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DOE STANDARD

TRITIUM HANDLING AND SAFE STORAGE

U.S. Department of Energy **AREA SAFT**
Washington, D.C. 20585

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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).

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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

1.0 INTRODUCTION

There are several tritium-handling publications, including International Atomic Energy Agency (IAEA) Technical Report Series, Number 324, *Safe Handling of Tritium*, published in 1991, in addition to the French Nuclear Safety Authority (ASN) and U.S. Department of Energy (DOE) publications. Furthermore there are internal documents associated with ITER. Most of the tritium publications are written from a radiological protection perspective. This Standard provides more extensive guidance and advice on the full range of tritium operations, shipping, and storage.

1.1 Purpose

This Standard can be used by personnel involved in the full range of tritium handling, from receipt to ultimate disposal. Issues are addressed at each stage of handling.

1.2 Scope

This Standard provides useful information for establishing processes and procedures for the receipt, storage, assay, handling, packaging, and shipping of tritium and tritiated wastes. It includes discussions and advice on compliance-based issues and adds insight to those areas in which DOE guidance is unclear. It is intended to be a “living document” that is revised periodically. For example, planning for and implementing contamination control as part of normal operation and maintenance activities are important functions in any tritium facility. The best practices from around the DOE complex are planned for inclusion in each revision of this Standard.

1.3 Applicability

DOE facilities range from small facilities engaged in operations using a few millicuries (mCi)to large-scale Hazard Category II Nuclear Facilities.. Guidance in this Standard applies to any scale of operations.

1.4 Key References

- DOE G 151.1-4, *Response Elements.*
- DOE O 151.1C, *Comprehensive Emergency Management System.*
- DOE O 232.2, *Occurrence Reporting and Processing of Operations Information*
- DOE O 243.1B, Chg. 1, *Records Management Program.*
- DOE O 410.2, *Management of Nuclear Materials*
- DOE O 420.1C, *Facility Safety.*
- DOE G 420.1-1A, *Nonreactor Nuclear Safety Design Guide for use with DOE O 420.1C*, Facility Safety.
- DOE O 435.1, Chg. 1, *Radioactive Waste Management.*
- DOE M 441.1-1, *Nuclear Material Packaging Manual.*
- DOE G 441.1-1C, Admin Chg. 1, *Radiation Protection Programs Guide for Use with Title 10, Code of Federal Regulations, Part 835, Occupational Radiation Protection*.
- DOE G 450.4-1C, *Integrated Safety Management System Guide.*
- DOE O 450.2, *Integrated Safety Management.*
- DOE P 450.4A, *Integrated Safety Management System Policy.*
- DOE O 458.1, Admin Chg. 3, *Radiation Protection of the Public and the Environment.*
- DOE O 460.1C, *Packaging and Transportation Safety.*
- DOE G 460.2-1, *Implementation Guide for Use with DOE O 460.2,* Departmental Materials Transportation and Packaging Management.
- DOE M 460.2-1A, *Radioactive Material Transportation Practices Manual.* DOE O 460.2A, *Departmental Materials Transportation and Packaging Management.*
- DOE O 461.1B, *Packaging and Transportation for Offsite Shipment of Materials of National Security Interest.*
- DOE O 461.2, *Onsite Packaging and Transfer of Materials of National Security* Interest.
- DOE O 474.2, Admin Chg. 3, *Nuclear Material Control and Accountability.*
- DOE O 5400.5, Radiation Protection of the Public and the Environment (10-8-09) (Archived Status)
- DOE-HDBK-1001-96, *Guide to Good Practices for Training and Qualification of Instructors.*

- DOE-STD-1020-2012, *Natural Phenomena Hazards Analysis and Design Criteria for Department of Energy Facilities.*
- DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23,* Nuclear Safety Analysis Reports, Change Notice 1, September 1997.
- DOE-STD-1070-94, *Guide for Evaluation of Nuclear Facility Training Programs.*
- DOE-STD-1098-2008, Chg. 1, *Radiological Control.*
- DOE-STD-1111-2013, *DOE Laboratory Accreditation Program Administration.*
- DOE-STD-1120-2005, Volume 2, *Integration of Environment, Safety, and Health into Facility Disposition Activities.*
- DOE-STD-1121-2008, *Internal Dosimetry.*
- DOE-STD-3009-2014, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analyses.*
- DOE-HDBK-3010-94, Change Notice 1, Volumes 1 and 2, *Airborne Release Fraction/Rates and Respirable Fractions for Nonreactor Nuclear Facilities.*
- DOE/TIC-11268, *A Manual for the Prediction of Blast and Fragment Loading on Structures.*
- DOE/MLM-3719, *Health Physics Manual of Good Practices for Tritium Facilities.*
- DOE Office of Nuclear and Facility Safety, Technical Notice 94-01, *Guidelines for Valves in Tritium Service.*
- DNFSB Recommendation 2005-1, *Nuclear Material Packaging.*
- DNFSB Letter dated August 19, 2011, Transmitting the Staff Report *Review of Safety Basis, Tritium Facilities, Savannah River Site.*
- DNFSB/TECH 34, *Confinement of Radioactive Materials at Defense Nuclear Facilities.*
- American National Standard. N 13.14 1994. *Internal Dosimetry Programs for Tritium Exposure -Minimum Requirements*
- ANSI N13-1-1999, *Sampling and Monitoring of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities*
- ANSI N13.12-1999, *Surface and Volume Radioactivity Standards for Clearance.*
- ANSI N14.5-2014, *Leakage Tests on Packages for Shipment.*
- ANSI Z88.2, *Practices for Respiratory Protection.*
- ASCE 7-95, *Minimum Design Loads for Buildings and Other Structures.*
- ASME, Nuclear Quality Assurance-1, multiple versions.
- ASN, Tritium White Paper, July 8, 2010 French Nuclear Safety Authority.

- 10 CFR Part 20, *Standards for protection against radiation.*
- 10 CFR Part 71, *Packaging and transportation of radioactive material.*
- 10 CFR 71 Subpart H, *Quality Assurance*
- 10 CFR Part 830, *Nuclear safety management.*
- 10 CFR Part 835, *Occupational radiation protection.*
- 36 CFR, Chapter XII, *National Archives and Records Administration.*
- 40 CFR Part 61, *National emission standards for hazardous air pollutants*.40 CFR Part 261, *Identification and listing of hazardous waste.*40 CFR Part 262, *Standards applicable to generators of hazardous waste.*
- 40 CFR 302.4, *Designation of hazardous substances.*
- 49 CFR Part 172, *Hazardous Materials Table, Special Provisions, Hazardous Materials Communication, Emergency Response Information, Training Requirements, and Security Plans.*
- 49 CFR Parts 100-180, *DOT Hazardous Materials Regulations.*
- 49 CFR 171.8, *Definitions and Abbreviations*.
- 49 CFR Part 173, *Shippers – general requirements for shipments and packagings.*
- 49 CFR Part 177, *Carriage by public highway.*
- 49 CFR Part 178, *Specifications for packaging's.*
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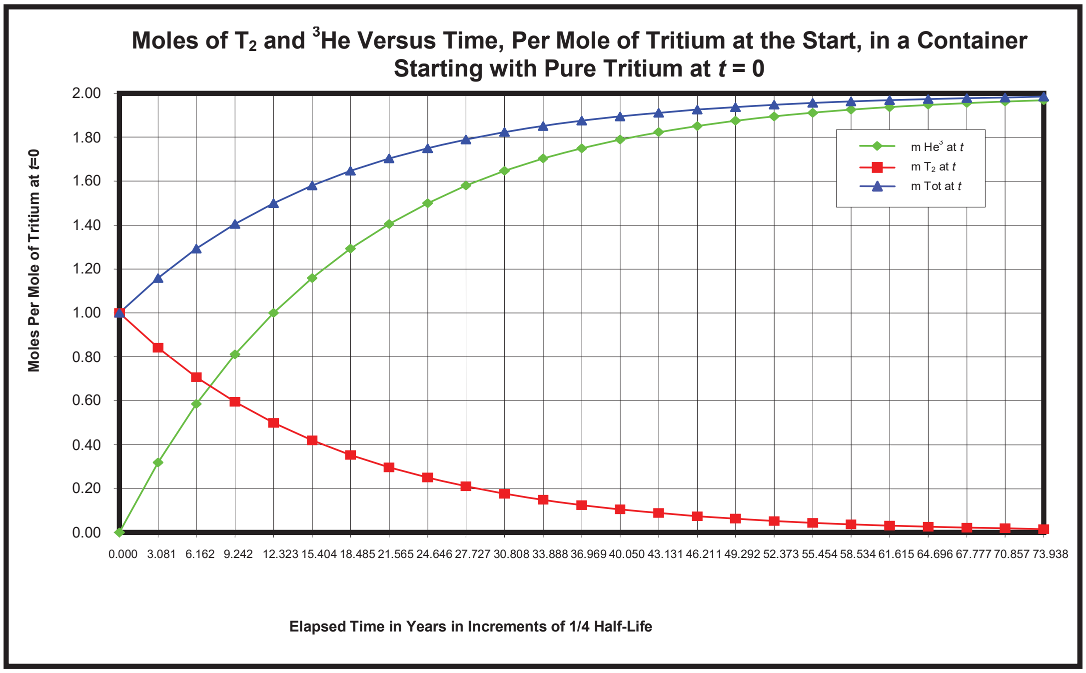
2.0 TRITIUM

Isotopes are elements that have the same atomic number (same number of protons in the nucleus) but different atomic mass (i.e., the total of protons and neutrons in the nucleus). There are three naturally occurring isotopes of hydrogen. Ordinary hydrogen, referred to as protium ($^{1}_{1}H$, atomic mass of 1), is the most abundant element in the universe and has one proton in the nucleus. Heavy hydrogen, referred to as deuterium ($^{2}_{1}H$ or D, atomic mass of 2), makes up about 0.015 percent of the hydrogen, and has one proton and one neutron in the nucleus. Radioactive hydrogen, referred to as tritium ($^{3}_{1}H$ or T, atomic mass of 3), has one proton and two neutrons in the nucleus. Refer to Appendix A for basic information on tritium, its properties, and compounds.

2.1 Radioactive Properties

Tritium is a beta emitter. It decays to ^{3}He by emitting a beta particle (electron) and an antineutrino from one of the neutrons in the nucleus. The energy of the beta particle varies from 0 to 18.6 kiloelectron-volts (keV) with an average energy of 5.69 keV. For scientific purposes, the generally accepted value for the half-life of tritium, as measured by Mound Laboratories, is 12.323 ± 0.004 years (4500.88 ± 1.46 days). For DOE accountability purposes, the half-life of tritium is 12.33 +/- 0.06 years. Figure 2-1 shows the rate of decay of one mole of tritium over six half-lives.

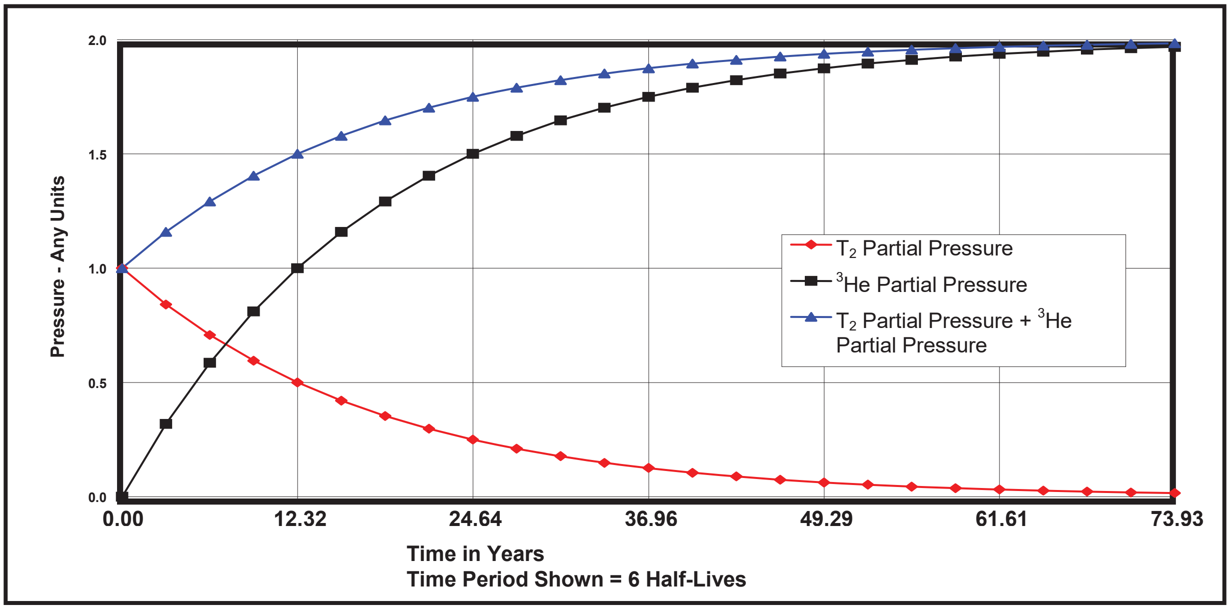


1) Figure 2-1: the rate of decay of one mole of tritium over six half-lives

2.2 Physical Properties

Tritium gas is colorless, odorless, tasteless, and radioactive. Tritium has a high coefficient of diffusion. It readily diffuses through porous substances such as rubber and can also diffuse through metals.

As tritium decays in a container of constant volume at a constant temperature, the tritium partial pressure decreases and the partial pressure of ^{3}He increase. The pressure in the container approaches twice that of the original container pressure. The rate of pressure change over time is shown in Figure 2-2.



2) Figure 2-2: Pressure versus time in a container of tritium

Other properties of tritium are listed below. Additional characteristics are given in Appendix A.

- Atomic weight = 3.01605
- Gram molecular weight = 6.03210
- 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
- Half-life = 12.323 +/− 0.004 years

In this Standard, tritium in the form of the oxide (HTO, DTO, and T_2O), unless otherwise specified, is HTO. Likewise, tritium in its elemental form (HT, DT, and T_2) is HT.

2.3 Chemical Properties

2.3.1 General Properties

The electronic configuration of tritium is the same as protium and deuterium. The chemical properties of the isotopes are also the same. The rates of reaction vary for the different isotopes due to the difference in the atomic masses. Additionally, the energy provided by the radioactive decay of tritium provides the activation energy required so that some reactions will occur with tritium that will not occur with deuterium or hydrogen.

Hydrogen is present in almost all materials. If tritium is present in a material containing hydrogen, the tritium atoms will exchange with hydrogen atoms to form a tritiated molecule of the material.

