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Exam Preview:

1. The advantage to transporting material as LSA is that, for domestic transportation of an exclusive use shipment that is less than an A_2 quantity, bulk packaging's may be used.
   a. True
   b. False
2. The WSRC PV is designed for use with the UC-609 to ship up to __ grams of gaseous tritium in a single valve, 21L-volume container at pressures up to 1,200 torr.
   a. 10
   b. 50
   c. 100
   d. 150
3. LSA is a quantity of Class 7 (radioactive) material with limited specific activity. Which LSA class matches the description: is defined as solids such as consolidated wastes or activated materials that meet the testing requirements of 49 CFR 173.468 (water leach test)?
   a. LSA-I
   b. LSA-II
   c. LSA-III
   d. LSA-IV
4. The advantage to transporting material as LSA is that, for domestic transportation of an exclusive use shipment that is less than an A_2 quantity, bulk packaging's may be used.
   a. True
   b. False

5. The UC-609 and BTSP are drum type packaging with a limited volume containment vessel. Their containment vessels are authorized for tritium gas, absorbed tritiated water vapor, or tritides and are limited to 100 and ___ grams of tritium, respectively
    a. 300
    b. 250
    c. 200
    d. 150
6. For airborne effluents, the annual discharge to the air for all radionuclides on site must be such that the offsite boundary dose is less than 50 mrem.
    a. True
    b. False
7. According to the reference material, type B (M) packages are those Type B package designs which have a maximum normal operating pressure of more than ___ kilopascals per square centimeter or a relief device which would allow the release of Class 7 (radioactive) material to the environment under the hypothetical accident conditions specified in 10 CFR Part 71.
    a. 300
    b. 500
    c. 700
    d. 900
8. According to Section 7.2.3 of the reference material, each lifting attachment that is a part of the package must be designed with a minimum safety factor of __, based upon breaking strength, when used to lift the package in the intended manner.
    a. 3
    b. 5
    c. 7
    d. 8
9. EPA uses the economic discard limit (EDL) to evaluate whether a hazardous material or residue should be classified as a waste. If it has sufficiently low economic value, the material should be considered for recovery and reuse.
    a. True
    b. False
10. According to the reference material, superfine clay will hold 73 percent water by volume without any free liquid. A DOT 7A, 55-gallon drum filled with Superfine□ clay will hold approximately __ gallons of water.
    a. 33
    b. 35
    c. 38
    d. 40

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FOREWORD

Tritium handling practices have evolved over several decades at Department of Energy (DOE) tritium facilities. The objective has been to accomplish required tritium work while minimizing and controlling the exposure of workers, the public, and the environment to tritium. This document provides guidance for the handling, storing, and shipping of tritium.

This Standard is approved for use by all DOE elements and their contractors. DOE technical standards, such as this Standard, do not establish requirements. However, all or part of the provisions in a DOE standard can become requirements if either they are explicitly stated to be requirements in a DOE requirements document or the organization makes a commitment to meet the Standard in a contract or in an implementation or program plan. The DOE implementation plan to address the Defense Nuclear Facilities Safety Board (DNFSB) Recommendation 2005-1 committed to implement interim storage packaging provisions for tritiated materials, which are discussed in Section 6.6. These packaging provisions are requirements for tritium facilities under the auspices of the National Nuclear Security Administration (NNSA) and the Environmental Management Program Office (EM).

The author of the Standard, Bill Weaver of DOE Office of the Chief of Nuclear Safety (CNS), wishes to acknowledge the contributions of the Savannah River National Laboratory (SRNL) and the Savannah River Site (SRS) staff, Paul Blanton, Paul Korinko, Greg Staack, and Steve Xiao; Ken Keeler and Mike Rogers of the Los Alamos National Laboratory (LANL); Diane Spencer of the Lawrence Livermore National Laboratory (LLNL); CNS staff members Steve McDuffie, Marlene Fitzpatrick, and Elaine Beacom; Nazir Kherani of the University of Toronto; Armando Antoniazzi of Kinectrics, Inc.; Genevieve Weaver of Penn State; Tracy Getz, Robin Henderson and Robert Waxman of Office of The General Counsel; and Steve Zobel, a staff member from the DOE Office of the Associate Under Secretary for Environment, Health, Safety and Security.

ACRONYMS

AEA	Atomic Energy Act of 1954
AI	Alveolar-Interstitial region
ALARA	As Low As Reasonably Achievable
ALI	Annual Limit on Intake
AMAD	Activity median aerodynamic diameter
AMD	Activity median diameter
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASN	French Nuclear Safety Authority
ANSI	American National Standards Institute
ARAR	Applicable or Relevant and Appropriate (CERCLA)
AU	Office of the Associate Under Secretary for Environment, Health, Safety and Security
AWQC	Ambient water quality criteria
bb	bronchiolar region
BB	Bronchial region
Bq	Becquerel
BS	Bone surface
BTSP	Bulk Tritium Shipping Package
BZA	Breathing zone air (sampler)
CANDU	CANada Deuterium Uranium pressurized reactor
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFR	Code of Federal Regulations
Ci	Curie
CMD	Count median diameter
CoC	Certificate of Compliance
CRC	Combustion Research Center
CWA	Clean Water Act
DCS	Derived Concentration StandardD&D Decontamination and Decommissioning
DAC	Derived Air Concentration
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DCF	Dose Conversion Factor
DCF_o	Dose Conversion Factor based on observed activity
DCG	Derived Concentration Guide
DNFSB	Defense Nuclear Facilities Safety Board
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation

DPM	Disintegrations per minute
DSA	Documented Safety Analysis
E_{50}	Committed Effective Dose
EDL	Economic Discard Limit
EPCRA	Emergency Planning and Community Right-to-Know Act
EPDM	Ethylene Propylene Diene Monomer
EPA	Environmental Protection Agency
EH	Office of Environment, Safety and Health
EIS	Environmental Impact Statement
EM	Office of Environmental Management
ET	Extrathoracic
f_1	fraction of radionuclide absorbed from the GI tract
FCA	Fire Control Area
FDTAS	Field Deployable Tritium Analysis System
FY	Fiscal Year
GI	Gastrointestinal
HEPA	High-Efficiency Particulate Air
HDPE	High-Density Polyethylene
HIVES	Highly Invulnerable Encased Safe
HMR	Hazardous Material Regulations
HSV	Hydride Storage Vessel
HSWA	Hazardous and Solid Waste Amendments
HTV	Hydride Transport Vessel
HVAC	Heating, Ventilation, and Air Conditioning
IAEA	International Atomic Energy Agency
IATA	International Air Transport Association
ICRP	International Commission on Radiological Protection
IMT	Insoluble Metal Tritide
INL	Idaho National Laboratory
ISM	Integrated Safety Management
ITER	International Thermonuclear Experimental Reactor
ITP	Insoluble Tritiated Particulate
keV	Kiloelectron volt
LANL	Los Alamos National Laboratory
LDPE	Low-Density Polyethylene
LDR	Land Disposal Restriction
LLD	Lower Limit of Detection
LLI	Lower Large Intestine
LLNL	Lawrence Livermore National Laboratory
LN	Lymph Node
LSA	Low Specific Activity
LSC	Liquid Scintillation Counting

LLW	Low-level waste
MAR	Material at Risk
mCi	Millicurie
MCL	Maximum Contaminant Level
mm	Millimeter
mrem	Millirem
NFPA	National Fire Protection Association
NMMSS	Nuclear Materials Management and Safeguards System
NNSS	Nevada National Security Site
NP	Nasal Passage Region
NPDWR	National Primary Drinking Water Regulation
NPH	Natural Phenomena Hazard
NRC	U.S. Nuclear Regulatory Commission
NRPB	National Radiological Protection Board
OBT	Organically Bound Tritium
OPI	Office of Primary Interest
ORR	Operational Readiness Review
OH-	Hydroxide
P	Pulmonary Parenchyma Region
PC	Performance Category
PCB	Polychlorinated biphenyl
PMR	Palladium Membrane Reactor
PPE	Personal Protective Equipment
PPPL	Princeton Plasma Physics Laboratory
psia	pounds per square inch absolute
psig	pounds per square inch gauge
PSO	Program Secretarial Officer
PTFE	Polytetrafluoroethylene
PVC	Polyvinyl chloride
PV	Product Vessel
RCRA	Resource Conservation and Recovery Act
RCS	Radiological Control Standard
RM	Remainder Organ
RMA	Radioactive Materials Area
RTF	Replacement Tritium Facility
RWP	Radiological Work Permit
S	Stomach or specific source organ (used with SEE)
SAES	Societá Apparecchi Elettrici e Scientifici
SAF	Self Absorption Factor
SAM	Surface Activity Monitor
SAR	Safety Analysis Report
SCO	Surface Contaminated Object

SDWA	Safe Drinking Water Act
SEE	Specific Effective Energy
SEL	Seismic Equipment List
SEM	Scanning Electron Microscope
SEP	Seismic Evaluation Procedure
SI	Small Intestine
SMT	Stable Metal Tritide
SNL	Sandia National Laboratory
SNLL	Sandia National Laboratory, Livermore
SNM	Special Nuclear Material
SRNL	Savannah River National Laboratory
SRS	Savannah River Site
SSCs	Structures, Systems, and Components
STC	Special Tritium Compound
Sv	Sievert
T	Tissue
TB	Trachea and Bronchial Region
TFG	Tritium Focus Group
TRL	Tritium Research Laboratory, Sandia National Laboratory
TWD	Technical Work Document
TSD	Treatment, Storage, and Disposal
TSR	Technical Safety Requirement
UB	Urinary Bladder
UHMWPE	Ultra-High-Molecular-Weight Polyethylene
ULI	Upper Large Intestine
WETF	Weapons Engineering Tritium Facility, Los Alamos National Laboratory
WSRC	Washington Savannah River Company

6.0 TRITIUM RECEIVING AND STORAGE

In the past, DOE has been a commercial supplier of tritium to industry. DOE ceased commercial sale of tritium at the end of fiscal year (FY) 1995 and now only supplies tritium to DOE-operated facilities. There is no charge to DOE facilities for the tritium. Costs may be incurred for the shipping container, the loading and shipping of the container, and any other incidental costs associated with shipping tritium from the supplying facility to the user.

6.1 Shipping Packages

The packages for shipment of tritium must meet regulatory requirements based on the amount of tritium being transported. The design requirements for radioactive materials shipping packages are specified in 10 CFR 71.43 and 49 CFR 173.410. See Section 7.2.3, *Minimum Requirements for Packaging*, for the basic shipping container design requirements.

6.1.1 Type A Shipping Packages

Type A quantities (less than 40 TBq) of tritium may be shipped in DOT 7A Type A shipping packages. Although several commercial manufacturers will design packages that meet Type A shipping criteria for solids, it is difficult at any given time to identify commercial manufacturers who are producing Type A packages for gas. There are no comparable requirements, as discussed in section 7.3.3, for leak-testing Type A packages as there are for Type B packages.

6.1.2 Type B Shipping Packages

Tritium in Type B quantities (greater than or equal to 40 TBq) is authorized to be shipped in the BTSP, UC-609, or H1616 packages. The method for shipment in each of these packages utilizes various “facility primary and secondary” process vessel designs that contain tritium in the state in which it is to be shipped; e.g., tritides, gas, or water vapor adsorbed on a molecular sieve. Tritium in the form of tritides is held in the WSRC-developed HTV or HSV or as a gas in the WSRC-developed Product Vessel (PV) or DOE-developed reservoirs. The three aforementioned Type B packages can be transported as non-exclusive (i.e., with other than radioactive cargo) by highway, rail, water, or air. Tritium in less than a Type B quantity (< 40 TBq) may be shipped in Type A packaging; e.g., a Commercial Tritium Containment Vessel (CTCV).

The LLNL-designed UC-609 shipping package is authorized to ship HTVs, PVs, and ALM-1s, along with other types of process vessels not described in this Standard.

The SRNL-designed BTSP shipping package is authorized to ship HTVs, HSVs, and PVs. A new revision to the BTSP SARP is being developed to include authorization to include ALM-1s.

When bulk forms of tritium are ordered from or are to be returned to a DOE facility, they are generally shipped in an HTV or PV. Caution is to be exercised when shipping “empty” HTVs, as the residual heel may contain > 40 TBq, thereby requiring a Type B shipment for the “empty” HTV. For design and analysis purposes, the primary process vessel shipped in the UC-609 and BTSP receives no credit for tritium containment. As discussed in section 3.3.1.b, tritium contained in certified Type B packages is credited to withstand credible facility accidents and thereby may be excluded from the inventory of a facility for DSA purposes.

Type B quantities of tritium may also be transported from one DOE site to another in the Sandia-designed H1616 shipping package of which there are two variations: the -1 and -2. DOE authorization is required before using the H1616 package. The primary tritium process vessels enclosed inside the H1616 must meet the SNL specification SS393217. The HTV is qualified for shipment in the H1616 packaging. Of the two H1616 variations, H1616-1 and H1616-2, qualification for transport in the H1616-1 automatically includes qualification for transport in the H1616-2.

Tritium may also be held as gas on LP-50s. Once a Type B package, it is no longer used for shipping off-site but is sometimes used as an onsite DOE storage/process vessel. Likewise,tritium-contaminated water vapor captured on adsorbent molecular sieve materials may be held onsite in ALM-1 process vessels.

6.2 Product Containers

6.2.1 WSRC Product Vessel

The WSRC PV is designed for use with the UC-609 to ship up to 10 grams of gaseous tritium in a single valve, 21L-volume container at pressures up to 1,200 torr. The PV is designed to meet the requirements of the ASME Boiler and Pressure Vessel Code, Section VIII, Division 1, 1991 Addenda, Lethal Service, Full Vacuum to 8.5 psig (1,200 torr) at 77°F.

Although the PV is designed to meet ASME code requirements, the PV is not ASME code stamped. The vessel is equipped with a single valve, Nupro® SS-4HS-TW valve, 316 SS body and is closed with a male Cajon® SS-4-VCR-4 nut and a Cajon® SS-4-VCR-CP cap.

6.2.2 WSRC Hydride Transport Vessel

The HTV[74] can be loaded with a maximum of 3 moles (18 grams) of tritium as uranium tritide. The main body of the HTV is fabricated from two four-inch Schedule 40 pipe caps, welded together and contains a free volume of approximately 690 cubic centimeters. The HTV weighs 9.3 pounds, has a maximum outside diameter of 4.560 inches, and is 9.980 inches high. The HTV is equipped with two

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[74] WSRC-RP-92-1161, SRS H1616 Hydride Transport Vessel Qualification Report (U), Revision 2, 1995

ports with a single Nupro® SS-4HS-TW closure valve on each port. Both ports are equipped with 2-micron nominal pore size, cup-shaped, porous stainless steel filters welded on the end of the port tube inside the vessel. The purpose of the filters is to confine the uranium powder. One filter is positioned in the volume above the uranium and is connected through a valve to the female port with a Cajon® SS-4-VCR-1 female nut. The second filter, connected through the valve to the male port, is positioned below the level of the uranium powder in the bottom of the vessel and is connected through the valve to the male port with a Cajon® SS-4-VCR-4 male nut. Both ports are used when the bed is used in a flowthrough mode of operation.

The maximum normal unloading temperature is 450°C, and the maximum unloading pressure at temperature is 2.9 psia. Exceeding the 2.9 psia pressure at temperature will impact the vessel service life.

When uranium is loaded to full stoichiometry with tritium to form UT_3, it retains the decay helium in the solid for about 300 days until the concentration of helium reaches approximately 0.134 He:U, after which some of the helium will be released from the solid. The helium release rate increases over a period of about 600 days, until it equalizes with the helium generation rate.

The HTV Dissociation Pressure [38], P is

$$\log P_{atm} = -(4038.2/T) + 6.074$$

where T = temperature (K)

which implies

$$P_{atm} = 10^{-(4038.2/T) + 6.074}$$

The second equation for the HTV Dissociation Pressure is

$$\log P_{psia} = -(4038.2/T) + 7.2413$$

which implies

$$P_{psia} = 10^{-(4038.2/T) + 7.2413}$$

WSRC conducted several tests on the HTV to determine the impact of an air ingress incident. The results indicate that in the dehydride state the uranium reacted with both oxygen and nitrogen in the air. The reaction stopped when the accumulation of argon and moisture in the vessel reached atmospheric pressure and prevented additional air from entering the vessel. The maximum temperature was approximately 200°C. If air is drawn through the uranium continuously, a temperature higher than 1,000°C can be reached, and damage to the vessel may occur.

6.2.3 Recommendations for Improvements for Product Containers

Both the PV and HTV containers are single-valve designs (although the HTV actually has two valves; one for filling and one for evacuating), employing an SS-4HS-TW valve. Selection of this valve was based in part on the favorable rating it received against valves of similar size. It is rated at 315°C and 1000 psig. Additionally, the valve stem tip is Stellite™ spherical design, a desirable tip configuration. Although the leak rate is verified annually at SRS, the employment of a single H-type valve close to the heat zone is not an optimal design, as described in EH Technical Notice 94-01, *Guidelines for Valves in Tritium Service*, and users should be prepared for potential leaks across the valve seat in addition to bellows leaks.

Modifications to existing designs could be incorporated into future revisions. In addition to valve over-torque protection and/or hardware upgrades, the HTV use of thin-walled SS 348 may lead to more than desired outgassing as the number of cycles accumulates. The upgrading of material or jacketed design are potential options. Other design improvements could include internal baffling and different (types, sizes, and locations) filters. An operational practice of tracking and segregating containers that are used for high-purity shipments and those used for other shipments (e.g., scrap shipments) would be beneficial in ensuring the capability to ship War Reserve gas by not contaminating the U bed with mixed isotopes.

6.3 Valve Container Operations

The use of two valves in series on containers filled with tritium that are to be disconnected from the tritium apparatus on a routine basis has been in common use for several years. When the failure of a single valve seat can result in the release of significant quantities of tritium, two valves in series should be used. If the seat of a single valve develops a leak during storage and the container port is uncapped, the tritium will be released into the containment system through the failed valve seat.

Valve seat failure is often associated with damage to the seat caused by long exposure to tritium or by misalignment resulting from improper operation or maintenance practices. Double valves are recommended in all cases where the container valve seat is exposed to tritium for long periods of time or is susceptible to misalignments. Additional guidance for tritium valve selection, operation, and maintenance is contained in the Office of Nuclear and Facility Safety's Technical Notice 94-01 and discussed in *Guidelines for Valves in Tritium Service.*[75] This document describes desirable features and material for various tritium applications and lists recommended practices.

It is assumed by experienced tritium handlers that valve seats can fail, and, therefore, the basic strategy in double-valve use is depicted in Figure 6-1 and described as follows:

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[75] William Weaver, *Guidelines for Valves in Tritium Service*, Fusion Technology, Vol. 25, July 1994.

- The container is connected, leak tested, and filled with the required quantity of tritium, and the second container valve is closed.
- The tritium between the second valve and the tritium supply is pumped away, and the section is isolated and monitored for pressure rise.
- If a pressure rise indicates that the second valve seat is leaking, then the container is emptied while connected to the manifold, and the valve seat is repaired.
- If no leak is detected, then the first valve is closed and the second valve is opened, and the tritium trapped between the first valve and the second valve is pumped away. This section is isolated and monitored for a pressure rise.
- If this test indicates that the first valve seat is leaking, the container is emptied, and the valve seat is repaired.
- If no leak is detected, then the second valve is closed, the container is disconnected, and the container port is capped and sealed.
- The container is transported to a new location or placed in storage as appropriate to the task.

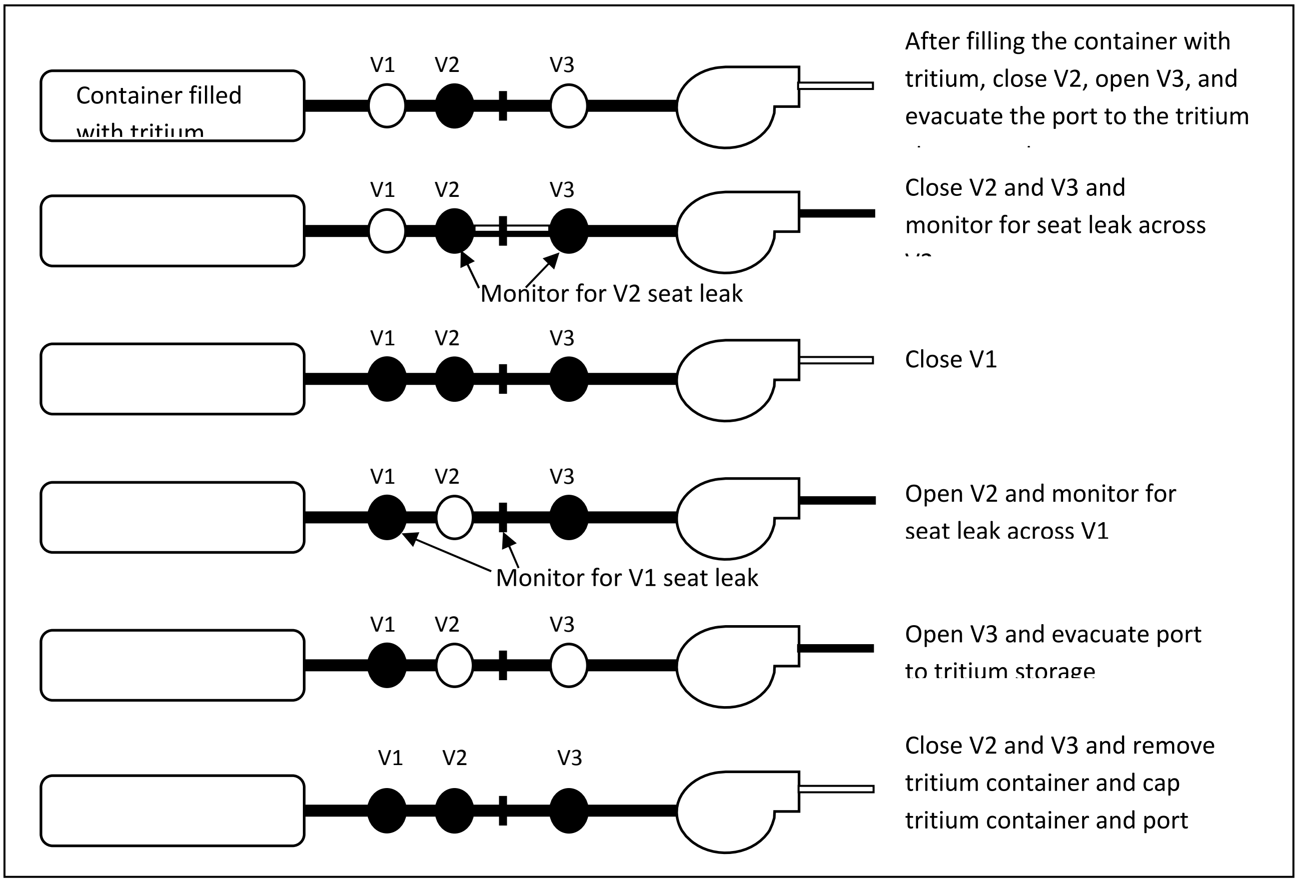
This leaves the container in the following condition:
The tritium is trapped inside the container by the first valve, and the valve seat of the first valve is exposed to tritium during storage or transport.

- The space between the first valve and the second valve has been evacuated and contains no significant quantity of tritium or other gases. The second valve seat is not exposed to tritium during the storage or transport cycle.
- The space between the second valve and the container port cap contains glovebox gas or air sealed into the void space by the port closure seal.

To make a reconnection to a manifold, the following steps apply:

- Before the container is reconnected to a manifold, the two valves are checked by hand to make sure they are closed.
- The port is uncapped and connected to the manifold.
- The gas trapped in the space between the port and the manifold is pumped out, and the connection is leak checked using a helium leak detector calibrated to detect a leak of $< 10^{-6}$ cc He/second.
- If the port connection is leaking, the leak is repaired, and the leak test is rerun.
- The first hand valve is opened and then closed to allow tritium from the container to enter the void space between the two valves.
- The second hand valve is opened and then closed to allow the trapped tritium to expand into the port volume.

