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DOE STANDARD

CRITERIA FOR PACKAGING AND STORING URANIUM-233-BEARING MATERIALS

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ABSTRACT

This Standard provides guidance for the packaging and long-term (50 years) storage of stabilized, separated uranium-233(^{233}U)-bearing metals and oxides. Metals are stabilized by removing liquids and pyrophoric materials, and oxides are stabilized by heating in air at an elevated temperature. Design, construction, and testing of the storage container, loading limits of metals and oxides for the storage container(s) and other safety-related requirements for the packaging process are specified. Package surveillance and record-keeping requirements during storage are outlined.

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FOREWORD

1. This Department of Energy (DOE) Standard is approved for use by all DOE organizations and their contractors holding inventories of special nuclear material (SNM) in quantities of at least low strategic significance (>15 g [0.033 lb_m] U-233) [10 CFR 70.4]. This Standard deals with stabilization, packaging, and storage of separated ^{233}U-bearing materials.
2. Comments (recommendations, additions, deletions) and pertinent data that may improve this document should be sent to the Technical Standards Project Office by letter or by using the self-addressed Document Improvement Proposal (DOE F 1300.3).
3. DOE Technical Standards do not establish requirements. However, all or part of the provisions in a DOE standard can become requirements under the following circumstances:
   (1) the provisions are explicitly stated to be requirements in a DOE requirements document; or
   (2) the organization makes a commitment to meet a standard in a contract or in an implementation plan or program plan required by a DOE requirements document.
4. Throughout this Standard, the word “shall” is used to denote actions that must be performed if the objectives of this Standard are to be met. If the provisions of this Standard become requirements through one of the ways discussed above, then the “shall” statements would become requirements.

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1. PURPOSE AND SCOPE

This Standard establishes the criteria for the packaging and safe storage of separated ^{233}U-bearing materials. Materials conforming to these criteria should be capable of being contained and stored safely for a nominal 50 years (pending disposition). It aims to obviate subsequent repackaging during their continuing storage until their respective dispositions are identified. Periodic inspections of ^{233}U packages shall be conducted in order to confirm the storage objectives covered by this Standard. This Standard does **not** apply to ^{233}U-bearing liquids, wastes, spent fuels, irradiated targets, in-process materials, or small quantities (= 15 g) involved in research and development studies. Furthermore, this Standard only applies to uranium-bearing materials in which ^{233}U (and associated ^{232}U) has the greatest radiological significance. Major particulars for the safe storage of separated ^{233}U are preventing nuclear criticality, containing the radioactive materials, protecting personnel from undue exposure to penetrating radiation, and safeguarding this SNM. The storage facility plays a primary role in addressing all of these safety elements except containment. The facility plays a principal backup role (i.e., defense in depth) in confining radioactive contaminants during upset conditions. Material stabilization, consolidation, access limitation, low maintenance storage, and reliability in verification of the inventory are the Department's present goals for the ^{233}U-bearing materials.

This Standard delineates the criteria for assuring that ^{233}U-bearing materials will be packaged in configurations (combinations of material form and packaging) promoting safe, long-term storage. The packages should require regular surveillance for safety under anticipated handling and storage conditions until their final disposition. Material control and accountability (MC&A) surveillances may occur simultaneously.

2. EQUIVALENCY AND EXCEPTIONS

This Standard provides criteria for ^{233}U-bearing materials that are to be newly packaged and materials that require repackaging for long-term safe storage. Currently stored material should be compared to the criteria in this Standard and repackaged as necessary. This standard allows using systems, methods, material forms, or devices that are functionally equivalent or superior in the place of those prescribed herein if demonstrated by technical documentation. Waivers and exemptions to this Standard shall be obtained in accordance with DOE TSPP-9.

3. APPLICABILITY

This Standard applies to all DOE organizations and their contractors as defined in their contracts.

4. REFERENCES

The following references are called out in this Standard.

10 CFR 20, “Standards for Protection Against Radiation”
10 CFR 70.4, “Domestic Licensing of SNM. Definitions”
10 CFR 830.120, “Nuclear Safety Management, Quality Assurance Requirements”
10 CFR 835, Appendix D, “Occupational Radiation Protection, Surface Radioactivity Values”
49 CFR 178, “Specifications for Packagings”
ANSI N14.5, “Standard for Radioactive Materials – Leakage Tests on Packages for Shipment,” American National Standards Institute, Inc., New York, NY
DOE O 420.1, “Facility Safety,” October 13, 1995
DOE O 440.1A, “Worker Protection Management for DOE Federal and Contractor Employees,: March 27, 1998
DOE O 470.1, “Safeguards and Security Program,” June 21, 1996
DOE O 471.2A, “Information Security Program,” March 27, 1997
DOE O 472.1B, “Personnel Security Activities,” March 24, 1997
DOE O 474.1, “Control and Accountability of Nuclear Materials,” August 11, 1999
DOE 5480.21, “Unreviewed Safety Questions,” December 24, 1991
DOE 5480.22, “Technical Safety Requirements,” Change 2, January 23, 1996
DOE 5480.23, “Nuclear Safety Analysis Reports,” Change 1, March 3, 1994
DOE 5632.7A, “Protective Force Program,“ Change 1, February 13, 1995
DOE-TSPP-9, “Maintenance of DOE Technical Standards,” June 1, 1999

5. ACRONYMS AND DEFINITIONS

5.1 Acronyms

ALARA	As Low As Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
AWE	Atomic Weapons Establishment, United Kingdom
CEUSP	Consolidated Edison Uranium Solidification Project
CFR	Code of Federal Regulations
DBA	Design Basis Accident
DOE	United States Department of Energy
HEU	Highly Enriched Uranium
INEEL	Idaho National Engineering and Environmental Laboratory
LOI	Loss-on-Ignition
LWBR	Light-Water Breeder Reactor
MBA	Material Balance Area
MC&A	Materials Control and Accountability
NDA	Non-destructive Assay
ORNL	Oak Ridge National Laboratory
PVC	Polyvinyl Chloride
SNM	Special Nuclear Material
SRS	Savannah River Site
TID	Tamper Indicating Device

5.2 Definitions

Conversion to Oxide	Chemical reaction of a ^{233}U-bearing material (e.g., fluoride, metal, or active oxide) to produce a stable uranium oxide.
Design Pressure	A characteristic of a sealed container which indicates its ability to withstand internal pressurization. In the language of the ASME Boiler and Pressure Vessel Code, it is the “Maximum Allowable Working Pressure.”
Drying	A heating process to remove adsorbed water or decompose hydrated compounds.
Engineered Materials/ Fabricated Fuel	Nuclear reactor fuel components (high-fired ceramic pellets, pins, plates, assemblies, etc.) consisting of ^{233}U-bearing material manufactured and maintained with a very high quality and quality assurance.
Free Gas Volume	That portion of the sealed package that is available to the fill gas and any gases generated during storage. See Appendix B for further discussion.
Irradiated Fuel	Nuclear material, including Fabricated Fuel, that in its existing form, has been subjected to irradiation in a nuclear reactor or accelerator and that consequently delivers an external radiation dose, and requires special containment and handling.
Loss-on-Ignition (LOI)	Mass loss measured after a weighed stabilized sample is heated in air for at least one hour to a material temperature in excess of the specified material stabilization temperature and high enough to drive off residual volatiles. This test measures any further weight change due to residual volatile species and verifies that the material has been stabilized.
Material Temperature	The lowest temperature within a mass of heated material. In other words, all of the material is at or above this temperature.