2.3.2 General Behavior

A chemical restatement of Newton's Third Law of Motion, Le Chatelier's Principle, states that when a system at equilibrium is subjected to a perturbation, the response will be such that the system eliminates the perturbation by establishing a new equilibrium. When applied to situations when the background tritium levels are increased in nature the reactions will be shifted to the right in order to adjust to the new equilibrium conditions by readjusting to the isotopic ratios.

$$CH_4 + 2T_2 \longleftrightarrow CT_4 + 2H_2$$

$$2H_2O + T_2 \longleftrightarrow 2HTO + H_2$$

The inverse situation also applies in that, when the background tritium levels are decreased, the reactions will be shifted back to the left, by again readjusting to the isotopic ratios; i.e.,

$$CH_4 + 2T_2 \longleftrightarrow CT_4 + 2H_2$$

$$2H_2O + T_2 \longleftrightarrow 2HTO + H_2$$

When Le Chatelier's Principle is applied to a sequential set of reactions like those depicted above, it shows that exchange reactions tend to behave as springs, constantly flexing back-and-forth, readjusting to changing energy requirements, in a constantly changing attempt to establish a new set of equilibrium conditions. Since elemental hydrogen, regardless of its form can be expected to dissolve to some extent in virtually any material, Le Chatelier's Principle can be applied to the solubility reactions as well on solubility reactions.

2.3.3 Behavior Model

Starting with the assumption that all three hydrogen isotopes are in equilibrium with each other, in the nominal isotopic ratios and applying Le Chatelier's Principle. From both, we get the relationship,

$$2H_2 + 2D_2 + 2T_2 \longleftrightarrow H_2 + HD + D_2 + HT + DT + T_2$$

In the Environment, virtually all of the tritium exists as water vapor. Applying this situation for the natural conversion to water and/or water vapor gives

$$2H_2O + 2D_2O + 2T_2O \longleftrightarrow H_2O + HDO + D_2O + HTO + DTO + T_2O$$

Furthermore, assume that the surfaces of all bound objects are coated with a series of mono-molecular layers of water vapor and that the innermost layers of water vapor are very tightly bound to the actual surface, that the intermediate layers of water vapor are relatively tightly to relatively loosely bound, and that the outermost layers of water vapor are very loosely bound.

When an overpressure of tritium is added to the system, a perturbation is introduced and Le Chatelier's Principle indicates that the tritium levels in the mono-molecular layers of water will be shifted to the right. Tritium is incorporated first into the loosely bound, outer layers, then into the intermediate layers, and finally into the very tightly bound, near surface layers.

When the overpressure is removed, the system experiences a new perturbation. In this case, however, the perturbation is in the negative direction, and the system becomes the entity that contains the excess tritium. Le Chatelier's Principle, in this case, indicates that the tritium levels in the mono-molecular layers of water will be shifted back to the left. The tritium that had previously been incorporated into the mono-molecular layers now begins to move out of the layers, in an attempt to return to background levels. The movement of tritium into the mono-molecular layers of water vapor is generically referred to as "plate-out." The movement of tritium out of the mono-molecular layers of water vapor is generically referred to as "outgassing."

2.3.4. Plate-Out

When the concentration gradients have been small and/or the exposure times have been short, only the outermost, loosely bound, mono-molecular layers of water vapor will be affected. Under such circumstances, the surface contamination levels will range from no detectable activity to very low levels; that is, up to a few tens of disintegrations per minute per 100 square centimeters (dpm/100 cm^2). Since only the outermost mono-molecular layers are affected, and since these layers are easily removed by a simple wiping, the mechanical efforts expended to perform decontamination on such surfaces will, if any, be minimal. When the concentration gradients have been relatively large and/or the exposure times have been relatively long, the affected mono-molecular layers will range down into the intermediately bound layers (i.e., the relatively tightly to relatively loosely bound layers). Under such circumstances, the surface contamination levels will range from relatively low to relatively high (i.e., from a few hundred to a few thousand dpm/100 cm^2). Because the tritium has now penetrated beyond those levels that would normally be easily removed, mechanical efforts expended to decontaminate such surfaces will become more difficult. When the concentration gradients have been large and/or the exposure times have been long, the affected mono-molecular layers will range all the way down into the very tightly bound layers. The tritium will have penetrated down into the actual surface of the material. Under such circumstances, the surface contamination will range from relatively high to very high levels (i.e., from a few tens of thousands to several hundred thousand dpm/100 cm^2), and that mechanical efforts expended to decontaminate such surfaces could be very difficult.

2.3.5 Outgassing

The phenomenon of outgassing is rarely a problem under the first of the exposure situations described above (i.e., situations in which the concentration gradients have been small and/or the exposure times have been short). However, when systems that have been exposed to even small amounts of tritium for long-to-very-long periods of time are suddenly introduced to room air, or any sudden change in its equilibrium situation, reactions can be thought of as springs, and the initial phenomenon of outgassing can be described as damped harmonic motion. Under such circumstances, therefore, a relatively large, initial “puff” of HTO will be released from the mono-molecular layers of water vapor, followed by a relatively long, much smaller trailing release. Because several curies of HTO can be released in a few seconds, and several tens of curies can be released in a few minutes, the speed of the “puff” portion of the release is quick. The duration of the trailing portion of the release can be lengthy. Depending on the concentration gradients involved and/or the time frames involved in the plate-out portion of the exposure, the trailing portion of the release can easily last from several days to several months or even years. The general derivation of tritium room concentration discussed in section 4.1.2 includes terms associated with a long outgassing period although they are commonly ignored in most practical applications. As the trailing portion of the release asymptotically approaches zero, the outgassing part of the release becomes too small to measure on a real-time basis, and the tritium levels involved in any given release can only be measured by surface contamination measurement techniques.

2.3.6 Material Contamination

When an overpressure of tritium is added to the system (i.e., the surface of an idealized material), Le Chatelier’s Principle indicates that the tritium levels in the mono-molecular layers of water will be shifted to the right; that is,

$$2H_2O + 2D_2O + 2T_2O \rightarrow H_2O + HDO + D_2O + HTO + DTO + T_2O$$

Tritium is incorporated first into the loosely bound, outer layers, then into the intermediate layers, and finally into the very tightly bound, near-surface layers. As the tritium loading in the near-surface layers builds, the disassociation processes that proceed normally as a result of the tritium decay make an overpressure of tritium available in a mono-molecular form (i.e., as T). Relative to the normal amounts of elemental hydrogen that can be expected to be dissolved in the material, the availability of excess tritium in the mono-molecular form represents a different type of perturbation on a system, and the available tritium begins to dissolve into the actual surface of the bulk material. As the local saturation sites in the surface of the bulk material begin to fill, the tritium dissolved in the surface begins to diffuse into the body of the bulk material; at that point, the behavior of the tritium in the body of the bulk material becomes totally dependent on the material in question. Elemental hydrogen, regardless of its form (H2, D2, T2, and all combinations thereof), can be expected to dissolve to some extent in virtually all materials. As a general rule, the solubility of tritium in pure metals and/or ceramics has a minimal effect, at normal room temperatures and pressures, except for the possibility of hydrogen embrittlement. For alloyed metals, such as stainless steel, similar considerations apply, again, at normal room temperatures and pressures. For alloyed metals, however, additional consideration is given to the possible leaching of impurities from the alloyed metal, even at normal room temperatures and pressures. In LP-50 containment vessels, for example, the formation of relatively large amounts of tritiated methane (i.e., up to 0.75 percent mole percent of CT4) has been noted after containers of high-purity tritium have been left undisturbed for several years. The formation of the tritiated methane, in this case, has long been attributed to the leaching of carbon from the body of the stainless steel containment vessel.

Under increased pressures (e.g., from a few tens to several hundred atmospheres), however, the general rules no longer apply for, in addition to the possibility of hydrogen embrittlement and possible leaching effects, helium embrittlement is also possible. Helium embrittlement tends to occur as a result of the dissolved tritium decaying within the body of the material, the resultant migration of the helium-3 atoms to the grain boundaries of the material, the localized agglomerations of the helium-3 atoms at the grain boundaries, and the resultant high-pressure build-ups at these localized agglomerations. Under increased temperature situations, the matrix of solubility considerations becomes even more complicated because virtually all solubility reactions are exponentially dependent on temperature. In the case of diffusional flow through the walls of a containment vessel, for example, it can be assumed that steady-state permeation will have been reached when

$$\frac{D * t}{L} = 0.045$$

where D = the diffusion rate in cm^2/sec, t = the time in seconds, and L = the thickness of the diffusion barrier. For type 316 stainless steel, the value for the diffusion rate is

$$D = 4.7 * 10^{-3} e^{(-\frac{12,900}{RT})}$$

and the corresponding value for R, in the appropriate units, is 1.987 cal/mole K. With a nominal wall thickness of 0.125 inches (i.e., 0.318 cm), this Equation indicates that it will take about 875 years to reach steady-state permeation, at a temperature of 25 ^{o}C. At 100 ^{o}C, the time frame will be reduced to about 11-years, and at 500 ^{o}C, it only takes about 12-hours.

2.4 Biological Properties

2.4.1 Gaseous Tritium and Tritiated Water

The body does not readily absorb H_2, HT, HD, D_2, DT, or T_2 from inhaled gases or through the skin. If inhaled in elemental form, almost all tritium in the gas is exhaled. Only a very small fraction is retained in the lungs.

Tritium in the form of water (HTO, DTO, and T_2O) is absorbed through the skin and in the lungs from inhaled gases. Tritium in water form is readily retained in the body and remains with a biological half-life of approximately 10 days. Due to the body's ready adsorption of tritium in the form of tritiated water, exposure to tritiated water in air is on the order of 10,000 times more hazardous than exposure to gaseous tritium (HT, DT, and T_2).

The Derived Air Concentration (DAC) for tritium is the airborne concentration that, if inhaled over a one-year period, would produce approximately a 5-rem dose to the "average" worker. The DAC is derived by the formula:

DAC = ALI/2400 DAC = derived air concentration (µCi/ml)
ALI = annual limit on intake (Ci)
2400 = breathing volume for the average worker over 1 year in m^3
= .02 m^3/min x 60 min/hr x 40 hr/wk x 50 wk/yr

The DACs for elemental tritium and tritiated water[1],[2] are listed in Table 2-1.

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[1] U.S. Environmental Protection Agency's Federal Guidance Report No. 11, Table 1 *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, September 1988

[2] 10 CFR Part 835, Appendix A

	µCi/ml	Bq/m^3
HT	2E-01	9E9
HTO	2E-05	7E05

1) Table 2-1: Derived air concentrations for tritium and tritiated water

2.4.2 Special Tritium Compounds

Special tritium compounds (STCs) are defined as any compound, except for H_2O and H_2, that contains tritium, either intentionally (e.g., by synthesis) or inadvertently (e.g., by contamination mechanisms). Examples of STCs are metal tritides and organically bound tritium.

Special tritium compounds differ from the more common forms of tritium (elemental tritium and tritium oxide) in a variety of characteristics, including particle sizes, chemical behavior, and biological properties when ingested, absorbed, or inhaled into the human body. The physical properties of special tritium compounds may make their detection, characterization, and subsequent assessments of hazards and exposure effects (i.e., individual dose assessments) difficult. As a result of these unique behaviors, specific guidance has been developed (see Appendix E) to facilitate the development and implementation of appropriate protective programs.

Although the DOE radiological protection community has been aware of STCs for many years, to date their impact has been limited by the design features that are incorporated into DOE facilities that handle significant quantities of tritium. These design features include various forms of material containment and control, such as gloveboxes and high-efficiency particulate air (HEPA)-filtered ventilation systems that effectively prevent significant releases of STCs to occupied areas of the workplace or the environment. Recently, sensitivity to STC contamination has been increased as a result of recent DOE activities involving decontamination and decommissioning of older facilities. These activities may compromise the effectiveness of the installed design features and allow releases of STC contamination to the surrounding areas. Such releases may cause exposures to individuals in the area and releases of STCs to the environment, both on- and off-site. In light of such experiences, DOE suggests that individuals setting up programs for radiological control of STCs contact sites that either work, with or have worked, with tritium to learn of their experiences with STCs.

Appendix E (formerly DOE-HDBK-1184-2004) has been prepared by DOE to assist its employees and contractors in developing and implementing radiation protection programs that will provide adequate protection against the hazards presented by special tritium compounds.

2.5 Preferred Forms

Most tritium in the DOE complex exists as a gas, in the form of tritiated water, or as a metal tritide. The preferred form of tritium is dependent upon its use in a process, length of storage, or its classification as a waste.

2.5.1 Characterization of Tritium Forms

2.5.1.a Gaseous Tritium

The use, transfer, storage, and shipment of gaseous tritium at or near atmospheric pressure have been safely used for over fifty years in the DOE complex. Gaseous tritium at or near atmospheric pressure occupies 22.414 L/mole at 0°C, and approximately 24.2 L/mole at room temperature, and requires approved packages for shipment in either Type A or B quantities. If the containers are not properly designed or if they are damaged, the gas can leak from the container into the environment.

Gaseous tritium at ambient pressure is easily handled by most gas handling systems and is a good source for general-purpose use. At low pressure and temperature, the tritium does not penetrate deeply into the container wall. Helium and tritium embrittlement of the container wall is not a significant issue at low pressures even after several years of exposure. As tritium decays, the pressure in the container increase (see Figure 2-2) due to the generation of the monatomic gas ^{3}He. This pressure increase, at most, would only be double the initial pressure. This factor has to accounted for during the initial design of the vessel so it does not become an issue later.

Gaseous tritium at high pressure takes up less space but is more difficult to contain in part due to the potential for tritium and helium embrittlement of the vessel materials. This embrittlement increases the probability of a tritium leak or catastrophic container failure. Unloading high-pressure gas requires specifically designed systems and experienced, skilled operators.

2.5.1.b Metal Tritides

Metal tritides reduce the overall volume of the stored tritium, but some of the finely divided metals used are pyrophoric. Some metals form low-melting-point alloys with the materials used in the construction of the metal tritide containers. Others require extremely high temperatures in order to recover tritium from the material. Depending on the individual design, the pressure in some vessels, containing tritium as a metal hydride, may increase by more than a factor of two during the life of the vessel. This can become problematic for shipping and disposal. Tritium as a hydride occupies very little volume but the decay helium-3, in some hydrides, will pressurize any open volume in the vessel. If the vessel is designed with little expansion volume then the helium-3 pressure can be significantly more than twice the loading pressure.

2.5.1.c Tritiated Water

Tritium in the form of T_2O may be difficult to store for long periods in part due to its corrosive properties. Experiments with T_2O indicate that pure T_2O is corrosive and as the percentage of T2O in the subject water increases so does its corrosivity. This corrosiveness is likely due to tritium oxide generating free radicals (OH^-) from radiolytic decomposition of water in addition to extra energy from beta decay impinging on surrounding molecules. Additionally, pure T_2O, like distilled H_2O, will dissolve many materials. No data currently exist that quantify the degree of corrosiveness; therefore, there is no basis to definitively state that the U. S. Environmental Protection Agency (EPA) threshold of corrosivity (i.e., a characteristic of hazardous waste), defined in Title 40 of the Code of Federal Regulations (CFR), Section 261.22, is not exceeded. The author believes, however, that only a high-purity product, and not waste, would have a reasonable chance of exceeding this threshold. Pursuant to the RCRA regulations, it is the responsibility of the waste generator to determine if that waste is subject to the hazardous waste requirements [40 CFR 262.11]. A broader discussion of the relationship between tritiated water and hazardous wastes is contained in Section 3.1.3. Note that for transportation purposes, DOT has specific defining criteria for corrosive materials that is different from that of the EPA; see 49 CFR 173.136-137. Dilute tritiated water recovered from tritium removal systems, although not at the EPA threshold has still proven to be somewhat corrosive, due in part to the other constituents. In a severe case, storage of tritiated water recovered from tritium removal systems in liquid form at concentrations as low as a few curies per milliliter has corroded through the weld area of stainless steel vessels after only a few days of exposure. In this specific example, it is probable that the extreme corrosive nature of this dilute tritiated water was due, in large measure, to chlorine contamination of the catalyst in the tritium removal system. This corrosion is evidently inhibited by absorption of the tritiated water on clay or in molecular sieve material.