- The outside of the port connection area can be sniffed with a tritium monitor. The containment system tritium monitor may be observed for an increase in tritium concentration to detect a tritium leak from the port connection into the containment volume.
- If no leak is detected, then the two series valves may be opened, and the reconnection is complete

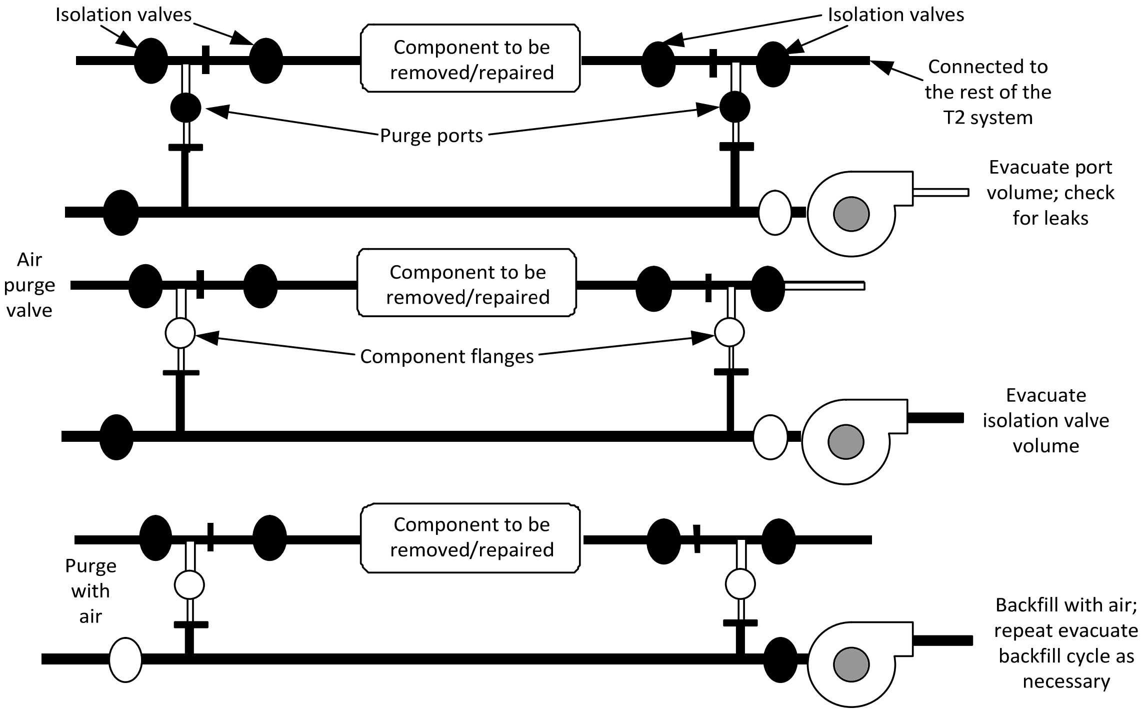


6.3.1 Tritium Apparatus, Isolation Valves, and Purge Ports

A “purge port” is a capped, sealed port connected through a valve to a potentially tritium-contaminated volume. The purpose of a purge port is to provide a path, which can be used to remove tritium-contaminated gases from the isolation volume prior to making a line break at the component flanges to remove a component. If the tritium-contaminated gases are not removed from the isolation volume, they will be released into the containment system when the flange is unsealed. The released gases and outgassing from the isolation volume surfaces will contaminate the containment volume gases. Following removal and replacement of a component, the port is used to leak test the new component and flange seals prior to placing the component in service.

Figure 6-2 is an illustration of two purge ports installed to allow evacuation of the volume and leak testing of the flanges between two sets of valves, thus allowing isolation and removal of a tritium-contaminated component. Note that the purge ports are part of the permanently installed system, and are not part of the component.

The component is isolated with two valves on each side of the component, and a purge port has been installed between each set of valves. The purge port allows evacuation of the volume between the two valves to remove the tritium-contaminated gases from the isolation volume prior to removal of the component.



15) Figure 6-2: Purge ports and isolation valves

In operation, the two sets of valves are closed to isolate the component from the rest of the tritium manifold. A vacuum pump is connected to the two purge ports. The purge port valves are opened and the gases trapped between the isolation valves are evacuated to remove the tritium-contaminated gases. In most applications, air is allowed to enter the purged volume, and the evacuation operation is repeated. Other gases, such as argon and nitrogen, may be used; however, air is usually more effective at decontaminating the surfaces. Ambient air entering the purged volume contains several thousand parts per million of normal water along with the nitrogen and oxygen. Some of the tritium and HTO on the internal surfaces of the purged volume exchange with the hydrogen and water in the ambient air and are pumped out during the purge cycle. This purge/backfill cycle is repeated three to six times to remove as much of the tritium as possible before disconnecting the component. Performance of three ambient air purge backfill cycles is typical, and, in practice, more than six purge backfill cycles has not proven to be beneficial.

After the component has been removed and replaced, the new component is leak-tested through the purge port before the isolation valves are opened. The sequence of operations is as follows:

- The four component isolation valves are closed.
- The purge ports are uncapped and connected to a vacuum pumping system.
- The port volume is evacuated and leak checked.
- If no leaks are found, the purge port valves are opened, and the gases in the volume between the isolation valves are evacuated to a tritium-contaminated waste gas collection and removal system.
- The evacuation valve is closed, and air is allowed to enter the isolation volume through a valve mounted on the maintenance manifold.
- The air inlet valve is closed. The ambient air is allowed to sit in the isolation volume for a few minutes to allow the exchange of hydrogen and tritium to take place.
- The evacuation valve is opened, and the volume is again pumped out to the cleanup system.
- This evacuation and/or backfill sequence can be repeated several times.
- The component is disconnected at the maintenance flanges and removed. The new component is installed.
- To control outgassing from the removed component, blind flanges with installed purge ports may be installed over the open end of the removed component.
- The new flange connections are leak tested through the purge ports. If the new flange connections are not leak tight, they are repaired or replaced and retested.
- When the component and flanges are leak tight, the purge valves are closed, disconnected from the leak test system, and capped.
- The component isolation valves are opened, and the new component is placed in service.

As is always the case when dealing with process quantities of tritium, adherence to procedures is particularly important. If a large source (e.g., container) of tritium is connected via valving to the piping used in evacuating, an inadvertent valve operation could result in a significant loss. One such incident occurred at Mound, in which a valve was inadvertently opened, resulting in a loss of approximately 10-grams of tritium to the cleanup system via the evacuation header.

6.4 Receiving Tritium

Tritiated material may be delivered to a receiving area, a receiving storage area, or directly to the tritium-handling facility. A brief description of the DOE regulatory requirement is as follows:

- The Receiving Area and/or Receiving Storage Area must be inside a posted Controlled Area.
- The Receiving Area and/or Receiving Storage Area must be inside a posted RMA.
- The Receiving Area and/or Receiving Storage Area must be posted to reflect the radiological hazard that exists due to the quantity and types of nuclear material received and/or stored.
- The boundary of the Controlled Area and the RMA can be the same boundary.

The area where the tritiated material is being kept while waiting to be transported from the Receiving Area must be posted as a Controlled Area and an RMA while the tritiated material is present. When the tritiated material is moved to a Storage Area, the signs can be removed. While in the Receiving Area and/or Storage Area, the material must remain in the certified shipping container. No tritiated material handling or unpacking can take place in this area.

As long as the material remains packaged in the approved container (includes certification), the quantities are not a facility issue during the receiving process. However, the total site quantities, including the packaged material, cannot exceed limits analyzed in the site's EIS.

Site and facility management may have internal requirements and limits associated with the receipt and storage of nuclear materials. In general, as long as the material remains in the approved shipping package it can be received, stored, and transported per requirements.

The container should be radiologically surveyed to determine radiological posting needs and to ensure it is safe for storage. When the material is removed from the receiving area, the RMA, and Controlled Area, any radiological sign may be removed.

6.5 Storage of Packaged Tritiated Materials

The packaged tritiated materials may be transported to an approved storage area while awaiting transport to a tritium-handling operations area or shipment. While in the Storage Area, the same rules apply as those in the Receiving Area.

- The Storage Area must be inside a posted Controlled Area.

- The Storage Area must be posted to reflect the radiological hazard that exists due to the quantity and types of nuclear material stored.
- Periodic surveys must be performed and be sufficiently detailed to determine that surface contamination levels are not exceeded, airborne releases are not occurring, and packages are not leaking.

6.6 Interim Storage of Tritiated Materials

6.6.1 Background

Tritiated material pending further handling—whether processing, recovery, repackaging, or shipping—is stored in various locations and configurations. These sections delineate the container configurations for packaging requirements applied to interim storage. Interim storage is on-site storage of tritium/tritiated materials outside of an approved engineered contamination barrier.

T The DNFSB issued in 2005 Recommendation 2005-1 on packaging of nuclear materials in interim storage. The DOE implementation plan to address the DNFSB Recommendation included interim storage provisions for tritium, initially published in DOE HDBK-1129-2008 and repeated below in Sections 6.6.2 and 6.6.3.These packaging provisions are requirements for tritium facilities under the auspices of the National Nuclear Security Administration (NNSA) and the Environmental Management Program Office (EM).

The primary focus of Recommendation 2005-1 is the protection of workers from dose exposure from deterioration of packaging or handling mishaps from nuclear material that are in interim storage. Due in part to the physical, chemical, and radiological differences between tritium and the actinides (which both the DNFSB Recommendation and DOE M 441.1-1 address), interim storage repackaging requirements for tritium consistent with the underlying philosophy, but not necessarily identical to the Manual, are described in section 6.6.2. Additionally, since historical data indicate that worker exposures from tritium have occurred as a result of inappropriate opening of containers, especially those containing legacy waste, operational restrictions were also considered in the development of the tritium requirements which exceed the scope of DOE M 441.1-1. Although not a requirement, a double-valve design, as discussed in section 6.2.3, is recommended for inclusion in the next generation of product vessel designs; this will allow the use of preferred container loading/unloading practices discussed in section 6.3.

6.6.2 Interim Storage Requirements

The goals of this section are compatible with the containment philosophy discussed in detail in Sections 4.1, 5.2, and in the Definitions section. The primary focus of the following goals is the protection of workers from unintended doses due to releases from containers. As such, in addition to mechanical-related failures of the containers, additional release mechanisms (such as opening gas containers in an unprotected area and handling mishaps such as knocking over and spilling tritides from hydride-bearing containers) were considered in formulating the following two objectives in this section.

1) Opening of primary tritium containers, especially those containing legacy material, should be done in a fume hood or glovebox. Other radiological controls can also be used if approved by the cognizant site radiological organization. These controls may include the use of tents, glove bags, elephant trunks, or supplied-air PPE.

2) Tritiated material residing in containers or locations not specifically listed in the "Exclusions to Packaging Requirements" section below must be packaged to meet interim storage goals. The function of the interim storage package is to retain the nuclear material during storage. The interim storage package is defined as an assembly of one or more containers that meets a DOE directive or Standard. The package can consist of a single container or be a double-container configuration with primary and secondary containers. Containers should be constructed consistent with the general design parameters of Chapter 5 of this Standard and must not violate the material compatibility principles. The package must meet both of the following two leak rates to satisfy interim storage requirements:

    a) prior to the drop test, the leak rate on the package as measured by a method comparable to those described in ANSI-N14.5-1977 must be equal to or less than 1 x 10^{-7} ref cm^3/sec, and
    b) leak no more than 6.5 Ci per hour after dropping the package, as a unit, from the working height (for the subject package) at the facility or four feet, at a minimum

6.6.3 Exemptions from Nuclear Material Packaging for Storage Requirements

- Tritium that is located within engineered contamination barriers as defined in DOE M 441.1-1 (e.g., within gloveboxes, or fume hoods (laminar flow)). Normally ventilated rooms, however, are not considered contamination barriers. Additionally, rooms or buildings serviced by an active tritium cleanup system are not considered contamination barriers for the worker. Engineered contamination barriers are on occasion defeated for operational reasons or maintenance acts. Due to the relatively short time frame (i.e., on the order of a few shifts) involved, (compared to interim storage) no compensatory measures are required for the tritium residing within the affected contamination barrier during these periods.

- Tritium that is located within certified[76] shipping containers (e.g., H1616 and UC-609 (and Bulk Tritium Shipping Package (BTSP), which is the UC-609 replacement upon certification) or in a PV container.

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[76] Certified means currently valid, and does not include those shipping containers whose certification period has lapsed.

- Packaged tritiated material that has been declared waste per the local site procedure and that has been packaged to meet DOT or Waste Acceptance Criteria.

- For tritium in containers known to possess less than A2 (1,080 Ci) releasable quantities. Container heels and tritium absorbed onto metal storage beds are not normally releasable and therefore are not included when determining whether A2 quantities are involved.

- Tritium in any form contained in weapons and prototypical designed weapons components.

Tritium packaging that is being handled under radiological (protective) control (e.g., containers that are in the process of connecting/disconnecting to a gloveport, such as tritium in a LP-50 connecting to a gloveport for unloading or for containers undergoing decontamination) are excluded from the packaging requirements for the period of the handling evolution itself. Handling is also inclusive of movement between engineered contamination barriers.

6.6.4 Priority for Working off Tritium Containers

Safe handling and storage of nuclear material at DOE facilities relies on the use of adequate containers to prevent worker contamination and uptake of radioactive material. The combinations of material and container configurations that were adequate for the originally anticipated period of storage or for a particular use may no longer be adequate. Facilities could prioritize the current inventory of tritiated material non-conforming containers (that do not meet any of the packaging exclusions above) based on the inventory and hazards of the material taking cognizance of the relevant physical, reactive, and radiological properties as well as the interactions with the existing and projected container conditions. Particular attention should be directed toward the condition of the container penetration including the weld, fittings and valve(s). Containers that cannot be verified to contain tritium compatible sealing surface material (see Section 5.1) in penetration fittings and valves could be dispositioned first. LP-50s containing substantial inventory would also be high priority containers to be worked followed by other non-conforming containers including molecular sieves. LP-50s and AL-M1-5s must have compensatory measures in place while awaiting repackaging, unloading or waste disposition, including:

a) associated secondary’s/ caps are in place and functioning and

b) the stored containers are under an approved surveillance program (the secondary’s/caps can be removed during surveillance checks) whose objective is the identification of early indication of container degradation. Leak rate testing (normally employing a sniff test method) must be an integral part of the surveillance program

Site plans are to be developed that implement a schedule based on risk for dispositioning tritium residing in non-conforming containers not meeting any of the five exclusions to the packaging requirements.

7.0 PACKAGING AND TRANSPORTATION

7.1 General Administrative Packaging and Transport Requirements

DOT and NRC have the primary responsibility for federal regulation of commercial radioactive material transportation. The DOT provides requirements for the transport of hazardous material in commerce through the Hazardous Materials Regulations (HMR) in 49 CFR Parts 171 through 180, such as requirements for marking, labeling, placarding, and packaging. The NRC provides requirements for the packaging and transportation of radioactive material in 10 CFR Part 71.

The Department of Energy has broad authority under the Atomic Energy Act of 1954 (AEA), as amended, to regulate activities involving radioactive materials that are undertaken by DOE or on its behalf, including the transportation of radioactive materials. In most cases that do not involve national security, DOE utilizes commercial carriers that undertake DOE shipments subject to regulation by DOT and NRC, as appropriate. However, DOE exercises its AEA authority to regulate certain DOE shipments, including shipments by government employees, shipments involving national security or other critical interests, and onsite transfers. As a matter of policy, all DOE packaging and transportation activities must be undertaken in accordance with the relevant DOT and NRC requirements and standards that apply to comparable commercial shipments, except where there is a determination that national security or another critical interest requires different action. In all cases, DOE packaging and transportation activities must be conducted in a manner that meets or exceeds the level of safety required by DOT and NRC for comparable commercial shipments. DOE’s transportation policy and requirements are set forth in DOE Order 460.1C, *Packaging and Transportation Safety*, Order 460.2A, *Departmental Materials Transportation and Packaging Management*, Manual 460.2-1A, *Radioactive Material Transportation Practices Manual*, Order 461.1B, *Packaging and Transportation for Offsite Shipment of Materials of National Security Interest*, and Order 461.2, *Onsite Packaging and Transfer of Materials of National Security Interest* (or their successor documents). Section 4 of DOE Order 460.1C requires in (a)(1) that "Each entity subject to this Order must perform packaging and transportation activities in accordance with the Department of Transportation (DOT) requirements of the Hazardous Materials Regulations (49 CFR Parts 171-180)". 49 CFR 173.416, Authorized Type B packages, cites NRC-certified packages as required. The NRC certification requirements meet the appropriate requirements of 10 CFR Part 71.

Tritium is considered a hazardous, radioactive material under the HMR, however, tritium in concentrations less than 1E+06 Bq/g (2.7E-05 Ci/g) and no more than 1E+09 Bq (2.7 E-02 Ci) in a consignment is considered an exempt quantity and not subject to DOT regulations.

Tritium may also meet the definition of other hazard classes in 49 CFR Part 173; e.g., tritium gas, in certain conditions, may meet the definition of a flammable gas. Tritium-contaminated hazardous waste will be subject to the requirements for hazardous waste shipments.

Organizational responsibility for the packaging and transport of hazardous materials such as tritium needs to be clearly defined. Written procedures should be developed for the operations that package and ship tritium. DOT hazardous materials regulations (49 CFR 172.704) require initial and recurrent general awareness/familiarization, safety-related, and function-specific training for hazardous materials workers. 49 CFR 172.704 also addresses the need for security awareness training; in-depth security training may be required based on the criteria in 49 CFR 172.800.

The EPA also provides pre-transport requirements applicable to hazardous/mixed wastes that address packaging, labeling, marking, placarding, and accumulation time provisions which can be found in 40 CFR 262.30 through 262.34

The guidance provided in this Standard assumes that tritium is the only radioactive material in the packages. If another radionuclide is present, additional requirements may apply to the packaging and transport of the combination of radioactive materials.

7.2 Selection of Proper Packaging

The package types for radioactive material are limited-quantity accepted package, low-specific-activity (LSA), surface-contaminated object (SCO), Type A, and Type B. The definitions of the package types are listed in order of increasing allowable radioactivity material limits. The Type A values related to these packages are found in the table of 49 CFR 173.435.

- Limited-Quantity Excepted Package: The requirements for a limited-quantity package of tritium are as follows:
  - The radiation level at any point on the external surface of the package does not exceed 0.005 milliSievert (mSv)/ hr (0.5 mrem/hr).
  - The amount of radioactive material is no more than the limit in 49 CFR 173.425 (see Table 7-1).
  - The package meets the general design requirements for radioactive materials packaging from 49 CFR 173.410 (see Section 7.2.3).

- The non-fixed (removable) contamination on the external surface of the package does not exceed the limits in 49 CFR 173.443(a) (0.41 Bq/cm^2 or 22 dpm/cm^2 for tritium).
- The outside of the inner or outer packaging is marked “Radioactive.”
- The outside of each package has to be marked with the applicable four digit UN identification number for the material preceded by the letters UN.
- The outside package be labeled with the designated International Air Transport Association (IATA) “Radioactive Material, Excepted Package” handling label.

The advantage to using the limited-quantity accepted packaging is that it is accepted from specification packaging, marking, labeling, and, if not a hazardous material or hazardous waste, it is also accepted from the shipping paper and certification requirements of the HMR.

- LSA is a quantity of Class 7 (radioactive) material with limited specific activity. Shielding materials surrounding the LSA material may not be considered in determining the estimated average specific activity of the package contents. The definition for LSA was revised into three groups: LSA-I, LSA-II, and LSA-III. The definitions for these groups are provided below.

  - LSA-I is defined as contaminated earth, concrete, rubble, other debris, and activated material in which the radioactive material is essentially uniformly distributed and the average specific activity does not exceed $10^{-6}A_2/g$. For tritium, the maximum allowable specific activity in these materials for LSA-I is 4 x 10^{-5} TBq/g (0.001 Ci/g, 1 Ci/kg, or 2.2 lb/Ci).
  - LSA-II is defined as either water with tritium concentration up to 0.8 TBq/L (20.0 Ci/L) or material in which Class 7 (radioactive) material is essentially uniformly distributed and the average specific activity does not exceed $10^{-4}A_2/g$ for solids and gases and $10^{-5}A_2/g$ for liquids. For tritium-contaminated solids or gases, the maximum allowable specific activity is 0.004 TBq/g (0.1 Ci/g, 100 Ci/kg). The resulting limit for tritium-contaminated liquid (other than tritiated water) is 4 x 10^{-4} TBq/g (0.01 Ci/g).
  - LSA-III is defined as solids such as consolidated wastes or activated materials that meet the testing requirements of 49 CFR 173.468 (water leach test) and the following requirements:

The Class 7 (radioactive) material is essentially uniformly distributed throughout a solid or a collection of solid objects or essentially uniformly distributed in a solid compact binding agent (such as concrete, bitumen, or ceramic).

The Class 7 (radioactive) material is relatively insoluble, or is intrinsically contained in a relatively insoluble material, so that, even under loss of packaging, the loss of Class 7 (radioactive) material per package by leaching when placed in water for seven days would not exceed 0.1 A_2. The resulting limit for tritium is 4 TBq (110 Ci).

The average specific activity of the solid does not exceed 2 x $10^{-3}A_2/g$. The resulting limit for tritium is 0.08 TBq/g (2.16 Ci/g).

The advantage to transporting material as LSA is that, for domestic transportation of an exclusive use shipment that is less than an A_2 quantity, bulk packaging's may be used. Also these shipments are accepted from the marking and labeling requirements of 49 CFR Part 173, Subpart I. However, the exterior of each package of radioactive materials must be stenciled or otherwise marked "RADIOACTIVE—LSA."

- SCO is similar to LSA but is limited to a solid object that is not itself radioactive but has Class 7 (radioactive) material distributed on any of its surfaces. SCO is divided into two groups, SCO-I and SCO-II. The definitions are provided below.

— SCO-I is defined as a solid object that meets all of the following requirements:

- The non-fixed contamination on the accessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 4 Bq/cm^2 (10^4 $\mu Ci/cm^2$) for beta emitters such as tritium.
- The fixed contamination on the accessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 4 x 10^4 Bq/cm^2 (1 $\mu Ci/cm^2$) for beta emitters such as tritium.
- The non-fixed plus the fixed contamination on the inaccessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 4 x 10^4 Bq/cm^2 (1 $\mu Ci/cm^2$) for beta emitters such as tritium.

— SCO-II is defined as a solid object on which the limits for SCO-I are exceeded and meets all of the following requirements:

- The non-fixed contamination on the accessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 400 Bq/cm^2 (10^{-2} $\mu Ci/cm^2$) for beta emitters such as tritium.
- The fixed contamination on the accessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 8 x 10^5 Bq/cm^2 (20 $\mu Ci/cm^2$) for beta emitters such as tritium.
- The non-fixed plus the fixed contamination on the inaccessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 8 x 10^5 Bq/cm^2 (20 $\mu Ci/cm^2$) for beta emitters such as tritium.

Also, as with bulk LSA, for domestic transportation of an exclusive use shipment of SCO that is less than an A_2 quantity, bulk packaging may be used. And these shipments are excepted from the marking and labeling requirements of 49 CFR Part 173, Subpart I. However, the exterior of each package must be stenciled or otherwise marked "RADIOACTIVE—SCO."

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 and • Activity limit for exempt consignment of 10^{-3} TBq (0.027 Ci)	Both conditions must be satisfied.
Solid	Limited quantity	0.04 TBq (1.1 Ci)	
	LSA-I	4×10^{-5} TBq/g (0.001Ci/g)	
	LSA-II	0.004 TBq/g (0.1 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 (10^{-4} μCi/cm^2). The maximum fixed contamination on accessible surfaces is 4×10^4 Bq/cm^2 (1 μCi/cm^2). The total surface contamination on the inaccessible surfaces is limited to 10^4 Bq/cm^2 (1 μCi/cm^2). The conveyance limit* is 4,000 TBq (1.08×10^5 Ci)
	SCO-II	Limit based on surface contamination	The maximum nonfixed contamination on accessible surfaces is 400 Bq/cm^2 (10^{-2} μCi/cm^2). The maximum fixed contamination on accessible surfaces is 8×10^5 Bq/cm^2 (20 μCi/cm^2). The total surface contamination on the inaccessible surfaces is limited to 8×10^5 Bq/cm^2 (20 μ Ci/cm^2). The conveyance limit* is 4,000 TBq (1.08×10^5 Ci)
	Type A	40 TBq (1080 Ci)	
	Type B	Limited by CoC for the package	The Type B UC-609 package is limited to 100 g of tritium; The BTSP package is limited to 150 g of tritium
Liquid	Limited quantity (tritiated water)	37 TBq (1000 Ci) 3.7 TBq (100 Ci) 0.037 TBq (1 Ci)	< 0.0037 TBq/L (0.1 Ci/L)0.0037 to 0.037 TBq/L(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 (0.01 Ci/g)	The conveyance limit* is 4,000 TBq (1.1×10^5 Ci)
	Type A	40 TBq(1080 Ci)	

	Type B	Limited by CoC for the package	The Type B UC-609 package is limited to 100 g of tritium; The BTSP package is limited to 150 g of tritium
Gas	Limited Quantity	0.8 TBq (21.6 Ci)	
	LSA-II	0.004 TBq/g (0.1 Ci/g	The conveyance limit* is 4,000 TBq (1.1 x 10^5 Ci)
	Type A	40 TBq (1080 Ci)	
	Type B	Limited by CoC for the package	The Type B UC-609 package is limited to 100 g of tritium; The BTSP package is limited to 150 g of tritium

6) Table 7-1: Allowable quantities of tritium per 49 CFR Part 173

Note that the 10/01/05 edition of 49 CFR Part 173 changed the A_2 quantity for tritium from 1,080 to ~1,100 Ci.