Oxide	Stable uranium-oxygen (U-O) compounds in the UO_2-UO_3 composition range (generally UO_2, U_3O_8, and UO_3). Uranium oxides are often a mixture of U-O compounds.
Package	The assembled combination of sealed containers required by this Standard (an inner and an outer), together with the contained stable ^{233}U oxides or metals and any additional interior convenience container(s) into which they initially have been placed.
Repackaging	A process whereby an existing storage container is opened and stabilized material transferred to a Package that meets the requirements of this Standard.
Specific Surface Area	The surface area per unit mass of solid.
Stabilization	A process to eliminate the tendency of a ^{233}U species to react with atmospheric components or reactive associated constituents. This generally means conversion to oxide and elimination of associated material that could undergo radiolysis during storage thus creating an overpressurization in the sealed package.
Waste	For the purposes of this Standard, waste has all of the following: 1. no existing or planned use, 2. no potential for weapons use (<12% ^{233}U in ^{238}U, or equivalent), and 3. no unacceptable criticality safety risk (e.g., <0.67% or equivalent).

6. PACKAGING AND STORAGE CRITERIA

6.1 Materials

Storable ^{233}U-bearing solid forms are metals, alloys, oxide powders, oxide monoliths, and engineered materials, such as ceramic oxide pellets and unirradiated fuels. There are significant differences in the chemical and physical properties of metals, various oxides, and engineered materials currently inventoried at Oak Ridge National Laboratory (ORNL) or Idaho National Engineering and Environmental Laboratory (INEEL). Criteria are provided for each material category.

6.1.1 Uranium-233-Bearing Metals and Alloys

1. Metal and alloy pieces to be packaged shall have a specific surface area of less than 0.005 m^2/g (24.4 ft^2/lb_m). Particles and metal pieces larger than 8 mesh meet this Criterion. Metal pieces with a specific surface area greater than 0.005 m^2/g (24.4 ft^2/lb_m), thin foils, turnings, and wires shall be converted to stable oxides for storage.
2. At the time of packaging, metals shall be visually free of non-adherent oxides, liquids, and organic materials such as plastics and oils.

6.1.2 Oxides

1. Stabilization Acceptance Criterion: The volatiles content (e.g., weight loss using the LOI method) of oxide to be packaged in any sealed container shall be less than 0.5 wt % at the time of packaging or repackaging.
2. Stabilization Conditions: Oxides that do not meet the Criterion in 6.1.2.1 shall be stabilized by heating the material in an oxidizing atmosphere to a material temperature of greater than 750°C (1382°F) for a time sufficient to meet the Criterion in 6.1.2.1, but not less than 1 hour.
3. Stabilization Verification: Verification that materials to be packaged have been stabilized to meet Criterion 6.1.2.1 shall be accomplished by: measuring the volatiles content of the material using a demonstrated technically appropriate method, such as LOI at 800°C (1472°F) for at least one hour.

6.1.3 Engineered Materials

1. Ceramic material, consisting of high-fired ^{233}U-thorium oxide pellets clad with Zircaloy or packaged in stainless steel and having a historical record of quality assurance meeting the intent of 10 CFR 830.120 or the applicable quality program at the time of manufacture, meets all the requirements of Section 6.1.2 without additional stabilization or testing. Pellets extracted from such rods are also considered to meet all the requirements of Section 6.1.2 at the time they are declad. Clad metal fuel with a similar assurance of cladding integrity meets all the requirements of Section 6.1.1.
2. Unclad, high-fired ceramic fuel pellets that satisfy Criterion 6.1.2.1 are considered to meet the requirements of Section 6.1.2 without additional treatment. Such mixed oxide pellets that cannot meet the requirements of Criterion 6.1.2.1 shall be stabilized according to Criterion 6.1.2.2, and shall meet Criterion 6.1.2.1 of this Standard prior to packaging.

6.1.4 Storage after Stabilization – Deferred Packaging

1. Oxides that previously have been stabilized as specified in Criterion 6.1.2.2 met the criteria specified in 6.1.2.1and 6.1.2.3 at the time of stabilization. These would be placed in a closed container (such as a convenience can) and be packaged into the inner and outer sealed containers described in this Standard without additional stabilization, provided the container and contents appear unchanged and the volatile content can be shown to be less than 0.5 wt%, for example by a re-weighing. The residual volatile content may be determined, for example, by measurement at the time of packaging into the inner container or by adding any weight gained during the time between stabilization and packaging into the inner container to the determined volatile content at the time of stabilization.

6.2 Containers – the “Packaging”

6.2.1 Container Design Concept

1. The container assembly shall consist of stabilized material in a minimum of two individually sealed, nested containers. One container provides the pressure boundary to prevent release of the contents. The other container provides an additional isolation boundary. The outer and inner containers shall be sealed by welding or by other techniques that meet or exceed the performance characteristics

and criteria of this Standard. The use of additional sealed or unsealed inner containers, sometimes referred to as material or convenience containers, is optional.

2. Interior containers shall be sized to fit in the next outer container with adequate clearance for welding the next outer container.
3. Both the inner and outer containers shall allow for non-destructive contents verification, inspection, and surveillance (such as by radiography and weighing).
4. The minimum Design Pressure of the inner and outer containers both shall be 2070 kPa (300 psia)[†].
5. The outer container shall be designed to the requirements outlined in DOE Order 440.1(current release at time of packaging), Attachment 1, Section 6.

6.2.2 Container Construction

1. Both the inner and outer containers shall be fabricated of low-carbon 300 (L-grade) series stainless steel or other materials of comparable or better performance (strength, corrosion resistance, etc). Closure welding shall be performed using accepted procedures that minimize sensitization of the stainless steel to stress corrosion cracking.
2. Neither the inner nor the outer container shall include combustible or organic material in its construction.
3. The loaded and assembled outer container shall be compatible with the dimensions required by the storage facility. For example, for Oak Ridge Building 3019:
   a) Diameter no more than 112 mm (4.4 in).
   b) Height no less than 102 mm (4.0 in).
4. Organic material (including elastomeric gaskets, organic coatings, or plastic bags) shall not be allowed in any of the containers, including the convenience container.

[†] Pressures expressed in kilopascals (kPa) are given in “absolute” terms. The conversion from pounds per square inch (psi) is that 1 psi is equal to approximately 6.9 kPa. Thus, 100 psig (psi gauge), or 115 psia (psi absolute) is equivalent to approximately 790 kPa.

6.2.3 Requirements for Container Testing

1. Design Qualification Testing

    a) The outer container shall remain leak-tight as defined by ANSI N14.5 after a free drop of the Package (outer container, inner container, and simulated contents) from a height consistent with the maximum credible accident for the storage facility (for example, from a height of 35 feet for Oak Ridge Building 3019) onto a flat, essentially unyielding, horizontal surface. The drop test shall follow the test procedures specified in applicable portions of 49 CFR 178.603 and 49 CFR 178.601.

    b) The inner container shall remain leak-tight as defined by ANSI N14.5 after a free drop of the container (including simulated contents) from a four foot height onto a flat, essentially unyielding, horizontal surface. The drop test also shall follow the test procedures specified in applicable portions of 49 CFR 178.603 and 49 CFR 178.601.

    c) The outer container shall remain leak-tight, as defined in ANSI N14.5, after a hydrostatic proof-test to 1.5 times the Design Pressure. The test shall be conducted as specified by 49 CFR 178.601.

2. Testing At Closure

    Both the inner and outer containers shall be tested for leak-tightness, as defined in ANSI N14.5, at their times of closure.

6.2.4 Other Requirements

1. Both the inner and outer containers shall have unique permanent identification markings, such as by etching or engraving.
2. The outer container and the exterior surface of the inner container shall conform to acceptable removable contamination requirements as defined by 10 CFR 835, Appendix D for transuranics at the time of assembly and closure of the outer container.

6.3 Contained Materials

6.3.1 Container Fill Gas

1. The atmosphere within any of the containers (including the convenience container, if used) shall not react adversely with the sealed containers or contained materials. Examples of atmospheres that may be used include argon, neon, nitrogen, and air.
2. The atmospheres within the inner and outer containers shall not preclude leak-testing of the containers.