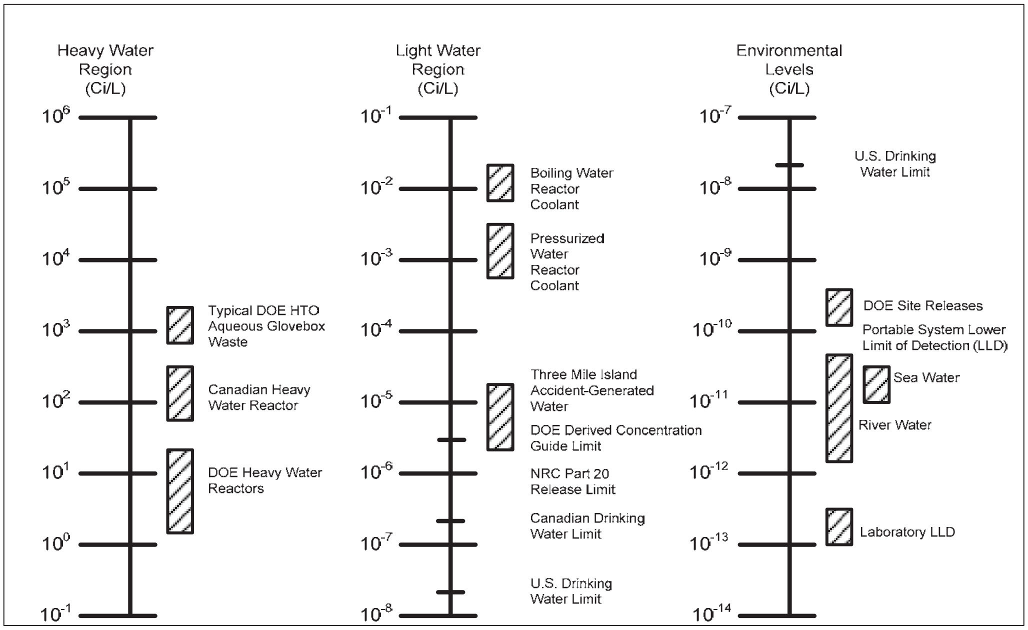
Figure 2-3 provides a comparison of the various concentrations of tritiated water found throughout the nuclear industry.

2.5.2 Identity of Common Forms

Tritium is usually supplied in gaseous or uranium tritide form. Other forms are also available but are not in common use for bulk shipment.

2.5.2.a Gas

In gaseous form, tritium is usually supplied at a purity of 90 to 95 percent tritium (99 percent in research applications) with deuterium and protium as the primary impurities.



3) Figure 2-3: Comparison of aqueous tritium levels found in the nuclear industry

2.5.2.b Metal Tritides

The use of metal tritide storage beds is one of the most convenient ways of handling tritium. The metal tritide beds have different operating parameters and characteristics, and there are advantages and disadvantages in use of the different materials.

2.5.2.b (1) Uranium

Uranium is currently the most useful material for general-purpose tritium storage beds. The equation form for the dissociation pressure of uranium tritide, deuteride, and hydride for the pressure in millimeters of mercury is, $P_{mm} = 10^{(-A/T + B)}$, where A and B are parameters listed in Table 2-3. Washington Savannah River Company (WSRC) has determined that in the hydride transport vessel (HTV), for a uranium-to-tritium ratio of 1:2.9 in the HTV vessel, the equation is

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

At room temperature, tritium in the presence of uranium powder forms uranium tritide. The tritium partial pressure in the bed is very low. As a result, at room temperature the bed acts as a vacuum pump that getters all of the hydrogen isotopes. The impurity gases that may be present, such as ^{3}He, N_2, O_2, or Ar, either remain in the overpressure gas in the bed or react with uranium to form stable compounds. Inert gases, such as Ar, will remain in the overpressure gas, and can be removed by pumping off with a vacuum pump after the pressure has stabilized; however, ^{3}He cannot be pumped off without first heating the bed. N_2 and O_2 will react chemically with the uranium to form stable uranium compounds in the bed, and, therefore, cannot be pumped off at all.

As the temperature of the bed is increased, the tritium partial pressure increases as a function of temperature. Depending upon the U:T ratio, it can reach a pressure of around 500 pounds per square inch absolute (psia) at 600°C. The tritium may be transferred into and out of manifolds, containers, etc., by heating the bed and then cooling it to room temperature. The general form of the equation for the dissociation pressure, P, in millimeters of mercury (mm) for uranium hydride, deuteride, and tritide is:

$$log\ Pmm = -A/T + B$$

or

$$Pmm = 10 - (A/T(K)) + B$$

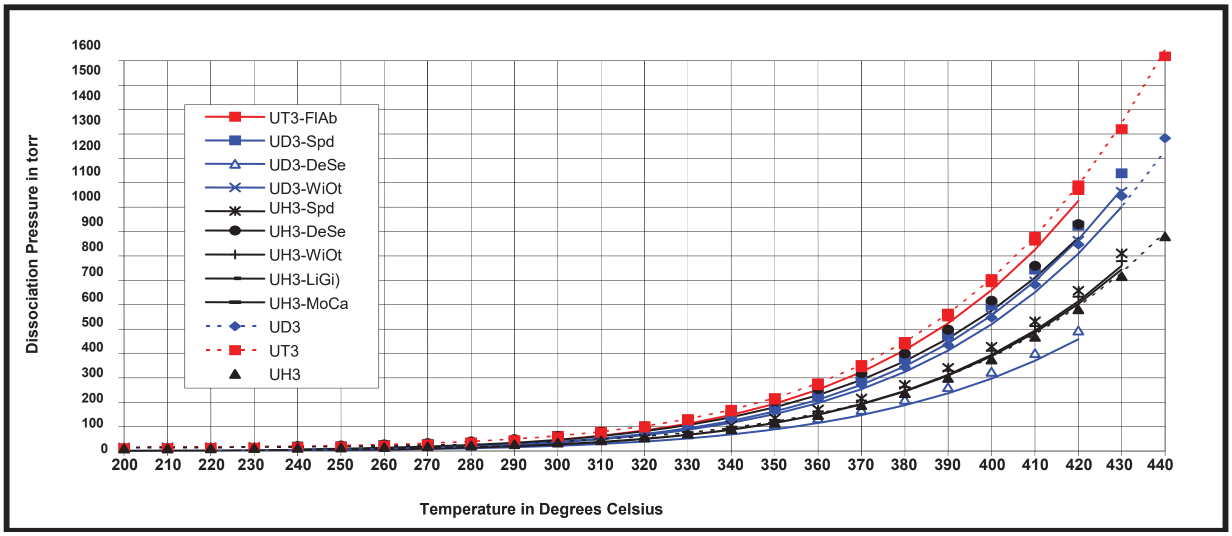
where

$$T = temperature\ (K)$$

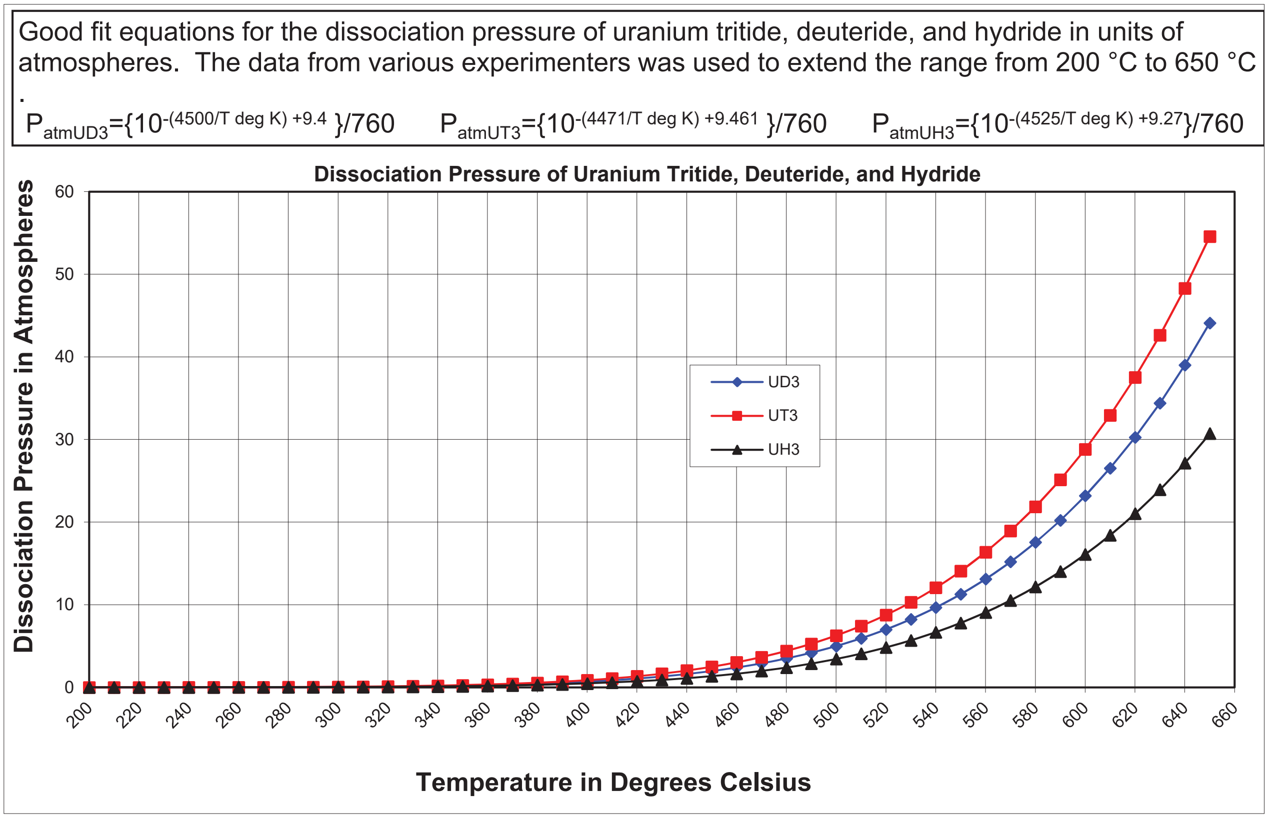
The values for A and B for hydrogen, deuterium, and tritium, determined by several different investigators, are listed in the following paragraphs, and the results are shown in Table 2-2 and plotted in Figure 2-4. Figure 2-5 is a plot of the general characteristics of uranium hydride, deuteride, and tritide.

Metal Tritide	Reference	Temperature Range (°C) Investigated	A (/Kelvin)	B	Temperature (°C) required to generate a pressure of 1 atmosphere
UH_3	Spedding, et al.	260 to 430	4500	9.28	430
	Destriau & Seriot	243 to 412	4255	9.08	415
	Wicke & Otto	200 to 430	4450	9.20	434
	Mogard & Cabane	500 to 650	4480	9.20	435
	Libowitz & Gibb	450 to 650	4410	9.14	432
UD_3	Spedding, et al.	unspecified	4500	9.43	414
	Destriau & Seriot	unspecified	4401	9.01	445
	Wicke & Otto	unspecified	4500	9.40	417
UT_3	Flotow & Abraham	unspecified	4471	9.461	408
	WSRC	unspecified	4038.2	6.074	unspecified

2) Table 2-2: Dissociation pressure equation parameters for uranium hydride, deuteride, and tritide



4) Figure 2-4: Dissociation pressure for uranium, hydride, deuteride, and tritide


Good fit equations for the dissociation pressure of uranium tritide, deuteride, and hydride in units of atmospheres. The data from various experimenters was used to extend the range from 200 °C to 650 °C .

$P_{atmUD3}=\{10^{-(4500/T \text{ deg K}) +9.4}\}/760$ $P_{atmUT3}=\{10^{-(4471/T \text{ deg K}) +9.461}\}/760$ $P_{atmUH3}=\{10^{-(4525/T \text{ deg K}) +9.27}\}/760$

5) Figure 2-5: Plot of a good fit curve for the dissociation pressure of uranium hydride, deuteride, and

Each time the tritium is cycled into the system manifolds, it picks up impurity gases. These impurities collect in the bed overpressure gas and may be pumped off to remove them after each heating/cooling cycle. Active impurity gases, such as oxygen and nitrogen, are irreversibly removed by reaction with the uranium.

Disadvantages to using uranium tritide beds are: 1) uranium powder is pyrophoric; 2) the generation of significant tritium pressure requires a high temperature that results in permeation of tritium through the vessel wall; and 3) the capacity is also permanently reduced by exposure to active impurity gases.

2.5.2.b (2) Palladium

Palladium is a metallic element of Group 8 in the Periodic Table. The symbol for palladium is Pd, the atomic number is 46, the atomic weight is 106.42, and the melting point is 1554.9°C.

At room temperature, palladium absorbs up to 900 times its own volume in hydrogen. It diffuses easily through heated palladium; this is one means of purifying the gas. Finely divided Pd is a good catalyst, and is used for hydrogenation and dehydrogenation reactions.

Palladium powder is currently the second most-used material for general-purpose tritium storage beds. Palladium can be obtained in powdered form and loaded directly into the container used for the metal tritide bed. Palladium was used extensively at both Lawrence Livermore National Laboratory (LLNL) and Sandia National Laboratories–Livermore (SNLL) in the tritium storage beds.

When the tritium is exposed to the powder, it dissolves in the palladium powder with a maximum Pd:T ratio of approximately 0.7. Palladium powder is not pyrophoric, but it has a higher tritium partial pressure than uranium at room temperature.