*The conveyance limit is the limit per trailer if highway transportation is used.

- Type A packages are limited to an A_2 quantity of radioactivity. The A_2 limit for tritium is 40 TBq (1,080 Ci) per package. The physical form of the package contents must be within the envelope of the contents used during testing of the package. Type A packages are significantly less expensive and much easier to obtain than Type B packages.
- Type B packages provide the highest level of safety features of all types of radioactive material packaging. The contents are limited by the requirements of the Certificate of Compliance (CoC) for the package. The CoC is issued by the agency that certifies that the package meets the Type B requirements.

7.2.1 Form and Quantity of Tritium

The form of the tritium affects the quantity of tritium that may be shipped in one package and the design and testing requirements for the packaging. Table 7-1 is a comparison of the allowable quantities of tritium in various package types. Quantities based on A_1 values for "Special Form" certified packages.

7.2.2 Evaluation of Approved Packaging

Tritium-contaminated solid waste is typically packaged in DOT Type A, Specification 7A open head 55-gallon drums or Type A metal boxes. The maximum amount of tritium that may be shipped in a Type A package is 40 TBq (1080curies).

Packages containing over 40 TBq of tritium must meet the Type B container requirements. These containers are few in number, and, compared to Type A packages are more expensive to manufacture and use. If Type B packages are to be considered for shipments, the CoC is to be reviewed carefully against the form and quantity of tritium to be shipped.

The UC-609, BTSP, H1616 and TRUPACT-II are the currently approved U.S. Type B containers for tritium and tritium-containing materials. The UC-609 and BTSP are drum type packaging with a limited volume containment vessel. Their containment vessels are authorized for tritium gas, absorbed tritiated water vapor, or tritides and are limited to 100 and 150 grams of tritium, respectively The H1616 also a drum style packaging with a limited CV volume is authorized only for gas and tritide. The TRUPACT-II is authorized for fourteen DOT Type A, Specification 7A 55-gallon drums or two Standard Waste Boxes. The authorized waste forms for the TRUPACT-II are solids or solidified liquids. There may be difficulties associated with shipping molecular sieve in Type B containers associated with pressure buildup. LANL is developing a path forward for this concern.

Tritium waste that meets the definition of LSA or SCO may be shipped domestically in industrial packaging or DOT Type A, Specification 7A packaging.

7.2.3 Minimum Requirements for Packaging

The design requirements for a radioactive materials package are located in 49 CFR 173.410. These requirements include the following:

- The package can be easily handled and properly secured in or on a conveyance during transport.
- The materials of construction of the packaging and any components or structure will be physically and chemically compatible with each other and with the package contents. The behavior of the packaging and the package contents under irradiation will be taken into account. Irradiation of materials is not a concern for tritium; however, material compatibility, as discussed in Section 5, should be considered.
- All valves through which the package contents could escape will be protected against unauthorized operation.
- Each lifting attachment that is a part of the package must be designed with a minimum safety factor of five, based upon breaking strength, when used to lift the package in the intended manner, and it must be designed so that failure of any lifting attachment under excessive load would not impair the ability of the package to meet other requirements of 49 CFR Subpart I. Any other structural part of the package, which 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.
- The external surfaces, as far as practicable, will be free from protruding features and will be easily decontaminated.
- The outer layer of packaging will avoid, as far as practicable, pockets or crevices where water might collect.
- Each feature that is added to the package will not reduce the safety of the package.
- The package will be capable of withstanding the effects of any acceleration, vibration, or vibration resonance that may arise under normal conditions of transport without any

deterioration in the effectiveness of the closing devices on the various receptacles or in the integrity of the package as a whole and without releasing the nuts, bolts, or other securing devices even after repeated use.

For transport by air, the following requirements apply:

- The integrity of containment will not be impaired if the package is exposed to ambient temperatures ranging from -40°C (-40°F) to 55°C (131°F).
- The temperature of the accessible surfaces of the package will not exceed 50°C (122°F) at an ambient temperature of 38°C (100°F) with no account taken for insulation;
- The surface temperature of packages is limited to 38°C (100°F) for nonexclusive shipments and 50°C (122°F) for exclusive shipments. This requirement is not be an issue with tritium shipments based on the low decay heat for tritium.
- Packages containing liquids will be capable of withstanding, without leakage, an internal pressure that produces a pressure differential of not less than 95 kPa (13.8 lb./square inch).

The requirements for Type A packages include the ability to withstand the following tests specified in 49 CFR 173.465:

- Water spray–simulation of 2 inches of rainfall per hour for at least one hour.
- Free drop–a drop of 1 to 4 feet, depending on the package mass, onto a hard surface.
- Stacking–compressive load of five times the package mass for a minimum of 24 hours.
- Penetration–this test is performed by dropping a 6 kilogram, 3.2 centimeter steel bar onto the weakest point of the package from a minimum height of one meter.

Packages designed for liquids and gases are required to pass a more rigorous free drop and penetration test than solids.

7.2.4 Onsite versus Offsite Shipments

Offsite shipments of radioactive material must meet the requirements of the DOT’s HMR and the relevant requirements for the mode of transport, unless a special permit has been granted by the DOT or an exemption has been granted under a DOE Order by DOE, depending on who is regulating the shipment. (See Section 7.1 for additional information.)

Onsite transfers of radioactive material must meet the requirements of the HMR and the relevant requirements for the mode of transport, or alternatively must comply with an approved DOE site- or facility-specific Transportation Safety Document that describes the methodology and compliance process to meet equivalent safety for any deviation from the federal regulatory requirements, unless an exemption has been granted by DOE.

Onsite transfers typically are not considered to be in commerce and are not subject to DOT jurisdiction, even when conducted by a contractor, so long as certain conditions are met, as described in 49 CFR 171.1(d)(4). In order for a shipment to be considered onsite and not in commerce, the rail or motor vehicle movements of a hazardous material must be exclusively within a contiguous (i.e., unbroken, continuous) facility boundary where public access is restricted. To the extent that the movement is within a contiguous facility but is on or crosses a public road or is on track that is part of the general railroad system of transportation, access to the public road must be restricted by signals, lights, gates, guards, or similar controls.

DOE transportation policy and requirements are set forth in DOE Order 460.1C, *Packaging and Transportation Safety*, Order 460.2A, *Departmental Materials Transportation and Packaging Management*, Manual 460.2-1A, *Radioactive Material Transportation Practices Manual*, Order 461.1B, *Packaging and Transportation for Offsite Shipment of Materials of National Security Interest*, and Order 461.2, *Onsite Packaging and Transfer of Materials of National Security Interest* (or their successor documents).

7.3 Package Loading and Preparation for Shipment

7.3.1 Disassembly and Inspection of the Package

DOE O 460.2A requires that DOE contractors inspect incoming hazardous materials shipments for damage or loss and evidence of leakage. Radioactive material shipments shall be inspected for external surface contamination and dose rate.

Additionally, pursuant to 49 CFR 173.474, prior to the first use of any radioactive materials packaging, the offerer must determine the following:

- The packaging meets the quality of design and construction requirements as specified in 49 CFR Part 173, Subpart I.
- The effectiveness of the shielding, containment, and, when required, the heat transfer characteristics of the package, are within the limits specified for the package design.

Additionally, for Type B packages, pursuant to 10 CFR 71.85, the following preliminary determinations prior to first use of any packaging are required:

- Ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce the effectiveness of the packaging.
- Where the maximum normal operating pressure will exceed 35 kPa (5 lb/in^2) gauge, the package user shall test the containment system at an internal pressure at least 50 percent higher than

the maximum normal operating pressure to verify the capability of that system to maintain its structural integrity at that pressure.

Finally, adherence to requirements of 10 CFR 835.405, "Receipt of Packages Containing Radioactive Material*",* must be observed upon receipt.

7.3.2 Package Loading and Assembly Operations

49 CFR 173.475 requires routine determinations during the loading of a radioactive materials package. Before each shipment of any Class 7 (radioactive) material, the offerer must ensure, by examination or appropriate tests, the following:

- The packaging is proper for the contents to be shipped.
- The packaging is in unimpaired physical condition, except for superficial marks.
- Each closure device of the packaging, including any required gasket, is properly installed, secured, and free of defects.
- Each special instruction for filling, closing, and preparation of the packaging for shipment has been followed.
- Each closure, valve, or other opening of the containment system through which radioactive material might escape is properly closed and sealed.
- The internal pressure of the containment system will not exceed the design pressure during transportation.
- For fissile material, each moderator and neutron absorber, if required, is present and in proper condition

Additionally, for Type B packagings,

- Any system for containing liquid is adequately sealed and has adequate space or other specified provision for expansion of the liquid.
- Any pressure relief device is operable and set in accordance with written procedures.
- 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 (see Section 7.2.3).
- Accessible package surface temperatures will not exceed the limits in 10 CFR 71.43 at any time during transportation. (See Section 7.2.3.)

7.3.3 Leak Testing

Leak testing (normally a helium leak test) is required for radioactive materials shipments in Type B packages for meeting the containment criteria for Type B packages in Title 10, Section 71.51, "Additional Requirements for Type B Packages," of the Code of Federal Regulations (10 CFR 71.51).

The regulations at 10 CFR 71.51 require licensees to ensure that Type B packages, following tests for normal conditions of transport and hypothetical accident conditions, meet the containment criteria to minimize radioactive contamination and dose rates to the public. The basis of leak testing is found in Regulatory Guide 7.4, Leakaage Tests on Packages for Shipment of Radioactive Material. This Guide describes an approach that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable. DOE uses conformance with this Reg Guide for satisfying the leak tightness in transport packages.

The package must be leak-tight (1×10^{-7} reference cm^3/s with a test sensitivity of $\leq 5 \times 10^{-8}$ reference cm^3/s) as defined by American National Standards Institute (ANSI) N14.5. There is no longer a requirement for a total integrated leakage test on Type B package containment vessels provided that the non-removable parts of the contaminated vessel, i.e., the seals, have not been modified or disturbed per the current version of the ANSI standard. Assembly verification and maintenance verification leak testing requirements will be noted in the CoC for the Type B package. Packages that contain liquid in excess of the A_2 quantity for tritium (40 TBq or 1,080 Ci) and intended for air shipment must be tested to show that they will not leak under an ambient atmospheric pressure of not more than 25 kPa, absolute (3.6 psia). The test must be conducted to determine compliance with this requirement on the entire containment system or on any receptacle or vessel within the containment system, per 10 CFR Part 71.

7.3.4 Preparation for Shipment

For each shipment of more than 40 TBq (1,080 Ci) of tritium, DOE O 460.2A requires the contractor to notify the shipment consignee of the dates of the shipment, the expected date of arrival, and any special loading or unloading instructions. Placarding for tritium shipments is required if the packages are labeled with Radioactive Yellow III labels or for exclusive use shipments of low specific activity tritium and surface contaminated objects with tritium transported in accordance with 49 CFR 173.427(b)(4) and (5) or (c). Marking, labeling, placarding and shipping paper requirements from 49 CFR Part 172 need to be evaluated for each shipment of hazardous materials.

7.4 Transportation Requirements and Records

Records may be the original or a reproduced copy. A microform may be used provided that authorized personnel authenticate the copy. The record may also be stored in electronic media with the capability for producing legible, accurate, and complete records during the required retention period. Each record must be legible throughout the retention period specified by regulation. Records such as letters, drawings, and specifications must include all pertinent information such as stamps, initials, and signatures. DOE O 414.1C and NQA-1 contain more detail on records and documentation.

Title 10 CFR 830.122(d) requires that "documents be prepared, reviewed, approved, issued, used and revised to prescribe processes, specify requirements, or establish design. Records must be specified, prepared, reviewed, approved and maintained."

7.4.1 Transportation Requirements for Packaging

Each shipper who transports licensed material outside of the confines of its plant or other place of use, or who delivers licensed material to a carrier for transport, shall comply with the applicable requirements of the regulations appropriate to the mode of transport of DOT in 49 CFR Parts 170 through 189.

- The shipper should particularly note DOT regulations in the following areas:
  - Packaging - 49 CFR Part 173, Subparts A and B and paragraphs 173.401 through 173.478.
  - Marking and labeling - 49 CFR Part 172, Subpart D and paragraphs 172.400 through 172.407; 172.436 through 172.440.
  - Placarding - 49 CFR 172.500 through 172.519, 172.556 and Appendices B and C.
  - Monitoring - 49 CFR Part 172, Subpart C.
  - Accident reporting - 49 CFR 171.15 and 171.16.
  - Shipping papers - 49 CFR Part 172, Subpart C.

- The shipper should also note DOT regulations pertaining to the following modes of transportation:
  - Rail - 49 CFR Part 174, Subparts A-D and K.
  - Air - 49 CFR Part 176, Subparts A-D and M.
  - Vessel - 49 CFR Part 176, Subparts A-D and M.
  - Public Highway - 49 CFR Part 177.

Additionally, 49 CFR 173.403 defines unilateral and multilateral approval for Type B packaging as

- Type B (U) package means a Type B packaging that, together with its radioactive contents, for international shipments requires unilateral approval only of the package design and of any stowage provisions that may be necessary for heat dissipation.

- Type B (M) packaging means a Type B packaging, together with its radioactive contents, that for international shipments requires multilateral approval of the package design, and may require approval of the conditions of shipment. Type B (M) packages are those Type B package designs which have a maximum normal operating pressure of more than 700 kilopascals per square centimeter (100 psi gauge) or a relief device which would allow the release of Class 7 (radioactive) material to the environment under the hypothetical accident conditions specified in 10 CFR Part 71.

Furthermore, 49 CFR 173.473 states that each shipper of a foreign-made Type B, Type B (U), Type B (M), or fissile material package for which a competent authority certificate is required by the IAEA *Regulations for the Safe Transport of Radioactive Materials, Safety Series No. 6* should also comply with the requirements in 49 CFR 173.473.

7.5 Quality Assurance/Control Requirements

7.5.1 DOT Quality Control Requirements

Quality control requirements for packaging and transportation of radioactive material are defined in 49 CFR Part 173.

49 CFR 173.474 – Quality control for construction of packaging

- Prior to the first use of any packaging for the shipment of radioactive material, the shipper shall determine that
  - The packaging meets the quality of design and construction requirements as specified in this subchapter; and
  - The effectiveness of the shielding, containment, and, when required, the heat transfer characteristics of the package, are within the limits specified for the package design.

49 CFR 173.475 – Quality control requirements prior to each shipment of radioactive materials.

- Before each shipment of any radioactive materials package, the shipper shall ensure by examination or appropriate tests, that
  - The package is proper for the contents to be shipped;
  - The package is in unimpaired physical condition, except for superficial marks;
  - Each closure device of the packaging, including any required gasket, is properly installed, secured, and free of defects;
  - Each special instruction for filling, closing and preparation of the packaging for shipment has been followed;
  - Each closure, valve, or other opening of the containment system through which the radioactive content might escape is properly closed and sealed;
  - Each packaging containing liquid in excess of an A_2 quantity and intended for air shipment has been tested to show that it will not leak under an ambient atmospheric pressure of not more than 0.25 atmosphere, absolute, (0.25 kilograms per square centimeter or 3.6 psia). The test must be conducted on the entire containment system, or on any receptacle or vessel within the containment system, to determine compliance with this requirement.

- The internal pressure of the containment system will not exceed the design pressure during transportation; and
- External radiation and contamination levels are within the allowable limits specified in this subchapter.

7.5.2 Other Quality Assurance Requirements
Both DOE (DOE O 460.1C which is currently in revision) and NRC (10 CFR Part 71, Subpart H) address other aspects of QA. Furthermore, many DOE sites invoke a specific version of NQA-1 in their contracts.

8.0 TRITIUM WASTE MANAGEMENT

The regulations governing waste are complex. It is important to understand the basic concept of "waste" and the difference between waste, low-level (radioactive) waste, sanitary land fill waste, reusable equipment, usable materials, and scrap. The following provides a guideline for defining waste. Note that tritium-containing materials, whether or not they are defined as waste, must be managed in accordance with applicable requirements, as discussed in Chapter 3.0.

- All materials, whether inside or outside a radiological or nuclear facility, which are of negligible or no economic value, considering the cost of recovery, are waste.

- Low-level radioactive waste is radioactive waste that is not high-level radioactive waste, spent nuclear fuel, transuranic waste, byproduct material (as defined in section 11e.(2) of the AEA, as amended), or naturally occurring radioactive material, and that has residual radioactive material concentrations above the approved DOE Authorized Limit defined in DOE O 458.1, Admin Chg 3.

- Sanitary landfill waste, which has no value and is below the unrestricted release limit, is waste and may be disposed of in a DOE sanitary landfill. Disposal of this waste at a non-DOE sanitary landfill may be done after notifying the appropriate state agency of the DOE authorized limits for this activity.

- Reusable equipment, such as computers, tools, electrical instruments, and drill presses, which can be used if released without restriction from the facility and has value, is not waste. If the equipment is contaminated above the unrestricted release limit and can be transferred and used at another radiological or nuclear facility, it is not waste.

- Usable materials, such as nails, bolts, nuts, sheetrock, plywood, plastic sheeting, barrels, decorative rock, sand, gravel, brick, and unused chemicals, which can be used if released without restriction from the facility and have value, are not waste. If the material is contaminated above the unrestricted release limit and can be transferred and used at another radiological or nuclear facility, it is not waste.

- Scrap, such as scrap nonprecious metal, scrap precious metal, scrap wood, and scrap chemicals, which can be sold as scrap if released without restriction from the facility and has value, is not waste.

8.1 Approved Limits for the Release of Contaminated Materials and Property Containing Residual Radioactive

For release and disposal, materials that exhibit radioactive surface contamination less than the values in the revised Figure IV-1 of DOE Order 5400.5 Chg. 2 (archived) fall into three categories.

- Category 1 Disposal in a DOE or non-DOE sanitary landfill

- Category 2 Disposal and/or treatment in a DOE or non-DOE CERCLA or RCRA treatment, storage, and disposal facility

- Category 3 Transfer of ownership (either by sale or other means) to members of the public

The Category 1 materials may be released or disposed of without further regard to residual radioactivity if they are at or below the pre-approved values referred to in DOE O 458.1. Section 4.k.(6)(f)1 b of DOE O 458.1 states that: "Previously approved guidelines and limits (such as the surface activity guidelines) may continue to be applied and used as Pre-Approved Authorized Limits until they are replaced or revised by Pre-Approved Authorized Limits issued under this Order." These can be found in the revised Figure IV-1 of archived DOE Order 5400.5 Chg 2 and DOE G 441-1C, *Radiation Protection Programs Guide for Use with Title 10, Code of Federal Regulations, Part 835,* Occupational Radiation Protection. Site policies and procedures should clearly describe the use of these values and how other requirements of the Order (such as ALARA, survey measurement recordkeeping, and compliance with the specific requirements of the disposal facility) are met. Disposal at a non-DOE sanitary landfill must meet the same requirements as waste disposed at a DOE sanitary landfill, and the appropriate state agency must be notified of the DOE-authorized limits for this activity.

Category 3 materials must meet similar requirements as Category 1 material, and the appropriate coordination with the state agency and/or NRC Region must be carried out.

8.1.1 Release Limit Requirements for Surface-Contaminated Material

There is one directive and one regulation that cover the control limits and methods of removable surface contamination measurement: 10 CFR Part 835, *Occupational radiation protection*, and DOE O 458.1. Additional guidance is contained in a DOE memorandum from A. Lawrence to

Distribution, dated May 1, 2002, which includes the draft DOE G 441.1 as an enclosure. The values specified for tritium are consistent with 10 CFR Part 835.

Appendix D to 10 CFR Part 835 specifies surface contamination values for use in determining whether a location needs to be posted as a contamination or high-contamination area (Subpart G), and if an item is considered to be contaminated and cannot be released from a contamination or high-contamination area to a controlled area (Subpart L). However, such items are excluded from the requirements of 10 CFR Part 835 if they have been documented to comply with the criteria for release established in a DOE authorized limit (10 CFR 835.1(b)(6)). 10 CFR Part 835 does not permit unrestricted release of contaminated items. The surface contamination value for removable tritium is 10,000 dpm/100 cm^2. In addition, Footnote 6 to Appendix D provides a requirement to consider the migration of tritium from the interior of an item to the surface when applying the surface contamination value for tritium. Per Appendix E of this Standard, "Radiological Control Programs for Special Tritium Compounds" is considered to apply to special tritium compounds, including metal tritides.

DOE O 458.1 concerns the clearance and release of materials. This document provides requirements for unrestricted release of objects with residual radioactivity. These requirements are further described in the document *Response to Questions and Clarification of Requirements and Processes: DOE 5400.5 Chg. 2, Section II.5 and Chapter IV Implementation (Requirements Relating to Residual Radioactive Materials)*[77]. This document recommends the use of 10,000 dpm/100 cm^2 as an interim guideline for removable tritium. This limit is also specified in Table 1, "Surface Activity Guidelines" (see Appendix D). This value is considered acceptable for special tritium compounds, including metal tritides.

ANSI/HPS Standard N13.12-2013, *Surface and Volume Radioactivity Standards for Clearance*, gives the screening level for unrestricted used items with residual levels of tritium on the surface of 600,000 dpm/100 cm^2, of which no more than 20% or 120,000 dpm/100 cm^2 should be removable (volume screening value of 3,000 pCi/g). These values are currently not endorsed by DOE.

Although not specifically addressed in the regulations, users are cautioned that, in some cases, the initial, relatively clean concrete surface measurements did not accurately characterize the bulk tritium contents.

8.1.2 Removable Surface Contamination Measurement Process

The regulatory requirements are not specific as to how the removable surface contamination wipe is to be done or whether or not it is to be wet or dry. Footnote 4 of Appendix D to 10 CFR Part 835

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[77] "Response to Questions and Clarification of Requirements and Processes: DOE 5400.5 Chg. 2, Section II.5 and Chapter IV Implementation (Requirements Relating to Residual Radioactive Material)," DOE Office of the Assistant Secretary for Environmental Safety and Health, Office of Environment, November 17, 1995

states that a dry wipe may not be appropriate for tritium, but does not provide an explanation as to what may be appropriate. As a result, it would seem that either a wet or dry wipe can be used and still meet the requirements of 10 CFR Part 835. The requirements stated in *Response to Questions and Clarification of Requirements and Processes…* include: "The measurement should be conducted by a standard smear measurement but using a wet swipe or piece of Styrofoam," although in Table 1 of this document, a dry wipe (i.e., a dry filter or soft absorbent paper) is specified. The "should" included in this sentence and the use of Styrofoam™ would seem to allow for the use of either dry or wet wipes.

The use of a wet or dry wipe to determine removable surface contamination levels is noteworthy due to possible differences in measured levels between the two techniques.

Appendix E of this Standard *Radiological Control Programs for Special Tritium Compounds*, provides guidance on monitoring surfaces for the presence of special tritium compounds.

The measurement of removable surface contamination has not been standardized by the technical disciplines, and, as a result, it is difficult to compare readings between facilities. Reasons for variations between facilities may be due to the following:

- Instrumentation: The instruments used for counting wipes are of different manufacturers and models providing varying accuracy and precision. The methodologies used are not standardized or cross-checked against each other to determine if results are similar.

- Units: The measurement units used at the different sites are either dpm/100 cm^2 or dpm/cm^2 with the former being the more common. Comparing results from one facility to another requires careful checking to make sure that the dpm values discussed are for the same area of surface.

- Wipe Material: The wipe materials most commonly used are dry filter paper, Styrofoam™, and Q-Tips®. There is no standardization of materials across facilities.

- Wipe Preparation: Dry wipes and wet wipes are used at the facilities. The wet wipes are prepared by soaking the Q-Tip® or the filter paper in water, counting solution, or other liquids.

- Wipe Technique: The technique used to determine the area wiped and pressure applied when using the different materials varies from site to site.

- Dry vs. Wet Wipes: Recent comparative tests performed at the LLNL Tritium Facility show that there are differences in the results obtained between a dry and a wet wipe of the same surface. In some cases, the dry wipe resulted in measured values higher than those measured by a wet wipe, while in other cases the wet wipe was higher. These tests, using de-ionized water as the wetting agent, indicate that the difference between a dry or wet wipe is a function of the type of surface and the level of contamination. Generally, the levels measured by wet and dry wipes of the same surface are within a factor of 3.