6.3.2 Mass of Contained Materials

1. The total mass of ^{233}U and other associated fissile species shall not exceed (a) 5.4 kg (11.9 lb$_m$) for metal or 9.1 kg (20 lb$_m$) for oxide per container, or (b) the limits specified in site-specific nuclear criticality safety programs, policies, and procedures for storage.
2. The mass of contained materials shall be limited further when needed to ensure that the bounding pressure of the inner and outer cans (calculated using the Aggregate Pressure Equation [Equation B.9] in Appendix B) is less than the respective container Design Pressure.
3. In cases where multiple limits exist, the most restrictive limit shall be applied.

6.3.3 Packaging Process

1. Readily removable foreign materials, such as metal fasteners and other debris, shall be removed from the material to be stored prior to packaging.
2. The oxide sample taken for verification (Criterion 6.1.2.3) shall be as representative as possible of the material placed in the sealed container at the time of packaging. If the material is to be stored in a convenience can for some time before packaging, and the provisions of Criterion 6.1.4.1 apply, the sample shall be as representative as possible of the material to be placed into a closed container.
3. Contained materials shall be packaged in such a way as to prevent corrosion or adverse effects on the structural integrity of the inner or outer container.
4. Only chemically and isotopically similar materials should be combined in an inner container or convenience container and packaged for storage.

6.4 Storage – Surveillance of Stored Packages for Safety

6.4.1 Surveillance Program

1. Surveillance Programs shall be site-specific and should be coordinated with MC&A surveys.
2. The Site Surveillance Program shall be risk-based, and shall include:
    a) A clearly defined approach (which may include statistical measures, anticipated failure rates, consideration of risks inherent in the package contents and other risks, and engineering judgement) by which package selection, frequency, and sample size shall be established, and may be adjusted;
    b) An initial safety surveillance frequency (or time between package inspections);
    c) An initial size and composition of the sample of packages to be surveyed;
    d) Non-destructive techniques to provide surveillance including initial baseline Package inspections within 30 days of package closure; and
    e) Documented safety surveillance throughout the storage period for information on continuing Package performance. The level of scrutiny over time may be adjusted based on observed Package behavior.
3. The Site Surveillance Program shall document safety inspection/surveillance methods and responsibilities.
4. The Site Surveillance Program shall require procedures or other definitive documentation that:
    a) Identify prerequisites for the surveillance (i.e., those actions that must be taken or conditions that must be satisfied before an inspection);
    b) Identify acceptance criteria and provide specific instructions for action when any of those criteria are not met; and
    c) Establish and maintain a documented safety surveillance schedule.

6.4.2 Surveillance Parameters

Each sampled Package shall be inspected and evaluated versus previous inspections for:

1. An indication of internal pressure build-up;
2. Transferable contamination on the outer container;
3. Signs of changes in material form within the container (e.g., by radiography, by weight change of metals); and
4. Signs of leakage and/or degradation of the container.

6.4.3 Evaluation of Surveillance Data

1. Data from a surveillance shall be compared against prior measurements (including the baseline) to identify changes.
2. An evaluation shall be performed and corrective action taken as appropriate when an unexpected change in a package is noted. It shall also be documented. This evaluation shall include, as appropriate, 1) options for opening the Package, 2) consideration for inspecting other similar Packages, based on factors such as contents, origin, and date of closure, and 3) assessment of potential consequences.

6.5 Documentation

6.5.1 Database

A database shall be maintained as a source of relevant information about stored materials and packages. For completeness, MC&A documentation should be coordinated with the database.

6.5.2 Database content elements

1. The database shall include, as a minimum, available information on the following *material characteristics*:
    a) Chemical and physical forms;
    b) Best available isotopic distribution including all actinides, and the date(s) of analysis;

   c) Quantity (mass) of material contents;
   d) Conditions of material testing, including date, temperature, processing duration, equipment used, atmosphere, and test results;
   e) Source of stored material (e.g., site, facility and Material Balance Area (MBA) that generated the material, chemical and physical form code, and material type code), if available;
   f) Other information relative to the contents such as expected major impurities with source of impurities data (e.g., process knowledge, destructive analysis, or X-ray fluorescence analysis).

2. The database shall include, as a minimum, identification of the following *Package characteristics*:
   a) Nominal fill gas composition of each container on sealing (e.g., air, helium, nitrogen, or argon);
   b) Leak test data record for each container in a Package;
   c) Number and type of containers in a Package;
   d) Date of packaging for each container;
   e) Initial radiation field [gamma and neutron at contact and 0.305 m (1 ft.)], including how it was measured;
   f) Baseline Package gross weight, tare weight, and dimensions;
   g) The unique identification number and tamper indicating device (TID) number, if any, associated with each container; and
   h) The manufacturer lot identification number for each container.
3. The database shall include, as a minimum, the following records from surveillances and inspections:
   a) Surveillance results;
   b) Records of tests performed;
   c) Dates of inspections; and
   d) Names of trained individuals performing inspections.

4. The database shall include, as a minimum, specific locations of stored packages in the storage facility.

6.6 Quality Assurance

Quality assurance to meet 10 CFR 830.120 shall be performed in accordance with site Quality Assurance Plans.

7. STORAGE FACILITY FEATURES

A facility used for the storage of ^{233}U should address the unique characteristics of the material and include nuclear criticality safety, confinement of radioactive materials, radiation shielding, and safeguarding SNM.

7.1 Nuclear Criticality Safety

Storage and handling of ^{233}U-bearing materials shall conform to the criticality safety requirements of DOE O 420.1. Criticality safety evaluations shall document that storage and handling activities shall remain subcritical during all normal and credible abnormal events. Criticality safety evaluations shall be performed for operations (under normal conditions) within any facility containing ^{233}U in excess of the limits specified in DOE O 420.1 or as specified in site-specific nuclear criticality safety program policies and procedures.

Special care should be exercised in validating calculation methods supporting criticality safety evaluations because of the paucity of data in the intermediate energy regime that may be important for some ^{233}U-bearing matrices under specified operational conditions.

7.2 Confinement of Contamination

The material physical form, material containers, or containment vessels serve as the principal barrier for confinement of contamination. Depending on the material storage system, the facility itself may serve as another confinement barrier.

The integrity of the storage facility shall be maintained through normal operations, anticipated operational occurrences, and all facility design basis accidents (DBAs). The adequacy of these confinement systems to effectively perform their required functions shall be demonstrated by the safety analysis. Requirements governing the safety analysis process include the applicable portions of DOE Orders 420.1, 5480.21, 5480.22, and 5480.23. The need for ventilation systems for confinement purposes shall be based on the results of the safety analysis. The combination of the material storage system and the storage facility represents a defense-in-depth safety confinement system.

7.3 Radiation Shielding

Owing to the presence of ^{232}U in ^{233}U inventories, radiation shielding may be required to attenuate the 2.6 MeV (4.17E-6 erg) photon emitted by the ^{232}U daughter, ^{208}Tl. Depending on the material form and material storage system used, the facility itself may serve as a radiation shield. The regulations pertaining to occupational radiation protection as specified in 10 CFR 835, shall be met.

7.4 SNM Safeguards

Uranium-233 is a weapons-usable material due to its fissile properties and its ability to be produced in sufficient quantities for manufacturing nuclear weapons. This material shall be protected from unauthorized access and unauthorized use. Safeguards measures shall meet the requirements of DOE O 470.1, DOE O 471.2A, DOE O 472.1B, DOE O 474.1, and DOE 5632.7A.