At room temperature, tritium, deuterium, and protium dissolve in the palladium powder and the tritium partial pressure in the gas over the powder is approximately 50 torr. The overpressure increases as a function of temperature. As the temperature of the palladium is increased by heating the bed, the tritium partial pressure increases as a function of the temperature and reaches a pressure of around 750 psia at 350°C. The general form of the equation for the dissociation pressure, P, in millimeters of mercury (mm) for palladium hydride, deuteride, and tritide is:

$$log\ Pmm = (-A/T + B)$$

or

$$Pmm = 10\ (-A/T + B)$$

where

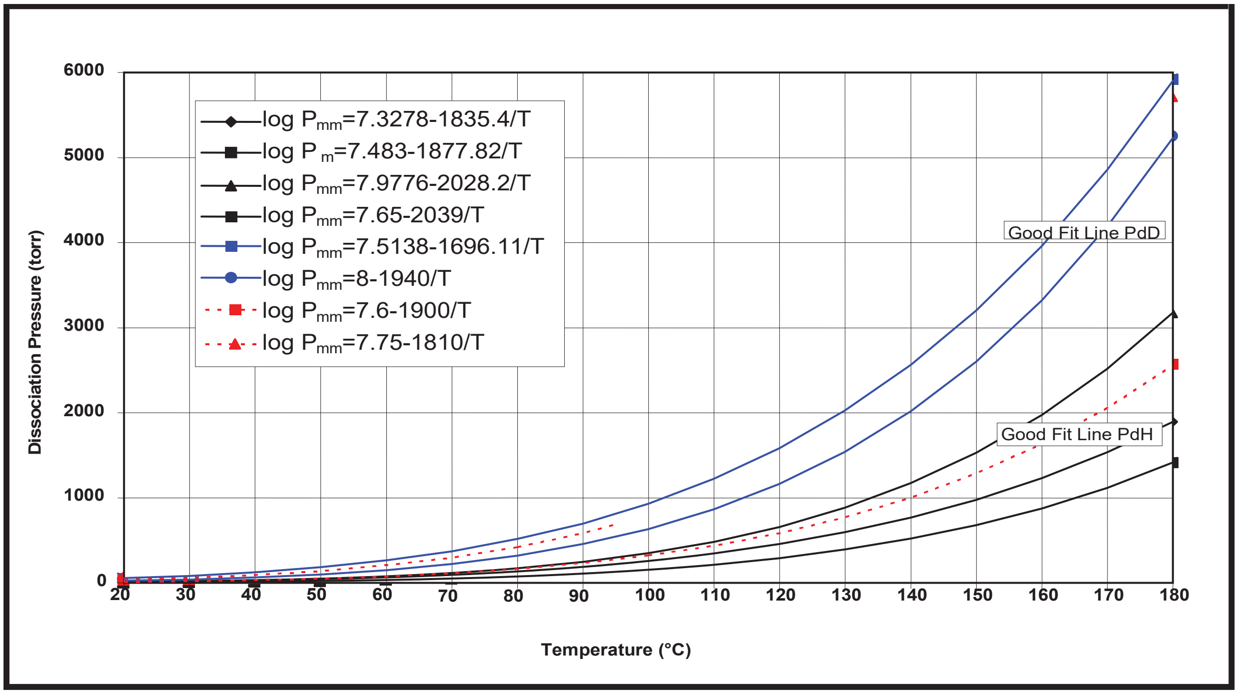
$$T = temperature\ (K)$$

The values for A and B for hydrogen and deuterium determined by different investigators are given in Table 2-3, and the equations developed by the different experimenters over the temperature range they investigated are plotted in Figure 2-6.

Metal Tritide	Reference	Temperature Range (°C) Investigated	A	B
PdH_x	Gillespe & Hall	0 to 180	1835.4	7.3278
	Gillespe & Hall	200 to 300	1877.82	7.483
	Ratchford & Castellan	unspecified	2028.2	7.9776
	Wicke & Nernst	-78 to 175	2039	7.65
PdD_x	Gillespe & Downs	to 300	1696.11	7.5138
	Wicke & Nernst	unspecified	1940	8.00

3) Table 2-3: Dissociation pressure equation parameters for palladium hydride and deuteride

The ^{3}He generated as a result of decay of the tritium absorbed in the palladium is trapped in the palladium and is not released until the bed is heated or until the T:^{3}He ratio reaches a particular value. ^{3}He generated as a result of decay in the overpressure gas is not absorbed in the palladium and remains in the overpressure gas. Most impurities do not react with, and are not gettered by, the palladium powder. These impurities accumulate in the overpressure gas as the bed is used to support operations.



6) Figure 2-6: Dissociation pressure of palladium hydride and deuteride

The generation of significant pressure at low temperature (750 psia at 350°C) is the primary advantage of palladium. The primary disadvantage of palladium is the high partial pressure of tritium over the powder at room temperature (50 torr at room temperature).

2.5.2.b (3) Titanium

Titanium is a metallic element in Group 4 of the Periodic Table. The symbol is Ti, the atomic number is 22, the atomic weight is 47.90, and the melting point is 1660°C. It is a low-cost metal, and can absorb and store tritium in a compact solid form at a tritium pressure of approximately $1E^{-7}$ torr.

Titanium hydride, TiH_2, in powder form, is a black metallic dust that is less prone to spontaneous ignition in air than the parent metal. Finely divided titanium hydride is reported to ignite at 440°C and its dust is an explosion hazard, which dissociates above 288°C. Titanium hydride is used in powder metallurgy, hydrogen production, foamed metals, glass solder, and refractories, and as a gas getter in the electronics industry. Titanium tritide in solid or massive form is stable in air for extended periods of time. Titanium tritide is considered an insoluable STC and must be completely contained. Titanium retains the decay helium up to a concentration of 0.3 He atom per Ti atom. SRS[3], Ontario Power Generation (formerly Ontario Hydro) [4] [5] [6] [7] [8], and Korea Hydro and Nuclear Power Company Limited (KHNP) have selected titanium as their long-term storage medium. The SRS titanium beds have an expected useful life of about 10 years, while the Ontario Power Generation and KHNP beds are expected to be in operation for over 20 years. The Canadians are not space-restricted, and therefore do not load their beds to the degree that SRS does. Additionally, longer times to maximum helium retention ratio (0.3 He:Ti) can be achieved by diluting the tritium concentration with deuterium or protium.

2.5.2.b (4) Zirconium

Zirconium is a metallic element of Group 4 in the Periodic Table. The symbol for zirconium is Zr, the atomic number is 40, and the atomic weight is 91.22. Zirconium is flammable as a powder and melts at 1850°C. Zirconium is a hard, lustrous, grayish metal that is strong and ductile, and is used in alloys, pyrotechnics, welding fluxes, and explosives.

Zirconium hydride ZrH_2 is a flammable gray-black powder and is used in powder metallurgy, nuclear moderators, and as a reducing agent. Finely divided zirconium hydride suspended in air will ignite at 430°C. Zirconium hydride contains about twice as many hydrogen atoms per unit of volume as liquid hydrogen. Massive zirconium hydride is stable in air for extended periods of time at

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[3] Heung, L.K., *Titanium for Long Term Tritium Storage*, WSRC-TR-94-0596, December 1994.

[4] Drolet, T.S., Wong, K.Y., and Dinner, P.J., *Canadian Experience with Tritium – The Basis of a New Fusion Project*, Nuclear Technology/Fusion, Vol. 5, January 1984.

[5] Kherani, N.P., and Shmayda, W.T., *Bulk Getters for Tritium Storage*, Ontario Hydro Research Division.

[6] Shmayda, W.T. and Kherani, N.P., *On the Unloading of Titanium Getter Beds*, Ontario Hydro Research Division, Report No. 85-118-K, October 2, 1985.

[7] Kherani, N.P., and Shmayda, W.T., *Titanium Sponge for Immobilization Tritium Containers*, Ontario Hydro Research Division, Report No. M85-120-K, December 19, 1985.

[8] Noga, J.O., *Investigations of Titanium and Zirconium Hydrides to Determine Suitability of Recoverable Tritium Immobilization for the Pickering Tritium Removal System*, Ontario Hydro Research Division, Report No. 81-368-K, November 12, 1981.

temperatures below 600°C. Additionally, SRNL has conducted research on the effect of over pressurization of hydrogen on Zircaloy-4[9]. Note that zirconium tritide is considered an insoluble STC and must be completely contained.

Certain DOE radioactive zirconium fines (which are destined for disposal) are managed as D001 mixed ignitable wastes. These radioactive zirconium fines are pyrophoric under 40 CFR 261.21(a)(2); i.e., they are capable of causing fire through friction. If zirconium used for tritium storage beds is destined for disposal (i.e., constitutes a waste), the Resource Conservation and Recovery Act (RCRA) hazardous waste characteristic of ignitability must be analyzed, per 40.CFR.261.21. A broader discussion of RCRA hazardous waste is contained in Section 3.1.3.

2.5.2.b (5) Societá Apparecchi Elettrici e Scientifici (SAES) Getters

During the late 1990s, investigations were conducted concerning the use of SAES getters to remove tritium from tritium-contaminated gaseous waste streams. These investigations have concentrated on getters that cracked the gases containing tritium and removed the resulting free tritium from the gas stream. The primary advantage of these getter systems is that tritium is not converted by the tritium removal system to the more radiotoxic tritiated water. Additionally, the tritium can be recovered in gaseous form from the getter, purified, and reused.

The materials tested include those manufactured in the form of pressed pellets that can be used in low–pressure, drop-packed bed reactors designed for the size required by the application flow rate and lifetime requirements. The basic strategy implemented in prototype systems was to crack the molecules on a hot getter and remove the nontritiated reactive impurities that interfere with the performance of the hydrogen gettering alloys. Following purification, the gas is passed through a hydrogen gettering bed to remove the hydrogen isotopes from the gas stream. This philosophy worked well, and has gained acceptance for use in inerted gloveboxes.

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[9] Morgan, Jr., G.A. and Korinko, P.S. *The Adsorption of Hydrogen on Low Pressure Hydride Materials* Conference Proceedings, Material Science and Technology 2011, October 16-20, 2011.

2.5.2.b (6) $LaNi_5$-Based Alloys

The use of lanthanum-nickel hydrides has been a continuing topic of interest; promising results for hydrogen storage have been reported[10]. Promising results were also reported in the literature in 1988[11]. Earlier research at Mound, however, in the 1970s and early 1980s indicated that lanthanum-nickel-based alloys were not appropriate for tritium service due, in part, to disproportionation. When cycled, the $LaNi_5$ had a tendency to separate to form the parent metals (La or Ni) or different alloys. The disproportionation tended to change the pressure, concentration, and temperature properties of the metal/alloy mix and increase the quantity of tritium bound in the heel that was not easily recoverable. However, research at Savannah River National Lab (SRNL) has shown that "Substitution of aluminum for a portion of the nickel in $LaNi_5$ was found to lower the hydrogen plateau pressure and allow LaNiAI alloys to be tailored to specific storage applications." SRNL found that the substitution of aluminum in $LaNi_5$ had the added benefit of stabilizing the alloy against disproportionation. Decay of absorbed tritium to ^3He does cause an "aging" effect, which limits the useful life of LaNiAI alloy as a tritium storage material to less than about 10 years[12]. Subsequently, SRNL determined that LaNiAl. 75 ($LaNi_{4.25}Al_{0.75}$) is a type F rather than a type S dissolution, thereby allowing relaxation of radiological controls in use for this hydride[13]. See Appendix E for further discussion of radiological controls associated with STCs.

2.5.2.c Absorbed Water

Molecular sieve material is used in tritium removal systems for removal of water contaminated with tritium. Systems such as tritium removal systems, effluent recovery systems, and cleanup systems remove tritium from a gas by oxidizing and/or cracking the tritium-containing components. The free tritium then combines with oxygen in the gas stream to form tritiated water. The gas stream is then cooled to room temperature, and the water contained in the gas stream, including the tritiated water, is removed by a molecular sieve trap.

A molecular sieve depending on type and size/mesh, can hold up to 25% water by weight., and the sieve may be regenerated to remove the water so it can be reused. Tritiated water absorbed on

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[10] *$LaNi_5$ Intermetallic Hydride*, extracted from *State-of-the-Art Review of Hydrogen Storage in Reversible Metal Hydrides for Military Fuel Cell Applications*, Gary Sandrock, Ph.D., for the Department of the Navy, Office of Naval Research, N00014-97-M-0001, July 24, 1997.

[11] *Hydrogen Isotope Sorption Properties of $LaNi_3Mn_2$ Alloy as a Candidate for the Tritium Storage Material*, T. Ide et al., Sumitomo Heavy Industries, Ltd. and H. Yoshida et al., Japan Atomic Research Institute, published in *Fusion Technology,* September 1988.

[12] Kirk Shanahan, et al., *Tritium Aging Effects in $LaNi_{4.25}Al_{0.75}$,* WSRC-MS-2002-00564, Rev. 1, November 12, 2002.

[13] Farfán, E.B, LaBone, T.R., Staack, G.C., Cheng, Y-S., Zhou, Y., and Varallo, T.P. *Determination of In Vitro Lung Solubility and Intake-to-Dose Conversion Factor for Tritiated Lanthanum Nickel Aluminum Alloy*. Health Physics, 103(3):249-254; 2012.

molecular sieve is not corrosive and may be stored in this way for long periods without damage to the container wall.

Water contaminated with HTO is also stored on clay. The common method of solidification of tritium-contaminated wastewater for disposal is to solidify the water on clay so that it can be classified as solid waste. Clay will hold approximately 60 percent water by volume. Waste disposal sites generally require the use of 100 percent more clay than required to solidify the water, and, as a result, the water is generally limited to 30 percent of the volume of the clay for waste solidification purposes. Water absorbed on clay is not corrosive and may be stored for long periods without damage to the container wall.

2.5.3 Summary

2.5.3.a Best for Storage Conditions

The decision on storage media is a function of the storage length and frequency of unloadings. Media range from gas (short timeframe, many movements) to titanium (long timeframe, few or no movements) with other media (e.g., uranium) in between. Although the preferred form for storage is a metal tritide and the least desirable form is a liquid, there are always exceptions to this rule. Factors to be considered include:

Solid (Metal Tritides): Unless already in solid form, tritium is not readily available as a solid metal tritide and requires conversion before storage. Metal tritides can store large quantities of tritium without occupying large volumes, but the storage containers are more complex than gaseous storage containers. Depending upon the metallic tritide chosen for storage, there are both advantages and disadvantages. Titanium tritide is very stable, even when exposed to air, but it is more difficult to recover tritium from the titanium than from other metals. Stored as uranium tritide, the tritium can be easily and quickly recovered and provides for the removal of most impurities that might accumulate during storage. However, uranium powder is also pyrophoric, and starts releasing ^{3}He after a few months.

Liquid (T_2O): Tritium is not readily available in water form and requires conversion before storage. Tritium is on the order of approximately 10,000 times more hazardous in oxide form than in elemental form. It takes very little space, but is difficult to store due in part to the potential corrosivity of the water. The tritiated liquid can be solidified on clay, molecular sieves, or polymers prior to disposal. In either case, the final decision is a function of the quantity of tritium and the tritium concentration of the water to be stored.

Gas (T_2): Tritium is readily available in gaseous form. A great deal of experience exists on the design of gaseous tritium storage systems. As a gas it takes up more volume than as a liquid or solid, but can be more easily released to the environment if the tritium container is breached. Gas also presents flammability vulnerability.

2.5.3.b Best for Operations/Process

Solid (Metal Tritides): When used as a gas in research where tritium is issued and returned as a gas, there are advantages to the use of hydride beds for storage. The heating/cooling cycle used to store and recover tritium from the bed results in routine removal of the ^{3}He and other impurities from the tritium supply. It can be reused in other processes at reasonably high purity. Additionally, storage as a metal tritide allows the bed to be used as a pressure generator and, in some cases, eliminates the need for mechanical pumps.

Liquid (T_2O): Unless the process itself uses tritium in the form of water, there are no advantages to storage of tritium in liquid form for operations.

Gas (T_2): Tritium is primarily used in gaseous form, purified in gaseous form, assayed in gaseous form, and is more useful in this form than any other form. As a result, storage in gaseous form for operations is appropriate.