Good practices in this area for consideration include:

- When the surface of the object is less than 100 cm^2, the activity per 100 cm^2 should be calculated based on the area wiped, and the entire object should be wiped.

- Records should be kept in units of dpm/100cm^2.

- Either dry or wet wipes can be used to satisfy regulatory requirements. The requirements do not identify a wetting agent, nor do they specifically require a wet wipe. Some facilities use wet wipes in the belief that this is a more conservative approach. In fact, depending on the specific surface conditions, either method may produce somewhat higher measured values. Given the high intrinsic variability of the wet swipe process and the fact that the dry method is less complex and can be standardized and performed more consistently, the dry process is the preferred method of this Standard.

- The methods used to measure removable surface contamination are not presently standardized and are not the same at all facilities. The following discussion describes a Generic Wipe Survey Technique that could be used at any DOE facility.

  - The amount of removable material should be determined by wiping an area of 100 cm^2 on the object using dry filter paper or a dry Q-Tip®, applying moderate pressure, and measuring the amount of radioactive material on the dry wipe.

  - Dry Filter Paper: The wiping technique, using dry filter paper, should consist of wiping a total path length of 16 inches using a single 16-inch lazy-S path or multiple shorter lazy-S paths that total 16 inches (tests show an average wipe width of 1 inch for dry filter paper).

  - Q-Tip®: The wiping techniques, using a dry Q-Tip®, should consist of wiping a total path length of 80 lineal inches of Q-Tip® path (tests show an average wipe width of 0.2 inches for a Q-Tip®).

- The Generic Wipe Survey Technique can be used for most applications; however, items with hidden surfaces or inaccessible areas require special consideration. For items that possess inaccessible areas that do not normally contact tritium, use of the Generic Wipe Survey Technique on the accessible surface area is sufficient in that these areas become contaminated only as a result of contamination on the surface area. However, for inaccessible areas such as pipes, tubes, drains, ductwork, valves, pumps, vessels, or transducers that potentially contact tritium, a two-step process is recommended:

- First Wipe Survey: The accessible surfaces of the item should be wipe-surveyed using the Generic Wipe Survey Technique.
- Second Wipe Survey: A second wipe survey should be made of the entry points into the item inaccessible areas (such as fittings, valve throats, ends of tubes, cracks, doors, and louvers). The second wipe survey should be made with a Q-Tip® inserted as far as reasonably possible into the openings of the items. The total area of wipe should be estimated and the results corrected to dpm per 100 cm^2.

8.1.3 Environmental Discharge Requirements

DOE O 458.1 Admin Chg. 3 makes use of a "best available technology" selection process to reduce effluent discharges; however, this process is not applicable to tritium. DOE O 458.1 Admin Chg. 3 relies on the ALARA philosophy in tritium operations to reduce effluent levels. The NRC release limit for 10 CFR Part 20, is 1.0 E-3 μCi/ml.

For airborne effluents, the annual discharge to the air (as stated in 40 CFR Part 61 Subpart H, *National Emission Standards for Emissions of Radionuclides Other Than Radon From Department of Energy Facilities*) for all radionuclides on site must be such that the offsite boundary dose is less than 10 mrem. Therefore, stacking and evaporation are methods that could be considered if this contribution and all others result in a dose under 10 mrem. Agreements with local regulators, however, may preclude or significantly reduce the release values allowed by these regulations, as could ALARA concerns.

All discharges to the environment that may contain tritiated wastes should be provided with monitoring systems. Future designs will likely incorporate real-time attributes to annunciate when the discharge concentration exceeds the specified limits. Soil column discharge is prohibited under DOE O 458.1.

The type and location of air monitoring stations are usually selected based on a number of factors, including cost. A passive environmental sampling system based on strategically located mole sieve (Zeolite 13X) samplers has been deployed in France. The tritium in the samplers is processed via desorption (heating in inert atmosphere, cryogenization of released vapors, and recovery of condensed water vapor) in about four hours and is then analyzed by liquid scintillation[78].

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[78] Pedro Caldeira Ideias, *Development of an Atmospheric Tritium Passive Sampler for Environmental Monitoring*. Presented at the Tritium 2013 Conference, Nice, France.

8.1.4 Tritium-Contaminated Wastewater

The generation of tritium-contaminated water is an inherent part of any tritium handling operation. Any water used in a facility that handles tritium has the potential of being contaminated. To ensure that wastewater exiting such a facility does not exceed regulatory limits, the facility should measure the tritium content of the wastewater generated and control its disposal.

8.1.4.a Tritium-Contaminated Wastewater Generation

The wastewater from a tritium facility falls into three categories:

- Sinks and Floor Drains: The least contaminated water is the water that runs into the facility sinks and floor drains. Unless water use is controlled, the volume of this water is typically in the tens of thousands of gallons per year. This water can generally be released to the sanitary sewer after the tritium content is measured to ensure that it is within regulatory limits.

- Wash and Decontamination Water: In a tritium handling area, contaminated wastewater is generated by routine activities such as mopping floors and decontamination of tools. Unless water use is controlled, the volume of water generated by these activities is typically in the hundreds to thousands of gallons per year. The tritium content of water from these activities is low, but is above background and is easily measured. Control and disposal of this low-level tritium-contaminated water is not industry standardized and, at present, is a function of the individual facility.

- Tritium Removal System Wastewater: Usually, the water collected by a facility Tritium Removal System will be the highest tritium concentration water collected at the facility. The volume of water generated by the tritium removal system is typically in the tens to hundreds of liters per year.

There are several potential methods for collection of the removal system wastewater. If possible, the system should be designed to collect water directly on a removable molecular sieve trap in less than 1,000 Ci quantities per trap. These traps can be disposed of as solid, low-level radioactive waste without further tritium handling if they are taken out of service (e.g., the date based on operational estimates of loading) prior to reaching 1,080 Ci. The molecular sieve trap can be valved off, certified to contain no pressure exceeding 1.5 atmospheres absolute at 20°C and no free liquids, and can be disconnected from the apparatus and placed directly into a DOT 7A package for shipment to the disposal site.

Type B quantities (1,080 curies/trap) may also be disposed of following this same technique except for overpacking the package in a Type B container for shipment to the disposal site. Due to radiolysis of the water, the increase of hydrogen and oxygen pressure in the shipping package has to be accounted for. Additionally, the pressure buildup in the container, due to decay of the tritium and radiolysis, has the potential to increase the pressure in the container above 1.5 atmospheres absolute.

The details of the packaging, stabilization of the waste item in the package, hydrogen generation in the package, and pressure buildup, and possible backfill of the void space with clay, should be explored with DOE waste site personnel before the first shipment is required. The DOE waste site criteria are subject to interpretation, and packaging details need to be discussed with DOE waste site personnel to make sure that the package meets their interpretation of the criteria.

8.1.4.b Tritium-Contaminated Waste Water Disposal

8.1.4.b (1) Solidification on Clay

Tritiated wastewater can be solidified on clay or Stergo superabsorbent (discussed in Section 8.1.4.b (2)), and disposed of as solid, low-level (radioactive) waste.

- Type A Quantities (< 40TBq)

  The waste site should be asked to approve solidification of Type A quantities of tritium in the form of water solidified directly on clay in DOT 7A, 55-gallon drums. Following the waste site guidance, which typically requires that 100 percent more clay be used to solidify the water than required, the following packaging method should meet the criteria.

  Clay will hold from 60 to 70 percent water by volume without any free liquid. The actual quantity that can be held is dependent upon the type of clay. Superfine® clay will hold 73 percent water by volume without any free liquid. Floroco® clay will hold approximately 65 percent by volume. Based upon this, a DOT 7A, 55-gallon drum filled with Superfine® clay will hold approximately 40 gallons of water. Following the typical guidance, which requires that 100 percent more adsorbent be used than required, a 55-gallon drum could be used to solidify 20 gallons of water.

  Other methods and techniques of wastewater solidification are available, such as clay-filled primary containers overpacked inside Type A containers. Mixtures of cement, clay, and other materials have been used with success. Unfortunately, most of the more complex solidification methods require the use of some type of aggressive mixing method, such as barrel mixers, which results in spreadable contamination problems and the resulting increased personnel exposures.

Type B Quantities (> 40TBq)

> Type B quantities of tritium in the form of water may also be solidified and packaged at the generation site in Type A containers for the purpose of storage at the waste site. However, Type B quantity waste, packaged in Type A containers, must be placed in a Type B shipping package during transport and shipping to the waste site.

The Trupact II shipping package is approved for shipment of solidified, tritium-contaminated water but users are cautioned to ascertain whether the hydrogen gas generated by radiolysis of the contents is bounded by the Trupact II SARP. LANL is currently working with SRNL to define any SARP changes required for shipment of some LANL legacy packages that contain HTO on molecular sieve.

8.1.4.b (2) Solidification on Polymers

PPPL shipped tritium to Hanford for burial and performed studies of solidification in various polymers. Three polymers are approved for absorption of radioactive liquids for land burial at the Hanford site. In order for these polymers to absorb the liquids, and, therefore, exhibit the properties of a solid, they have to be used in accordance with EPA test method 9096, *Liquid Release Test*, so that they pass a pressure test of 20 psi (this is a specific Hanford burial requirement; for example, Nevada National Security Site (NNSS) does not require a 20 psi test, but instead relies on vibration testing to simulate transport conditions in accordance with NNSS waste acceptance requirements. This testing is performed for each solidification medium and associated liquid). These polymers provide significant advantages over clay-based absorbents in that they do not require mixing, and that they exhibit less than 1% expansion by volume.

PPPL performed independent evaluations and testing of these polymers to verify the ratios and absorption capability. Stergo® (Corpex Technologies) passed the 20 psi test at a ratio of 20:1 water to polymer; additionally, Stergo® and SP-400 (Waterworks).passed the test at a pressure of 50 psi (the normal test pressure), using ratios of 10:1.

Users may want to perform their own additional independent testing to determine the correct ratio(s) for their product (if different from the aforementioned) and/or for their specific liquid characteristics. The PPPL testing was performed on tap water with a neutral pH.

8.1.4.b (3) Evaporation to the Environment

It is possible but expensive to solidify several thousand gallons of water containing only tens of curies of tritium. When properly permitted by State and Federal regulations, tritiated wastewater can be evaporated to the environment. An example of an evaporator, which was permitted by EPA, is the evaporator used at the now closed TRL facility at SNL. This tritium-contaminated wastewater evaporator was permitted to evaporate up to 100 Ci/yr to the environment. Over several years, this system evaporated tens of thousands of gallons of extremely low-level tritium-contaminated water.

Evaporation may not be a reasonable or feasible method of wastewater disposal at every site; however, it was the preferred disposal option at Three Mile Island. Over 2.2 million gallons of tritiated water were collected for storage and treatment at the TMI site after the 1979 accident. This waste water, with tritium concentrations ranging from $1.6E^{-1}$ μCi/mL to $2.4E^{-2}$ μCi/mL, resulted from many sources including the primary coolant, spent fuel pool, submerged demineralized system, and the EPICOR I and II ion-exchange processing systems. Lesser sources included wastewater from the decontamination of systems and components including steam generators and flushing of auxiliary systems. The principal constituents of this processed water were boric acid and tritium. The water was accumulated in large Process Water Storage Tanks (about 600,000 gallons) and processed through a vacuum evaporator system. The majority of the tritiated water was released to the environment. The boric acid and residual radioactive contamination was concentrated into a powder form and disposed of at a LLW disposal facility in Barnwell, SC. In the past, these low-level concentrations of tritiated water were diluted to meet the 10 CFR Part 20, Appendix B limit of $< 1.0E^{-3}$ μCi/mL and released. However, in this case, the estimated 1,000 Ci of tritiated water was released to the environment by evaporation. See Figure 2-3 for a comparison of the TMI accident-generated tritiated water concentrations with those found throughout the tritium complex.

8.1.4.b (4) Release to the Sanitary Sewer

Very-low-level tritium-contaminated water, which is generated by hand washing, showering, and normal facility water use, may be stored and analyzed to see that it meets state and Federal release criteria. If the water meets these regulations and the DOE requirements in DOE O 458.1 Section 4.g.(8), it may be released to the local sanitary sewer. For illustration, the NRC release limit for tritium, as defined in 10 CFR Part 20, Appendix B, Table 3, for release to sanitary sewers is 1.0 E-2 µCi/mL. This release limit is based on the monthly average (total quantity of tritium divided by the average monthly volume of water released to the sanitary sewer). If more than one radionuclide is released, the sum of the fractions rule applies. Also, DOE O 458.1, Section 4.g.(8)(a)4. specifies that releases do not result in an annual discharge (above background) into sanitary sewers in excess of 5 Ci (185 GBq) of tritium.

Regulations regarding the packaging for disposal and disposal of tritium-contaminated water are complex. The available choices for disposal are dependent on the tritium content of the water and the equipment available. Facility planning should include provisions for disposal of tritium-contaminated water at all levels of contamination down to a level that may be released to the local sanitary sewer system.

8.1.4b (5) Novel Approaches

Research into novel approaches continues to address the recovery of tritium from lightly contaminated tritiated water. Standard processes used in enrichment are prohibitively expensive with a large amount of slightly contaminated tritiated water, but, as environmental awareness continues to grow, new approaches need to be developed. Interesting research using mesoporous compounds has been conducted, and removal by Organic-Functionalized SBA-15 has been reported with promising results to date[79]. Additional research with multiple application areas has been pursued at SRNL. SRNL's work with porous-wall hollow glass microspheres[80], initially funded through DOE's Plant-Directed Research and Development Program, has earned a R&D 100 Award in Materials Science and has been licensed for production to the Mosci Company. In addition to storage, promising applications with tritium separation and purification are being investigated using palladium encapsulated in the spheres[81]. Currently, smaller scale processes including nanotemplating and medical applications have been identified for use[82]. Larger scale applications are yet to be pursued.

8.2 Waste Characterization

The purpose of hazardous waste characterization is to determine the applicability of the RCRA hazardous waste management requirements and to demonstrate compliance with these requirements (in accordance with 40 CFR Parts 261 through 268). Radioactive waste characterization must satisfy the requirements in DOE O 435.1 Chg 1. Mixed waste characterization must satisfy both. Waste characterization is the process used to determine the physical, chemical, and radiological properties of the LLW. As a general rule, if a hazardous waste component is reasonably expected to be present in a radioactive waste stream, it is advisable to manage the waste as mixed waste until the waste is characterized by chemical analysis or acceptable knowledge to document that the waste contains no hazardous wastes as identified in 40 CFR Part 261.

Waste characterization is the process of identifying, assessing, and documenting the physical, chemical, and radiological properties of process wastes and wastes generated from cleanup and removal activities. The waste profile created during characterization is the foundation for waste segregation, management, and certification operations. Characterization programs should be developed to address waste certification requirements dictated by the waste management, disposal, or treatment option selected for each waste stream, in addition to the applicable regulatory requirements. The primary characterization determinations to be considered for tritium-contaminated waste are the low-level and hazardous waste determinations.

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[79] Akira Taguchi, *Tritium Removal from Tritiated Water by Organic-Functionalized SBA-15*. Presented at the Tritium 2013 Conference, Nice, France.

[80] Heung, L. K., G. G. Wicks, and R. F. Schumacher. "Encapsulation of Palladium in Porous Wall Hollow Glass MIcrospheres." *Materials Innovations in an Emerging Hydrogen Economy: Ceramic Transactions* 78 (2009): 143.

[81] Wicks, G. G., L. K. Heung, and R. F. Schumacher. *Microspheres and microworlds*. *American Ceramic Society Bulletin* 87.6 (2008): 23.

[82] Li, Shuyi, et al. *Porous-wall hollow glass microspheres as novel potential nanocarriers for biomedical applications*. *Nanomedicine: Nanotechnology, Biology and Medicine* 6.1 (2010): 127-136.

Defining process waste streams is one of the most overlooked aspects of waste characterization. Many generators believe characterization involves the assessment of inventory containers. Effective waste management has to begin at the point of waste generation. The generator first identifies all of the waste-generating processes. Information such as process inputs, sub processes, equipment, and chemical use is reviewed to generate an adequate profile for each waste stream.

Documenting the waste stream assessments is also an important part of the characterization process. It is critical that the personnel generating waste be provided with the information required to properly manage waste generated in their area. Additionally, an auditable record is created that allows for review of the characterization process and determinations. Waste-generating processes are then reviewed periodically to identify changes that could result in the addition of new streams or changes to existing streams. In addition, the regulations should be monitored to address promulgated and proposed changes that affect current or future characterization programs.

RCRA, an amendment to the Solid Waste Disposal Act, was enacted in 1976 to address the management of municipal and industrial wastes. RCRA includes a number of subtitles. Subtitle C of RCRA directs the management of hazardous waste. This subtitle imposes administrative provisions to ensure accountability for hazardous waste management, as well as substantive requirements designed to protect human health and the environment from the effects of improper management of hazardous waste. Under the RCRA implementing regulations, hazardous waste is defined as any solid waste exhibiting one or more of the characteristics of hazardous waste: ignitability, corrosivity, reactivity, and metal or organic compound toxicity [40 CFR 261.20-261.24]; or is included on one of three lists of hazardous wastes [40 CFR 261.30-261.33]. These lists include hazardous wastes generated by nonspecific (F-listed wastes) and specific sources (K-listed wastes), in addition to commercial chemical products (P- and U-listed wastes). RCRA was amended significantly in 1984 by the Hazardous and Solid Waste Amendments (HSWA), which expanded the scope and requirements of RCRA. Among other things, HSWA required EPA to evaluate all listed and characteristic hazardous wastes, and to develop requirements (i.e., treatment standards) that must be achieved prior to land disposal of these wastes. The implementing regulations for accomplishing this statutory requirement are established within the Land Disposal Restrictions (LDR) program [40 CFR Part 268].

In conjunction with the hazardous waste and LDR-related waste determinations, additional physical and chemical information may be required depending on the treatment or disposal options being considered. These parameters may include, but are not limited to, the assessment of the waste for free liquids, headspace gases, chelating agents, anions/cations, PCBs, particulate, explosives, pyrophorics, and waste matrix composition.

8.2.1 Waste Knowledge

Waste knowledge or acceptable knowledge refers to information used to support waste characterization activities. In recent regulatory preambles and guidance, EPA uses the terms “waste knowledge” or “acceptable knowledge” in place of the term “process knowledge.” The terms "waste knowledge" or "acceptable knowledge" are broader terms that include process knowledge, waste analysis data obtained from generators, and existing records of analysis, or a combination of this information supplemented with chemical analysis. Process knowledge is the common terminology for the RCRA regulatory language "applying knowledge of the hazard characteristic of the waste in light of the materials or the processes used" [40 CFR 262.11(c)(2)]. EPA has defined process knowledge to include detailed information on the wastes obtained from existing published or documented waste analysis data or studies conducted on hazardous wastes generated by similar processes [see *Joint NRC/EPA Guidance on Testing Requirements for Mixed Radioactive and Hazardous Waste*, November 20, 1997 (62 FR 62079); and *Waste Analysis at Facilities that Generate, Treat, Store, and Dispose of Hazardous Wastes: A Guidance Manual*, OSWER 9938.4-03, April 1994]. Process knowledge includes any documentation that describes or verifies a facility's history, mission, and operations, in addition to waste stream-specific information used to define generating process, matrix, contaminants, and physical properties. Waste-generating facilities should maintain and utilize waste knowledge to characterize waste streams (i.e., in lieu of sampling and analyzing the waste), whenever possible, to avoid unnecessary exposures to radioactivity and eliminate needless or redundant waste testing. This requires the generator to maintain auditable records that document the composition of the waste, including waste matrix and contaminants.

The documentation should be constructed in such a way that independent review would result in consistent characterization conclusions. The collection, review, management, and dissemination of the information should be standardized and documented to ensure data integrity and that the information is defensible during future assessments.

The use of waste knowledge to characterize wastes may be applicable in a number of situations, including:

- Waste stream is difficult to sample because of the physical form.
- Sampling analysis would result in unacceptable risks of radiation exposure.
- Waste is too heterogeneous in composition.
- Waste sampling and analysis is not feasible or necessary.
- Waste stream results from well-documented specific processes, such as with standard laboratory operations.

8.2.2 Tritium Disposition Options

Tritium exists in various concentrations and forms throughout the complex. The tritium contained in waste streams could be stored, released to the environment (i.e., in compliance with applicable regulatory limits or permitted levels), or recovered for reuse. This section provides an overview of a process to help make an informed decision on tritium disposition.

The diagram depicted in Figure 8-1 provides a simplified flow path for ultimate disposition of tritiated material. The first fork is associated with the production source of the tritium. As discussed in Section 3.1.3, the RCRA regulations include an exclusion from the hazardous waste management requirements for source, special nuclear and byproduct material as defined by the AEA [40 CFR 261.4(a)(4)]. U.S. reactor and non-weapons production accelerator produced tritium wastes meet this definition of "byproduct" material, and, as such, are excluded from the RCRA requirements applicable to solid waste and hazardous waste. Additionally, EPA has applied a regulatory policy in certain cases [83] [84] [85] [86] [87] by which residuals, derived from the management of RCRA exempt or excluded waste, retain the exemption or exclusion (even if they subsequently exhibit hazardous characteristics). These considerations provide the basis for not including a hazardous waste determination step, relative to assessing the tritium itself, in the reactor-produced tritium fork presented in Figure 8-1.

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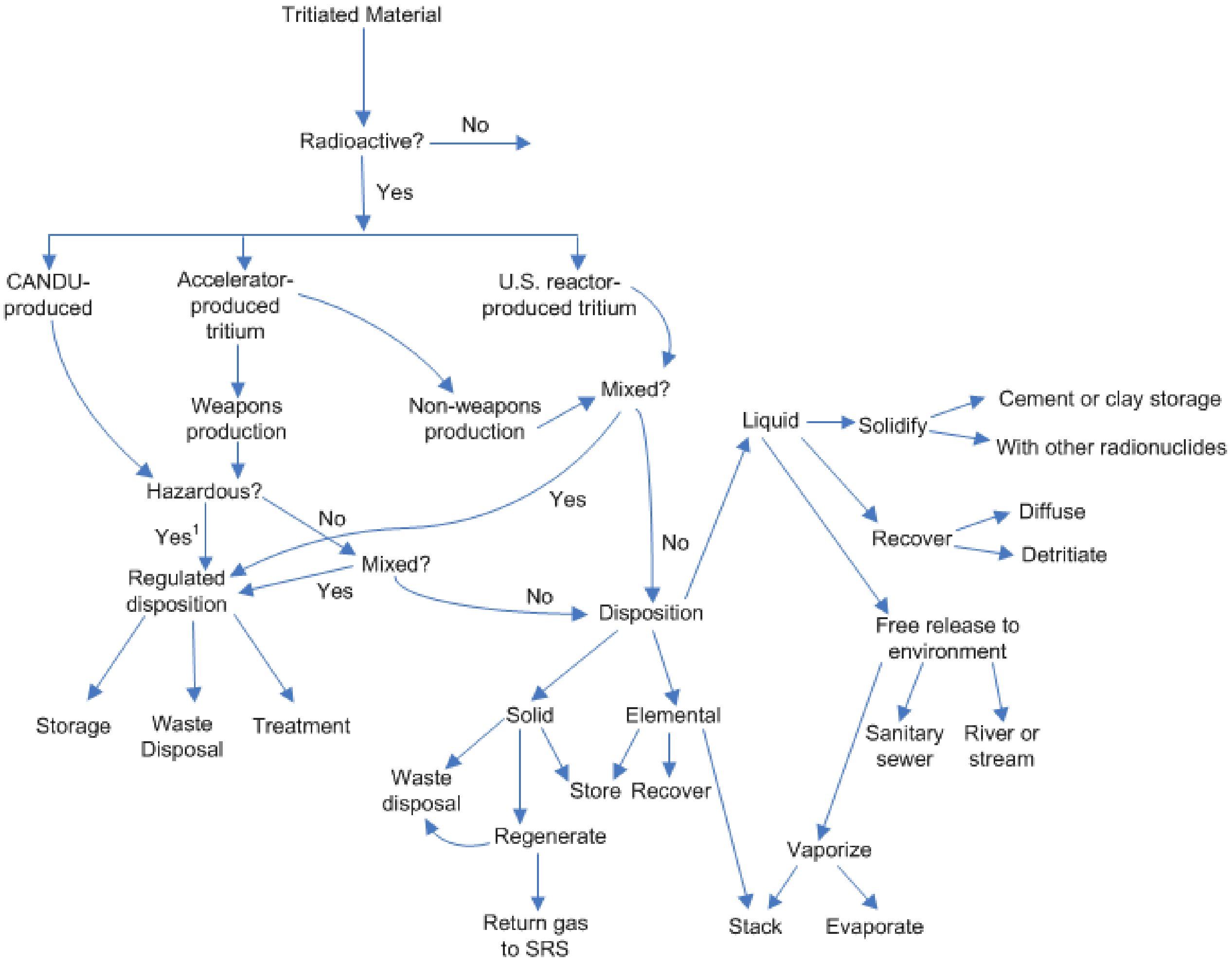
[83] Preamble of the First Third Land Disposal Restrictions Final Rule, 53 FR 31149, August 17, 1988.

