General Shipping and Limited Life Component Data (LLC)*
- TP 45-51A, *Transportation of Nuclear Weapons Materiel (Supplement), Shipping and Identification Data for Stockpile Major Assemblies*
- TP 45-51B, *Transportation of Nuclear Weapons Materiel (Supplement)—Palletized Cargo*

These documents are classified. As a result no information regarding the contents of these documents will be included in this reference guide.

d. Discuss the maximum allowable dose rates and package locations for HAC, and briefly discuss why the transportation safeguards system (TSS) limits radiation dosage to no greater than the nonexclusive use limits in 10 CFR 71.

The following is taken from NNSA SG-100.

The maximum dose rate at one meter from any external surface position of the package under HAC must not exceed 1,000 mrem/h, as specified in 10 CFR 71.51. For HAC, the dose rate of the external package surface is assumed to be that of the CV.

18. NNSA package certification engineers must demonstrate a familiarity level knowledge of the Price-Anderson Amendment Act of 1988 (PAAA) and its relationship to subparts A and B of 10 CFR 830.

a. Describe the purpose of the Price-Anderson Amendment Act.

The following is taken from the American Nuclear Society, *The Price-Anderson Act Background Information*.

The main purpose of the PAAA is to ensure the availability of a large pool of funds to provide prompt and orderly compensation of members of the public who incur damages from a nuclear or radiological incident no matter who might be liable.

The Act provides omnibus coverage, that is, the same protection available for a covered licensee or contractor extends through indemnification to any persons who may be legally liable, regardless of their identity or relationship to the licensed activity. Because the Act channels the obligation to pay compensation for damages, a claimant need not sue several parties but can bring its claim to the licensee or contractor.

b. Discuss the general applicability to the Department's nuclear safety activities.

The following is taken from the DOE Office of Health, Safety, and Security (HSS), Office of Enforcement and Oversight (HS-40), *Safety and Security Enforcement Process Overview*.

For nuclear safety noncompliances, the determination of safety significance is based on the DID approach to nuclear safety embodied in DOE's nuclear safety regulations:

- The extent or severity, or both, of an actual adverse nuclear safety event or condition, or the potential that it could occur
- The extent to which the safety barriers intended to prevent an abnormal or accident condition have been violated, defeated, or not properly established
- The extent to which mitigating safety features intended to protect workers or the public in an abnormal or accident condition have been violated, defeated, or not properly established

c. Describe the general indemnity that DOE offers to contractors.

The following is taken from HSS, HS-40, *Safety and Security Enforcement Process Overview*.

The Atomic Energy Act (AEA) provides indemnification to DOE contractors who manage and operate nuclear facilities in the DOE complex; associated subcontractors and suppliers are included under this coverage. In 1988, the PAAA was signed into law to continue this indemnification. The rules that implement the PAAA subject DOE-indemnified contractors, subcontractors, and suppliers to potential civil penalties for violations of DOE rules, regulations, and compliance orders relating to nuclear safety requirements. As part of its agreement to continue the indemnification coverage, Congress required that DOE-indemnified contractors, subcontractors, and suppliers be made subject to civil penalties for violations of DOE's nuclear safety requirements. On August 17, 1993, DOE published its nuclear safety enforcement procedural rules and enforcement policy, which has since been amended several times. The Director of HS-40 has the responsibility to carry out the statutory enforcement authority provided to DOE in the PAAA.

The Bob Stump National Defense Authorization Act for Fiscal Year 2003 extended previously approved indemnification levels until December 31, 2004, and required DOE to promulgate a final rule to establish and provide for enforcement of worker safety and health requirements. The Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 extended indemnification until December 31, 2006. The Energy Policy Act of 2005 extended indemnification of DOE contractors until December 31, 2025, increased liability coverage to approximately $10 billion (plus inflation adjustments) per incident, and repealed waivers or exclusions for remission of civil penalties for nonprofit organizations upon the signing of a new contract.