2.5.3.c Best for Disposal Conditions

Disposing of tritium in the form of a liquid waste or gaseous waste is difficult. Generally speaking, waste tritium is converted to solid form so that the material can be disposed of as a solid low-level (radioactive) waste, assuming there is no RCRA hazardous component.

Solid (Metal Tritide): It is possible to dispose of gaseous tritium by converting it to a solid metal tritide. However, the disposal sites require that the metal tritide not be pyrophoric. If it is in particulate form, the metal tritide must be contained to meet disposal site requirements.

Liquid (T_2O): If the waste is in gaseous form, the tritium is normally removed from the gas mixture and reused. If the concentration of tritium in the waste gas is too low to make recovery of the tritium economically worthwhile, the waste gas is sent to an effluent processing system, where the tritium is removed before the gases are released to the environment. Most current effluent processing systems remove tritium from the waste gas by converting it to water. The water is then solidified on molecular sieve, clay, mixtures of clay and cement, or Stergo® superabsorbent (discussed in Section 8.1.4.b (2)), and is then packaged as solid waste and shipped to the disposal site. Noted that tritium contaminated water on molecular sieve will develop increasing pressure over time due to the radiolysis of the water by tritium decay. Depending on the initial pressure, the water loading on the sieve and the tritium concentration in the water, the overpressure can exceed 1.5 atmosphers within months. LANL is developing a path forward concerning transportation and disposal associated with this issue on molecular sieve.
Gas (T_2): Waste disposal sites generally will not accept packages containing pressurized gases or those in which a potential exists for generating 1.5 atmospheres (absolute) of pressure over time.

3.0 BASIC TRITIUM REGULATORY INFORMATION

Due to its more hazardous profile, most of the regulatory interest in tritium is concerned with the oxide form. Figure 2-3 pictorially illustrates various concentrations and regulatory setpoints. The radiological materials inventory for tritium accounting purposes may not coincide with the radiological materials inventory for documented safety analysis (DSA) purposes, which may not coincide with the radiological materials inventory for Environmental Impact Statement (EIS) purposes. This is due to prescribed allowances for excluding various portions of the inventory, as discussed in Sections 3.1 and 3.3. DOE regulations concerning tritium (along with other radionuclides) can be found in 10 CFR Part 830, which contains associated Quality Assurance (QA) Program requirements to be met.

3.1 Tritium Accountability and Environmental Considerations

3.1.1 Radiological Materials Inventory

The Atomic Energy Act of 1954, as amended (AEA) describes three categories of materials: byproduct, source, and special nuclear material (SNM). DOE O 474.2, Admin Chg. 3, *Nuclear Material Control and Accountability describe two categories of accountable nuclear material*: SNM and Other. Tritium is listed in Attachment 2, Table B in the Order, as an "other" category of accountable nuclear material that must be controlled and accounted for financial and nuclear materials management purposes and protected in a graded manner consistent with its strategic and monetary importance. Since tritium is not listed in Table A of DOE O 474.2, Attachment 2, it is not managed as SNM, and Table C, Graded Safeguards Table is not used to determine categorization and attractiveness levels. The reportable quantity of tritium is one gram. The quantity of tritium in accountable items must be reported to the hundredth of a gram (when at or above the reportable quantity level) for NMMSS input. Personnel should always check the latest revision to the Order to verify the current limits. Tritium contained in water (H_2O or D_2O) that is used as a moderator in a nuclear reactor is not an accountable nuclear material..

Material management at DOE facilities is different than Nuclear Material Management. DOE O 410.2, *Management of Nuclear Materials*, provide requirements for managing nuclear materials. Disposition of nuclear material can control how tritium can be released. If it falls below accountable levels it can be treated as waste, otherwise it must be dispositioned according to DOE O 410.2T. The Office of Nuclear Materials has issued an integration guidance document, *Nuclear Material Disposition Process Guidance*, issued September 2013,[14] for use in this topical area. See section 8.2.3 for further discussion.

Quantities of tritium contained in waste may be part of the facility accountable nuclear materials inventory until it is removed from the facility to a waste accumulation area for storage or to a waste packaging area for packaging.

DOE requires tritium facilities with more than a reportable quantity of tritium (1-gram) establish material control and accountability systems to provide accurate nuclear materials inventory information, DOE Order 410.2. Transactions exceeding the reporting quantity for tritium (1-gram) must be reported to NMMSS, per DOE O 474.2. The facility tritium inventory and scrap levels of tritium must be minimized consistent with the operational needs and safeguards practices of the facility. A well-designed facility materials control and accountability system includes, in part, the following:

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[14] Nuclear Materials Guidance document phase 1 for implementing DOE Order 410.2, *Nuclear Material Management*, Office of Nuclear Materials Integration, NNSA. September 2013.

- Accounting system database;
- Account structure;
- Records and reports;
- Physical inventories
- Periodic physical inventories;
- Special inventories;
- Inventory verification/confirmation measurements;
- Measurements and measurement control;
- Organization;
- Selection and qualification of measurement methods;
- Training and qualification of measurement personnel;
- Measurement systems;
- Measurement control;
- Material transfers;
- External transfers;
- Internal transfers;
- Material control indicators;
- Shipper/receiver difference assessment;
- Inventory difference evaluation;
- Evaluation of other inventory adjustments;
- Documentation and reporting forms;
- Procedures and requirements.

3.1.2 Limits for Tritium in Drinking Water and the Environment

The current National Primary Drinking Water Regulation (NPDWR) for beta- and photon-emitting radionuclides is 4 millirem (mrem) per year. The maximum contaminant level (MCL) for tritium in drinking water systems, used to demonstrate compliance with the 4 mrem/year level regulatory criterion as determined by the EPA (40 CFR Part 141) is 20,000 pCi/L (740 Bq/L). Assuming that one drank 2 L per day, using the ICRP 30-based Derived Concentration Guides (DCGs), a concentration of 80,000 pCi/ L (2,960 Bq/L) would produce a dose of 4 mrem per year. Other international standards are based on 10 percent of the public dose limits as recommended by ICRP 60 and ICRP 103; however, implementation varies by nation. Example drinking water values include: the World Health Organization, based on ICRP 60 of 270,000 pCi/ L (10,000 Bq/L) and the Canada Nuclear Safety Commission of 189,000 pCi/ L (7,000 Bq/L). The European Commission (EC) has a 2,700 pCi/L (100 Bq/L) screening value for tritium to determine if more detailed evaluations are necessary and to determine if tritium in combination with other radionuclides may exceed the 0.1 mSv/year (10 mrem/y) recommended dose criterion.

The EPA MCL of 20,000 pCi/L for tritium was promulgated in 1976 based on radiological risk estimates from the Biological Effects of Ionizing Radiation I report (BEIR I), dose factors from

Handbook 69, *Maximum Permissible Concentrations of Radionuclides in Air and Water for Occupational Exposure*, published by the National Bureau of Standards in 1959 and amended in 1963, and various assumptions regarding drinking water exposures. In 1991, as part of an effort to update and revise the drinking water standards contained in 40 CFR Part 141, the EPA evaluated potential doses from 20,000 pCi/L using Federal Guidance Report No. 11 dose factors based on ICRP 30, and determined that it would result in a 69,000 pCi/L limit; however, based on its policy for implementing the Safe Drinking Water Act Amendments of 1996, the EPA left unmodified the 20,000 pCi/L limit, over DOE objections.

NRC requirements (not applicable to DOE facilities) for their licensees include compliance with NRC public dose limits by calculating total effective dose equivalent to the individual likely to receive the highest dose and demonstrate that it does not exceed public dose limits or demonstrate that annual average concentration of effluents at the boundary of the unrestricted area does not exceed values in 10 CFR Part 20, Appendix B, Table II. The values for liquid effluents are based on an assumed ingestion of 2 L/day over the year and a dose constraint of 50 mrem/ year. DOE O 458.1, *Radiation Protection of the Public and the Environment,* similarly requires DOE contractors to monitor releases and calculate potential doses to members of the public to demonstrate that DOE activities are being managed so that public dose is as far below the 100 mrem in a year primary dose limit as is practical. DOE Technical Standard DOE-STD-1196-2011, *Derived Concentration Technical Standard*, contains a table of Derived Concentration Standards (DCSs) (which replaced DCGs), which are concentrations of radionuclides in water that would cause a 100 mrem in a year dose if an individual drinks 730 L (2 L/ day). The tritium DCS is 0.0019 µCi/ ml (approximately 2,000,000 pCi/ L); however, unlike NRC, DOE does not permit the use of concentration guidelines as a means of compliance. Dose estimates are necessary to demonstrate compliance and the DCSs are provided as a tool to assist in that evaluation. Although it is expected that discharges would be significantly less than DCS values, exceeding the DCS does not necessarily violate DOE O 458.1 if dose calculations demonstrate that public doses, as a result of all pathways, are as low as reasonably achievable (ALARA) below the primary dose limit (this includes all sources and pathways, and a 25 mrem/yr constraint, which is usually applied to single sources). DOE O 458.1 also requires that DOE facilities manage discharges so as not to cause public drinking water systems downstream to exceed the drinking water maximum contamination limits in 40 CFR Part 141, *National Primary Drinking Water Regulations*.

Neither DOE (DOE O 458.1) nor NRC (10 CFR Part 20, Subpart E) requirements for cleanup or clearance of real property have specific concentrations or dose limits for tritium in groundwater. Tritium must be considered along with all other radionuclides and pathways when demonstrating that the dose to the public will be as far below 25 mrem/ year as is reasonably achievable, per DOE O 458.1. However, DOE's requirement that DOE activities not cause existing public drinking water systems to otherwise exceed drinking water standards also applies to cleanup and clearance.

Although rarely a health concern, the detection of tritium in groundwater is of interest to many, and is sometimes a contentious issue both within and outside the DOE complex. For example, the identification of an onsite groundwater tritiated plume at Brookhaven National Laboratory contributed to the decision to close the High Flux Beam Reactor (the source of the tritium in the plume), even though the plume did not present a significant public risk and the spent fuel pool had a stainless steel liner installed to prevent future leakage. Environment, Safety and Health Safety Advisory 2006-04, *Stakeholder Sensitivity to Tritium Releases*[15], describes issues encountered at commercial nuclear power plants and non-RCRA-regulated landfills that have found tritium in their leachate (e.g., the Pennsylvania Department of Environmental Protection has identified tritium in the leachate of most of its landfills). Both NRC and the Nuclear Energy Institute are sensitive to tritium groundwater issues, as are the States. Effective monitoring programs and good outreach and communication programs are key to preventing and mitigating actual and perceived problems with tritium control. The accumulation and storage of tritiated water awaiting the disposition decision at Fukushima highlights the public's apprehension and interest with radioactivity in general and even tritium specifically.

3.1.3 RCRA Applicability

Under the implementing regulations of RCRA, specifically at 40 CFR 261.4(a)(4), source, SNM, and byproduct material, as defined by the AEA, is excluded from the definition of solid waste, and thus from the RCRA hazardous waste management requirements. The AEA definition of byproduct material includes "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." Tritium produced in a U.S. reactor meets the definition of byproduct material; therefore, the waste streams derived from U.S. reactor-produced tritium are excluded from RCRA regulation (provided such waste streams do not also contain a RCRA hazardous waste component in addition to the byproduct material component).

Accelerator-produced tritium, on the other hand, did not qualify for this exclusion, since the tritium is produced by a linear accelerator, and does not involve the production or utilization of SNM or the extraction or concentration of source material. However, the AEA was amended by the Energy Policy Act of 2005 (EPACT 2005), which amended the definition of byproduct material to include "any material that has been made radioactive by use of a particle accelerator, and is produced, extracted, or converted after extraction, before, on or after the date of enactment of [EPACT 2005] for use for a commercial, medical, or research activity." Therefore, accelerator-produced tritium used for weapons production does not qualify for the RCRA material exclusion, while other uses for the accelerator-produced tritium do meet the RCRA exclusion. Additionally, tritium produced in the original Canada Deuterium Uranium (CANDU) design, employing natural uranium as fuel, may not, meet the definition of byproduct material; therefore, it is prudent to assume that it also does not qualify for RCRA exclusion.

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[15] DOE/EH-0699, DOE Environment, Safety and Health Advisory 2006-4, *Stakeholder Sensitivity to Tritium Releases*, July 2006.

Thus, for U.S. reactor-generated or accelerator non-weapons production tritium waste to be considered hazardous waste, the waste stream also would have to contain a RCRA-listed or non-tritium-derived characteristic hazardous waste component. The waste stream from CANDUs and weapons-related accelerator tritium would not automatically be excluded from RCRA; however, unless such tritium wastes exhibit one of the characteristics of RCRA hazardous waste (ignitability, corrosivity, reactivity, or toxicity [40 CFR Part 261, Sections 21 through 261.24]) or contain a RCRA-listed hazardous waste, the waste streams would not need to be managed as RCRA hazardous or mixed waste. Pursuant to the RCRA regulations, it is the responsibility of the waste generator to determine if that waste is subject to the hazardous waste requirements [40 CFR 262.11]. Categories of characteristic hazardous waste (and associated properties) that appear to have some potential to apply to certain accelerator-produced tritium wastes are as follows:

- Ignitability [40 CFR 261.21] – Ignitable wastes are solid wastes that exhibit any of the following properties: liquids with a flashpoint of less than 60°C (140°F); solids that are capable of causing fires through friction, absorption of moisture, or spontaneous chemical changes; ignitable compressed gases, as defined in 49 CFR 173.300; or oxidizers, as defined in 49 CFR 173.151.

- Corrosivity [40 CFR 261.22] – Corrosive wastes are solid wastes that exhibit any of the following properties: an aqueous material with pH ≤ 2 or ≥ 12.5; or a liquid that corrodes steel at a rate greater than ¼ inch per year at a temperature of 55°C (130°F).

- Reactivity [40 CFR 261.23] – Reactive wastes are solid wastes that exhibit any of the following properties: (1) they are normally unstable and readily undergo violent change without detonating; (2) they react violently with water; (3) they form potentially explosive mixtures with water; (4) when mixed with water, they generate toxic gases, vapors, or fumes in a quantity sufficient to present a danger to human health or the environment; (5) they are a cyanide- or sulfide-bearing waste which, when exposed to pH conditions between 2 and 12.5, can generate toxic gases, vapors, or fumes in a quantity sufficient to present a danger to human health or the environment; (6) they are capable of detonation or explosive reaction if subjected to a strong initiating source or if heated under confinement; (7) they are readily capable of detonation or explosive decomposition or reaction at standard temperature and pressure; (8) they are a forbidden explosive, as defined in 49 CFR 173.54, a Class A explosive, as defined in 49 CFR 173.53, or a Class B explosive, as defined in 49 CFR 173.88.