[84] Memorandum, Environmental Protection Agency, Marcia E. Williams and Christina Kaneen to Robert L. Duprey, "Applicability of Bevill Amendment to the American Natural Gas Coal Gasification Facility," September 1987.

[85] Letter, Marcia E. Williams, EPA, to G. N. Weinreich, ANG Coal Gasification Co., June 16, 1986.

[86] Memorandum, Environmental Protection Agency, John H. Skinner to Harry Seraydarian, "Clarification of Mining Waste Exclusion," May 16, 1985.

[87] RCRA Superfund Hotline Report, February 1985.



16) Figure 8-1: Ultimate disposition of tritiated material

NOTE: Naturally-occurring and accelerator-produced radioactive material (NARM) mixed with hazardous waste is managed by DOE as mixed low level waste. In other words, the management of such waste is the same as for the relevant hazardous waste mixed with source, SNM, or byproduct material.

CANDU and accelerator-produced weapons production tritium waste do not qualify for the RCRA source, special nuclear, and byproduct material exclusion. As such, the tritium waste stream that exhibits one or more of the characteristics of hazardous waste (e.g., ignitability, corrosivity) would be subject to the RCRA hazardous waste provisions. For this reason, the left fork in Figure 8-1 includes a determination of whether the material (i.e., the tritium waste itself) is hazardous. While it is unlikely that forms of tritiated materials in the container configurations used in the DOE complex would exhibit any of the four characteristics of hazardous waste, it is still possible (e.g.,

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[1] This pathway appears unlikely to exist in DOE; however, final determinations are always subject to local regulatory authority and interpretation.

compressed tritium gas in a container configuration satisfies the characteristic of ignitability), and is therefore illustrated in Figure 8-1.

Both forks of the figure include a mixed waste determination. Specifically, this determination step considers whether the tritiated material also contains a RCRA hazardous waste component (i.e., in addition to the tritium/radioactive component). If a tritium waste stream also contains a hazardous waste component, the waste stream would need to be managed as a radioactive mixed waste in accordance with the AEA and RCRA. The determination that a mixed waste exists will result in the more stringent disposition option depicted. Note that even though naturally-occurring and accelerator-produced radioactive materials do not constitute a source, special nuclear or byproduct material, (except for accelerator produced tritium meeting the commercial, medical or research uses is now byproduct) all DOE waste containing naturally-occurring and accelerator-produced radioactive materials mixed with a hazardous component has to be managed as hazardous waste under RCRA [pursuant to DOE O 435.1 Chg. 1]. The disposition options for non-hazardous, non-mixed tritium from both forks are similar, with the biggest difference resulting from the storage requirements for RCRA solid material. These disposition options are a function of the form (solid, liquid, or gas).

8.2.3 Economic Discard Limit for Tritiated Water

EPA uses the economic discard limit (EDL) to evaluate whether a hazardous material or residue should be classified as a waste. If it has sufficiently high economic value, the material should be considered for recovery and reuse. The intent is not to allow facility (government-owned and commercially-owned) operations to define material as not being waste and thereby not be required to meet the current Federal statutes on permitting of hazardous waste material for treatment, storage, and disposal (TSD).

In a simplistic form, the EDL is the threshold value in determining when a material's economic value ($/unit weight or volume) exceeds the associated cost for the material's TSD. If the economic value, based on current market value or replacement value is determined to be of greater value than the cost of TSD, the material can be considered non-waste. The governing statues and requirements for non-waste are less extensive, and the disposition times are less severe.

There are many ways to determine the economic value of a material; the most common is market price. The current market price is, in large measure, established by Canadian commercial sales, and is approximately 3.00 $US/Ci base price Historically, this value has been fairly constant in U.S. dollar terms; for example, in 2008, it was $3.15 US/Ci ($3.50 CAD/Ci), and in 1999, it was $3.15 USD/Ci ($4.61 CAD/Ci) at 1 USD= 1.4653 CAD. The Canadian price though is just the base price and there are handling fees etc. that drive the price of smaller purchases to about $12.00 US. The quantity of tritium transacted also affects the price. In 2007, PNNL obtained quotes from three sources for a 1,000-Ci purchase. The Canadian price was 3.50 CAD/Ci for greater than Type A quantities, adding a 4,000-CAD fixed fee for any purchase less than Type A, resulting in a cost of 7.5 CAD/Ci. Private firm cost was quoted at 10 USD/Ci, and the DOE cost was “free” for the tritium, but the associated costs of labor, safeguards and security, and overhead costs factored in the resulting value of 15 USD/Ci[88]. These data suggest the current value of about 12 USD/Ci for smaller quantities of tritium as fairly accurate.

It appears that the value of the recovered tritium (i.e., from waste streams; high-Ci-content process streams are more economical) is overstated in many proposals to recover tritium, and that recovery options, even the most cost-effective such as the PMR, are difficult to justify on economics alone. Any recovery argument should also have an environmental impact component as well.

In the absence of a Department-wide policy on tritium recovery, DOE sites with tritium oxide and tritiated water inventories should consider their EDLs based on their site-specific conditions and needs. The implication of the preceding discussion is that once tritium goes into the oxide form, it is difficult to justify on an economic basis alone using any recovery method. The fact that economic recovery of tritium from oxide form is not easily justified should be a factor in the local DOE Site Office determination of waste classifications.

The NNSA Office of Nuclear Materials Integration has issued guidance[89] for implementing DOE O 410.2, *Nuclear Material Management*. The phase one version of the guidance does not include discard limits; phase two is developing discard limit guidance for uranium and plutonium materials, but is not planned to contain guidance on tritium disposition; however, the TFG and the point of contact for the guidance document will evaluate potential inclusion in the next revision of the document. The disposition process is described in the reference, starting with the “No Defined Use” (NDU) site declaration and the guidance includes a Nuclear Material Characterization Form that can be used to seek concurrence from program line management and ONMI for disposal of nuclear material.

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[88] Dave Baldwin, Pacific Northwest Laboratory, Personal Communication to Bill Weaver, 2008.

[89] NNSA Office of Nuclear Materials Integration, Nuclear Material Disposition Guidance Document, Phase II, April 2015.

8.3 Waste Packaging

DOE, DOT, and NRC requirements for radioactive waste packaging are presented in Chapter 7.

Barriers, in addition to the outer packaging, should be considered to inhibit tritium migration from waste packages. The only significant difference associated with packaging of waste for shipment versus packaging of any other quantity of nuclear material for shipment is that Type B quantities of waste may be packaged in Type A containers at the waste generation site and shipped to the waste site overpacked in Type B containers. During shipment, the waste package must meet the current DOE and DOT regulations for shipment of the form and quantity of radioactive material; i.e., Limited Quantity, Type A Quantity, Type B Quantity. After receipt at the waste disposal site, the Type A package containing the Type B quantity of low-level (radioactive) waste is removed from the Type B package. The Type B package is then returned to the shipper for reuse, and the Type A package containing the Type B quantity of tritium is stored at the waste disposal site along with the other low-level waste.

The waste acceptance requirements are a little different at each waste disposal site. The following is a list of things to be considered before waste is packaged.

- Free Liquids: Waste should contain as little free liquid as reasonably achievable, but under no conditions shall the free liquid volume exceed 0.5 percent by volume of the external container.
- Absorbent: If absorbent material is used to solidify liquid in the waste, the quantity of absorbent material added to the waste should be sufficient to absorb a minimum of twice the volume of the liquid.
- Particulate: Waste in particulate form should be immobilized, and, if immobilization is impractical, other acceptable waste packaging shall be used, such as overpacking; drum containing particulate enclosed in another drum; and wooden or steel box with particulate enclosed in a 6-mil sealed plastic liner inside the box.
- Gases: Pressure in the waste box shall not exceed 1.5 atmospheres absolute at 20^{o}C. Compressed gases as defined by 49 CFR 173.300 are generally not accepted for disposal unless the valve mechanism has been removed or it is obvious that the container has been punctured.
- Containers, Vessels, Manifolds: Containers, vessels, and manifolds that have been exposed to tritium and should remain sealed, must be certified to contain no pressure greater than 1.5 atmospheres absolute prior to packaging as waste. DOE Order 5820.2A, from which the 1.5 atmospheres requirement is taken, has been superceded by DOE O 435.1, but in the revised Order the 1.5 atmosphere requirement only applies to initial gaseous forms ("Low-level waste in a gaseous form must be packaged such that the pressure does not exceed 1.5 atmospheres absolute at 20°C.") Another requirement from DOE O 435.1 states, "When waste is packaged, vents or other measures should be provided if the potential exists for pressurizing containers or generating flammable or explosive concentrations of gases within the waste package."

- Stabilization: Where practical, waste should be treated to provide a more structurally and chemically stable waste form.
- Etiologic Agents: LLW containing pathogens, infectious wastes, or other etiologic agents as defined in 49 CFR 173.386 is generally not accepted.
- Chelating Agents: Chelating or complexing agents at concentrations greater than 1 percent by weight of the waste form are generally not accepted.
- Polychlorinated Biphenyls: Polychlorinated Biphenyls contaminated LLW is generally not accepted unless the Polychlorinated Biphenyls concentration is less than 50 ppm
- Explosives and Pyrophorics: Material in a form that may explode spontaneously or combust if the container is breached is generally not accepted for disposal.

8.4 Waste Shipping

The shipping requirements for Class 7 (radioactive) material can be found in Section 6.1. If the tritium waste is mixed with RCRA hazardous waste, a hazardous waste manifest and Land Disposal Restriction notification is required for offsite shipments. See Section 7.1 for RCRA pre-transport requirements associated with mixed waste. If the tritium waste is mixed with RCRA hazardous waste, all the characteristics of the resultant material must also be reviewed against the DOT hazardous materials defining criteria (different from RCRA). In addition, any containers less than 110 gallons in capacity must be marked as hazardous waste in accordance with 40 CFR 262.32.

During shipment, the waste package must meet the current DOE and DOT regulations for shipment of the form and quantity of radioactive material; i.e. Limited Quantity, Type A Quantity, and Type B Quantity.

APPENDIX A: USEFUL NUMERICAL VALUES

A.1 General Data

- 1 becquerel (Bq) = 1 disintegration/second = 2.7×10^{-11} Ci
- 1 Ci = 3.7×10^{10} disintegrations/second = 3.7×10^{10} Bq = 37 GBq
- Avogadro's Number = 6.023×10^{23} molecules/mole
- STP conditions: 760 Torr, and 0°C (1 atm and 273K)
- 1 mole of ideal gas at STP = 22.414 L at 0° C, and approximately 24.2 L/mole at room temperature
- 1 Sievert (Sv) = 83.8 roentgens = 100 rem
- A1 quantity-special form
- A2 quantity –other than special form or normal form

A.2 General Tritium Data

- Tritium decays to ^{3}He + beta + neutrino
- Half-life of tritium (Scientific purposes) = 12.323 ± 0.004 years (4500.88 ± 1.46 days)
  Half-life of tritium (Accountability purposes) = 12.33 +/- 0.06 years (DOE M 474.1-2, Figure IV-2)
- Tritium decay factor = 0.99984601/day
- Maximum beta energy of decay (E max.) = 18.6 keV
- Mean beta energy of decay (E mean) = 5.69 keV
- Volume of 1 Ci of tritium (T_2) at STP = 0.386 mL
- Tritium (T_2) gas = 9619 Ci/g
- T_2 gas contains 58023 Ci/mole
- 2.589 Ci/cm^3 of T_2 at STP
- Diameter of a tritium atom (approximate) = 1.1 Angstroms
- Dissociation energy, T_2 to 2T = 4.59 eV
- Ionization energy, T to T+ + e- = 13.55 eV
- Atomic weight = 3.01605
- Gram molecular weight of tritium = 6.0321 g
- Grams tritium/liter at STP = 0.269122 g/L
- Liters/gram of tritium at STP = 3.71579 L/g
- Boiling point of tritium at 1 atmosphere = 25.0 K
- T_2 gas, at 1 atmosphere pressure and 25°C = 2.372 Ci/cm^3
- Tritiated water, T_2O = 3200 Ci/cm^3

A.3 Regulatory Quantities

- Type B Quantity = > 1080 Ci
- Type A Quantity = > 0.11 to <1080 Ci depending on form
- Limited Quantity = < 21.6 Ci of tritium in gaseous form
  < 1.1 Ci of tritium in solid form
  < 1 Ci of tritium at a concentration of > 1 Ci/L liquid
  < 100 Ci at a concentration of > 0.1 to < 1 Ci/L liquid
  < 1000 Ci at a concentration of < 0.1 Ci/L liquid
- A2 Quantity = 40 TBq (1080Ci)
- Limited Quantity Excepted Package Requirement for tritium
  - Radiation level < 0.005 mSv/hr (0.5 mrem/hr)
  - Quantity of radioactive material < limit in 49 CFR 173.425
  - Package meets general design requirements for radioactive material packaging from 49 CFR 173.410
  - Nonfixed/ removable contamination on the external surface of the package is < limit in 49 CFR 173.443 (0.41 Bq of tritium/cm^2 or 22 disintegrations/min (dpm)/cm^2 for tritium)
  - Outside of inner or outer packaging is marked “Radioactive”
- Low Specific Activity (LSA) A quantity of Class 7 (radioactive) material with limited specific activity
  - LSA-I
    - Earth, concrete, rubble or other debris = 4 x 10^5 TBq of tritium/g (0.001 Ci of tritium/g, 1 Ci of tritium/kg or 2.2 lb/Ci of tritium)
  - LSA-II
    - Tritium-Contaminated Water up to: 0.8 TBq of tritium/L (20.0 Ci/L)
    - Solids and Gases up to 0.004 TBq of tritium/g (0.1 Ci/g, 100 Ci/kg)
    - Liquids 4 x 10^4 TBq of tritium/g (0.01 Ci/g, 10 Ci/kg)
  - LSA-III
    - Solids, consolidated wastes, or activated material that meet 49 CFR 173.468 water leach test; and
    - Uniformly distributed in a collection of solid objects or uniformly distributed in solid compact binding agent; i.e., concrete, bitumen, ceramic, etc.
    - Relatively insoluble or intrinsically contained in relatively insoluble material such that, under loss of packaging and placed in water for seven days, would not exceed 4 TBq of tritium (110 Ci of tritium)
    - The average specific activity of the solid does not exceed 0.08 TBq of tritium/g (2.16 Ci/g)
- SCO (Surface Contaminated Object) Non-radioactive solid objects with Class 7 (radioactive) materials distributed on the surfaces
  - SCO-I
    - Nonfixed contamination on accessible surface averaged over 300 cm^2 is < 4 Bq of tritium/cm^2 (10^4 µCi/cm^2)

    - Fixed contamination on the accessible surface averaged over 300 cm^2 is < 4 x 10^4 Bq of tritium/cm^2 (1 µCi/cm^2)
    - Nonfixed plus fixed contamination on the inaccessible surface averaged over 300 cm^2 is < 8 x 10^5 Bq of tritium/cm^2 (20 µCi/cm^2)
  - SCO-II

    Solid object on which limit for SCO-I is exceeded and meets the following:
    - Nonfixed contamination of accessible surface averaged over 300 cm^2 is < 400 Bq of tritium/cm^2 (10^{-2} µCi/cm^2).
    - Fixed contamination of accessible surface averaged over 300 cm^2 is < 8 x 10^5 Bq of tritium/cm^2 (20 µCi/cm^2).
    - Nonfixed plus fixed contamination of accessible surface averaged over 300 cm^2 is < 8 x 10^5 Bq of tritium/cm^2 (20 µCi/cm^2).
- Type A Packages limited to 40 TBq of tritium (< 1080 Ci) per package
- Type B Packages for quantities > 40 TBq of tritium (> 1080 Ci) per package
- Graded Safeguards Program
  - Category III: Weapons or test components, containing reportable quantities > 50 g T_2 with isotopic fraction T_2 > 20 percent
  - Category IV: All other reportable quantities
-
- Facility Categories
  - Hazard Category 1: Category A reactors and facilities designated by PSO.
  - Hazard Category 2: > 30 g of tritium
  - Hazard Category 3: > 1.6 but < 30 g of tritium
  - Radiological Facility:< 1.6 g of tritium

A.4 Tritium Dose and Exposure Data

- Biological half-life = 8 to 12 days (oxide); biological half-life of tritides is currently being researched
- Derived Air Concentration (DAC)
  - DAC for HTO = 20 µCi/m^3 = 2 x 10^{-5} µCi/mL = 7 x 10^5 Bq/m^3
  - DAC for HT = 200,000 µCi/m^3 = 2 x 10^{-1} µCi/mL = 9 x 10^9 Bq/m^3
  - DACs for STCs –see Appendix E
- Dose Conversion Factor (DCF)
  - DCF = 0.067 mrem/µCi (inhalation)
  - DCF = 0.1 mrem/µCi (inhalation plus 50 percent allowance for skin absorption)
- Annual Limit on Intake (ALI)
  - ALI for HTO = 80,000 µCi
  - An initial exposure equilibrium urine count of 1 µCi/L equates to approximately a 3-mrem dose.
  - A urine count of 50 µCi/ L for a year equates to approximately a 5,000-mrem dose.

  - Breathing 20 µCi/m^3 HTO in air for 8 hours/day, 50 week/year will result in a dose of approximately 5,000 mrem
  - Tritiated water is approximately 10,000 times more hazardous than tritium gas because of rapid uptake mechanisms.
- Tritium Beta Particles
  - Range in Air = 4.5 to 6 mm
  - Range in Water = 0.0005 cm
  - Range in Tissue = 0.0007 cm
  - Radiation, 1 mCi in man (70 kg) = 0.0044 rem/day
  - Maximum Penetration = 0.6 mg/cm^3

A.5 Tritium Container Data

- SRS Hydride Transport Vessel (HTV)
  - Reusable container for transporting up to 18 grams of tritium
  - Shipped in UC-609/BTSP
  - Dual port, flow through capable
  - One female, one male port, Cajon® SS-4-VCR
  - Tritium stored as uranium tritide
  - Contains 493 g depleted uranium
  - Stoichiometry Maximum 1:2.9
  - Weight 9.3 lbs.
  - Height 9.995 in
  - Diameter 4.6 in
  - Volume 690 cm^3
  - Maximum normal operating temperature = 450°C
  - Pressure Limit at Maximum normal operating temperature = 2.9 psia
  - Tritium vapor pressure as a function of temperature at U:T = 1:2.9
  - Dissociation Equations
  - $\log P_{atm} = -4038.2/T + 6.074$, $P_{atm} = 10^{-4038.2/T + 6.074}$
    - $\log P_{psia} = -4038.2/T + 7.2413$, $P_{psia} = 10^{-4038.2/T+7.2413}$
  - Maximum specified tritium leak rate in std cc = <1.26 x 10^{-7} std cc/s.
- SRS Product Vessel (PV)
  - Reusable container for transporting 10 g of tritium in gas form
  - Shipped in UC-609/BTSP
  - Volume 21 L
  - Maximum pressure 1,200 torr
  - Height 30.5 in
  - Diameter 9.875 in
  - Weight 44 lb
  - Single valve, Nupro® SS-4-HS-TW

  - Male Nut, Cajon® SS-4-VCR-4
  - Female Cap Cajon® SS-4-VCR-CP
  - Maximum helium leak rate = 1 x 10^{-7} STP cc/s He with 8.5 psig helium internal pressure by belljar method
- SRS Hydride Storage Vessel (HSV)
  - Container for storing up to 1600 STP liters of hydrogen isotopes
  - Capable of being shipped in UC-609/BTSP
  - Dual port, flow through capable
  - One female, one male port, Cajon® SS-4-VCR
  - Tritium stored as titanium tritide
  - Contains 4400 g titanium (Ergenics HY-STOR 106)
  - Weight 45 lb
  - Height 16.3 in
  - Diameter 6.6 in
  - Maximum normal operations temperature = 760°C
  - Pressure Range at normal operating temperature = 192 psig to Full Vacuum
  - Pressure Limit at 120°C = 1000 psig
  - Maximum helium leak rate = 1x 10^{-7} STP cc/s He with 30 psig helium internal pressure by belljar method
- SRS Bulk Tritium Shipping Package (BTSP)
  - Type B(M) Reusable Shipping Package
  - Authorized Content; Up to 150 grams of Tritium as a gas, solid or adsorbed tritiated water
  - Vapor on molecular sieve materials.
  - Package Gross Weight 650 lb
  - Maximum Content Weight 120 lb
  - Packaging Overpack
  - Height 50.5 in
  - Diameter 24.5 in
  - Containment Vessel (CV)
  - Height 37.5 in
  - Diameter 15 in
  - Available Volume 2,943 in3
  - 30.625 in high x 10 in ID
  - CV Weight ~153 lbs
  - CV Design Pressure Limit at 400ºF @ 500 psig
- SRS LP-50 (No longer certified for shipping).
  - Container for storing tritium in gas form
  - Volume 50 L
  - Aluminum shell for contamination control and valve protection

  - Maximum initial pressure at loading 1,200 torr
  - Height 28.3 in
  - Diameter 13.2 in
  - Product Container Weight 32 lb
  - Aluminum Shell Weight 43 lb
  - Full Package Weight including drum 260 lb
  - Single valve, Hoke Model 4213X2 packless valve
  - Male Nut, Cajon® SS-4-VCR-4
  - Female Cap Cajon® SS-4-VCR-1-BL
  - Maximum helium leak rate = 7×10^{-8} STP cc/s He with 22.7 psia helium internal pressure by belljar method
- Mound AL-M1-5
  - Up to 100,000 Ci absorbed on molecular sieve, silica gel or commercial clay
  - Used for storage and shipment of absorbed tritiated water.
  - Cylindrical vessel 6 5/8 in O.D. by 23 7/8 height
  - 316 Stainless Steel
  - Cap which is screwed onto threaded center post of container containing two self-sealing quick disconnect fittings used for connection to tritium monitors to check for tritium leak from container
  - Cap has two functions; physical protection of valves, fittings and pressure transducer at top of container and also secondary containment. The cap is sealed with four O rings; one face seal at the bottom circumference of the cover, another near the top of the center post and two sealing the quick disconnect fittings in their wells at the top of the cap.
  - Organic compounds should be limited to 1% or less of the water content of the container.
  - Halogen compounds are to be avoided

A.6 Other Data

- Calorimeter Factor
  - 3.0657 +/- 0.009 g of tritium per Watt
  - 0.3240 +/- 0.0009 Watts/g of tritium
- ANSI N14.5-87 Leakage Test
  - $< 1 \times 10^{-7}\ \text{cm}^{3}/\text{s}\ {}^{/}\text{He}$

APPENDIX B: DEFINITIONS

Actual Activity – The total quantity of radioactive material within a particulate; Sometimes referred to as true activity

Airborne radioactivity area: Any area where the measured concentration of airborne radioactivity, above natural background, exceeds or is likely to exceed the derived air concentration (DAC) values listed in Appendix A or Appendix C of 10 CFR Part 835, or where an individual present in the area without respiratory protection could receive an intake exceeding 12 DAC-hrs in a week (10 CFR Part 835).

As low as reasonably achievable (ALARA): A phrase (acronym) used to describe an approach to radiation protection to control or manage exposures (both individual and collective to the work force and the general public) and releases of radioactive material to the environment as low as social, technical, economic, practical, and public policy considerations permit (DOE 5400.5 Chg 2).

Below regulatory concern : A definable amount of low-level waste that can be deregulated with minimal risk to the public (DOE O 435.1 Chg. 1)

Best available technology for radioactive effluent control (BAT): The preferred technology for a particular activity, selected from among others after taking into account factors related to technology, economics, public policy, and other parameters. As used in this Order, the BAT is not a specific level of treatment, but is the conclusion of a selection process in which several alternatives are evaluated (DOE O 435.1).

Biokinetic Model: A mathematical model that describes in quantitative terms the retention and transport of a material in the body. The biokinetic model used in ICRP 78 is recommended to evaluate intakes of STCs.

Breathing Zone Air Sampler (BZA): An air sampler that draws air from the area close enough to the nose so the sample can be considered representative of the air a person breathes (ANSI Z88.2-1992). An example of a breathing zone air sampler is a lapel monitor.