The National Defense Authorization Act for Fiscal Year 2000 added new Section 234B to the AEA; on January 26, 2005, DOE published 10 CFR 824, "Procedural Rules for the Assessment of Civil Penalties for Classified Information Security Violations," to implement this new section. Section 234B provides that a DOE contractor or subcontractor who violates any rule, regulation, or Order relating to the safeguarding or security of restricted data and/or other classified or sensitive information shall be subject to a civil penalty. 10 CFR 824 provides that civil penalties will be assessed for violations of requirements for the protection and control of classified information

On February 9, 2006, DOE issued the Worker Safety and Health Program Rule, 10 CFR 851, which includes, in Subpart E, the enforcement process to be applied to violations of the worker safety and health regulation, and, in Appendix B, the enforcement policy for such violations. 10 CFR 851 provides that, beginning May 25, 2007, no work may be performed at a covered workplace unless an approved worker safety and health program is in place.

d. Demonstrate an understanding of the topics below, associated with the PAAA:

- **Documented safety analyses (10 CFR 830, Subpart B)**
- **Unreviewed safety questions (10 CFR 830, Subpart B)**
- **Quality assurance requirements (10 CFR 830, Subpart A)**
- **Technical safety requirements (10 CFR 830, Subpart B)**

Documented Safety Analyses

The following is taken from 10 CFR 830.204.

The contractor responsible for a hazard category 1, 2, or 3 DOE nuclear facility must obtain approval from DOE for the methodology used to prepare the DSA for the facility unless the contractor uses a methodology set forth in table 2 of Appendix A to 10 CFR 830.

The DSA for a hazard category 1, 2, or 3 DOE nuclear facility must, as appropriate for the complexities and hazards associated with the facility

- describe the facility, including the design of safety SSCs) and the work to be performed;
- provide a systematic identification of both natural and man-made hazards associated with the facility;
- evaluate normal, abnormal, and accident conditions, including consideration of natural and man-made external events, identification of energy sources or processes that might contribute to the generation or uncontrolled release of radioactive and other hazardous materials, and consideration of the need for analysis of accidents which may be beyond the design basis of the facility;
- derive the hazard controls necessary to ensure adequate protection of workers, the public, and the environment, demonstrate the adequacy of these controls to eliminate, limit, or mitigate identified hazards, and define the process for maintaining the hazard controls current at all times and controlling their use;
- define the characteristics of the safety management programs necessary to ensure the safe operation of the facility, including (where applicable) QA, procedures, maintenance, personnel training, conduct of operations, emergency preparedness, fire protection, waste management, and radiation protection; and
- with respect to a nonreactor nuclear facility with fissionable material in a form and amount sufficient to pose a potential for criticality, define a criticality safety program that
  - ensures that operations with fissionable material remain subcritical under all normal and credible abnormal conditions;
  - identifies applicable nuclear criticality safety standards; and
  - describes how the program meets applicable nuclear criticality safety standards.

Unreviewed Safety Question

The following is taken from 10 CFR 830.203.

The contractor responsible for a hazard category 1, 2, or 3 DOE nuclear facility must establish, implement, and take actions consistent with an unreviewed safety question (USQ) process that meets the requirements of 10 CFR 830.203.

The contractor responsible for a hazard category 1, 2, or 3 DOE existing nuclear facility must submit for DOE approval a procedure for its USQ process by April 10, 2001. Pending DOE approval of the USQ procedure, the contractor must continue to use its existing USQ procedure. If the existing procedure already meets the requirements of this section, the contractor must notify DOE by April 10, 2001 and request that DOE issue an approval of the existing procedure.

The contractor responsible for a hazard category 1, 2, or 3 DOE new nuclear facility must submit for DOE approval a procedure for its USQ process on a schedule that allows DOE approval in an SER issued pursuant to section 207(d) of 10 CFR 830.