The discussion in Section 2.5.1.c provides only a qualitative argument for the determination that for tritiated waste, the characteristics of corrosivity typically are not generally exhibited as little data exists in this area. Moreover, little data are currently available to confirm whether or not the vapor space of some tritium containers (e.g., tritium oxide adsorbed on molecular sieves) would exhibit the hazardous characteristic of ignitability or reactivity over time due to radiolytic decay. As explained above, reactor-generated or accelerator non-weapons-production tritium waste streams that do not contain a hazardous waste component may be excluded from the RCRA hazardous waste regulations pursuant to 40 CFR 261.4(a)(4). This may be the case even if sufficient quantities of both hydrogen and oxygen are present to exhibit characteristics of ignitability or reactivity. This is based on a regulatory policy that EPA has applied in certain cases whereby residuals derived from the management of exempt or excluded waste retain the exemption or exclusion.[16] [17] [18] [19] [20]. However, as indicated above, if a tritium waste (irrespective of its source) also contains a distinct hazardous waste component, the waste stream should be managed as a radioactive mixed waste under the AEA and RCRA.

The application of RCRA to certain tritium waste streams may be subject to regulatory interpretation and enforcement discretion. With this in mind, it is recommended that determinations as to whether or not certain tritium wastes constitute RCRA hazardous waste be discussed and validated with the appropriate regulatory agency (e.g., the EPA Region or RCRA-authorized State agency). Section 8.2.2 provides a flow diagram and expanded discussion on this issue in addition to the definitions and options for tritium recovery and disposal.

The RCRA characteristics are vital for purposes of disposal, but DOT has specific and (often) distinct defining criteria for the purposes of transportation. See 49 CFR Part 173, Subpart D, *Definitions, Classification, Packing Group Assignments and Exceptions for Hazardous Materials Other Than Class 1 and Class 7*.

3.1.4 CERCLA Applicability

Under the CERCLA, cleanups are required to attain applicable or relevant and appropriate (ARAR) standards. With respect to contaminated aquifers, CERCLA designates MCLs (20,000 for tritium) established under the SDWA as the "relevant and appropriate" standard for the restoration of contaminated ground waters that serve as a current or potential future drinking water source. However, in situations where multiple contaminants are present and have the potential for additive effects, the cleanup goal may be set lower than the MCL. In general, discharges to surface water bodies require compliance with AWQC, although if the surface water body be used as a drinking water source, the MCL may be applied to such discharges. If RCRA hazardous listed or characteristic wastes be present, project managers will need to determine which RCRA requirements will apply as discussed in section 3.1.3.

3.2 Tritium Safeguards and Security

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

[17] 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.

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

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

[20] RCRA Superfund Hotline Report, February 1985.

Tritium is a nuclear material of strategic importance and must be safeguarded from theft or diversion. The current DOE safeguards and security directives do not explicitly grade or categorize tritium as predecessor directives had.

3.3 Tritium Facility Safety Analysis and Regulatory Quantity Limits

A documented safety analysis is required by 10 CFR Part 830, *Nuclear Safety Management*, for all Hazard Category 1, 2, and 3 nuclear facilities. Irrespective of these requirements, the good practices associated with the implementation of Integrated Safety Management (ISM) principles necessitate that hazards be identified and controlled, which is a major step in the safety analysis process.

3.3.1 Safety Analysis

Safety analyses address various accident scenarios. Some typical examples of accident categories include fire, explosion, tritium leaks, cooling water breaks, and failure of detritiation system components.

3.3.1.a Facility Requirements

There are a few fundamental assumptions normally made when performing safety analyses on tritium facilities, which, if not satisfied, require more detailed analyses or development of corrective actions. These include:

- The integrity of the primary container should be ensured for all normal operations, anticipated operational occurrences, and for the design basis accidents (DBAs) it is required to withstand.

- If the facility structure is not part of the secondary barrier, its failure as a result of severe natural phenomena or other postulated DBAs should not prevent the primary container or the secondary containment or confinement system from performing their necessary safety functions.

- When secondary containers (secondaries) are used, a tritium effluent removal system to handle tritium leakage from primary containers is recommended by this Standard, but not required. .

3.3.1.b Radiological Materials Inventory

Attachment 1 to DOE-STD-1027-92, Change Notice 1, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23,* Nuclear Safety Analysis Reports, states,

Additionally, material contained in [Department of Transportation (DOT)] Type B shipping containers (with or without overpack) may 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 [see Section 6.1] without an overpack should have heat protection provided by the facility's fire suppression system. Further discussion expands this clarification as follows:

The Material at Risk (MAR) is the bounding quantity of hazardous material that is available to be acted on by a postulated accident. MAR values used in hazard and accident analysis shall be consistent with the values noted in hazard identification/evaluation, and shall be bounding with respect to each accident being evaluated. While DOE-STD-1027-92 excludes material in Department of Transportation Type B qualified containers from consideration for the purposes of hazard classification, the existence of such material shall be acknowledged in the DSA and excluded from the source term for a particular event scenario only if the containers can be shown to perform their safety functions under the accident conditions. If the containers are not currently qualified (qualifications expire on a fixed schedule), then their contents must be included in the summation of facility inventory and be included in accident analyses, irrespective of accident conditions. If, however, the container is currently certified, then a comparison of the conditions resulting from transportation and facility accidents is performed. For example, consider a currently certified Type B container that has been qualified to withstand fire and crush loads (i.e., Type B Hypothetical Accident Crush Test loads) associated with a transportation accident. If these transportation accident conditions are more severe than the associated credible facility accident conditions (e.g., fire and seismic crush loads), than the inventory in these containers can be excluded from source terms for those associated accident scenarios (irrespective of fire suppression system coverage). If the facility fire conditions exceed the transportation fire conditions, the inventory is included. Note that credit for safety-class or safety-significant fire suppression affects the selection of the credible facility fire scenario, and therefore, the fire conditions for comparison with the transportation fire conditions.

If the transportation accident conditions are less severe than the facility accident conditions, then the contents are included in the accident scenarios. For example, if the seismic crush loads exceed the transportation crush loads, the contents cannot be excluded from the facility seismic accident scenarios. They could, however, still be excluded from the facility fire scenarios if the transportation fire conditions bound the facility fire conditions.

DOE-STD-1027-92, Change Notice 1 does not provide explicit exemption criteria for other than Type B containers. General guidance for other than Type B containers would be to include the tritium in the facility inventory. Specific guidance concerning the degree and amount of residual tritium inventory to include in safety analyses based on facility-specific configurations, can be obtained from AU.

The Hazard Category of a nuclear facility is based on threshold quantities of radiological material inventory. Gram quantities are rounded values based on the calculated number of curies. [21] The definitions for each Hazard Category are as follows:

- Hazard Category I (generally limited to nuclear reactors): Regardless of the quantity of tritium in the inventory, a facility that handles only tritium is not classified, by tritium quantity alone, as a Hazard Category I. The Program Secretarial Officer (PSO) may designate a tritium facility as Category I if the potential for significant offsite consequences exists.

- Hazard Category II: To be classified as a Category II nuclear facility, the facility tritium inventory must be > 30 grams.

- Hazard Category III: To be classified as a Category III nuclear facility, the facility tritium inventory must be > 1.6 and < 30 grams.

- Less than Hazard Category III: (formally called Radiological) Facilities that have less than 1.6 grams of tritium in the facility radiological material inventory.). [10 CFR Part 830]

- Some DOE sites have adopted a lower threshold limit for less than Haz Cat III Nuclear Facilities at the RQ level for the associated radionuclide, under which the facility would be considered non-radiological. This approach has also been endorsed in the past by various DOE Program Offices at HQ.[22] Using this approach for tritium facilities for which the RQ value is 100 Ci, any facility possessing less than this would be considered non-nuclear or non-radiological. This approach, however, is contrary to the legal definition as stated in the Atomic Energy Act and DOE Standard 1027 in which there is no lower threshold identified to become a non-radiological facility for a facility that contains radiological inventory and users are cautioned not to treat tritium facilities with less than 100 Ci as non-nuclear. Unlike the existence of processes for downgrading Haz Cat II and III Nuclear Facilities there are none for a Haz Cat III Nuclear Facility to non-nuclear.

3.3.1.c Material Release Assumptions

Once the total inventory available for release is known, the appropriate source term can be calculated. The components of the source term, as described in DOE-HDBK-3010-94, *Airborne Release Fraction/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*, are material at risk, damage ratio, airborne release fraction, respirable fraction, and leak path factor. The factors for airborne release and respirable fractions are normally assumed to be 1.0 for elemental tritium and oxide. The other factors are facility-specific. For DSA fire scenarios, the fraction of the release

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[21] DOE-STD-1027-92, Change Notice 1, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23,* Nuclear Safety Analysis Reports, September 1997.

[22] *Implementation Guidance for Authorization Basis*, DOE Office of Defense Programs, Revision 1, August 21, 1995.

assumed to be oxide is normally 100 percent. Typical scenarios to review for inclusion in safety analyses are area fires (single and multiple Fire Control Areas (FCAs)), full facility fire, leaks or spills, hydrogen explosions, and Natural Phenomena Hazards (NPH) events (the design basis earthquake (DBE) may have a seismic-induced fire, which usually results in the bounding accident). DOE-STD-3009-94, Change Notice 3, *Preparation Guide for U.S. DOE Nonreactor Nuclear Facility Safety Analysis*, provides detailed guidance for performing accident analyses.

Several dose methodologies have been used for safety analyses throughout the DOE complex. However, with the implementation of DOE G 414.1-4, *Safety Software Guide for Use with 10 CFR Part 830, Subpart A,* Quality Assurance Requirements*, and DOE O 414.1C,* Quality Assurance, some computer modeling methodologies are designated as toolbox codes for safety analysis, while others are not. Safety analysts should check the current DOE Safety Software Control Registry before using a specific program for safety analysis calculations.

The value associated with dry deposition velocity for tritium oxide used in safety basis calculations has been the subject of Defense Nuclear Facilities Safety Board (DNFSB) and SRS discussions since 2011. The DNFSB, in the Staff Issue Report of March 17, 2011[23], indicated that use of the value of 0.5 cm/s was non-conservative. Analyses by SRNL[24] to address this and other safety basis calculations concluded that 1) the Pasquill stability classification for regulatory application of SRS is based on measurements of the standard deviation of the vertical component of wind direction fluctuations, collected from the 61m level of the SRS meteorological towers and processed in full accordance with EPA-454/R-99-005, *Meteorological Monitoring Guidance for Regulatory Modeling Applications*[25]; and that 2) meteorological databases used as input for MACCS2 calculations should contain hourly data for five consecutive annual periods from the most recently available 10 years.

Although it is likely that the realistic deposition velocity is approximately in the range of 0.1 cm/s, there is not currently data to support this supposition. Current toolbox codes do not model re-emission and as such SRS research into re-emission has been proposed. The results of this research may allow credit for a portion of the re-emission to the canopy as oppose to assuming 100% re-emission to the atmosphere, which will affect the value chosen for deposition velocity. Several models have been proposed for washout, which is difficult to model; however research is continuing in this area. A paper presented at the 10th International Conference on Tritium Science and

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[23] August 19, 2011, letter from P. Winokur, DNFSB, to T. D'Agostino, NNSA Administrator, transmitted the March 17, 2011, Staff Issue Report *Review of Safety Basis, Tritium Facilities, Savannah River Site*.

[24] C.H. Hunter. SRNL-STI-2012-00055, *A Recommended Pasquill-Gifford Stability Classification Method for Safety Basis Atmospheric Dispersion Modeling at SRS*. Rev. 0, March 2012.

[25] U.S. Environmental Protection Agency. EPA-454/R-99-005, *Meteorological Monitoring Guidance for Regulatory Modeling Applications*, February 2000.

Technology (TRITIUM 2013) examined the relationship between exchange rate and washout coefficient[26].

The values for parameters (e.g., relative concentration, deposition velocity) selected for use in dose consequence analyses would ideally be site-specific. In the absence of the data to quantify re-emission 0.00 cm/sec should be the default value to be used for deposition velocity.

Another program used primarily for modeling prior to environmental release, the Tritium Migration Analysis Program (TMAP) [27] was developed by EG&G Idaho, Inc., to dynamically analyze tritium transport in fusion reactor facilities. TMAP solves conservation equations for any number of gaseous species in a system composed of solid structures or walls, and related gas filled enclosures by including the phenomena of surface molecular dissociation and atomic recombination, bulk diffusion with the possibility of trapping by material defects produced by neutron damage, chemical reactions within enclosures, and convective flow between enclosures. In TMAP, the movement of hydrogen species across structure surfaces is not only governed by molecular surface kinetics but by solution laws, such as Sieverts' or Henry's laws, or inhibited from crossing the surface. Movement in solids (may be a layered composites) is modeled by one-dimensional Fick's law of diffusion equations. The thermal response of these structures is obtained from the solution of one dimensional heat conduction equations that account for applied heat or boundary temperature loadings conditions. The code was specifically developed to predict the inventories of tritium in and release from fusion reactor systems that experience both plasma driven permeation at very high implanted fluxes (10^{24} ions/m^2-s) and pressure driven permeation at very low tritium pressures (< 10 Pa). However, it is evident that TMAP has application to a much wider variety of problems, including modeling experiment facilities developed to measure tritium material transport properties.

TMAP4 can be obtained from the Department of Energy's (DOE's) Energy Science and Technology Software Center (ESTSC).

Several Federal environmental laws (e.g., Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), and the Clean Water Act (CWA)) require that releases of hazardous substances above reportable quantities (for tritium, this is 100 curies in 24 hours) be reported to the National Response Center. In many cases, the curie levels are back-calculated from offsite dose receptor requirements. DOE O 231.1A, DOE M 231.1-2, DOE O 151.1C, and DOE G 151.1-4 make distinctions between "normal" or "routine" releases and "abnormal" or "accidental" releases, and suggest reporting abnormal or accidental releases, even if they are below Federally permitted levels.

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[26] A.V. Golubev et al. *The Field Experiments on the HTO Washout from the Atmosphere*. Presented at the 10th International Conference on Tritium Science and Technology (TRITIUM 2013), Nice, France.

[27] G. R. Longhurst, D. F. Holland, J. L. Jones, B. J. Merrill, TMAP4 User's Manual, EGG-FSP-10315, Idaho National Engineering Laboratory, July 1992

3.3.2 Integrated Safety Management

All tritium-related operations and activities, including design, construction, system acceptance and turnover, operations, shutdown, deactivation, should follow commitments identified in DOE ISM directives of the 450.4 series, which were an outgrowth from the Defense Nuclear Facilities Safety Board (DNFSB) Recommendation 95-2 Implementation Plan. The objective of ISMS, as articulated in 48 CFR § 970.5223-1, *Integration of Environment, Safety, and Health into Work Planning and Execution*, is that, in performing work for DOE, the Contractor shall perform work safely, in a manner that ensures adequate protection for employees, the public, and the environment, and shall be accountable for the safe performance of work. The Contractor shall exercise a degree of care commensurate with the work and the associated hazards. The Contractor shall ensure that management of environment, safety and health (ES&H) functions and activities becomes an integral but visible part of the Contractor's work planning and execution processes.

This clause goes on to address ISMS in more detail. The commitments for ISM follow the seven Guiding Principles:

1. Line Management Responsibility for Safety;
2. Clearly Defined Roles and Responsibilities;
3. Personnel Competence Commensurate with Responsibilities;
4. Balanced Priorities;
5. Identification of Safety Standards and Requirements;
6. Hazard Controls Tailored to the Work Being Performed;
7. Operations Authorization.