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Appendix A. Technical Bases for ^{233}U Packaging and Storage Criteria

This Appendix summarizes the technical bases for the criteria in this Standard. Section numbers in this Appendix correspond to the section numbers in the body of the Standard and provides guidance where applicable. The reader is directed to the primary technical source information and the ^{233}U Technical Handbook [Storch 1999] for further details.

Development of this Standard was modeled on the Standard for plutonium [DOE 1999]. Although plutonium and uranium have significant physical/chemical differences, many of the storage issues are similar. Consequently, this Standard makes use of plutonium studies for issues that are material independent or for issues where the plutonium behavior bounds the ^{233}U behavior (e.g., alpha radiolysis).

A.1 PURPOSE AND SCOPE

This Standard establishes criteria for packaging, and long-term safe storage of separated ^{233}U-bearing metals and oxides. Storage packages that meet these criteria should maintain their integrity (i.e., should not require further repackaging) for a minimum of 50 years. The bulk of these materials are stored at ORNL and INEEL as solid forms including metals, oxide powders, ceramic oxide pellets, and oxide monoliths. This Standard does not apply to ^{233}U-bearing liquids, residues, wastes, spent fuels, irradiated targets, in-process materials or small quantities involved in research and development studies. Those materials are either addressed by other storage programs or are not germane to the intended safe storage activity.

With the exception of oxide from the Consolidated Edison Uranium Solidification Project (CEUSP) , the material covered by this Standard is nearly isotopically pure ^{233}U with trace amounts of ^{232}U. Isotopes of uranium that may be present (with their half-lives in parentheses), include ^{238}U (4.5 x 10^9 y), ^{236}U 2.4 x 10^7 y), ^{235}U (7.0 x 10^8 y), ^{234}U (2.4 x 10^5 y), ^{233}U (1.6 x 10^5 y), and ^{232}U (69 y). Uranium-233 and its associated isotope ^{232}U are man-made. They present potentially more severe radiological hazards than the naturally occurring uranium isotopes do. An isotopic level of 0.66 wt % ^{233}U (excluding ^{232}U) in weapons-grade highly enriched uranium (HEU) is the ^{233}U isotopic concentration at which the specific activity of ^{233}U exceeds that for uranium highly enriched in the ^{235}U isotope [Bereolos et al. 1998]. Any ^{232}U present will lower this bound even further.

Although disposition of ^{233}U is part of the planned activities for DOE's fissile materials disposition program, it is recognized that various factors could potentially delay disposition. Fifty years is selected as a reasonable upper time limit that material might have to be stored because of such delays.

A.2 EQUIVALENCY AND EXCEPTIONS

The basis for equivalency shall be a technical justification for any departure from specific provisions of the Standard. This technical justification will be subject to oversight by the authorizing official.

A.3 APPLICABILITY

No further basis provided.

A.4 REFERENCES

No further basis provided.

A.5 ACRONYMS AND DEFINITIONS

No further basis provided.

A.6 PACKAGING, AND STORAGE CRITERIA

A.6.1 Materials

A.6.1.1 Uranium-233 Metals

1. Potentially pyrophoric metals are not acceptable storage forms because this could lead to fires and dispersal of the uranium. Uranium metal in massive form presents little fire hazard, but it will burn if exposed to a severe, prolonged fire. By contrast, finely divided uranium metal powder is pyrophoric [Bretherick 1986], and can ignite spontaneously, even if confined in a container without liquid or without air movement. The presence of moisture in the gas phase over exposed chips increases this possibility [Totemeier 1995]. The flammability of uranium metal depends almost entirely on its specific surface area. Finely divided uranium metal ignites spontaneously upon exposure to air and burns rapidly to form an oxide. For uranium foils and wires, the experimentally determined ignition temperatures are somewhat higher than for powders having the same specific surface area. The specific surface

area should not exceed 0.005 m^2/g (24.4 ft^2/lb_m), based on a study of ignition temperature versus surface area of uranium powder [Baker 1960]. The lowest ignition temperature measured in the study was 255ºC (491ºF) for powder having a specific surface area of 0.00512 m^2/g (25 ft^2/lb_m). For foils and wires, experimental ignition temperatures were somewhat higher than for powders having the same surface area [Leibowitz and Bingle 1959]. Thus, the most readily ignitable form is powder making the powder limits the most appropriate baseline. The 0.005 m^2/g (24.4 ft^2/lb_m) limit was chosen since this was the smallest surface area found to ignite and the corresponding ignition temperature of about 255ºC (491ºF) is far above temperatures expected to be achieved during storage [Thein 1999].

Uranium metal pieces larger than sieve mesh size 8 (0.00238 m or 0.00781 ft) are assured of having a specific surface of less than 0.005 m^2/g (24.4 ft^2/lb_m) and may be stored in tube vaults. Thus, uranium metal of less than sieve mesh size 8 (0.00238 m or 0.00781 ft), powders, thin foils, and turnings of uranium are potentially pyrophoric and must be converted to stabilized oxide prior to storage or must be stored in a sealed container with an inert atmosphere.

2. Loose removable oxide associated with uranium metal also may be pyrophoric. An adherent oxide layer on stored metal is generally beneficial because it tends to retard further oxidation. However, as UO_2 (the first oxide produced), this coating may be pyrophoric. Therefore, prior to repackaging ^{233}U metal, readily removable loose oxide must be removed from outer metal surfaces. The pyrophoricity hazard from easily removable oxide is mitigated by light brushing with a soft bristle or wiping, but not a wire brushing, prior to packaging the metal. The loose oxide generated by brushing should be treated and stored according to this Standard. Oxide removal should not be so aggressive that the adherent oxide layer on the metal surface is removed. Since ^{233}U metal allowed by this Standard has low specific surface area (see Criterion 6.1.1.1 for details) and is therefore easily examined, visual inspection for free water and organic materials with the unaided eye during packaging is sufficient to assure that unsafe quantities are not present in the storage environment.

A.6.1.2 Oxides

The uranium-oxygen system is one of the most complex systems known. These oxides generally are reported as one of three formulae: UO_2, U_3O_8, or UO_3. The system is generally a mixture of oxides with a O/U ratio between 2 and 3. Uranium oxides in this range are considered stable and suitable for storage.

1. The criterion of 0.5 wt % volatiles provides a reasonable balance between the difficulty of achieving lower volatiles contents (primarily moisture) and the cost (of both the container and any ancillary impact on storage facility size) of providing a container that will withstand the overpressure theoretically postulated by a higher volatiles content. Correlation of the volatiles criterion with bounding pressures in storage containers is established in Appendix B (Derivation of Pressure Equation) and other sections of this Appendix.

2. Most of the stored ^{233}U oxide powders were assayed for volatiles content at varying heating temperatures at the times they were formed, then stored in closed containers. Testing each batch for volatiles content according to Criterion 6.1.2.3 will verify if the historic measurements are still reliable. Existing material failing to meet the 0.5 wt% requirement must then be treated as specified in Section 6.1.2.2 and be subject to subsequent verification.

3. The stabilization requirements of this Standard accomplish the following primary objectives:

   - reduce the residual moisture (volatiles) content to less than 0.5 wt% and similarly reduce equivalent quantities of species such as hydrates that might produce pressurizing gases during long-term storage via radiolysis;
   - minimize potential for water readsorption above the 0.5 wt% threshold; and
   - stabilize reactive volatile ^{233}U species.

   There are also secondary objectives that are desirable, but not absolutely necessary. If the material is treated to meet a primary objective, then the following secondary objectives should also be considered:

   - convert to U_3O_8
   - enhance particle size

   If the volatiles content cannot be shown to be less than 0.5 wt %, this Standard specifies that oxide material will be placed in a continuously oxidizing atmosphere at a material temperature of greater than 750°C (1382°F) for a minimum of one hour.