The contractor responsible for a hazard category 1, 2, or 3 DOE nuclear facility must implement the DOE-approved USQ procedure in situations where there is a

- temporary or permanent change in the facility as described in the existing DSA
- temporary or permanent change in the procedures as described in the existing DSA
- test or experiment not described in the existing DSA
- potential inadequacy of the DSA because the analysis potentially may not be bounding or may be otherwise inadequate

A contractor responsible for a hazard category 1, 2, or 3 DOE nuclear facility must obtain DOE approval prior to taking any action determined to involve a USQ.

The contractor responsible for a hazard category 1, 2, or 3 DOE nuclear facility must annually submit to DOE a summary of the USQ determinations (USQD) performed since the prior submission.

If a contractor responsible for a hazard category 1, 2, or 3 DOE nuclear facility discovers or is made aware of a potential inadequacy of the DSA, it must

- take action, as appropriate, to place or maintain the facility in a safe condition until an evaluation of the safety of the situation is completed;
- notify DOE of the situation;
- perform a USQD and notify DOE promptly of the results; and
- submit the evaluation of the safety of the situation to DOE prior to removing any operational restrictions initiated to meet the requirements of 10 CFR 830.

Quality Assurance Requirements

The following is taken from 10 CFR 830.121.

Contractors conducting activities, including providing items or services, that affect, or may affect, the nuclear safety of DOE nuclear facilities must conduct work in accordance with the QA criteria in 10 CFR 830.122, “Quality Assurance Criteria.”

The contractor responsible for a DOE nuclear facility must

- submit a QAP to DOE for approval and regard the QAP as approved 90 days after submittal, unless it is approved or rejected by DOE at an earlier date;

- modify the QAP as directed by DOE;
- annually submit any changes to the DOE-approved QAP to DOE for approval. Justify in the submittal why the changes continue to satisfy the QA requirements; and
- conduct work in accordance with the QAP.

The QAP must
- describe how the QA criteria of 10 CFR 830.122 are satisfied;
- integrate the QA criteria with the safety management system, or describe how the QA criteria apply to the safety management system;
- use voluntary consensus standards in its development and implementation, where practicable and consistent with contractual and regulatory requirements, and identify the standards used; and
- describe how the contractor responsible for the nuclear facility ensures that subcontractors and suppliers satisfy the criteria of 10 CFR 830.122.

Technical Safety Requirements
The following is taken from 10 CFR 830.205.

A contractor responsible for a hazard category 1, 2, or 3 DOE nuclear facility must
- develop TSRs that are derived from the DSA;
- prior to use, obtain DOE approval of TSRs and any change to TSRs; and
- notify DOE of any violation of a TSR.

A contractor may take emergency actions that depart from an approved TSR when no actions consistent with the TSR are immediately apparent, and when these actions are needed to protect workers, the public or the environment from imminent and significant harm. Such actions must be approved by a certified operator for a reactor or by a person in authority as designated in the TSRs for nonreactor nuclear facilities. The contractor must report the emergency actions to DOE as soon as practicable.

A contractor for an environmental restoration activity may follow the provisions of 29 CFR 1910.120 or 1926.65, "Hazardous Waste Operations and Emergency Response," to develop the appropriate hazard controls provided the activity involves either
- work not done within a permanent structure
- the decommissioning of a facility with only low-level residual fixed radioactivity

e. Demonstrate an understanding that violations of applicable nuclear safety rules and regulations are enforceable criminally and civilly.

The following is taken from 48 CFR 952.250-70.

Civil Penalties
The contractor and its subcontractors and suppliers who are indemnified are subject to civil penalties, pursuant to 234A of the AEA, for violations of applicable DOE nuclear safety related rules, regulations, or Orders.

Criminal Penalties

Any individual director, officer, or employee of the contractor or of its subcontractors and suppliers who are indemnified are subject to criminal penalties, pursuant to 223(c) of the AEA, for knowing and willful violation of the AEA of 1954, as amended, and applicable DOE nuclear safety related rules, regulations, or Orders which violation results in, or, if undetected, would have resulted in a nuclear incident.

f. Discuss the role of Federal line management with respect to implementing the requirements of the PAAA.

The following is taken from HSS, HS-40, *Safety and Security Enforcement Process Overview*.