Buffer zone: The smallest region beyond the disposal unit that is required as controlled space for monitoring and for taking mitigative measures, as may be required (DOE O 435.1 Chg 1).

Byproduct material: (1) any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material, and (2) the tailings or wastes produced by the extraction or concentration of uranium or thorium from any ore processed primarily for its source material content (Atomic Energy Act of 1954, 42 USC 2011). The Energy Policy Act of 2005 revised this definition as follows: The term ''byproduct material'' means

(1) any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material;

(2) the tailings or wastes produced by the extraction or concentration of uranium or thorium from any ore processed primarily for its source material content;

(3) (A) any discrete source of radium-226 that is produced, extracted, or converted after extraction, before, on, or after the date of enactment of this paragraph for use for a commercial, medical, or research activity; or

(B) any material that –

(i) has been made radioactive by use of a particle accelerator; and

(ii) is produced, extracted, or converted after extraction, before, on, or after the date of enactment of this paragraph for use for a commercial, medical, or research activity; and

(4) any discrete source of naturally occurring radioactive material, other than source material, that –

(A) the Commission, in consultation with the Administrator of the Environmental Protection Agency, the Secretary of Energy, the Secretary of Homeland Security, and the head of any other appropriate Federal agency, determines would pose a threat similar to the threat posed by a discrete source of radium-226 to the public health and safety or the common defense and security; and

(B) before, on, or after the date of enactment of this paragraph is extracted or converted after extraction for use in commercial, medical, or research activity.

Certified waste: Waste that has been confirmed to comply with disposal site waste acceptance criteria (e.g., the Waste Isolation Pilot Plant-Waste Acceptance Criteria for transuranic waste, the DOE/NVO-325 criteria) under an approved certification program (DOE O 435.1 Chg 1)

Confinement system: Any equipment, structure, or system, which limits the release and/or dispersion of a hazardous/radioactive material within a facility. Examples are fume hoods, air locks, ventilation systems, and may include containment and recovery systems. Confinement systems may consist of multiple techniques and barriers depending upon the quantity of tritium involved and the consequences of an uncontrolled release (U.S. DOE Tritium Focus Group).

A collection of barriers that can satisfy a specified leak criterion contingent upon operation of its ancillary (active) system. Examples of confinement systems include: a glovebox and its associated cleanup system, and a room with its associated cleanup system. Note that in the context of this definition, a glovebox with an associated glovebox cleanup system is a confinement system. A glovebox structure itself is a containment system if, and only if, the specified leak criterion can be met by the structure itself (DOE-HDBK-1129-99).

An area having structures or systems from which releases of hazardous materials are controlled. Primary confinement systems are process enclosures (gloveboxes, conveyors, transfer boxes, and other spaces normally containing hazardous materials). Secondary confinement areas surround one or more primary confinement systems (operating area compartments) (DOE O 435.1 Chg 1).

Containment system: Any equipment, structure, or systems that serve as an integral and essentially leak tight barrier against the uncontrolled release of hazardous/radioactive material to the environment and other areas within the facility. Examples include process piping, sealed containers, tanks, gloveboxes, and any other closed loop system, which holds the material for possible recovery of tritium (U.S. DOE Tritium Focus Group).

A collection of passive barriers that can satisfy a specified leak criterion without operation of any ancillary equipment. An example of a containment system is a series of piping and vessels enclosing tritium gas operations. An example of a simple double containment system is a container within another container with each container acting as a separate and independent containment system; more intricate double containment systems have the capability to monitor the volume between the containers for leak detection of the inner container (DOE-HDBK-1129-99).

The assembly of components of the packaging intended to retain the radioactive material during transport (10 CFR Part 71).

Controlled Area: Any area to which access is managed by or for DOE to protect individuals from exposure to radiation and/or radioactive material (10 CFR Part 835).

Derived Air Concentration (DAC): For the radionuclides listed in Appendix A of 10 CFR Part 835, the airborne concentration that equals the Annual Limit on Intake (ALI) divided by the volume of air breathed by an average worker for a working year of 2,000 hours, assuming a breathing volume of 2,400 m^3 (10 CFR Part 835).

Derived Concentration Guide (DCG): The concentration of a radio nuclide in air or water that, under conditions of continuous exposure for one year by one exposure mode (i.e., ingestion of water, submersion in air, or inhalation), would result in an effective dose equivalent of 100 mrem (1 mSv). DCGs do not consider decay products when the parent radionuclide is the cause of the exposure (DOE Order 5400.5 Chg 2).

DOE/DOT Type A, approved shipping package: For the purpose of this text, these are packages that can be used for the transport of Type A quantities of radioactive materials. The two typical packages used for solids are metal 55-gallon drums with full removable lids and metal boxes 4 feet wide by 4 feet high and 7 feet long with removable lids. DOT 7A packages may be fabricated in almost any size to fit special needs like packaging of gloveboxes.

Documented Safety Analysis (DSA): Documented safety analysis means a documented analysis of the extent to which a nuclear facility can be operated safely with respect to workers, the public, and the environment, including a description of the conditions, safe boundaries, and hazard controls that provide the basis for ensuring safety (10 CFR Part 830). Successor to the Safety Analysis.

Dose Assessment: Process of determining radiation dose and uncertainty included in the dose estimate, through the use of exposure scenarios, bioassay results, monitoring data, source term information and pathway analysis.

Dose Conversion Factor (DCF): Dose per unit intake.

Dosimetric Model: A mathematical model that prescribes how to use the biokinetic and radiation transport models to quantify the dose to specific organs and tissues and how to calculate effective dose. The dosimetric models for STC are described primarily in ICRP publications 67 and 71.

Facility segmentation: The concept of independent facility segments should be applied where facility features preclude bringing material together or causing harmful interaction from a common severe phenomenon. Therefore, the standard permits the concept of facility segmentation provided the hazardous material in one segment could not interact with hazardous materials in other segments. For example, independence of HVAC and piping must exist in order to demonstrate independence for facility segmentation purposes. This independence must be demonstrated and places the "burden of proof" on the analyst. Additionally, material contained in DOT Type B shipping containers (with or without overpack) may also be excluded from summation of a facility's radioactive inventory if the Certificates of Compliance are kept current and the materials stored are authorized by the Certificate. However, Type B containers without overpack should have heat protection provided by the facility's fire suppression system (DOE-STD-1027-92, Chg 1, Rev. 1).

Free liquids: Liquids that readily separate from the solid portion of a waste under ambient temperature and pressure (DOE O 435.1 Chg 1)

Gastrointestinal (GI) Tract Model: A mathematical representation of the behavior of radionuclides in the contents of the human gastrointestinal tract.

Hazard Category 1: A facility in which the hazard analysis shows the potential for significant off-site consequences. Category A reactors and facilities designated by PSO. Regardless of the quantity of tritium, a facility that handles only tritium is not a Hazard Category 1 facility unless it is designated so by the PSO (DOE-STD-1027-92, Chg 1, Rev. 1).

Hazard Category 2: A facility in which the hazard analysis shows the potential for significant on-site consequences. Facilities with the potential for nuclear criticality events or with sufficient quantities of hazardous material and energy, which would require on-site emergency planning activities. The threshold tritium inventory for a tritium facility to be designated as a Hazard Category 2 facility is 30 grams or 300,000 Ci (DOE-STD-1027-92, Chg 1, Rev. 1)

Hazard Category 3: A facility in which the hazard analysis shows the potential for only significant localized consequences. Facilities included are those with quantities of hazardous radioactive materials that meet or exceed values in Table A.1 of DOE-STD-1027-97. The threshold for tritium is specified as 1.6 grams or 16,000 Ci (DOE-STD-1027-92, Chg 1, Rev. 1)

Hazardous materials: Those materials that are toxic, explosive, flammable, corrosive, or otherwise physically or biologically health-threatening (DOE-STD-3009)

Hazardous wastes: Those wastes that are designated hazardous by EPA regulations (40 CFR Part 261) (DOE O 435.1 Chg 1) The DOT Hazardous Material Regulations define hazardous materials and hazardous waste differently; see 49 CFR 171.8.

Insoluble Metal Tritide (IMT): A type of insoluble special tritium compound in which tritium has formed a chemical bond to a metal

Insoluble Special Tritium Compound: A special tritium compound, for which the tritium cannot be rapidly taken up by the systemic compartment of the body

Insoluble Tritiated Particle (ITP) – Any tritiated particle from which the tritium is not readily released in air or aqueous solutions during the time interval over which the sample is collected and initially analyzed. This time interval may vary significantly, typically ranging from minutes to days.

Leak-tight: A leakage rate, which, “in a practical sense, precludes any significant release of radioactive materials. This degree of containment is achieved by demonstration of a leakage rate that is $\leq$ to 1×10^{-7} ref cm^3/s of air, at an upstream pressure of 1 atmosphere (atm) absolute, and a downstream pressure of 0.01 atm abs, or less. Note: A leakage rate of 1×10^{-7} ref cm^3/s is equal to 4.09E-12 gram-moles of dry air or helium, and is equivalent to a helium leakage rate, under the same conditions, of approximately 2×10^{-7} cm^3/s.” (ANSI N14.5-2014))

Less than Hazard Category 3 Nuclear Facilities (replaces the term Radiological Facilities):

Facilities that do not meet or exceed Category 3 threshold criteria specified in DOE STD-1027-92, Rev. 1, Table A.1, but still possess some amount of radioactive material are still Nuclear Facilities. The Category 3 threshold quantity of tritium is 16,000 curies. Less than Hazard Category 3 Nuclear Facilities are exempt from some requirements such as development and maintenance of DSAs but are not exempt from other safety requirements (e.g., 10 CFR Part 820 Subpart A, 10 CFR Part 835). Less than Hazard Category 3 Nuclear Facilities are those with an inventory of radiological materials below the levels as defined in DOE-STD-1027-92, Chg. 1, (currently 16,000 curies). There is no official or legal lower threshold limit to be become a non-nuclear facility for the Less than Hazard Category 3 nuclear facilities.

Low-level radioactive waste or Low-level waste: radioactive material that is not high-level radioactive waste, spent nuclear fuel, or AEA section 11.e.2 byproduct material. The phrase is defined in the (Low-Level Radioactive Waste Policy Amendments Act of 1985 and . DOE M 435.1-1)

Mixed waste: Waste containing both radioactive and hazardous waste components as defined by the Atomic Energy Act and the Resource Conservation and Recovery Act, respectively (DOE O 435.1-1)

Nonreactor nuclear facility: those facilities, activities or operations that involve, or will involve, radioactive and/or fissionable materials in such form and quantity that a nuclear or a nuclear explosive hazard potentially exists to workers, the public, or the environment, but does not include accelerators and their operations and does not includeactivities involving only incidental use and generation of radioactive materials or radiation such as check and calibration sources, use of radioactive sources in research and experimental and analytical laboratory activities, electron microscopes, and X-ray machines (10 CFR Part 830).

Nuclear facility: means a reactor or a nonreactor nuclear facility where an activity is conducted for or on behalf of DOE and includes any related area, structure, facility, or activity to the extent necessary to ensure proper implementation of the requirements established by this Part (10 CFR Part 830). DOE-STD-1027-92, Chg. 1 defines tritium inventory thresholds for nuclear facilities.

Observed Activity: The apparent quantity of radioactive material within a particulate as determined by liquid scintillation counting, without attempting to correct for beta particle self-absorption, bremsstrahlung, or the emissions from HT or HTO.

Organically Bound Tritium (OBT): A type of tritiated material in which the tritium has formed a chemical bond with an organic material – typically via a carbon-tritium bond.

Primary containment: The first barrier to an uncontrolled release of hazardous/radioactive material to the environment and/or other areas in the facility. The barrier may/may not serve for containment of the radioactive material (U.S. DOE Tritium Focus Group).

Protective Action Guides (PAGs): Projected numerical dose values established by EPA, DOE, or States for individuals in the population. These values may trigger protective actions that would reduce or avoid the projected dose.

Radioactive waste: Solid, liquid, or gaseous material that contains radionuclides regulated under the Atomic Energy Act of 1954, as amended, and of negligible economic value considering costs of recovery (DOE O 435.1 Chg. 1).

Radiological area: Any area within a controlled area which must be posted as a “radiation area,” “high radiation area,” “very high radiation area,” “contamination area,” “high contamination area,” or “airborne radioactivity area” in accordance with (10 CFR 835.603)

Receiving and/or shipping area: An area or two different areas that have been designated as radioactive materials receiving and/or Shipping Areas and have been posted for the receipt and shipment of packaged radioactive materials.

Receiving and/or Shipping, Storage Area: An area or two different areas that have been designated as the Receiving and/or Shipping Area, Storage Area and have been posted for the storage of packaged incoming and outgoing shipments of radioactive materials.

Reference man: A hypothetical aggregation of human (male and female) physical and physiological characteristics arrived at by international consensus (ICRP Publication 23). These characteristics may be used by researchers and public health workers to standardize results of experiments and to relate biological insult from ionizing radiation to a common base. The “reference man” is assumed to inhale 8,400 cubic meters of air in a year and to ingest 950 liters of water in a year (International Commission on Radiological Protection (ICRP) Publication 89, "Basic Anatomical and Physiological Data for Use in Radiological Protection: Reference Values," September 2001).

Release of property: As used in DOE Order 5400.5 Chg. 2, it is the exercising of DOE’s authority to release property from its control after confirming that residual radioactive material (over which DOE has authority) on the property has been determined to meet the guidelines for residual radioactive material in Chapter IV of DOE Order 5400.5 Chg. 2 or any other applicable radiological requirements. There may be instances in which DOE or other authority will impose restrictions on the management and/or use of the property if the residual radioactive material guidelines of Chapter IV of DOE Order 5400.5 Chg. 2 are not met or if other applicable Federal, state, or local requirements cause the imposition of such restrictions (DOE Order 5400.5 Chg. 2).

Safety analysis: A documented process that: Provides systematic identification of hazards within a given DOE operation, describes and analyzes the adequacy of the measures taken to eliminate, control, or mitigate identified hazards, and analyzes and evaluates potential accidents and their associated risks (DOE Standard 3009).

Safety-class structures, systems, and components (safety-class SSCs): the structures, systems, or components, including portions of process systems, whose preventive or mitigative function is necessary to limit radioactive hazardous material exposure to the public, as determined from safety analyses (10 CFR Part 830).

Safety-significant structures, systems, and components (safety-significant SSCs): the structures, systems, and components which are not designated as safety-class structures, systems, and components, but whose preventive or mitigative function is a major contributor to defense in depth and/or worker safety as determined from safety analyses (10 CFR Part 830).

Secondary containment: The second barrier to an uncontrolled release of hazardous or radioactive materials to the environment and/or other areas in the facility. The barrier may/may not serve for containment of the radioactive material (U.S. DOE Tritium Focus Group).

Self-Absorption Factor for Beta Particles (SAFβ): The fraction of beta particles emitted from within a particulate that escapes the particulate.

Self-Absorption Factor for Energy (SAFe): The fraction of energy emitted from within a particulate that escapes the particulate.

Soil column: An in-situ volume of soil through which liquid waste streams percolate from ponds, cribs, trenches, drain fields, or other areas or facilities used for the primary purpose of removing or retaining the suspended or dissolved radionuclides contained within the liquid process waste stream (DOE O 435.1).

Soluble Tritiated Particle: Any tritiated particle from which the tritium is readily released in air or aqueous solutions during the time interval over which the sample is collected and initially analyzed. This time interval may vary significantly, typically ranging from minutes to days.

Source material: (1) uranium, thorium, or any other material which is determined by the commission pursuant to the provisions of section 61 (42 USC 2091) to be source material; or (2) ores containing one or more of the foregoing materials in such concentration as the Commission now DOE and NRC may by regulation determine from time to time (Atomic Energy Act of 1954, 42 USC 2011 et seq.).

Special nuclear material: (1) plutonium, uranium enriched in the isotope 233 or in the isotope 235, and any other material which the Commission, pursuant to the provisions of section 51 (42 USC 2071) determines to be special nuclear material, but does not include source material; or (2) any material artificially enriched by any of the foregoing, but does not include source material (Atomic Energy Act of 1954, 42 USC 2011 et seq.).

Special Tritium Compound (STC): means any compound, except for H2O, that contains tritium, either intentionally (e.g., by synthesis) or inadvertently (e.g., by contamination mechanisms) (10 CFR Part 835).

Stochastic effects: means malignant and hereditary diseases for which the probability of an effect occurring, rather than its severity, is regarded as a function of dose without a threshold, for radiation protection purposes (10 CFR Part 835).

Technical safety requirements (TSRs): the limits, controls, and related actions that establish the specific parameters and requisite actions for the safe operation of a nuclear facility and include, as appropriate for the work and the hazards identified in the documented safety analysis for the facility: Safety limits, operating limits, surveillance requirements, administrative and management controls, use and application provisions, and design features, as well as a bases appendix (10 CFR Part 830).

Tritiated Material: Any material containing at least an accountable amount of tritium

Tertiary containment: The third barrier to an uncontrolled release of hazardous or radioactive materials to the environment and/or other areas in the facility. The barrier may/may not serve for containment of the radioactive material (U.S. DOE Tritium Focus Group).

Treatment: Any method, technique, or process designed to change the physical or chemical character of waste to render it less hazardous, safer to transport, store or dispose of, or reduced in volume (DOE O 435.1 Chg. 1).

Uptake: For STCs, the process by which the tritium atoms in STCs are taken into the systemic compartment of the body. This process includes the uptake of tritium that has been dissociated from the host molecule as well as the uptake of an entire STC molecule

Waste container: A receptacle for waste, including any liner or shielding material that is intended to accompany the waste in disposal (DOE O 435.1 Chg 1)

Waste package: The waste, waste container, and any absorbent that are intended for disposal as a unit. In the case of surface contaminated, damaged, leaking, or breached waste packages, any overpack shall be considered the waste container, and the original container shall be considered part of the waste (DOE O 435.1 Chg. 1).

APPENDIX C: ASSAY METHODS

There are a number of different assay methods used at the DOE tritium facilities. Most facilities need assay equipment capable of measuring tritium in gaseous, solid, and liquid form.

- Gas Analysis: For assay of gaseous tritium, most facilities use some form of mass spectrometer ranging from quadrapole to large sophisticated light isotope drift tube systems. Gas analysis equipment; especially gas analysis equipment which will measure the low molecular weight gases like H_2, HD, HT, D_2, DT and T_2 accurately; is very expensive and requires a high degree of expertise to operate. These systems can cost from fifty to several hundred thousand dollars.

- Solid (Metal Tritide): Tritium stored in solid form such as a metal tritide must either be decomposed to return it to the gas form for analysis, or the heat output of the solid caused by the decay of tritium can be measured in a constant heat flow calorimeter. In order to use calorimetry to measure the tritium quantity, it must be known that the item being assayed does not contain any other radioactive component and that no chemical reactions are taking place in the container. Constant heat flow calorimeters vary in chamber size from a few cubic centimeters up to a few liters, and the item to be assayed must be small enough to fit inside the calorimeter chamber. The constant heat flow assay process is the most accurate assay method available if the chamber size and item to be assayed are well matched.

- Liquid (HTO, DTO, T_2O): Tritium at high concentrations in liquid form is generally measured using calorimetry. Low concentrations of tritium in liquid form are generally measured by using a scintillation counter. A sample of the liquid is mixed in a scintillation cocktail, and the quantity of tritium in the sample is measured.

C.1 Measurement Accuracy and Safeguards and Security

Past DOE directives required that tritium be accounted for to the hundredth of a gram and was a problem in that most of the equipment and the techniques used cannot accurately determine the tritium quantity to a hundredth of a gram once the quantity assayed exceeds about one half gram. Sophisticated equipment was justified mostly for special process needs. Even sophisticated equipment does not measure the quantity of tritium accurately to a hundredth of a gram once the quantity exceeds about five grams. The assay technique to be used in an operation or facility should be discussed with DOE safeguards and security to make sure that it will meet the DOE needs for the facility safeguards and security category and the activities performed in the facility. Current requirements for accountability have been raised to the gram level as discussed in Section 3.1.

C.2 Tritium Assay Analysis by PVT Mass Spectrometer

The most common method of assaying tritium in gaseous form, i.e., T_2, HT, and DT, mixed with other gases such as Ar, N_2,, O_2, and ^3He, is referred to as "PVT mass spec." The total number of moles of gas, n, in a container is calculated using the equation

$$n = PV/zRT$$

where P = pressure in the container in torr
V = volume of the container in liters
z = compressibility factor (See Table C.1)
R = constant = 62.3631 (See below)
T = temperature (K)

In the formula, the container volume (V) is determined ahead of time by measurement using a volume measuring system, or, if no other means is available, it may be calculated from the physical dimension of the container. The gas pressure (P) and temperature (T) are determined by measurement with available instruments at the time of the mass spec sampling. R is a constant, which is a function of the units of pressure and volume used in the equation and is equal to 62.3631 for pressure in torr and volume in liters.

The compressibility factor (z) is a function of gas type, pressure, and temperature and is either determined from a compressibility table for tritium or estimated using a standard equation such as

$$z_{(T2)} = 1 + [(P_{(torr)} \times 0.000832)/1000]$$

This equation is for a temperature of 295 K.

Most operating facilities have established methods for determining the compressibility factor. For those facilities that do not already have these methods established, Table C-1, *Table of Tritium Compressibility Factors*, may be helpful.

As an example, for a container with a volume of 22.414 L at a pressure of 760 torr and a temperature of 273.15 K (Standard Temperature and Pressure (STP) conditions) the number of moles is calculated as follows:

$$\begin{aligned} N &= PV/zRT \\ &= (760 \times 22.414)/(1.000 \times 62.3631 \times 273.15) \\ &= 1.0000 \text{ mole} \end{aligned}$$

Note: $z_{(T2)}$ at 273.15 K = 1.000.