This ensures that the stabilization requirements of the Standard will accomplish the above objectives.

a) **Reduce the moisture/volatiles content, which might produce pressurizing gases during long-term storage via radiolysis, to less than 0.5 wt % of contained solids content.**

The three stable forms of uranium oxide are UO_2, UO_3, and U_3O_8. Heating the material above 750°C (1382 °F) is sufficient to eliminate free and hydrated waters.

Water also might be present in ^{233}U oxides as chemisorbed water. However, since chemisorbtion is a surface phenomenon it will have little impact on the 0.5 wt% limit for ^{233}U oxides. A simple model for H_2O suggests that 0.2 mg (4.4E-7 lb_m) H_2O forms a monolayer on a square meter surface [Haschke and Ricketts 1995]. Thus, to be able to chemisorb more than 0.5 wt% would require specific surface areas in excess of 25 m^2/g (122,000 ft^2/lb_m). To achieve such large surface areas would require special conditions not commonly found in ^{233}U oxide production environments. As an example, UO_2 produced at ORNL was made to a surface area specification of less than 6 m^2/g (29,300 ft^2/lb_m) [Parrott, Sr. et al. 1979]. Surface areas of ^{233}U material found in ORNL records are generally well below 10 m^2/g (48,800 ft^2/lb_m) with the largest area found to be 12.5 m^2/g (61,000 ft^2/lb_m).

b) **Reduce potential for moisture/volatiles readsorption above the 0.5 wt% threshold;**

Moisture does not tend to readsorb in significant quantities on stabilized U_3O_8. Calcining the material at 750°C (1382°F) will convert the oxides to U_3O_8 thus eliminating the concern of readsorption and will ensure that the material will have gotten to the stable U_3O_8 phase.

c) Stabilize volatile species

The principle intermediary in the purification of ^{233}U oxides from thorium fuel is uranyl nitrate $UO_2(NO_3)_2$ generally appearing as the dihydrate $UO_2(NO_3)_2 \cdot 2H_2O$ [Storch 1999]. This compound melts at 184°C (363 °F) and denitration begins above 300°C (572°F) [Ullman 1996]. Thus, stabilizing to 750°C (1382°F) will be sufficient to drive off any residual nitrates.

Process histories indicate no known significant quantities of organics in existing ^{233}U oxides. Any organics used in the separation of ^{233}U from fission products (for example, in the Purex or Thorex processes) would be decomposed and driven off below 750°C (1382°F). Examples of such materials include tributyl phosphate (boiling point = 289°C [552.2°F]), diethyl benzene (boiling point = 200°C [392°F]), and kerosene (boiling point = 175°C -325°C [347-617°F]).

d) Convert to U_3O_8

U_3O_8 is the most attractive storage form because it is the most stable uranium oxide (the others convert to U_3O_8 upon heating). Thus, if materials are to be treated, then converting to U_3O_8 is desirable.

e) Enhance particle size

Enhancing particle size is desirable because it reduces the inhalation hazard of powdered material. However, it was not included as a primary objective because the two sealed containers provide multiple barriers between the hazard and targets (workers, public, and the environment). Thus, enhancing particle size would be defense-in-depth rather than a fundamental requirement. Furthermore, the effect of heating on particle size is highly dependent on the method of preparation of the oxide [Clayton 1961]. As an example, much of the oxide currently in inventory was prepared at ORNL by heating pregranulated ammonium diuranate in a furnace for 6 hours at 800°C (1472°F) or greater (not including preheating to reach temperature or cool down). Samples of this material indicate that it has a particle size distribution with greater than 90% of the particles being less than 10 microns (3.28E-5 ft) in equivalent spherical diameter. In fact, greater than 90% at less than 10 microns (3.28E-5 ft) was a specification for the material prepared for the light-water breeder reactor (LWBR) program [Parrott, Sr. et al, 1979]. Given that the material has already seen elevated temperatures for extended time, it is unlikely that further heating will show extensive enhancement.

4. The Standard requires verification that materials have been adequately stabilized. Stabilizing above 750°C (1382°F) and appropriate handling prior to packaging ensure that the only significant mechanism for container pressurization is radiolysis of readsorbed water into pressurizing gases. Thus, verification of adequate stabilization requires measurement (e.g., LOI) to ensure that residual volatiles in the packaged material is below the threshold specified in Criterion 6.1.2.1. The LOI test is accomplished by heating the oxide sample to at least 800°C (1472°F) for at least one hour and determining the resulting weight loss. The temperature selected must be higher than the stabilization temperature. However, the temperature should not be so high as to decompose U_3O_8, which forms gaseous UO_3 from 900-1500ºC (1652-2732°F) depending on surface area [Kirk-Othmer 1997]. Therefore, 800ºC (1472°F) was chosen as the LOI temperature. The LOI test has great attractiveness because of its simplicity and low cost. However, LOI does not directly measure the parameter of greatest interest – hydrogenous material content. Indeed, because of the chemistry of uranium oxides, LOI may give false results when applied to the forms UO_3 and UO_2. At temperatures greater than 450ºC (842°F) in air, UO_3 will begin to convert to U_3O_8. Thus, a portion of the LOI result will be from oxygen loss and the LOI measurement will be greater than the volatiles loss. UO_2 will also convert to U_3O_8 at temperatures greater than 450ºC (842°F) in air. In the case of UO_2 the LOI measurement will be less than the volatiles loss and may even show a "gain on ignition." Even U_3O_8 heated above 500-700ºC (932-1292°F) may give off oxygen to form substoichiometric U_3O_{8-x} [Gmelin 1978]. This Standard therefore encourages use of LOI with mass spectroscopy analysis of the off gas. Note that if the material is stabilized as part of the repackaging process, it should be U_3O_8 (see Section A.6.1.2.3).

A.6.1.3 Engineered Materials

1. For purposes of this Standard, fabricated fuel made from metals or high-fired oxides are considered to be stabilized and contained if their cladding has retained its integrity. When there is assurance of cladding integrity, the stabilization requirements of this Standard are deemed satisfied.

2. High-fired oxide fuel pellets qualified for nuclear fuel are quite pure, have controlled stoichiometry, and have been formed at higher temperatures than specified in this Standard for stabilization. Consequently, unirradiated pellets need only meet Criterion 6.1.2.1 to be eligible for packaging. Pellet materials that do not meet the criterion should be treated according to the provisions of Criterion 6.1.2.2 of this Standard.

A.6.1.4 Storage after Stabilization – Deferred Packaging

1. The verified stabilization step provides certainty that Criterion 6.1.2.1 was satisfied at a point in time. A high degree of confidence that the material still meets Criterion 6.1.2.1 is provided by the observation that the container and contained material appear unchanged and the volatiles content remains acceptably low. Evidence of change may include, for example, corrosion or substantial pitting of the container, significant discoloration of the contents, or change in structural integrity of the container. To provide assurance of stabilization, verification of the volatiles content is required, by a measurement (e.g., LOI or re-weighing).

A.6.2 Containers – the “Packaging”

A.6.2.1 Container Design Concept

1. A design goal for the storage Package is that it be maintenance free without further reprocessing or repackaging.

   A sealed container design, rather than a container design with a gas filter, was selected for two reasons: 1) gas filters allow the entry of moist air which could interact with stored materials; and 2) if the container was not always oriented properly, stored powder could plug the filter and later “blow out” causing, at a minimum, a local spread of contamination.

A welded closure is preferred because it is believed to provide the best combination of features such as design qualification test performance, ease of assembly under production conditions, container (package) payload capacity, and achievement of a 50-year life. Other closure techniques, such as those involving metal gaskets or metal O-rings are also acceptable. However, these techniques must demonstrate a 50-year life and pass the design qualification tests described in Section 6.2.3.1.