DOE and contractor personnel are expected to ensure strong safety and classified information security compliance and performance; an effective compliance assurance process; timely and proper identification, reporting, and resolution of noncompliances; and effective interface with the enforcement program community.

The overall structure of the DOE safety and security enforcement program includes roles and responsibilities for the HSS, HS-40, DOE line management, and contractors, which include the following:

- The Director of HS-40, within HSS, has program responsibility for the DOE safety and security enforcement programs. To maintain effective interfaces, HS-40 works closely with DOE program office, field element, and contractor management, primarily through individuals serving as enforcement coordinators.
- DOE program office and field element managers have line management responsibility for safety and security and should designate enforcement coordinators to serve as the principal interface with HS-40 and contractors on all enforcement matters.
- Contractor management is responsible for implementing DOE requirements and designating individuals who serve as the principal interface with the corresponding DOE field element enforcement coordinator and HS-40. Enforcement coordinators serve as the principal lead in the contractor organization for issues related to implementation of safety and security regulations, identification and reporting of noncompliances, and enforcement proceedings.

Director, Office of Enforcement and Oversight

The Director manages all enforcement activities, directs technical and legal reviews, oversees the investigative process and the determination/preparation of appropriate enforcement outcomes, and refers potential criminal actions to the Department of Justice and the Inspector General. The Director is authorized to issue enforcement correspondence and levy sanctions, except for those cases involving NNSA contractors where the sanction requires the signature of the Administrator, based on the recommendation of the Director. The Director regularly communicates to senior DOE and contractor management the state of the enforcement program and observations on safety and classified information security compliance issues. The Director also provides guidance for outreach and training related activities that help to facilitate the implementation of DOE’s enforcement program.

HS-40 Enforcement Staff

STAFF MEMBERS

Maintain operational awareness of assigned sites and regularly interface with program office, field element, and contractor enforcement coordinators.

Review and evaluate information on noncompliances, including information reported to the noncompliance tracking system (NTS) and the safeguards and security information management system.

Identify significant noncompliant conditions and recommend investigation, inspection, and/or the outcome of the enforcement proceeding.

Conduct investigations or inspections associated with potential violations of DOE safety and classified information security requirements, and prepare reports and/or technical evaluations.

Participate in enforcement conferences.

Provide recommendations during pre- and post-conference, DOE-only discussions and deliberations.

Prepare initial drafts of enforcement products.

Inform DOE personnel of their obligation to maintain confidentiality on the details of planned enforcement activities and communications.

Conduct regulatory assistance reviews of noncompliance screening, reporting, and self-assessment processes.

Maintain the NTS.

Maintain docket files and retrieval system for enforcement proceedings and other activities requiring an administrative record.

Conduct periodic enforcement outreach, including workshops and site-specific training/familiarization visits, for DOE and contractor enforcement coordinators and managers.

DOE and Contractor Line Management

For effective coordination and to ensure that DOE achieves a high level of safety and security performance, DOE and contractor line management perform several important functions directly related to successful execution of DOE's enforcement program, including the following:

- Demonstrating strong support for the noncompliance screening and reporting process, assessment programs, and the corrective action process
- Designating an individual to serve as the enforcement coordinator, and placing that individual at a sufficiently senior reporting level to facilitate management awareness of regulatory compliance issues
- Maintaining regular and open communication with the contractor, program office, and HS-40 on safety and security, noncompliance conditions, and noncompliance report resolution

- Working with HS-40 to facilitate expeditious resolution and closure of enforcement matters
- Ensuring that staff are available to support and participate in HS-40 investigations or reviews

DOE Enforcement Coordinator

Although DOE's enforcement program applies only to indemnified DOE contractors (and associated subcontractors and suppliers), the DOE headquarters and field element enforcement coordinators play an important role in helping HS-40 enforcement staff understand DOE line management's perspectives on events, program deficiencies, and contractor performance. In addition, DOE enforcement coordinators assist in coordinating site visits, document requests, and other interactions with site contractors.