The total moles of gas in the container at any time, t, is the sum of the moles of the individual gases present in the mixture at that time or in equation form:

$$PV/zRT = n_{(\text{Moles Total})} = n_{(T2)}+n_{(HT)}+n_{(DT)}+n_{(CT4)}+n_{(qTw)}+n_{(He\text{-}3)}+n_{(N2)}+ n_{(O2)}+n_{(Ar)}+n_{(etc.)}$$

The $n_{(qTw)}$ represents a generic tritium-containing component. The "q" in, $n_{(qTw)}$, represents any other element which may be present, and the "w" represents the number of tritium atoms in the molecule. The $n_{(etc.)}$, represents any generic, non-radioactive, non-tritium component.
From this equation, it follows that

$$PV/zRT = n_{(\text{Moles Total})} = n_{(\text{Total Moles of Tritium Containing Gases})} + n_{(\text{Total Moles of Non-tritium Gases})}$$

where

$$n_{(\text{Total Moles of Tritium Containing Gases})} = n_{(T2)} + n_{(HT)} + n_{(DT)} + n_{(CT4)} + n_{(qTw)} + \text{etc.}$$

and

$$n_{(\text{Total Moles of Non-Tritium Gases})} = n_{(He\text{-}3)} + n_{(N2)} + n_{(O2)} + n_{(Ar)} + n_{(etc.)} + \text{etc.}$$

The number of moles of tritium in the container is the sum of the number of moles of tritium in each tritium component. The number of moles of tritium in each tritium component is equal to the number of moles of the component multiplied by the moles of tritium per mole of component. The moles of tritium per mole of component, is defined as the ratio of the number of tritium atoms in the component chemical formula to the number of tritium atoms in T_2, i.e., 2.

$$n_{(\text{Total Moles of Tritium})}= \underline{2}n_{(T2)} + \underline{1}n_{(HT)} + \underline{1}n_{(DT)} + \underline{4}n(CT_4)/2 + \underline{w}\, n_{(qTw)} + \text{etc.}$$

TABLE C-1: Table of tritium compressibility factors at 295 K

$$Z_{(T2)} = 1 + \{[(P_{atm} \times 760) \times 0.000832]/1000\}$$

P (atm)	**$z(T_2)$**	**P (atm)**	**$z(T_2)$**	**P (atm)**	**$z(T_2)$**	**P (atm)**	**$z(T_2)$**	**P (atm)**	**$z(T_2)$**	**P (atm)**	**$z(T_2)$**
0.0	1.0000	3.0	1.0019	6.0	1.0038	9.0	1.0057	12.0	1.0076	15.0	1.0095
0.1	1.0001	3.1	1.0020	6.1	1.0039	9.1	1.0058	12.1	1.0077	15.1	1.0095
0.2	1.0001	3.2	1.0020	6.2	1.0039	9.2	1.0058	12.2	1.0077	15.2	1.0096
0.3	1.0002	3.3	1.0021	6.3	1.0040	9.3	1.0059	12.3	1.0078	15.3	1.0097
0.4	1.0003	3.4	1.0021	6.4	1.0040	9.4	1.0059	12.4	1.0078	15.4	1.0097
0.5	1.0003	3.5	1.0022	6.5	1.0041	9.5	1.0060	12.5	1.0079	15.5	1.0098
0.6	1.0004	3.6	1.0023	6.6	1.0042	9.6	1.0061	12.6	1.0080	15.6	1.0099
0.7	1.0004	3.7	1.0023	6.7	1.0042	9.7	1.0061	12.7	1.0080	15.7	1.0099
0.8	1.0005	3.8	1.0024	6.8	1.0043	9.8	1.0062	12.8	1.0081	15.8	1.0100
0.9	1.0006	3.9	1.0025	6.9	1.0044	9.9	1.0063	12.9	1.0082	15.9	1.0101
1.0	1.0006	4.0	1.0025	7.0	1.0044	10.0	1.0063	13.0	1.0082	16.0	1.0101
1.1	1.0007	4.1	1.0026	7.1	1.0045	10.1	1.0064	13.1	1.0083	16.1	1.0102
1.2	1.0008	4.2	1.0027	7.2	1.0046	10.2	1.0064	13.2	1.0083	16.2	1.0102
0.1	1.0001	4.3	1.0027	7.3	1.0046	10.3	1.0065	13.3	1.0084	16.3	1.0103
1.4	1.0009	4.4	1.0028	7.4	1.0047	10.4	1.0066	13.4	1.0085	16.4	1.0104
1.5	1.0009	4.5	1.0028	7.5	1.0047	10.5	1.0066	13.5	1.0085	16.5	1.0104
1.6	1.0010	4.6	1.0029	7.6	1.0048	10.6	1.0067	13.6	1.0086	16.6	1.0105
1.6	1.0010	4.7	1.0030	7.7	1.0049	10.7	1.0068	13.7	1.0087	16.7	1.0106
1.8	1.0011	4.8	1.0030	7.8	1.0049	10.8	1.0068	13.8	1.0087	16.8	1.0106
1.9	1.0012	4.9	1.0031	7.9	1.0050	10.9	1.0069	13.9	1.0088	16.9	1.0107
2.0	1.0013	5.0	1.0032	8.0	1.0051	11.0	1.0070	14.0	1.0089	17.0	1.0107
2.1	1.0013	5.1	1.0032	8.1	1.0051	11.1	1.0070	14.1	1.0089	17.1	1.0108
2.2	1.0014	5.2	1.0033	8.2	1.0052	11.2	1.0071	14.2	1.0090	17.2	1.0109
2.3	1.0015	5.3	1.0034	8.3	1.0052	11.3	1.0071	14.3	1.0090	17.3	1.0109
2.4	1.0015	5.4	1.0034	8.4	1.0053	11.4	1.0072	14.4	1.0091	17.4	1.0110
2.5	1.0016	5.5	1.0035	8.5	1.0054	11.5	1.0073	14.5	1.0092	17.5	1.0111
2.6	1.0016	5.6	1.0035	8.6	1.0054	11.6	1.0073	14.6	1.0092	17.6	1.0111
2.7	1.0017	5.7	1.0036	8.7	1.0055	11.7	1.0074	14.7	1.0093	17.7	1.0112
2.8	1.0018	5.8	1.0037	8.8	1.0056	11.8	1.0075	14.8	1.0094	17.8	1.0113
2.9	1.0018	5.9	1.0037	8.9	1.0056	11.9	1.0075	14.9	1.0094	17.9	1.0113
3.0	1.0019	6.0	1.0038	9.0	1.0057	12.0	1.0076	15.0	1.0095	18.0	1.0114

Table C-1: Table of Tritium Compressibility Factors at 295 K *(continued)*

$$Z_{(T2)} = 1 + \{[(P_{atm} \text{ x } 760) \text{ x } 0.000832]/1000\}$$

P (atm)	$z(T_2)$	P (atm)	$z(T_2)$	P (atm)	$z(T_2)$	P (atm)	$z(T_2)$	P (atm)	$z(T_2)$	P (atm)	$z(T_2)$
18.0	1.0114	21.0	1.0133	24.0	1.0152	27.0	1.0171	30.0	1.0190	33.0	1.0209
18.1	1.0114	21.1	1.0133	24.1	1.0152	27.1	1.0171	30.1	1.0190	33.1	1.0209
18.2	1.0115	21.2	1.0134	24.2	1.0153	27.2	1.0172	30.2	1.0191	33.2	1.0210
18.3	1.0116	21.3	1.0135	24.3	1.0154	27.3	1.0173	30.3	1.0192	33.3	1.0211
18.4	1.0116	21.4	1.0135	24.4	1.0154	27.4	1.0173	30.4	1.0192	33.4	1.0211
18.5	1.0117	21.5	1.0136	24.5	1.0155	27.5	1.0174	30.5	1.0193	33.5	1.0212
18.6	1.0118	21.6	1.0137	24.6	1.0156	27.6	1.0175	30.6	1.0193	33.6	1.0212
18.7	1.0118	21.7	1.0137	24.7	1.0156	27.7	1.0175	30.7	1.0194	33.7	1.0213
18.8	1.0119	21.8	1.0138	24.8	1.0157	27.8	1.0176	30.8	1.0195	33.8	1.0214
18.9	1.0120	21.9	1.0138	24.9	1.0157	27.9	1.0176	30.9	1.0195	33.9	1.0214
19.0	1.0120	22.0	1.0139	25.0	1.0158	28.0	1.0177	31.0	1.0196	34.0	1.0215
19.1	1.0121	22.1	1.0140	25.1	1.0159	28.1	1.0178	31.1	1.0197	34.1	1.0216
19.2	1.0121	22.2	1.0140	25.2	1.0159	28.2	1.0178	31.2	1.0197	34.2	1.0216
19.3	1.0122	22.3	1.0141	25.3	1.0160	28.3	1.0179	31.3	1.0198	34.3	1.0217
19.4	1.0123	22.4	1.0142	25.4	1.0161	28.4	1.0180	31.4	1.0199	34.4	1.0218
19.5	1.0123	22.5	1.0142	25.5	1.0161	28.5	1.0180	31.5	1.0199	34.5	1.0218
19.6	1.0124	22.6	1.0143	25.6	1.0162	28.6	1.0181	31.6	1.0200	34.6	1.0219
19.7	1.0125	22.7	1.0144	25.7	1.0163	28.7	1.0181	31.7	1.0200	34.7	1.0219
19.8	1.0125	22.8	1.0144	25.8	1.0163	28.8	1.0182	31.8	1.0201	34.8	1.0220
19.9	1.0126	22.9	1.0145	25.9	1.0164	28.9	1.0183	31.9	1.0202	34.9	1.0221
20.0	1.0126	23.0	1.0145	26.0	1.0164	29.0	1.0183	32.0	1.0202	35.0	1.0221
20.1	1.0127	23.1	1.0146	26.1	1.0165	29.1	1.0184	32.1	1.0203	35.1	1.0222
20.2	1.0128	23.2	1.0147	26.2	1.0166	29.2	1.0185	32.2	1.0204	35.2	1.0223
20.3	1.0128	23.3	1.0147	26.3	1.0166	29.3	1.0185	32.3	1.0204	35.3	1.0223
20.4	1.0129	23.4	1.0148	26.4	1.0167	29.4	1.0186	32.4	1.0205	35.4	1.0224
20.5	1.0130	23.5	1.0149	26.5	1.0168	29.5	1.0187	32.5	1.0206	35.5	1.0224
20.6	1.0130	23.6	1.0149	26.6	1.0168	29.6	1.0187	32.6	1.0206	35.6	1.0225
20.7	1.0131	23.7	1.0150	26.7	1.0169	29.7	1.0188	32.7	1.0207	35.7	1.0226
20.8	1.0132	23.8	1.0150	26.8	1.0169	29.8	1.0188	32.8	1.0207	35.8	1.0226
20.9	1.0132	23.9	1.0151	26.9	1.0170	29.9	1.0189	32.9	1.0208	35.9	1.0227
21.0	1.0133	24.0	1.0152	27.0	1.0171	30.0	1.0190	33.0	1.0209	36.0	1.0228

To determine the component in the gas and the number of moles of each component in the gas, a sample of the container gas is analyzed, and the mole percent of each gas is determined. This gas analysis results in a number for each component in the mixture, which represents the mole percent of each gas at the time of the analysis. The mole percent (m%) is calculated by

$$m\%_{(component)} = (\text{Moles of a component/Moles Total}) \text{ x } 100$$

Therefore, the Mole Percent Total ($m\%_{(Total)}$) is

$$m\%_{(\text{Total})} = 100 = m\%_{(T2)} + m\%_{(HT)} + m\%_{(DT)} + m\%_{(CT4)} + m\%_{(qTw)} + m\%_{(He\text{-}3)} + m\%_{(N2)} + m\%_{(O2)} + m\%_{(\text{etc.})}$$

The number of moles of each gas component in the container is calculated by

$$n_{(\text{Moles of Component})} = (m\%_{(\text{component})}/100) \times n_{(\text{Total Moles})}$$

The grams of each component can then be calculated using the formula

Grams of Component = $(m\%_{(\text{component})}/100) \times (n_{(\text{Total Moles})}) \times$ (Gram Molecular Weight of Component)

The following is the process used to determine the number of moles and grams of tritium in the HT component in a container.

1. Calculate the total moles of material in the container.
2. Analyze the sample using a mass spectrometer to determine the mole percent of HT.
3. Calculate the number of moles of HT in the container.
4. Calculate the number of grams of HT in the container.
5. Multiply the moles of HT by ½ to determine the moles of T_2 in the HT component.
6. Multiply the moles of HT by ½ and by the gram molecular weight of tritium to determine the grams of tritium in the HT component.

In steps 5 and 6, the ½ is used because it is the ratio of the number of tritium atoms in HT to the number of tritium atoms in T_2 ($HT/T_2 = 1/2$).

The percent tritium in the container is the sum of the mole percent of each tritium component multiplied by the number of moles of tritium per mole of the tritium component. The moles of tritium per mole of component is equal to the ratio of the number of tritium atoms in the component chemical formula to the number of tritium atoms in T_2; i.e., 2. This can be expressed as the following formula:

$$m\%_{(\text{Tritium Per Mole Total})} = \underline{2}\, m\%_{(T2)} + \underline{1}\, m\%_{(HT)} + \underline{1}\, m\%_{(DT)} + \underline{4} m\%_{(CT4)} + \underline{2w}\, m\%_{(qTw)} + \text{etc.}$$

The moles of tritium in the container are the sum of the moles of tritium contained in each component in the container. This is calculated using the following equation, which is the ratio of the tritium atoms in the component chemical formula to the tritium atoms in T_2; i.e., 2. In the formula qT_w/T_2 this would be w/2, or in HT/T_2 this would be1/2.

$$\text{Moles of Tritium} = [(m\%_{(T2)}/100) \times (T_2/T_2) \times n_{(\text{Moles Total})}] + [(m\%_{(HT)}/100) \times (HT/T_2) \times n_{(\text{Moles Total})}] + [(m\%_{(DT)}/100) \times (DT/T_2) \times n_{(\text{Moles Total})}] + [(m\%_{(CT4)}/100) \times (CT_4/T_2) \times n_{(\text{Moles Total})}] + [(m\%_{(qTw)}/100) \times (qT_w/T_2) \times n_{(\text{Moles Total})}] + \ldots$$

where x/T_2 is the number of moles of tritium per mole of the tritium component.

Factoring out $n_{(\text{Moles Total})}/100$ and rearranging, the equation becomes

$$(\text{Moles of tritium} \times 100)/(n_{(\text{Moles Total})}) = (m\%_{(T2)} + m\%_{(HT)} \times 1/2 + m\%_{(DT)} \times 1/2 + m\%_{(CT4)} \times 2 + m\%_{(qTw)} \times w/2 + \text{etc.})$$
$$= m\%_{(\text{Tritium Per Mole Total})}$$

Rearranging the equation becomes

$$\text{Moles of tritium} = (n_{(\text{Moles Total})} \times m\%_{(\text{Tritium Per Mole Total})})/100$$

The amount of tritium in grams is obtained by multiplying the moles of tritium obtained in the last equation by the gram molecular weight of tritium (6.0321g).

As an example, the grams of tritium in a shipment of research grade tritium are determined using the following process.

An analysis of a shipment of research grade gaseous tritium shows the following:

Component	Percent
T_2	99.704
D_2	0.115
DT	0.079
HT	0.050
HD	0.023
^3He	0.016
Ar	0.012
N_2	0.010
H_2	0.005

The container pressure is 742 mm, temperature is 20°C, and volume is 49.348 liters. This gives a compressibility factor ($z_{(T2)}$) of 1.0006 and the constant, R, i+s 62.3631.

Calculating the percent tritium is as follows:

$$
\begin{aligned}
m\%_{(\text{Tritium Per Mole Total})} &= m\%_{(T2)} + m\%_{(HT)} \text{ x } 1/2 + m\%_{(DT)} \text{ x } 1/2 \\
&= 99.704 + (0.050 \text{ x } 1/2) + (0.079 \text{ x } 1/2) \\
&= 99.704 + 0.025 + 0.0395 \\
&= 99.7685 \text{ percent}
\end{aligned}
$$

The number of moles of gas in the container is calculated by

$$
\begin{aligned}
n_{(\text{Moles Total})} &= PV/zRT \\
&= (742 \text{ x } 49.348)/(1.0006 \text{ x } 62.3631 \text{ x } 293.15) \\
&= 2.002 \text{ moles total}
\end{aligned}
$$

The amount of tritium in grams is

$$
\begin{aligned}
\text{Grams of tritium} &= (n_{(\underline{\text{Moles Total}})} \text{ x } m\%_{(\underline{\text{Tritium Per Mole Total}})} \text{ x } 6.0321)/100 \\
&= (2.002 \text{ x } 6.0321 \text{ x } 99.7685)/100 \\
&= 12.046 \text{ grams tritium}
\end{aligned}
$$

The determination of the amount of tritium is only for the point in time that the sample was analyzed as a result of the decay of tritium. Over time, the number of moles of the gases containing tritium decreases. The pressure and number of moles of non-tritiated gases increase with time due to the ^{3}He produced and the molecules formed by the atoms of other materials released when the tritium decays. Therefore, the mole percent and number of moles of gas in the container and in the sample change with time. Figure C-1 shows an example of the changing moles in a gas mixture.

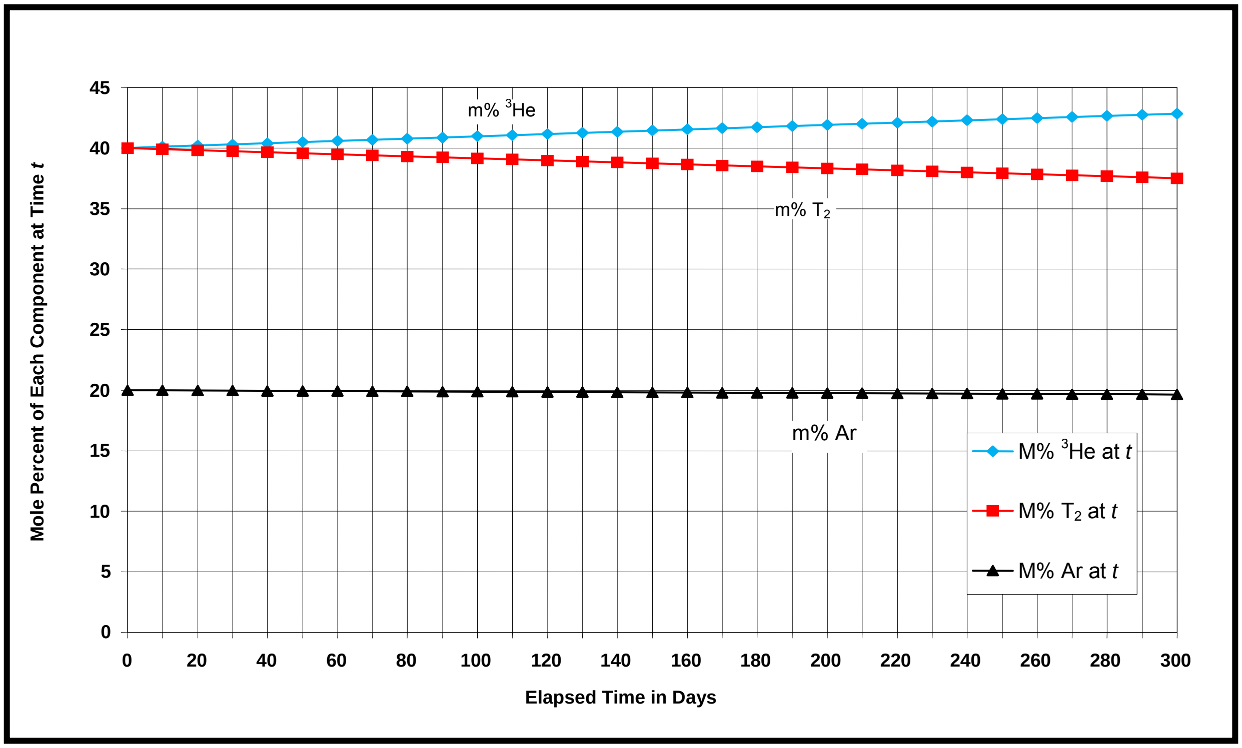

FIGURE C-1. Change in mole percent with time in a mixture of 40% T_2, 40% ^{3}He, and 20% Ar

The total number of moles in the container is known at the time of the sampling but not at the time of the analysis. If the gas analysis is performed on the same day as the sampling, then, since the half-life of tritium is 4500.88 days, the error caused by the decay is small (< 0.015 percent for 1 day, approximately 1 percent in 67 days). Some facilities are equipped to perform the sample analysis within a few minutes or on the same day as the sampling. Others facilities depend upon collecting a gas sample, on the sampling date, for analysis at a later date.

The following equations include three significant dates. The Sample Date is the day the gas sample for analysis is collected, and the pressure, volume, and temperature of the container are measured and recorded to determine the total moles of gas. The Analysis Date is the date the gas sample is analyzed to determine the mole percent of tritium per mole in the sample. The Book Value Date is a date on which the quantity of tritium in the container is known from a previous assay. Additionally,

it assumes a gaseous mixture at the start of 100% tritium and other stable gases; i.e., it does not contain HT, DT, HTO, or other radioactive or non-inert gases.
The process steps are as follows:

1. Calculating the number of moles of gas in the container on the Sampling Date.

2. Determining the percent of tritium in the collected sample on the Analysis Date by gas analysis.

3. Using the two values in steps 1 and 2, the time period, in days, between the two dates and the half-life of tritium, calculating the number of grams of tritium in the container on the Sampling Date.

Once the quantity of tritium in the container on the Sampling Date is known, the quantity of tritium in the container for any other date, including the Book Value Date can be calculated.

The number of moles of gas in the container on the Sampling Date ($n_{(TotCSmplDa)}$) is the sum of the number of moles of the non-tritiated gases present in the container, which do not change with time due to decay, ($n_{(Non\text{-}tritium)}$), plus the number of moles of tritium in the container at the time of the sampling ($n_{(T2CSmplDa)}$). In equation form

$$
\begin{aligned}
PV/zRT &= n_{(TotCSmplDa)} \\
&= n_{(Non\text{-}tritium)} + n_{(T2CSmplDa)}
\end{aligned}
$$

Rearranging the equation it becomes

$$n_{(Non\text{-}tritium)} = n_{(TotCSmplDa)} - n_{(T2CSmplDa)} \qquad \{1\}$$

The total number of moles of material in the container on the Analysis Date, ($n_{(TotCAnlDa)}$) is

$$n_{(TotCAnlDa)} = n_{(Non\text{-}tritium)} + n_{(T2CSmplDa)} \text{ x } e^{((t \ln(0.5))/4500.88)} + 2 \text{ x } (n_{(T2CSmplDa)} - n_{(T2CSmplDa)} \; e^{((t \text{ x } \ln(0.5))/4500.88)})$$

where $n_{(Non\text{-}tritium)}$ = sum of the moles of the other gases
$(n_{(T2CSmplDa)} \; (e^{((t \ln(0.5))/4500.88)})$ = number of moles of T_2 decayed
$2 \text{ x } (n_{(T2CSmplDa)} - n_{(T2CSmplDa)} \; e^{((t \ln(0.5))/4500.88)})$ = number of moles of ^{3}He created
t = time between the Sampling Date and Analysis Date in days

Factoring, the equation becomes

$$n_{(\text{TotCAnlDa})} = n_{(\text{Non-tritium})} + n_{(\text{T2CSmplDa})} \text{ x } (2 - e^{((t \text{ x } \ln(0.5))/4500.88)}) \qquad \{2\}$$

On the day the gas analysis is performed, the mole percent of tritium ($m\%_{(\text{T2MAnlDa})}$) is determined by gas analysis. The equation is

$m\%_{(\text{T2MAnlDa})}$ is equal to the number of moles of tritium on the sampling date decayed to the analysis date; i.e., ($n_{(\text{T2CSmplDa})} \text{ x } e^{((t \text{ x } \ln(0.5))/4500.88)}$) divided by the number of moles in the sampled container on the analysis date; i.e., ($n_{(\text{TotCAnlDa})}$) multiplied by 100 to convert it to percent.

$$m\%_{(\text{T2MAnlDa})} = \{(n_{(\text{T2CSmplDa})} e^{((t\text{x}\ln(0.5))/4500.88)})/n_{(\text{TotCAnlDa})}\} \text{ x } 100$$

where ($n_{(\text{T2CSmplDa})} \text{ x } e^{((t \text{ x } \ln(0.5))/4500.88)}$) = number of decayed moles of T
($n_{(\text{TotCAnlDa})}$) = number of moles in the sample on the Analysis Date
t = time between the Sampling Date and Analysis Date in days

Rearranging the equation, it becomes

$$n_{(\text{TotCAnlDa})} = \{(n_{(\text{T2CSmplDa})} e^{((t \text{ x } \ln(0.5))/4500.88)})/m\%_{(\text{T2MAnlDa})}\} \text{ x } 100 \qquad \{3\}$$

Substituting Equation {1} into Equation {2}, we have

$$n_{(\text{TotCAnlDa})} = n_{(\text{TotCSmplDa})} - n_{(\text{T2CSmplDa})} + n_{(\text{T2CSmplDa})} \text{ x } (2 - e^{((t \text{ x } \ln(0.5))/4500.88)})$$

Rearranging the equation, it becomes

$$n_{(\text{TotCAnlDa})} = n_{(\text{TotCSmplDa})} + n_{(\text{T2CSmplDa})} \text{ x } (1 - e^{((t \text{ x } \ln(0.5))/4500.88)}) \qquad \{4\}$$

Substituting Equation {3} into Equation {4}, we have

$$100\,(n_{(\text{T2CSmplDa})} e^{((t \text{ x } \ln(0.5))/4500.88)})/ m\%_{(\text{T2MAnlDa})} = n_{(\text{TotCSmplDa})} + n_{(\text{T2CSmplDa})} (1 - e^{((t \text{ x}\ln(0.5))/4500.88)})$$

Multiplying through by $m\%_{(T2MAnlDa)}$, it becomes

$$100\,(n_{(T2CSmplDa)}\, e^{((t \times \ln(0.5))/4500.88)}) = m\%_{(T2MAnlDa)}\, n_{(TotCSmplDa)} + m\%_{(T2MAnlDa)}\, n_{(T2CSmplDa)}\, (1 - e^{((t \times \ln(0.5))/4500.88)})$$

Rearranging and factoring, we have

$$n_{(T2CSmplDa)} = \quad (n_{(TotCSmplDa)} \times m\%_{(T2MAnlDa)})/((e^{((t \times \ln(0.5))/4500.88)} \times (100 + m\%_{(T2MAnlDa)})) - m\%_{(T2MAnlDa)})$$

The quantity of tritium in grams is then the number of moles of tritium on the Sampling Date, $n_{(T2CSmplDa)}$,,multiplied by the gram molecular weight of tritium (6.03210 g) or

$$\text{Grams } T_2 \text{ On Sampling Date} = \frac{6.0321 \times (PV/zRT) \times (m\%_{(T2MAnlDa)})}{[e^{((t \times \ln(0.5))/4500.88)} \times (100 + m\%_{(T2MAnlDa)})] - m\%_{(T2MAnlDa)}}$$

If a container has pure tritium mixed with other non-decaying and non-chemically reacting gases on the Book Value Date, sampled on the Sample Date and analyzed on the Analysis Date, then the gram of tritium in the container on the Sample Date can be calculated using the derived formula, the PV/zRT data and the $m\%_{(T2MAnlDa)}$ data measured on the Analysis Date.