The material container (convenience container) is a container that is used to transfer ^{233}U-bearing material. A material container is not required in packaging and is not considered an isolation barrier by this Standard. Use of a material container can reduce the potential for contamination during loading and closure of the inner container, facilitate packaging, and may provide an additional material barrier.

2. These requirements simply provide functionality in the design.
3. Storage of ^{233}U-bearing material must comply with existing safeguards and security, physical inventory, and audit and surveillance directives, which rely on nondestructive assays as a technique for validation. These requirements call for routinely assaying stored materials for process, accountability, and inventory controls. Safeguards and security, physical inventory, and audit and surveillance procedures should be done concurrently with packaging and storage.
4. The specified design pressure of 2070 kPa (300 psia) is sufficient to contain the pressure generated under conditions described in Appendix B, Section B.6. These conditions bound the planned repackaging operations at ORNL.
5. The outer can does not qualify as an American Society of Mechanical Engineers (ASME) pressure vessel under the Boiler and Pressure Vessel Safety Code because the diameter of the can is less than 15.24 cm (6 in.). Paragraph 6, Pressure Safety, of Attachment 1 to DOE Order 440.1, requires that the ASME code or an alternative design code equal or superior to the intent of the ASME code be used when national consensus codes are not applicable. In accordance with this Standard, the pressure containment vessel is designed to ASME requirements and the fabricator manufactures the vessel according to code, but does not stamp the vessel as complying with the code. This approach should be used in application of this Standard by designing and manufacturing the outer storage container to ASME

specifications (for example, ASME VIII) with exceptions documented to show safety equal to or superior to the intent of the ASME code.

It should be noted that designation as a pressure vessel can arise simply because of the need to contain the internal pressure generated by radioactive decay and by operation at a temperature higher than that at which it was filled and sealed. Beyond that, its function as the primary containment requires that it be able to contain the pressures that might conceivably be generated by all credible processes during storage.

Finally, it should be noted that the pressure estimates derived in Appendix B using the pressure equation are highly conservative bounding estimates. Current data indicate that it is unlikely that container pressures will exceed 207 kPa (30 psia) under normal storage conditions during a 50-year storage period [Icenhour et al. 2000].

A.6.2.2 Container Construction

1. Corrosion-resistant material should be used for the containers. Use of 304L and 316L stainless steels are recommended, with 316L being preferable to 304L because of its greater corrosion resistance. Both materials are justified based on extensive experience in this and similar types of service. Stainless steels 301, 302, and 303 are not recommended due to their relatively low concentrations of alloying additions. The use of any corrosion resistant alloy recognized as suitable pressure vessel material under the criteria of the ASME Boiler and Pressure Vessel Code or equivalent recognized safety standards is permissible. To reduce the chance of galvanic corrosion, contact between different metals should be minimized as much as practical.

2. Polyethylene and polyvinyl chloride (PVC) have been used as bagging materials. Both deteriorate under heat and radiation and generate undesirable gaseous products [Kazanjian et al. 1985]. Experiments show that low-density polyethylene can be used to temperatures as high as 300ºC (572ºF) without excessive degradation to form hydrogen. The maximum service temperature for PVC is ~85ºC (185ºF). In addition to H_2, thermal degradation of this plastic produces gaseous HCl, which corrosively reacts with other materials to form hydrogen. Radiolysis of the two plastics yields large amounts of the same gaseous products. The radiolysis rate is a

function of the surface area of the plastic in intimate contact with radioactive material [Friedlander et al. 1966]. Because of the limited escape depth of alpha particles from dense materials, a film of contamination of fine oxide particles deposited on the plastic is comparable to a massive piece of metal in promoting radiolytic degradation. If the plastic is outside of a sealed can, degradation of the plastic is reduced because the only source of uranium-containing particles is contamination on the exterior of the can.

Nevertheless, as good operating practice, plastic in any portion of the packaging is forbidden by this Standard. The “bagless loadout” system has been developed to obviate the need for bagging out of material and will be put into operation as part of repackaging operations at ORNL.

3. The diameter of the outer container is sized to fit into the existing storage tube vaults at the ^{233}U National Repository at ORNL Building 3019. A minimum height is specified to ensure that the package will not tumble when placed in a tube vault. Containers should not be so tall that they do not fit within the storage configuration or are unwieldy to handle. For example, cans for repackaging operations at ORNL are approximately 1 ft. (0.3048 m) tall.

   This design will minimize future handling and avoid unnecessary additional personnel exposure, operational risk, and waste generation, so these dimensions will need to be factored into any future design of shipping containers.

4. Elastomeric seals on food-pack cans have been used for storage of ^{233}U. Although such containers have been used successfully with little or no significant seal degradation, this Standard conservatively excludes them from use.

A.6.2.3 Requirements for Container Testing

1. Design Qualification Testing

   a) The purpose of the drop test of the entire package is to ensure that a storage package accidentally dropped from the maximum storage height would not release any material. The number of tests, the number of samples per test, and the drop orientation of the samples are specified in 49 CFR 178.603(a). The target for the drop tests is defined in 49 CFR 178.603(d). The distance of the drop is measured from the target to the lowest point on the sample container. The drop height specified in the criterion is to be used instead of the heights indicated in 49 CFR 178.603(e). The criterion for passing the test is that it retain its function, (i.e., that it remain leak tight as defined by ANSI N14.5 [ANSI 1997]).

b) The purpose of the four-foot drop test for the inner container is to ensure that a loaded inner container accidentally dropped from the maximum packaging height would not release any material. See A.6.2.3.1 for details of the tests.

c) The hydrostatic proof test provides verification that the container will remain leak tight under maximum design conditions, plus a safety margin.

2. Testing During Use

ANSI N14.5, Leakage Tests on Packages for Shipment, specifies that the acceptable maximum leak rate is 1×10^{-4} std. mm^3/sec (1×10^{-13} std. m^3/sec or 3.5×10^{-12} ft^3/sec) of dry air at one atmosphere [ANSI 1997]. Full penetration weld closures provide the highest integrity and longest life seals possible. Welds eliminate gaskets, which may degrade and leak. Mechanical seals using bolts or screwed connections are susceptible to wear, creep relaxation, seizure, or other mechanical failure.

A.6.2.4 Other Requirements

1. Identification markings are required on all storage containers to facilitate maintenance of an inventory database and management of stored materials.

2. External surfaces of the outer container shall be as free from removable contamination as practical. Exterior surface contamination may be evidence of potential leakage of radioactive materials. Limits for ^{233}U are not specified in 10 CFR 835. The limits for transuranics are used because they are the most restrictive and have the most similar nuclear characteristics to ^{233}U.

The sealed inner container is the primary barrier to release of radioactive materials. To ascertain that this goal has been accomplished, the outer surface of the inner container must be within the removable contamination limits for transuranics in Appendix D of 10 CFR 835 at the time that the loaded inner container is placed into the outer container.

A.6.3 Contained Materials

A.6.3.1 Container Fill Gas

1. The stored material condition should not change significantly because of reactions with the container atmosphere. If material stabilization has to be repeated, there would be additional handling and worker radiation exposure. For example, the container atmosphere for the repackaging system at ORNL will be inert.

2. The container atmosphere must not act to mask leak testing and must support leak testing. For example, the inert backfill in the ORNL repackaging system will be used to detect leaks.

A.6.3.2 Mass of Contained Materials

1. The mass limit for fissile materials is based on criticality safety limits for ^{233}U. The single parameter subcritical limits for ^{233}U are 6.0 kg (13.2 lb_m) for metal and 10.1 kg (22.3 lb_m) for oxide (the most limiting oxide is UO_2) [ANSI/ANS 1983]. The limits in this Standard are 90% of these single parameter limits to provide a margin of safety. It should be noted that these limits are the most restrictive limits for the prevalent fissile nuclides (i.e., ^{233}U, ^{235}U, ^{237}Np, and ^{241}Am) except for plutonium. If plutonium is present, further restrictions on the mass limit should be considered on a case-by-case basis. Criticality safety analyses also may impose more restrictive limits.