Contractor Enforcement Coordinator

This individual typically is responsible for key aspects of the contractor's processes for identifying, screening, and reporting noncompliances, and also serves as the primary regulatory-compliance interface with HS-40 for the sharing of information, facilitating site visits (where warranted), and acting as a liaison with senior contractor management to keep them informed of enforcement proceedings. As such, the contractor enforcement coordinator plays a critical role in facilitating the execution of DOE's enforcement program.

19. NNSA package certification engineers must demonstrate the technical writing and assessment/performance skills necessary to execute the responsibilities of a package certification engineer.

<u>Mandatory Performance Activities:</u>

a. **Participate, as a member of a TSRP, in the review of a SARP.**

b. **Write a SER to document review of a HAR.**

c. **Write a SER to document review of a SARP.**

d. **Write an OTA and an OTC.**

e. **Participate in a packaging and transportation appraisal that evaluates DOE/NNSA site performance.**

f. **Participate in the development or revision of a DOE/NNSA packaging safety guidance document.**

The KSAs of this competency statement are performance-based. The Qualifying Official will evaluate their completion.

20. **NNSA package certification engineers must demonstrate a working level knowledge of assessment techniques and DOE O 226,1B, *Implementation of Department of Energy Oversight Policy.***

   a. **Describe the assessment requirements and limitations associated with the interface with contractor employees.**

The following is taken from NNSA/Service Center, Environment, Safety and Health Department, *Assessment Handbook*.

The assessor must be independent of the particular area being assessed with respect to the contractor. The assessor must not have a vested interest in the outcome of an assessment with respect to his or her area of the assessment.

The assessor has no line management oversight over the contractor and particular assessment area and is not in any way a supervisor over those contractors with whom he/she interfaces during the course of the assessment. If a contractor fails to give the assessor information or documents requested, the issue should be turned over to the team leader who will then interface with appropriate DOE/NNSA line management and their contractor counterparts.

The assessment results are a government work product. Aside from the contractor response during factual accuracy processing, the results are not subject to negotiation by the contractor.

   b. **Discuss the essential elements of a performance-based assessment, including:**
      - **Assessment planning**
      - **Defined assessment criteria/checklists**
      - **Investigation**
      - **Fact-finding**
      - **Exit interview**
      - **Reporting**
      - **Follow-up**
      - **Closure**

Assessment Planning

The following is taken from DOE G 414.1-1C.

Performance-based assessments take an approach that focuses first on the adequacy of the process that produced a product or service, and then on the product itself. If problems are found in the product or work processes, the assessor evaluates the methods and procedures used to implement the applicable requirements in an effort to find the failure that led to the problems.

The assessor is expected to determine whether a noncompliance or series of noncompliances with procedures could result in a failure to satisfy top-level requirements. Results of prior compliance assessments may help the assessor in determining the focus areas for planning performance-based assessments.

In performance-based assessments, great emphasis is placed on getting the full story on a problem before coming to a conclusion. If an assessor sees a problem with the execution of a process, the next step should determine the extent of the problem. Is it limited to one area? Is it limited to one process? Can the problem be traced to the qualification program for the operator or to the qualification program for the process? Or is there a problem with the material, indicating a problem such as engineering or procurement?

While the assessor should be familiar with requirements and procedures, in performance-based assessments the assessor's experience and knowledge play an integral part in determining whether requirements are satisfied. Therefore, participants in performance-based assessments should be technically competent in the areas they are assessing.