The process for planning and conducting pre-work hazards analysis for all work operations at tritium facilities should be consistent with the Guiding Principles listed above and should follow a structured approach commensurate with the risks and hazards involved. Several methods for enhanced work planning and related work planning strategies used throughout the DOE complex follow five steps:

1) Define the Scope of Work;
2) Analyze the Hazards;
3) Develop and Implement Hazard Controls;
4) Perform Work within Controls; and
5) Provide Feedback and Continuous Improvement.

This sequence is one of many ways to ensure a structured approach in work planning with a focus on worker safety and reduced environmental risks.

3.3.3 Facility Segmentation

Facility segmentation is discussed in DOE-STD-1027-92. Activities conducted in the same facility may be separated into different areas for analysis purposes, but only if independence (in accident space) can be shown. Potential areas for segmentation include:

- Receiving area;
- Receiving storage area;
- Shipping area;
- Shipping storage area;
- Tritium unpackaging, handling, and packaging operations;
- Low-level waste accumulation area;
- Low-level waste packaging area; and
- Packaged low-level waste storage area.

The independence of the segments, however, is not easily demonstrated. Common piping and heating, ventilation, and air conditioning (HVAC) cannot exist to use this segmentation process. Additionally, common-cause initiating events (e.g., fire, seismic) affect multiple segments, thereby placing an additional burden of proof on the analyst to demonstrate independence. A successful segmentation analysis, however, would not affect the applicability of Price-Anderson enforcement of the Nuclear Safety Rules to the facility.

3.4 Radiological Material Quantity Limits

3.4.1 Tritium Shipping, Radioactive Material Inventory, Quantity Limits

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

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

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

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

- Limited Quantity (< 21.6 Ci for gas; see Table 7-1 for solids, liquids, and applicable 49 CFR Part 173 requirements): Limited quantities of tritium can be packaged and shipped in strong, tight containers (paper boxes, paint cans) with proper markings.

- Type A Quantity (21.6 to < 1080 Ci): Type A quantities of tritium use DOT Specification 7A containers, properly marked and surveyed prior to shipment. A number of different packages are available from small cans, 55- and 85-gallon drums, 4 x 4 x 7-foot steel boxes, up to and including oversized, specially designed containers. These containers are relatively inexpensive.

- Type B Quantity (≥ 1080 Ci): Type B quantities of tritium must be shipped in a certified Type B package. There is only a limited number of these Type B packages available for tritium shipment (e.g., BTSP, UC-609), and special routing (i.e., prescribed routes employing highway route control such as using major highways or bypassing cities) is required.

- Type B-Quantity, Low-Level Radioactive Waste: For the purposes of storage at the waste site, Type B-quantity solid waste can be stored in Type A containers.

At the waste generation location, items containing greater than 1080 Ci of tritium are normally stored in Type A containers. Type A containers containing over 1080 Ci must then be placed into Type B containers for shipping to DOE waste sites. At the DOE waste site, the Type A containers can be removed from the Type B package and stored in the Type A package. This allows the expensive Type B package to be returned to the shipper and reused for another Type B shipment. A radiological materials inventory must still meet the requirements as discussed in Section 3.3.1.b. Additionally, 10 CFR Part 835 contains labeling requirements.

3.4.2 Tritium Receiving Area, Shipping Area, Quantity Limits

There are no limits (other than local administrative limits) on the inventory of approved containers that can be stored in a properly marked shipping area, receiving area, shipping storage area, and receiving storage area. There are, however, site-wide limits that may be imposed in the EIS that could restrict the total inventory quantities.

These areas must be inside an area posted as a Controlled Area, per 10 CFR Part 835. The Controlled Area encloses an area posted as a Radioactive Materials Area (RMA). The Controlled Area and the RMA can be the same area. The RMA must be periodically surveyed and appropriately marked to indicate the nuclear hazard associated with the area. These areas may have more than one incoming or outgoing certified package containing Type A or B packages waiting for transport.

3.5 Tritium Unpackaging, Handling, and Packaging Areas, Quantity Limits

The tritium is removed from the specification package/certified package in unpackaging, handling, and packaging areas, and, as such, these areas if not emcompassed by the nuclear facility will be designated as a nuclear facility (Hazard Cat 2, 3 or less than 3), depending upon the quantity of tritium in the inventory.

3.6 Tritium Waste Collection and Waste Packaging Area, Quantity Limits

The waste collection/packaging area will have, in process, the collection and packaging of low-level waste. The waste collection/packaging area should likewise be reviewed for designation as a nuclear facility as defined by the quantity of tritium.

3.7 Tritium Radworker Training

Subpart J of 10 CFR Part 835, *Occupational Radiation Protection*, requires that individuals complete radiation safety training commensurate with the hazards in the area and the required controls. DOE-STD-1098-2008, *Radiological Control* lists Appendix F of this Standard, “Radiological Training for Tritium Facilities,” as a reference to be considered in addition to DOE’s core training material for radiological workers at tritium facilities. This Appendix provides a recommended implementation process for conducting the radiation safety training at tritium facilities as required by 10 CFR Part 835 and as recommended in DOE-STD-1098-2008.

3.8 Tritium Focus Group (TFG)

The TFG comprises both DOE Federal and contractor personnel associated with tritium operations. It was formed in 1991, in response to the Secretary of Energy's Task Group on tritium operations, and has expanded in scope and stature since that time. Its charter is contained in Appendix H.

4.0 FACILITY DESIGN

DOE tritium facilities, as a subset of DOE nonreactor nuclear facilities, must conform to the design requirements contained in DOE O 420.1C, *Facility Safety*. Implementation guidance for this Order can be found in DOE G 420.1-1A. The general design philosophy of nuclear facilities in general, and tritium facilities specifically, follows the tenets of multiple levels of protection; preventive vs. mitigative design features, passive vs. active safety systems, engineered vs. administrative controls, and judicious use of zoning throughout the facility.

4.1 Confinement of Tritium Philosophy

It is desirable to have tritium construction projects managed by a team that includes tritium operations expertise on staff. The building/systems would be designed to meet the needs of the user, and costly retrofits after completion could be avoided.

During the first 25 years of tritium technology, the handling techniques in use were designed to protect the worker from exposure to tritium. Worker protection was provided primarily by the use of single-pass ventilation systems designed to rapidly remove any tritium released in the breathing space from the area occupied by the workers. The ventilation gases were released through an elevated stack at high velocity to massively dilute the gases before they could reach ground level. Single-pass ventilation systems and high-velocity hoods were used extensively and quite successfully for worker protection during these early years. These high-velocity ventilation, high-velocity air hood, and elevated release techniques are still used for worker protection, but generally as a supplement to improved barriers that better protect the environment.

In those early years, the room or building enclosing the tritium activity was equipped with a single-pass ventilation system that did not recirculate the air back into the facility. Outside air was brought in by the ventilation fans, conditioned for comfort, passed through the building spaces one time, and was then released to the environment through an elevated stack. The room air exchange rate generally accepted to be adequate for worker protection was 6 to 10 room air changes per hour.

The tritium apparatus was enclosed in a high-velocity air hood, and the worker worked through gloves in the doors or reached in through hood openings to operate the equipment. The high-velocity air hoods were maintained at a pressure negative to the room spaces, and the natural air flow was from the room through the hood opening and then out the ventilation duct work and up the stack to the environment.

Tritium releases that occurred due to normal operations, component failure, and worker error were inside the hood. The high-velocity air flowing through the hood swept any released tritium away from the worker and out the stack to the environment where the tritium was massively diluted in concentration. These techniques protected the worker; however, tritium was released to the environment.

As concern for the environment increased, tritium technologists first attempted to control releases by increasing design and material selection requirements, adding and enforcing regulations, and increasing worker training and awareness. Time would prove that these techniques, although helpful, were not completely successful. The tritium workers were already operating at a high performance level and the tritium releases associated with equipment design and material selection were not measurably decreasing as a result of the more stringent design requirements and reviews. Tritium releases continued to occur. The expectation of faultless materials and errorless workers was unrealistic. By the late 1960s and early 1970s, another philosophy, one of capture, containment, and cleanup evolved.

4.1.1 Tritium Capture, Contain, and Cleanup Process

The capture, contain, and cleanup process encloses the primary or first wall tritium container inside a secondary container such as a double-walled container, glovebox, room, or building so that any tritium escaping from the primary container is captured in the secondary container. A tritium removal system associated with the secondary container then removes the tritium from the secondary by circulating the captured gases through a cleanup system.

4.1.1.a Containment and Confinement Systems

There are many different technical descriptions of the terms containment and confinement. The simple dictionary definitions are: containment – being contained, which in turn is to hold or enclose; and confinement – being confined, which in turn is to restrict, to keep within limits. It is beneficial to define how these words are used in the tritium field. In some facilities and applications, the terms are used interchangeably. The TFG definitions are defined in Appendix B, *Definitions*. This Standard as noted in the Definitions makes a distinction between these terms as follows:

Containment system: 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.

Confinement system: 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.

Simply stated, containment consists of some arrangement of physical barriers that do not need any other equipment or operator action to operate to meet a leak rate criterion, whereas confinement consists of an arrangement of barriers (any kind) that need an active system or action to meet its leak rate criterion. Note that in the context of these definitions, 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.

As mentioned, the terms containment and confinement are not well defined in the DOE complex. For example, DOE M 435.1-1 defines confinement as follows:

> Confinement: The control or retention of radioactive materials within a designated boundary. Primary confinements are process enclosures and other spaces normally containing radioactive material. Secondary confinement surrounds one or more primary confinement systems.

Adding to the confusion are requirements for ventilation systems. DOE G 420.1-1A for use with DOE O 420.1C does not define containment or confinement, but defines the terms primary, secondary, and tertiary “confinement barriers” as:

- Primary confinement. Provides confinement of hazardous material to the vicinity of its processing. This confinement is typically provided by piping, tanks, gloveboxes, encapsulating material, and the like, along with any off-gas systems that control effluent from within the primary confinement.

- Secondary confinement. Consists of a cell or enclosure surrounding the process material or equipment along with any associated ventilation exhaust systems from the enclosed area.

- Tertiary confinement. Typically provided by walls, floor, roof, and associated ventilation exhaust systems of the facility. It provides a final barrier against the release of hazardous materials into the environment.

The Guide also references DNFSB/TECH 34, *Confinement of Radioactive Materials at Defense Nuclear Facilities*, in which the terms “active confinement” systems and “passive confinement” systems are discussed. The terms “active confinement” and “passive confinement” roughly translate to confinement and containment, respectively, as used in this Standard.

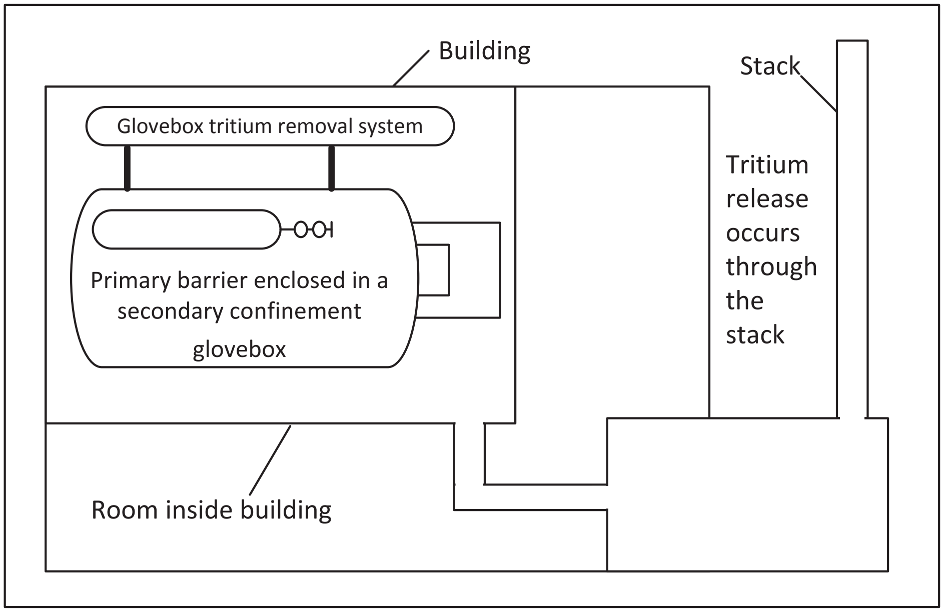
4.1.1.a(1) Primary Containers

Operations are conducted with tritium enclosed inside leak tight primary containers consisting of parts such as valves, tubing, pipe, components, transducers, pumps, and vessels. The leak rate of these primary or first wall containers, at operating pressure, is generally certified to be less than 10^{-6} to 10^{-7} cm^3 of helium per second (cm^3 He/sec). The quantity of tritium released during all normal operations is extremely small and can be estimated from the engineering specifications. Many primary tritium systems are designed with pressure relief protection. These devices should not relieve directly into the environment, but rather into holding tanks designed with sufficient capacity to retain the entire contents of the primary system.

4.1.1.a(2) Secondary Containers

In modern tritium operations, the primary container is enclosed inside a secondary barrier such as a glovebox. The secondary system is only exposed to tritium, if it is released from the primary barrier.

Secondary containers in the DOE complex vary in size, shape, leak rate, and quality depending upon the age, projected use, and the quantity of tritium at risk. Figure 4-1 illustrates an example of secondary containment.



7) Figure 4-1: Secondary containment enclosing a primary container filled with tritium inside a building equipped with single-pass ventilation to a stack

4.1.1.a(2)(a) High-Quality Secondary Containers

Some operations are equipped with high-quality, double-walled pressure vessels. The outer pressure vessel is the secondary containment system, and the inner container is the primary container. This type of secondary containment system is generally used for storage of large quantities of tritium of very high quality, and is certified at operating pressure to leak rates of less than 10^{-6} to 10^{-7} cm^3 He/s.

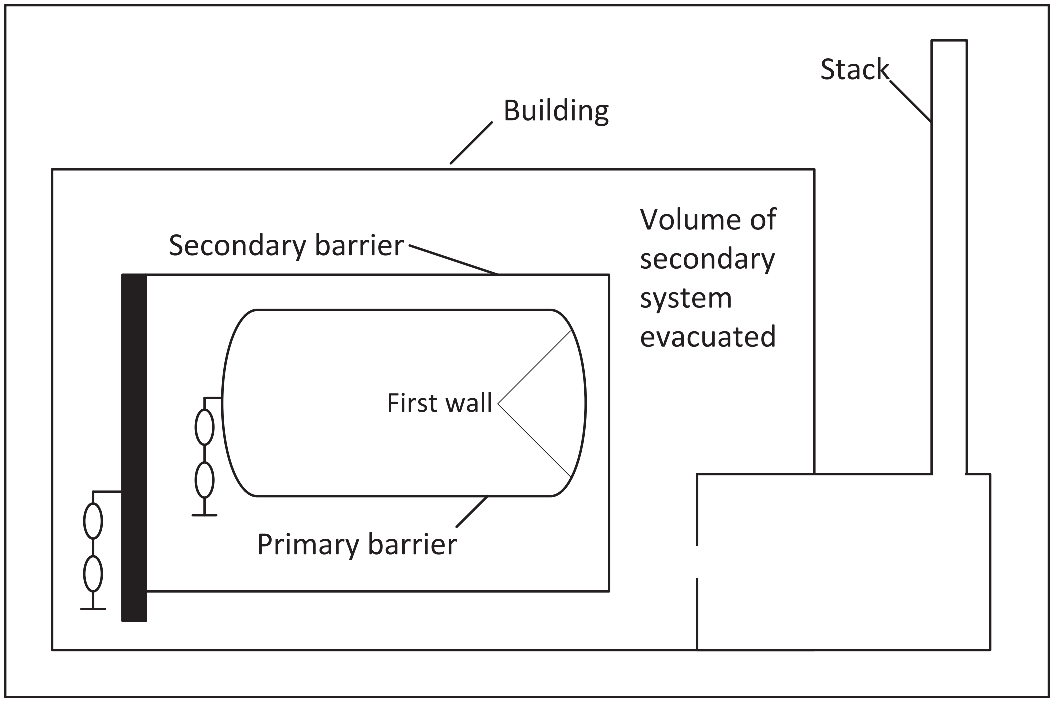
These high-quality secondary containment systems safely store any tritium released from the primary container for several days or weeks without a significant release to its environment. The maximum quantity of tritium released from these high-quality systems during a significant primary container leak can be accurately estimated from the secondary containment system engineering specifications.