   It is also recommended that the mass of material in single containers be kept below the Safeguards Category II limits (< 400 g [0.882 lb_m] total U for metals and < 2 kg [4.4 lb_m] total U for oxides [DOE M 474.1-1]). This requirement will ensure that single cans of material will not require expensive Safeguards Category II security measures during the surveillance phase or during future transportation.

   Storage and handling of ^{233}U-bearing materials shall conform to the criticality safety requirements of DOE O 420.1. Criticality safety evaluations shall document that storage and handling activities shall remain subcritical during all normal and credible abnormal events. Criticality safety evaluations shall be performed for operations (under normal conditions) within any facility containing ^{233}U in excess of the limits specified in DOE O 420.1 or as specified in site-specific nuclear criticality safety program policies and procedures.

2. The assurance of safe storage embodied in this Standard rests on the defense-in-depth premise that there are two barriers (containers) that are each capable of withstanding credible pressure, corrosion, etc. In the case of pressurization, this is accomplished by ensuring that the design pressure of the container is greater than the theoretical ability of the contained materials to pressurize, as determined by the bounding pressure calculation derived in Appendix B.
3. Using the lowest of multiple limits is the most conservative approach.

A.6.3.3 Packaging Process

1. Foreign materials such as metal items and processing debris are not expected in ^{233}U oxide packages. Any such materials shall be removed from the oxide prior to packaging. Items may be removed manually or by screening the powder, and can be removed either before or after stabilization.
2. Because the oxide (including contained impurities) may pick up atmospheric water, it is important that the sample taken for volatiles analysis be representative of the material actually packaged into the storage container. This can be done, for example, by controlling the glovebox atmosphere and/or packaging within a very few minutes of sampling.
3. This Standard prohibits packaging materials that may corrode the containment system. The primary corrosion mechanisms of interest are general corrosion and stress corrosion cracking. A recent evaluation of both types of corrosion using PuO_2 concludes that neither of these mechanisms is likely to be significant under storage environments anticipated by this Standard. [Kolman, 1999]

   General corrosion is not a credible problem because the quantity of oxidizer (oxygen or water) available to react with the containers is too limited to be significant. The initial amount of elemental oxygen available if an air atmosphere was present during packaging is very small, and little elemental oxygen is expected to result from catalytic or radiolytic decomposition of water (see Section A.6.1.2 of this appendix). This also makes corrosion pitting highly unlikely.
4. To promote material homogeneity for facilitating MC&A measurements and stabilization tests (such as for adsorbed residual volatiles), and to preserve

characterization information, only similar materials (comparable in form and composition, for example) should be packaged together.

A.6.4 Storage – Surveillance of Stored Packages for Safety

Surveillance is to be non-destructive to preserve storage container integrity. Examples of non-destructive testing methods include:

- Radiography to observe physical changes in the stored material (e.g., oxide growth on uranium metal) and dimensional changes of the inner container (internal pressure change);
- Eddy current or ultrasonic testing of the integrity of the outer container and its welds; or
- Weight measurement change, which would indicate a breach in the package.

Note that the destructive examination of a very limited number of storage packages is not precluded when the cost, including personnel exposure, is outweighed by the value of the information to be obtained.

A.6.4.1 Surveillance Program

1. Plans need to be responsive to site policies and practices.
2. During the course of packaging and storage of ^{233}U metal and oxide, there is a possibility that some container manufacturing defects may not be detected, that some batches of material may be improperly stabilized or packaged, or that some of the packages may be damaged during handling and storage. One function of the surveillance program is to identify these potential threats to package integrity and ensure their mitigation. Flaws in initial packaging are expected to be detected by inspection of every package (primarily for weld integrity) within 30 days of packaging. Ordinarily, this inspection would be done immediately after packaging, but 30 days delay reasonably accommodates operational considerations. This initial inspection should provide baseline information on the leak rate of both welded containers (the inner container should be inspected after it is closed and before insertion into the outer container, and the outer container inspected after it is closed), verification of contents through non-destructive assay (NDA) measurements, and any other information deemed desirable and attainable through non-destructive measurements such as radiography. This initial inspection may be part of the quality program for verifying package integrity.

Along with initial inspections, the Surveillance Program may consist of other elements that provide enhanced surveillance to certain subpopulations of the package inventory. These subpopulations may be determined on a random or non-random basis

3. Inspection/surveillance methods must be documented to ensure consistency. Delineation of responsibilities is needed to ensure a consistent management approach and awareness of responsibilities.
4. No further basis is provided.

A.6.4.2 Surveillance Parameters

The parameters specified by the criterion are the measurable indicators identified in Sections 6.2 and 6.3, above. These parameters provide an indication of material stability and package integrity.

A.6.4.3 Evaluation of Surveillance Data

These evaluations confirm the continuing safety of the packages.

A.6.5 Documentation

A.6.5.1 Database

An electronic database is recommended because a manual database could be overly cumbersome. The architecture is not specified here to allow maximum flexibility to interface with existing databases and files. Some data will be classified, partly because Category I quantities of SNM will be stored.

A.6.5.2 Database Content Elements

1. These parameters allow the most appropriate characterization practical. The intent is to capture all available relevant information, and not to require additional characterization beyond that which is already available.
2. Package data can meet a number of needs. For example, if a Package exhibits unexpected behavior, these data can help identify other, similar Packages that may require inspection. These data also allow disposition processing to be optimized.

3. No further basis is provided.
4. No further basis is provided.

A.6.6 Quality Assurance

No further basis is provided.

A.7 STORAGE FACILITY FEATURES

A.7.1 Nuclear Criticality Safety

A principal safety consideration for the safe storage of ^{233}U is eliminating the possibility of the material reaching a configuration that would result in criticality. Criticality avoidance is a prime priority in safety considerations in the design and operation of a ^{233}U storage facility. In addition to providing an array that is criticality safe, the packages and facility shall be engineered, constructed, controlled, and monitored to avoid the occurrence of accidental criticality for all credible natural phenomena events such as fires, flooding, earthquakes, and tornadoes. Because criticality safety is considered the dominant safety concern in the design and operation of a ^{233}U storage facility, the vault area should be designed with consideration of water sources such as fire sprinklers. Co-existing combustible materials should be minimized or eliminated from the facility in order to minimize the potential for fires and the need for fire suppression systems.

A majority of the ^{233}U in inventory consists of mixtures of ^{233}U and ^{232}U or mixtures whose properties are dominated by the ^{233}U and ^{232}U content. Uranium-233 has different nuclear criticality properties than the other two SNMs, ^{235}U and ^{239}Pu. Therefore, facilities designed for ^{235}U and plutonium may not be acceptable for comparable activities involving ^{233}U from a nuclear criticality safety standpoint and these facilities shall be evaluated to meet the requirements for criticality safety specified in DOE O 420.1.

A.7.2 Confinement of Contamination

The matrix of the ^{233}U-bearing material and/or the sealed inner container provides the first barrier against spread of contamination; the outer container and the tube vaults provide additional barriers. The packaging should be designed to maintain mechanical integrity, including its seal, during normal handling. However, this package is not expected to provide protection against all perils such as major fires and earthquakes; design of the facility and of the storage array are expected to address these considerations.

A.7.3 Radiation Shielding

Uranium-233, with its associated contaminant isotope ^{232}U, presents more severe external radiation hazards than any of the naturally occurring uranium isotopes do. Massive biological shielding is required, where high concentrations of ^{232}U occur, to protect personnel from the 2.6 MeV (4.17E-6 erg) gamma emission of ^{232}U daughter product, ^{208}Tl. The occupational radiation exposure should be kept as low as reasonably achievable (ALARA) and radiation protection be provided as specified in 10 CFR 835, "Occupational Radiation Protection." Dose rates are dependent on the source (e.g., activity, geometry, and matrix), shielding, and source-to-detector configuration, so expected dose rates for actual conditions should be determined on a case-by-case basis.