Assessment programs should be developed to the level of rigor and detail required to ensure adequate review of programs, systems, and processes. An assessment program provides the structure for the overall process and ensures assessments are conducted in a cost-effective, efficient manner. Items considered essential for a comprehensive assessment program include the following:

- Assessment scheduling, planning approach, and logic
- Methodology for determining/developing performance criteria
- Recognition and use of third-party assessment results
- Assessment ethics and behaviors
- Qualification and training of assessment team personnel
- Protocols for conduct, including interfaces and meetings
- Format/review of assessment plans and agendas
- Reporting methods/procedures for concerns, findings, observations, and improvement opportunities, including distribution, and mechanisms for addressing imminent danger issues
- Procedures/processes for concern verification and follow-up
- Assessment records management program, including identification of records that will be retained, retention periods, and protection
- Central repository or website to ensure assessment supporting documentation can be retrieved

Assessments should be conducted commensurate with the hazards, status, and importance of the program goals, systems, or work processes and should be focused on worker health and safety, public health and safety, environmental protection, community concern, strategic planning, organizational resources, compliance and liability, and business efficiency and productivity. Complexity, reliability, risk, and economic issues should also be considered when planning and scheduling assessments. The application of a graded approach using a risk-based decision-making process will ensure resources are applied in a manner that provides the greatest benefit to the assessed organizations and their customers.

Defined Assessment Criteria/Checklists

Assessments seek to ensure performance expectations defined by management and process owners are being met. Assessors should clearly understand the programs, systems, or processes being assessed, including their goals, and associated objectives and requirements for efficient, effective performance of operations. Performance requirements can be found in the following source documents:

- Federal and state regulatory requirements
- Appropriate codes and standards
- Contract requirements
- DOE Orders, manuals, and notices
- Implementation plans
- Implementation procedures
- Facility safety documents
- Policy and mission statements
- DOE-approved "work smart" standards
- Standards/requirements identification documents
- Plans and programs
- Management, business, operating, and/or strategic plans
- Applicable standards, permits, authorizations, and regulatory agreements

Much information about performance and additional performance requirements may be available to assessors in existing documents and reports, such as

- reports from outside regulators;
- facility operations/activity/metrics reports;
- performance reviews;
- previous assessment reports, including self-assessment reports;
- internal inspections, reviews, and reports;
- corrective action plans and status reports;
- concerns and occurrence reports;
- performance indicators;
- monitoring and survey data, and modeling data and analyses; and
- PAAA NTS reports.

Requirements contained in these documents are selected based on impact on the assessed organization's mission and the relationship to the scope of the assessment. From selected requirements, objective statements (performance measures) are developed for determining whether a program, system, or process is working efficiently and effectively. From these measures, the specific performance criteria are developed and tools selected for conducting the assessments. In developing performance criteria, assessment personnel should not reinterpret or redefine requirements specified in the source documents. It is critical for a successful assessment that the requirements are understood and clearly explained in the assessment documentation.

Assessment planning tools such as checklists are an essential element of an effective assessment. They vary in format, content, and level of detail, but all have one thing in common: they help focus the assessor on the mission and objectives of the program, system, or process being assessed. Application of planning tools before an assessment ensures that time will be used effectively and that the assessment's focus is identified and maintained. Assessment planning tools are often used to relate the performance criteria to the established assessment scope and may include lists of interview questions, major elements of programs, or detailed process work steps. Similar to a road map, each tool is used to remind the assessor of where he/she is going and the items likely to be encountered along the way. Planning tools are extremely useful when the assessment basis is complex or the requirements come from multiple sources. Typical planning tools include matrices, flowcharts, cause-effect diagrams, tree diagrams, checklists, and information systems.

Investigation

The following is derived from DOE G 414.1-1A (Archived).

Effective assessments use a combination of tools and techniques to maximize the productivity of the assessment team and resources. Such assessment techniques include document reviews, interviews, observation, inspection, and performance testing. Investigations (using these techniques) should be sufficiently thorough and information gathered with sufficient diligence that accurate, detailed conclusions and issues can be provided to assist the organizations that will receive the final report.

Fact-Finding

The following is derived from DOE G 414.1-1A (Archived).

Techniques that may be used in fact-finding are discussed below.

INTERVIEWS

Interviews provide the means of verifying the results of observation, document review, inspection, and performance testing; allow the responsible person to explain and clarify those results; help to eliminate misunderstandings about program implementation; and provide a venue where apparent conflicts or recent changes can be discussed and organization and program expectations can be described.