Figure C-2 is a graph of the changes in the moles of material taking place in a container of pure T_2 versus time over a period of six tritium half-lives. The graph shows the moles of tritium decreasing from 1.0 and approaching 0.0, the ^{3}He increasing from 0.0 and approaching 2.0, and the total moles increasing from 1.0 and approaching 2.0.

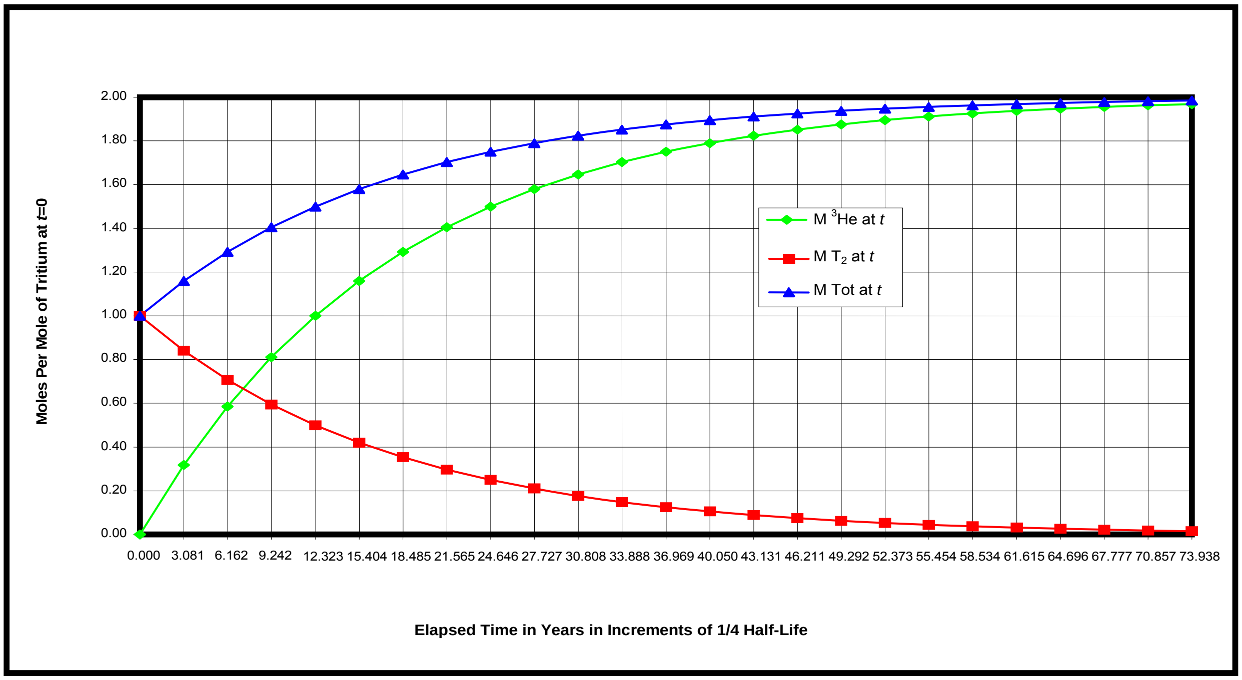

FIGURE C-2. Moles of T_2 and ^{3}He versus time

Similar formulas can be derived for other two-component gases, such as HT mixed with other non-tritiated gases and DT mixed with other non-tritiated gases. See Figures C-3 to C-6 for other examples.

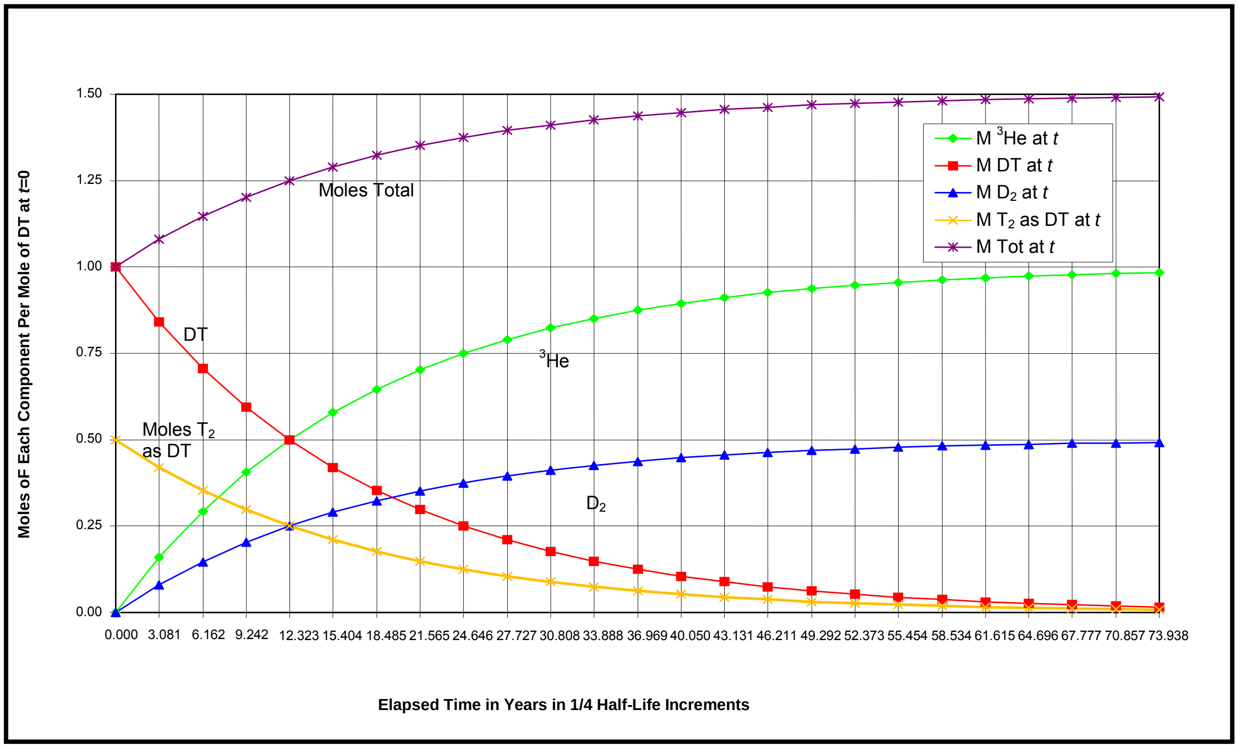

FIGURE C-3: Moles of DT, ^{3}He, and D_2 versus time

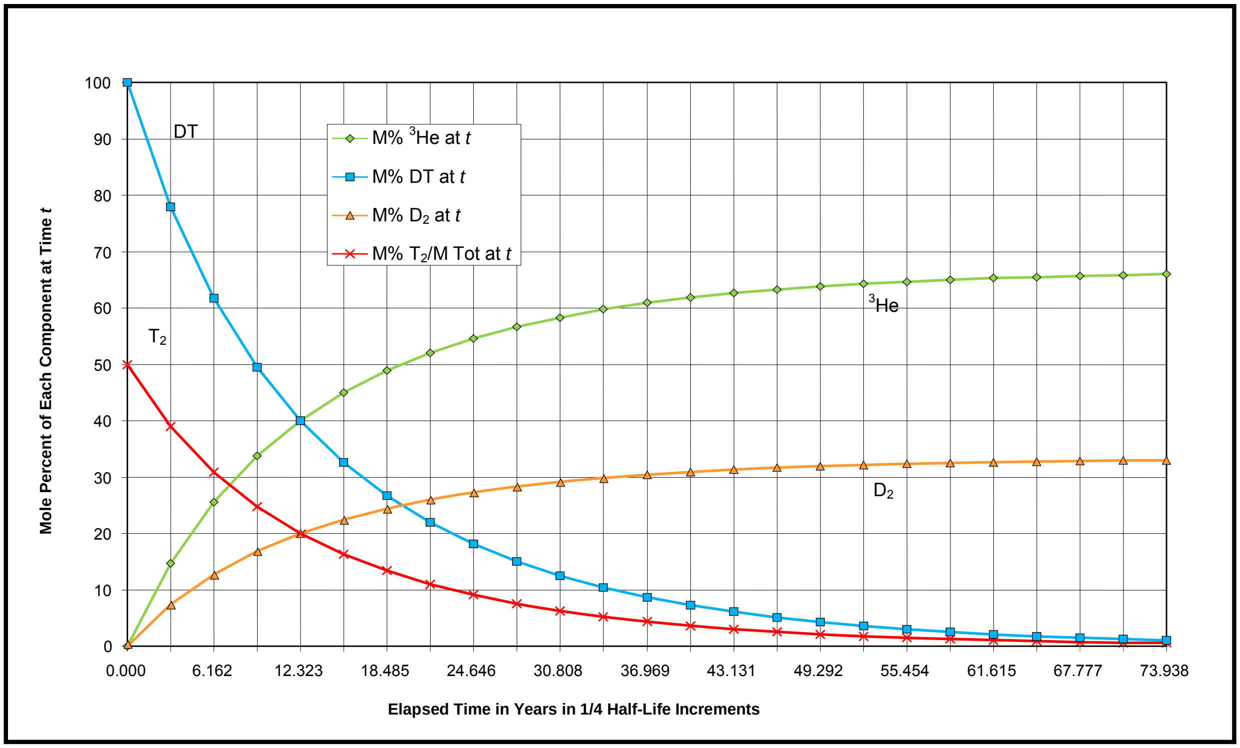

FIGURE C-4: Mole percent of T_2, D_2, DT, and ^{3}He versus time

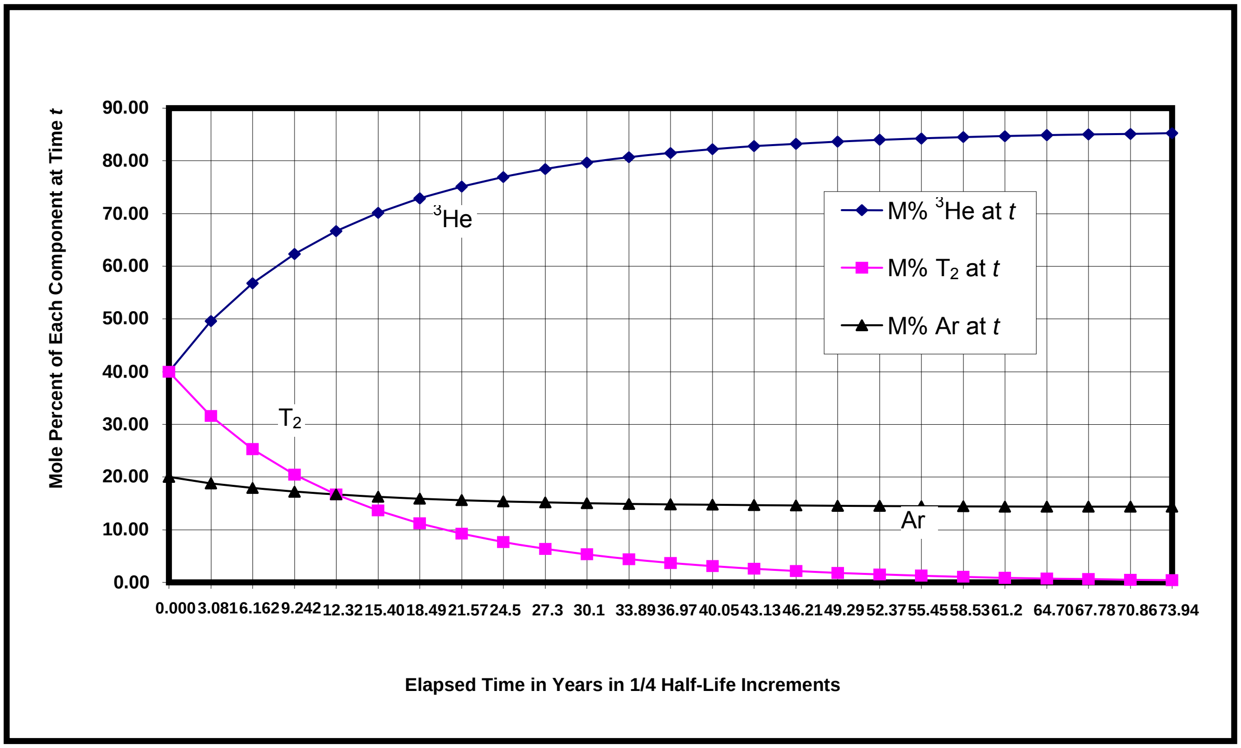

FIGURE C-5: 40% T_2, 40% ^{3}He, and 20% Ar - Changes in mole percent of components versus time

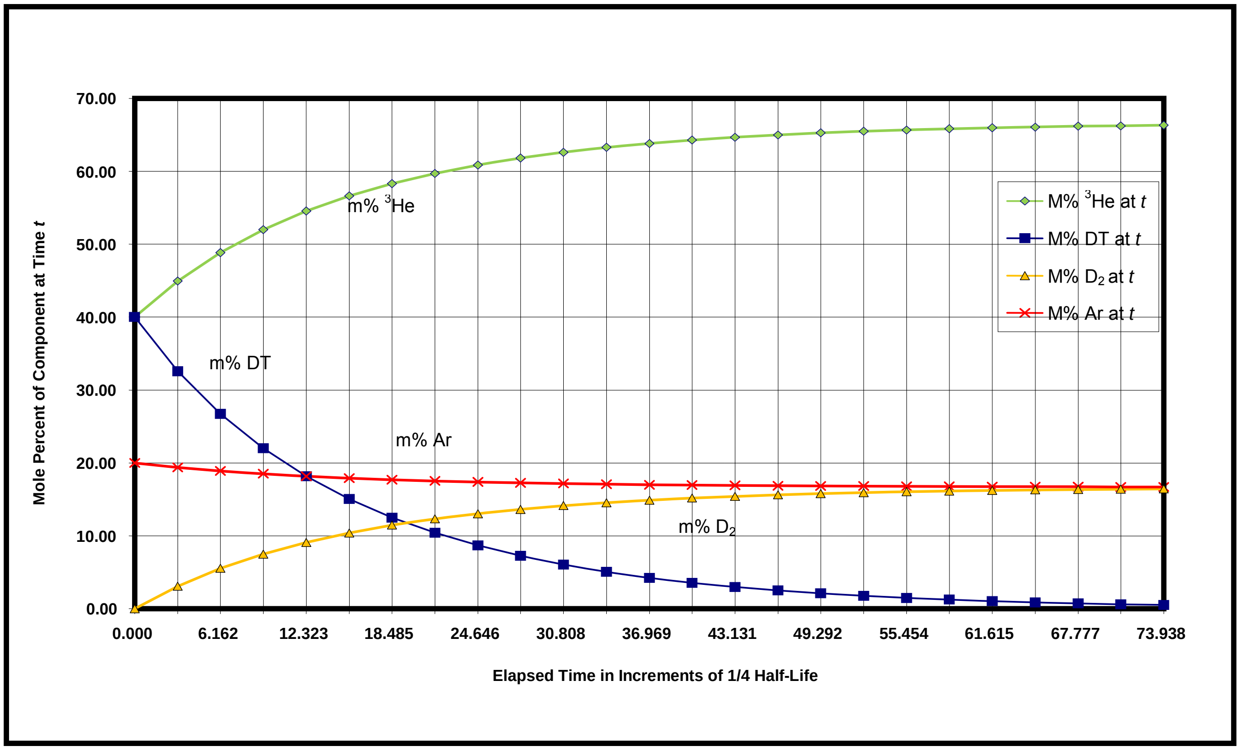

FIGURE C-6: 40% DT, 40% ^{3}He, and 20% Ar, Change in mole percent of each component versus time

C.3 Calorimetry Assay

Calorimetry is the quantitative measurement of heat. A calorimeter is an apparatus for measuring heat quantities generated in or emitted by materials in processes such as chemical reaction, changes of state, and formation of solutions. Heat is generally measured in calories or joules. A calorie is a unit of heat energy equal to the heat energy required to raise the temperature of a gram of water from 14.5 to 15.5°C, at a constant pressure of 1 atmosphere. A calorie is equal to 4.186 joules.

A calorimeter designed to be used in processes that continually generate heat (power sources) and measures power instead of heat is called a Constant Heat Flow (CHF) calorimeter. A CHF calorimeter measures the power (joules/second) of a source not the heat output (joules) of a source. The power is usually measured in Watts, which is a unit of power equal to 1 joule/second.

A radioactive material is a power source, which deposits the energy due to decay in the radioactive material itself and in the materials surrounding the radioactive material. The power generated by the decay of tritium has been measured and is equal to 0.3240 ± 0.0009 watts/gram of tritium.

Mound Laboratory has been the leader in the design, fabrication, calibration, and operation of CHF calorimeters for many years. Mound has specialized in the development of CHF calorimeters to be used in the measurement of radioactive material quantities by measuring their power output. CHF calorimeters are generally designed to meet the specific needs of the items to be assayed and are limited in application by the following:

- Physical size of the calorimeter measurement chamber,
- Wattage range of the measurement system,
- Precision and accuracy of the measurement for the size and wattage range of the item to be measured,
- Throughput or number of samples to be measured per day.

CHF calorimeters have been designed in many different configurations, such as over/under, and twin. Most CHF systems in use today use digital control systems operated by a stored program and are easy to operate. The steps in making a CHF measurement are generally as follows:

- Install a dummy mass in the calorimeter container, pack steel or copper wool around the dummy mass, and install it in the measurement chamber.
- Make a zero baseline run at a wattage level (W_{zbl}), which is at a wattage level greater than the unknown wattage level of the sample to be measured.
- During the baseline run, the digital control system establishes a calorimeter bridge voltage value for a known (W_{zbl}) wattage input.
- Remove the calorimeter container from the measurement chamber, remove the dummy sample from the container, and replace it with the sample to be measured, place it back in the measurement chamber, and make an unknown sample run.
- During the unknown sample run, the digital control system decreases the power in the calorimeter until the bridge voltage is the same as that measured in the zero baseline run.
- The power input to the calorimeter during this unknown sample run (W_{usr}) is measured.
- The power of the sample being measured (W_s) is calculated by subtracting the wattage value measured during the zero baseline run from the wattage measured during the unknown sample run to find the wattage of the sample. In equation form:

$$W_s = W_{zbl} - W_{usr}$$

The calorimeter factor for tritium used at most DOE sites for the purposes of reporting accountable quantities of tritium to DOE is 0.3240 +/- 0.0009 Watts/g of tritium.

CHF calorimetry can be used to measure tritium in solid form. CHF is the most accurate method available for the measurement of tritium quantities if the chamber size and wattage level of the item to be measured are well matched to the specifications of the CHF system being used. CHF systems, however

- Do not provide any information about the different gases present in a container (e.g., HT, DT, H_2, D_2, ^3He),
- Only measure the quantity of tritium in the container,
- Are not currently available for items larger than 11 inches in diameter and 16 inches long, and
- Take several hours to complete a single measurement.

APPENDIX D: CONTAMINATION AND SURFACE ACTIVITY THRESHOLDS

D.1 Appendix D to Part 835 - Surface Radioactivity Values

The data presented in Appendix D are to be used in identifying contamination and high contamination areas as defined in Sec. 835.2(a), identifying the need for surface contamination monitoring and control in accordance with Sec. 835.404, identifying the need for radioactive material controls in accordance with Sec. 835.1101.

Surface Radioactivity Values {1}
[In dpm/100 cm^2]

Radionuclide	Removable {2}, {4}	Total (fixed + removable) {2}, {3}
U-nat, U-235, U-238, and associated decay products	1,000{7}	5,000
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	20	500
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	200	1,000
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above {5}	1,000	5,000
Tritium and tritiated compounds {6}	10,000	See footnote 6

{1} The values in this Appendix, with the exception noted in footnote 6, apply to radioactive contamination deposited on, but not incorporated into the interior of, the contaminated item. Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides apply independently.

{2} As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

{3} The levels may be averaged over one square meter provided the maximum surface activity in any area of 100 cm^2 is less than three times the value specified. For purposes of averaging, any square meter of surface should be considered to be above the surface radioactivity value if (1) from measurements of a representative number of sections it is determined that the average contamination level exceeds the applicable value; or (2) it is determined that the sum of the activity of all isolated spots or particles in any 100 cm_2 area exceeds three times the applicable value.

{4} The amount of removable radioactive material per 100 cm2 of surface area should be determined by swiping the area with dry filter or soft absorbent paper, applying moderate pressure, and then assessing the amount of radioactive material on the swipe with an appropriate instrument of known efficiency. (Note—The use of dry material may not be appropriate for tritium.) When removable contamination on objects of surface area less than 100 cm^2 is determined, the activity per unit area should be based on the actual area, and the entire surface should be wiped. It is not necessary to use swiping techniques to measure removable contamination levels if direct scan surveys indicate that the total residual surface contamination levels are within the limits for removable contamination.

{5} This category of radionuclides includes mixed fission products, including the Sr-90 which is present in them. It does not apply to Sr-90 which has been separated from the other fission products or mixtures where the Sr-90 has been enriched.

{6} Tritium contamination may diffuse into the volume or matrix of materials. Evaluation of surface contamination shall consider the extent to which such contamination may migrate to the surface in order to ensure the surface contamination value provided in this appendix is not exceeded. Once this contamination migrates to the surface, it may be removable, not fixed; therefore, a "Total" value does not apply. In certain cases, a "Total" value of 10,000 dpm/100 cm$_2$ may be applicable either to metals of the types from which insoluble special tritium compounds are formed, that have been exposed to tritium, or to bulk materials to which insoluble special tritium compound particles are fixed to a surface.

{7} These limits apply only to the alpha emitters within the respective decay series.

D.2 Response to Questions and Clarification of Requirements and Processes: DOE Order 5400.5 Chg. 2, Section II.5 and Chapter IV Implementation (Requirements Relating to Residual Radioactive Materials), DOE Office of Assistant Secretary for Environment, Safety and Health, dated November 17, 1995

TABLE 1 SURFACE ACTIVITY GUIDELINES

Allowable Total Residual Surface Activity (dpm/100 cm^2)[4]

Radionuclides[5]	Average[6/7]	Maximum[9/8]	Removable[9]
Group 1 – Transuranics, I-125, I-129, Ac-227, Ra-226, Ra-228, Th-228, Th-230, Pa-231	100	300	20
Group 2 – Th-natural, Sr-90, I-126, I-131, I-133, Ra-223, Ra-224, U-232, Th-232	1,000	3,000	200
Group 3 – U-natural, U-235, U-238, and associated decay products, alpha emitters	5,000	15,000	1,000
Group 4 – Beta-gamma emitters (radionuclides with decay modes other than alpha emission or spontaneous[10] fission) except Sr-90 and others noted above[7]	5,000	15,000	1,000
Tritium (applicable to surface and subsurface)[11]	N/A	N/A	10,000

4 As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by counts per minute measured by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

5 Where surface contamination by both alpha- and beta-gamma-emitting radionuclides exists, the limits established for alpha- and beta-gamma-emitting radionuclides should apply independently.

6 Measurements of average contamination should not be averaged over an area of more than 1 m^2. For objects of smaller surface area, the average should be derived for each such object.

7 The average and maximum dose rates associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/ h and 1.0 mrad/ h, respectively, at 1 cm.

8 The maximum contamination level applies to an area of not more than 100 cm^2.

9 The amount of removable material per 100 cm^2 of surface area should be determined by wiping an area of that size with dry filter or soft absorbent paper, applying moderate pressure, and measuring the amount of radioactive material on the wiping with an appropriate instrument of known efficiency. When removable contamination on objects of surface area less than 100 cm^2 is determined, the activity per unit area should be based on the actual area, and the entire surface should be wiped. It is not necessary to use wiping techniques to measure removable contamination levels if direct scan surveys indicate that the total residual surface contamination levels are within the limits for removable contamination.

10 This category of radionuclides includes mixed fission products, including the Sr-90 that is present in them. It does not apply to Sr-90 that has been separated from the other fission products or mixtures where the Sr-90 has been enriched.

11 Property recently exposed or decontaminated should have measurements (smears) at regular time intervals to ensure that there is not a build-up of contamination over time. Because tritium typically penetrates material it contacts, the surface guidelines in Group 4 are not applicable to tritium. The Department has reviewed the analysis conducted by the DOE Tritium Surface Contamination Limits Committee (“Recommended Tritium Surface Contamination Release Guides,” February 1991), and has assessed potential doses associated with the release of property containing residual tritium. The Department recommends the use of the stated guideline as an interim value for removable tritium. Measurements demonstrating compliance of the removable fraction of tritium on surfaces with this guideline are acceptable to ensure that non-removable fractions and residual tritium in mass will not cause exposures that exceed DOE dose limits and constraints.