Except for spontaneous fission, neutrons are not directly produced during the radioactive decay of any of the uranium isotopes or the sequential decays. However, alpha-neutron reactions, in which alpha particles react with low-Z isotopes such as ^{6}Li, ^{7}Li, ^{9}Be, ^{10}B, and ^{19}F, (and to a lesser extent ^{27}Al and ^{28}Si), generate neutrons. Depending on the material storage system used, the facility itself may serve as a shield.

A.7.4 SNM Safeguards

DOE requirements for safeguards are given in DOE O 470.1, DOE O 471.2A, DOE O 472.1B, DOE 474.1, and DOE 5632.7A.

Appendix B. Derivation of Pressure Equation

This Appendix provides a derivation of the equation used to bound the internal pressure of the storage package. It also provides guidance on use of the equation. It is assumed that the Ideal Gas Law applies to the conditions and gases important to the calculations. According to that law

$$PV = n\mathrm{R}T \tag{B.1}$$

where P is absolute pressure, V is volume, T is absolute temperature, n is the number of moles of gas, and R is a constant with units consistent with those chosen for P, V, and T. If a gas is at some standard condition, described by P_0, V_0, and T_0, then the quantity nR can be evaluated as

$$n\mathrm{R} = \frac{(P_0)(V_0)}{T_0} \tag{B.2}$$

At constant volume the pressure under different temperature conditions can be determined by

$$P_1 = \frac{nR}{V}T_1 = P_0\left(\frac{T_1}{T_0}\right) \tag{B.3}$$

In the above equation, T_1 is the temperature at which P_1 is to be evaluated. In the case of a storage can, the volume, V, will simply be the interior volume of the outer container, less the volume occupied by internal containers, less the volume occupied by the ^{233}U oxide material. This volume can be calculated as

$$V = V_c - \frac{m}{r} \tag{B.4}$$

where V_c is the volume of the container, m is the mass of the oxide and r is the density of the oxide. Densities for uranium oxides are given in Table B.1. For cases where the stoichiometry is not known, the most limiting density (7300 kg/m^3 or 456 lb$_m$/ft^3) should be used.

Table B.1. Theoretical densities of uranium oxides

Uranium oxide	Theoretical Density (g/cm^3) [lb_m/ft^3]
UO_2	10.970 [685]
U_3O_8	8.380 [523]
UO_3	7.300 [456]

For ideal gases, the pressure of a mixture of gases can be determined as the sum of the partial pressures of the individual gases. Four gases that require consideration in a ^{233}U storage container:

(1) the container fill gas, P_f;
(2) any gases evolved through radiolysis, chemical reactions, or desorption, P_{H2};
(3) helium generated by alpha decay of the contained radioactive species, P_{He}, and
(4) radon generated in the decay chains of any ^{232}U and ^{238}U, which may be stored with the ^{233}U.

B.1 CONTAINER FILL GAS

The appropriate equation is simply the gas law at constant volume:

$$P_f = \frac{(P_0)(T_1)}{T_0} \tag{B.5}$$

where P_0 is the pressure at which the container was loaded (usually atmospheric), and T_0 is the temperature of the fill gas when the container was sealed.

B.2 EVOLVED GASES

There are two significant sources of evolved gases in ^{233}U containers, water and plastics. As described in section A.6.1.2.1 water may react with stoichiometric UO_2 to form hydrogen. Water may also undergo radiolysis, forming a variety of compounds and radicals. However, this is a reversible process, and thermodynamics favors the reverse reactions. Metal oxides, including uranium oxides may also act as a sink for oxygen and hydrogen. Therefore, the number of moles of evolved gas over time will be approximately bounded by assuming complete reaction of the water to form hydrogen and complete scavenging of oxygen to prevent the reverse reaction. Organics are forbidden by Sections 6.2.2.2 and 6.2.2.4 and will not be considered here. Combining the hydrogen component with the Ideal Gas Law gives,

$$P_{H2} = \frac{RT_1}{V}\left(\frac{(m)(X_{H2O})}{MW_{H2O}}\right) \tag{B.6}$$

where X_{H20} is the mass fraction of water in ^{233}U and MW_{H2O} is the molecular weight of water.

B.3 HELIUM GENERATION

The mass fraction of a radionuclide that is converted to helium through alpha decay may be calculated using a program such as ORIGEN 2. This fraction, b, is shown in Table B.2 for a 50-year period for isotopes of uranium which are found in ^{233}U bearing materials.

Table B.2 Mass fraction of uranium isotopes converted to helium after 50 years

Isotope	b
^{232}U	0.0381
^{233}U	3.8e-6
^{234}U	2.4e-6
^{235}U	8.4e-10
^{236}U	2.5e-8
^{238}U	1.3e-10

For the ^{233}U-bearing material in storage, only the terms for ^{232}U and ^{233}U need be considered since the other isotopes will not be present in large enough quantities to have significant contributions.

Incorporating this data into the gas law results in

$$P_{He} = \left(\frac{m}{MW_{He}}\right)\sum(b_i)(X_i)\left(\frac{RT_1}{V}\right) \tag{B.7}$$

B.4 RADON GENERATION

The longest-lived isotope of radon, ^{222}Rn, has a half-life of 3.8 days. Over a 50-year period, any Rn intermediate on a decay chain will reach a steady-state concentration that is insignificant by comparison to the helium that is generated by alpha decay of other radionuclides in the same decay chain. Thus, the partial pressure contribution of all isotopes of Rn may be neglected.

B.5 AGGREGATE PRESSURE EQUATION

Summing the partial pressures from Equations B.5, B.6, and B.7 and evaluating at a storage temperature, T_1, leads to

$$P = \frac{(P_0)(T_1)}{T_0} + \left(\frac{(R)(T_1)}{V}\right)\left[\frac{(m)(X_{H2O})}{MW_{H2O}}\right] + \left(\frac{m}{MW_{He}}\right)\sum (b_i)(X_i)\left(\frac{(R)(T_1)}{V}\right) \tag{B.8}$$

Substituting Equation B.4 for volume and combining like terms leads to

$$P = \left(\frac{T_1}{T_0}\right)\left[P_0 + \left(\frac{RT_0}{V_c - \frac{m}{r}}\right)\left[\frac{(m)(X_{H2O})}{MW_{H2O}} + \left(\frac{m}{MW_{He}}\right)\sum (b_i)(X_i)\right]\right] \tag{B.9}$$

B.6 BOUNDING PRESSURE CALCULATION USING SRS BAGLESS LOADOUT CAN

This section presents an example calculation of the bounding pressure expected in a SRS bagless loadout can. A typical can has an inside diameter of 0.874 m (0.287 ft.) and a height of 0.0254 m (0.083 ft.). A loading of 1.5 kg (3.3 lb_m) ^{233}U as UO_3 (the least dense of the uranium oxides to minimize free volume), with 0.5 wt % moisture is assumed.

Based on these conditions, the bounding hydrogen partial pressure is 1108 kPa (160.7 psia), and the bounding helium partial pressure is 9.38 kPa (1.36 psia). In practice, a great deal of this hydrogen will be adsorbed by oxides or react with oxygen containing species to reform water. Further assuming the material was packaged at atmospheric conditions and that the temperature during storage increases by 10% on an absolute basis [Thein 1999], Equation B.9 yields P=1340 kPa (194 psia). This pressure is well below the specified 2070 kPa (300 psia) criteria from Section 6.2.1.4.

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CONCLUDING MATERIAL

Review Activity:

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