DOCUMENT REVIEWS

Document reviews provide the objective evidence to substantiate compliance with applicable requirements. A drawback is that the accuracy of the records cannot be ascertained by review alone. This technique should be combined with interviews, observation, inspection, and/or performance testing to complete the performance picture. Records and documents should be selected carefully to ensure they adequately characterize the program, system, or process being assessed.

OBSERVATION

Observation, the viewing of actual work activities is often considered the most effective technique for determining whether performance is in accordance with requirements. Assessors should understand the effect their presence has on the person being observed and convey an attitude that is helpful, constructive, positive, and unbiased. The primary goal during observation is to obtain the most complete picture possible of the performance, which should then be put into perspective relative to the overall program, system, or process.

INSPECTIONS

Inspections are performed in accordance with acceptance criteria to verify the condition of physical facilities, systems, equipment, and components.

PERFORMANCE TESTING

Performance testing is used to observe the response of personnel or equipment by creating a specific situation and noting the resulting performance. This technique is especially helpful when activities of interest would not normally occur during an assessment visit. It is also useful when the timeliness and appropriateness of the response are critical.

Exit Interview

The following is taken from DOE G 414.1-1C.

This meeting is used primarily by the assessment team to present the assessment summary. Reasonable time should be allowed to discuss any concerns, but this meeting should not be used to argue the assessment findings or methodology. There should be no surprises during the exit meeting, since the assessment team should have taken every effort possible during the conduct of the assessment to ensure the assessed organization was aware of the team's findings and concerns. Prior to the exit meeting, the assessment team should consider combining related weaknesses or performance issues into a smaller number of well-supported findings to help focus management's follow-up actions. A written summary of the assessment conclusions and results should be provided at the exit meeting.

Reporting

The following is derived from DOE G 414.1-1C.

Assessment reports are required for documentation of assessment results. Assessment team leaders have the overall responsibility for preparing the report and obtaining appropriate approval for its release as applicable. The report may be formal or informal depending on the level of assessment performed, but should provide a clear picture of the results in terms of the programs, systems, and processes assessed. The assessment report should be clear, concise, accurate, and easy to understand, and should include only facts that directly relate to assessment observations and results. It should include sufficient information to enable the assessed organization to develop and implement appropriate improvement plans.

Note: A management assessment report may not require all of the content listed below and may only require an executive summary.

Specific report formats may vary considerably from one organization to the next. An independent assessment report usually includes the following sections:

- Executive summary
- Assessment scope
- Identification of team members
- Identification of personnel contacted
- Documents reviewed
- Work performance observed
- Assessment process and criteria
- Results of the assessment including identification of areas for improvement, and/or strengths

Follow-up
The following is taken from DOE G 414.1-1C.

A follow-up assessment with special focus may be performed and should be completed in accordance with applicable corrective action documents. Particularly, this follow-up assessment should evaluate the effectiveness of corrective actions. A reasonable subset of corrective actions should be reviewed for effectiveness. To increase the validity of the effectiveness review, a sufficient amount of time for implementation of the corrective action should be allowed before performing the review (e.g., six months).

The results of past assessments, which identified areas of good/noteworthy performance, should be used to reduce the frequency and depth of future assessments. Areas of poor performance should receive increased attention, especially if there are indications that management has been unable to correct identified problems. This is because recurring and cumulative deficiencies, even in a low-hazard operation, may indicate systemic problems and may decrease the likelihood of the organization achieving its mission.

Closure
The following is taken from DOE G 414.1-5 (Archived).

An integral part of a successful corrective action program is the capability to maintain a systematic approach for tracking and reporting the status of the corrective actions to successful closure and implementation. This may be accomplished manually or electronically.

Maintaining and updating this information provides consistent data for tracking and analyzing program status and trends. The process used to track and report corrective action progress should be readily accessible and provide sufficient data to appraise, analyze, and report the status of corrective actions affecting the safety, mission performance, and security of the site/organization.

c. Describe the following assessment methods and the advantages or limitations of each method:
- **Document review**
- **Observation**
- **Interview**

The following is taken from DOE G 414.1-1C.

Document Review
Document reviews provide the objective evidence to substantiate compliance with applicable requirements. A drawback is that the accuracy of the records cannot be ascertained by review alone. This technique should be combined with interviews, observation, inspection, and/or performance testing to complete the performance picture. Records and documents should be selected carefully to ensure they adequately characterize the program, system, or process being assessed.

Observation
Observation, the viewing of actual work activities, is often considered the most effective technique for determining whether performance is in accordance with requirements. Assessors should understand the effect their presence has on the person being observed and convey an attitude that is helpful, constructive, positive, and unbiased. The primary goal during observation is to obtain the most complete picture possible of the performance, which should then be put into perspective relative to the overall program, system, or process.

Interview
Interviews provide the means of verifying the results of observation, document review, inspection, and performance testing; allow the responsible person to explain and clarify those results; help to eliminate misunderstandings about program implementation; and provide a venue where apparent conflicts or recent changes can be discussed and organization and program expectations can be described.

d. Describe the methods by which noncompliance is determined and communicated to the contractor and Department management.

The following is taken from 48 CFR 970.5223.1.

The contractor shall promptly evaluate and resolve any noncompliance with applicable ES&H requirements and the system. If the contractor fails to provide resolution or if, at any time, the contractor's acts or failure to act causes substantial harm or an imminent danger to the environment or health and safety of employees or the public, the CO may issue an order stopping work in whole or in part. Any stop work order issued by a CO shall be without prejudice to any other legal or contractual rights of the government. In the event that the CO issues a stop work order, an order authorizing the resumption of the work may be issued at the discretion of the CO. The contractor shall not be entitled to an extension of time or additional fee or damages by reason of, or in connection with, any work stoppage ordered in accordance with this clause.

e. Describe the action to be taken if the contractor challenges the assessment findings, and explain how such challenges can be avoided.

The following is taken from NNSA/Service Center Environment, Safety, and Health Department, *Assessment Handbook.*

All reports of assessment results must be reviewed for factual accuracy. The contractor being assessed must concur that the reported results are factual, or detailed objective evidence must be assembled to provide indisputable evidence that reported information is factual. Factual accuracy review can be performed in parallel with assessment performance, or a period of time following completion of the data-gathering portion of the assessment may be allotted to the contractor to provide formal comments. Comments should be restricted to those of a technical nature. The contractor being assessed must provide documented objective evidence to address discrepancies documented within the report. Inclusion of a disputed nondiscrepancy within the report is solely at the discretion of the team leader. It is very important that the review plan identify the process to be used for factual accuracy verification, and that the process is clearly understood by the contractor and site office.

Failure to adequately verify factual accuracy of information contained within the final report often results in corrective action weaknesses, and greatly reduces the effectiveness of any type of assessment.

Because the goal of any assessment is to identify and implement actions to improve the assessed activity, the contractor being assessed must fully understand any identified problems, and must agree that the problem identification was based upon valid information. It is not necessary that the assessed contractor agree that the problem identified is valid, since requiring this agreement can result in contentious relations between the contractor and the team.

f. Discuss NNSA Business Operating Procedure 10.003, *Site Integrated Assessment Plan*.

NNSA Business Operating Procedure 10.003 establishes a standard process for the annual development, updating, and reporting of site integrated assessment plans across the NNSA Nuclear Security Enterprise as described in Chapter 9 of NNSA Policy Letter *Transformational Governance and Oversight* (NAP-21). It also supports DOE O 226.1B, *Implementation of Department of Energy Oversight* Policy, which requires written plans and schedules for planned assessments.

g. Discuss NNSA NAP-21, *Transformational Governance and Oversight*.

The purpose of NNSA NAP-21 is to identify the principles, responsibilities, processes, and requirements that the NNSA will utilize to transform and improve Federal governance and oversight of management and operating contractors. The information in this document is to be used as the foundation for governance transformation and the basis for how NNSA conducts business.

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