The space between the primary vessel and the secondary container in these systems is usually evacuated during service. If tritium is released into these spaces, there are no dilution gases present, and the gas leaking from the secondary container is in the same form as the gas in the primary container. If the gas released into this high-quality secondary is 90 percent tritium, less than 1 Ci of tritium will be released to the surrounding area for each four to 40 storage days.

Following a release into a high-quality secondary container, the tritium can be recovered by pumping it from the secondary container into another primary container. Several days can elapse during the recovery process without a significant release of tritium to the environment.

4.1.1.a(2)(b) Medium-Quality Secondary Container

It is not practical or possible to enclose all primary tritium containers inside high-quality, non-diluting, evacuated secondary containers. Gloveboxes, as discussed in 4.1.1.a, can be utilized in both secondary containment and secondary confinement systems. Figure 4-2 illustrates one example.



8) Figure 4-2: Secondary confinement

Most primary containers can be enclosed in gloveboxes. The box gives access to the primary containers for ease of operation and maintenance.

The leak rate of a glovebox can generally be certified to be no more than 10^{-3} to 10^{-4} cm^3 He/s. In a typical working environment, tritium gloveboxes are operated at near room pressure but slightly negative, on the order of a few tenths of an inch of water column. Tritium gloveboxes do not operate at the large differential pressures common with other gloveboxes through the DOE complex such as those that house actinides and fission products. In Tritium gloveboxes it is important to minimize the in-leakage of air (oxygen) for flammability concerns (however no gloveboxes have been operated with positive differential since 1991, when the Tritium Task Group [28] recommended against this practice which was observed at the Salt Facility at Los Alamos). The low differential pressure allows for less potential for in-leakage than for the larger differentials gloveboxes but the oxygen monitors in the tritium gloveboxes are usually still credited in the safety basis. Although the glovebox is still negative with respect to the room, the tritium levels in the box are usually several orders of magnitude greater than the tritium levels in the room and, due to the laws of partial pressures, the movement of tritium due to permeation alone will always be from the box to the room. To minimize the permeation of tritium from the box to the room, gloveports should be covered and evacuated when they are not in use.

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[28] U.S Department of Energy, *Report of the Task Group on Operation of Department of Energy Tritium Facilities*. October 1991.

In addition to oxygen monitors, most tritium gloveboxes are connected to cleanup systems designed to remove undesired impurities from the glovebox gases and return clean gases back to the box. One of the undesired impurities is tritium (as T2, HT, and/or HTO) itself. Although cleanup systems are be designed to remove free tritium down to the part-per-million to part-per-billion level the return gases from such cleanup systems will never be completely devoid of free tritium. In addition, cleanup systems designed to remove free tritium from the glovebox gases are not capable of removing tritium that has plated-out on the interior surfaces of the glovebox or any of the equipment that is inside the glovebox. Thus, when the box is opened to room air for maintenance purposes, a "puff" type of tritium release will occur, such as that described in Section 2.3.5 "Outgassing". The chemical form of the tritium release will be HTO.

Tritium permeation and diffusion through the elastomeric seals and gloves is reasonably well known, and the tritium released from the glovebox during a primary container leak can be accurately estimated.

If tritium is released into the glovebox, the gases in the glovebox are mixed with and dilute the tritium. The gases leaking from the glovebox are then in the form of tritium mixed with the glovebox gases. Assuming a glovebox with a volume of 1 m^3 and a release of 10 grams of tritium, the glovebox concentration following the release will be 0.1 Ci/cm^3. Even at the relatively high leak rates of 10^{-3} to 10^{-4} cm^3 He/s, approximately 0.1 Ci of tritium will be released for every 1,000 to 10,000 seconds of elapsed time. This is a release of approximately 1 Ci from the facility stack for every 3 to 30 storage hours, which generally does not pose a significant health risk.

Following a release into a glovebox, the tritium is recovered in a tritium removal system. This low-level waste is in the form of water contaminated with tritium. Several hours can elapse during the recovery process without a significant release of tritium to the environment.

4.1.1.a(2)(c) Low-Quality Secondary Containers

Other facilities are equipped with lower-quality systems such as rooms or buildings. It is difficult to determine the leak rate of these low-quality systems and, therefore, it is difficult to estimate the quantity of tritium that will be released to the environment during a primary container leak.

If the secondary container is a room or building, the ventilation system could be shut off to reduce the release of the tritium to the environment through the building stack when a release occurs. In this case the released tritium contaminates the building. If a tritium cleanup system is available (in a room or a building), the ventilation system is switched over to the recirculation mode to reduce the amount of contamination. If a cleanup system is not available then a choice is made (ahead of time so it is proceduralized and personnel are trained) whether to stack or contain the release.

For example, if there is no cleanup system and the ventilation system is shut off, assuming a 100 m^3 (20 × 18 × 10 feet) building and a 1-gram release to the building, the tritium concentration in the building gases will be 0.0001 Ci/cm^3. If the building is the containment barrier, it will leak approximately 5 percent of its volume to the environment per hour; therefore, an environmental tritium release of 500 Ci/hr will result. Cleanup of the building also is required in this scenario.

Even with a room or building cleanup system, a substantial fraction of the tritium released from the primary containment system will be released to the environment before recovery is accomplished due to the high leak rate of these low quality secondary systems. However, these low-quality systems still have a place in the tritium containment strategy. These room- or building-type systems, equipped with high flow rate tritium removal systems, are used in facilities where there are no other feasible alternatives due to the large size of the equipment being enclosed.

4.1.1.a(3) Primary and Secondary Containers

The container-within-a-container concept introduced above is applied in the tritium community in various configurations. Figure 4-3.a shows a building confinement system. Figure 4-3.b shows an arrangement consisting of a higher-quality primary container with a lower-quality secondary container with indication or sampling capability from the annulus area to help detect whether the primary is leaking. Before opening the secondary, the annulus area should be sampled and if a leaking primary is suspected from the sample or from the pressure indicator, then the container is moved to within a glovebox or compensatory measures taken prior to opening. Depending on the application and quality of the primary container, the secondary may not need to be a containment barrier itself. The secondary's for high quality primary containers are most often employed as a physical protection barrier for the primary container. For example, the LP-50 SARP[29] states…"no credit is taken for the seal on the shell in the safety analysis and no test of its seal tightness is made. The secondary vessel is a mechanical protection barrier. For primary containers of questionable or low quality, the secondary should provide a containment function as the probability of primary container failure is higher." Section 6.6 discusses the container requirements for interim storage of tritium/ tritiated materials.

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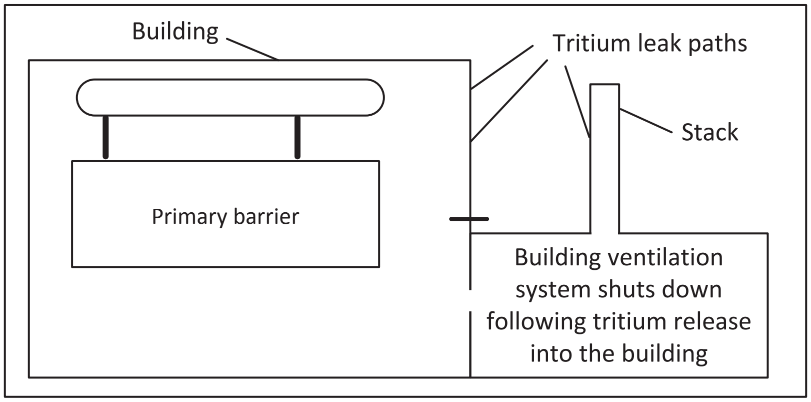
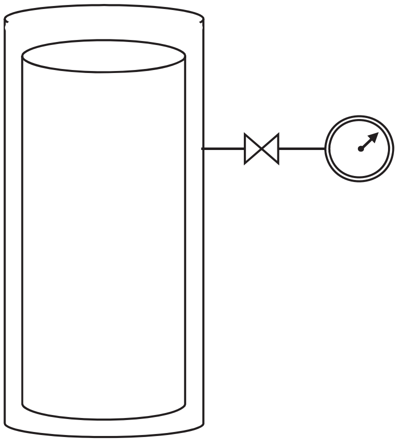
[29] LP-50 SARP, *Safety Analysis Report for Packaging (SARP): USA/9507/BFL (ERDA-AL), Model AL-M1*, MLM-2447, September 30, 1977.



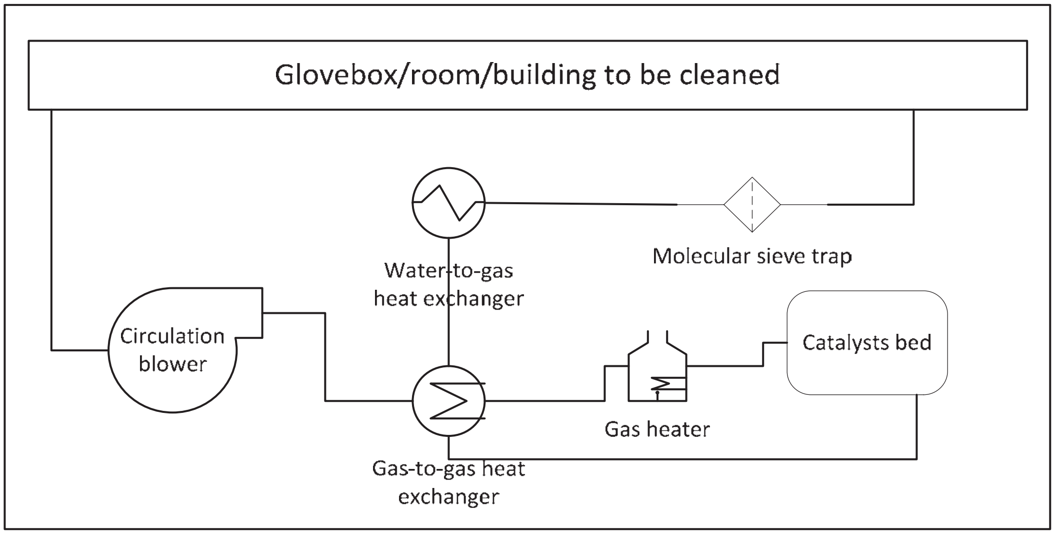
10) Figure 4-3.a: Building confinement system



9) Figure 4-3.b: Typical double containment configuration

4.1.2 Tritium Cleanup and Removal Systems

Current designs employ tritium removal systems. When tritium is released into a secondary, the associated cleanup system starts, and the gases containing tritium are circulated through the cleanup system, and the tritium is removed. See Figure 4-4 for a diagram of a typical gas-to-water conversion tritium removal system.



11) Figure 4-4: Typical gas-to-water tritium removal system flow schematic

If the cleanup system is associated with a high-quality barrier (leak rate of less than 1 Ci of tritium in a period of 4 to 40 days), tritium transfers to another container can take several days to complete without a significant release of tritium to the environment. If the cleanup system is associated with a medium quality barrier (leak rate of less than 1 Ci in 3 to 30-hours), the cleanup system flow rate needs to be high enough to remove tritium within a few hours in order to prevent a significant release of tritium to the environment. If the cleanup system is associated with a low-quality barrier (leak rate of 8 Ci or more per minute), the cleanup system flow rate needs to be very high so that tritium is removed from the gas before it is released to the environment.

The captured tritium is generally removed by circulating the gas through a system that removes tritium down to the part-per-billion level. Typical present-day cleanup systems remove tritium by cracking the molecules on a hot catalyst. The free hydrogen atoms combine in the catalytic reactor with the oxygen present to form water. The tritiated water is then removed from the gas by molecular sieve traps. One example of this type of cleanup system that has been used extensively in several different tritium facilities works as follows:

(1) Tritium is released into the secondary containment system (i.e., the glovebox) as a result of a primary system failure

(2) The cleanup system is started and the tritium-containing gases captured in the glovebox are circulated through the cleanup system

(3) The cleanup system removes the tritium from the gas stream by breaking down the hydrogen (i.e., H, D, and T) containing molecules on a hot, precious metal catalyst

(4) The free H, D, and T atoms are recombined with oxygen in the catalytic reactor to form water vapor

(5) The hot water vapor is cooled to a suitable temperature by passing the gas stream through one or more heat exchangers

(6) The water is removed from the gas stream by passing the gas stream through molecular sieve traps

Catalyst/molecular sieve tritium removal systems of this type can be very effective. Depending on the tritiated species, the reduction in tritium concentration for such systems has been measured at ratios of 10^6:1 to 10^8:1 when operated in a once-through flow mode, in which the gas stream passes from the glovebox, through the cleanup system, and out the stack to the environment. In most situations, these types of cleanup systems are operated in the continuous flow mode, where the gas stream is moved from the glovebox, through the cleanup system, and back to the glovebox. When operated in the continuous flow mode, these types of cleanup systems can easily reduce the tritium concentration in the gas stream to the parts-per-billion level (i.e., 2.6 mCi/ m^3), or lower, as long as they are operated for a sufficiently long period of time.

The tritium released to the environment during this process is a function of the quantity of tritium released, the volume and leak rate of the container, and the cleanup rate of the tritium removal system.

There are several considerations in determining the adequacy of the tritium removal system. First is the volume: the larger the volume, the longer it will take to remove tritium from the gases. Second is the tritium removal system flow rate. This rate affects the time required to remove tritium from the gases. The lower the flow rate, the longer it will take to remove tritium from the volume. Third is the cleanup rate of the tritium removal system. This system is typically operated in a recirculating mode, but may also be operated in a single-pass mode. The tritium-contaminated gases are pumped through the tritium removal system, and the cleaned gases are either returned to the volume or released to the environment.

In a large, complex system filled with equipment, it is difficult to know exactly how the returned or make-up gases will mix with gas in the system. The gas exit port should be spaced several feet away from the return port. It could be assumed that gas flows through in a slug, like a piston, and that only a single pass would be required to remove all of the tritium from the gases. Slug or piston displacement flow, however, is unlikely, and a more common assumption is to assume that the incoming tritium free gases exponentially dilute the gases.

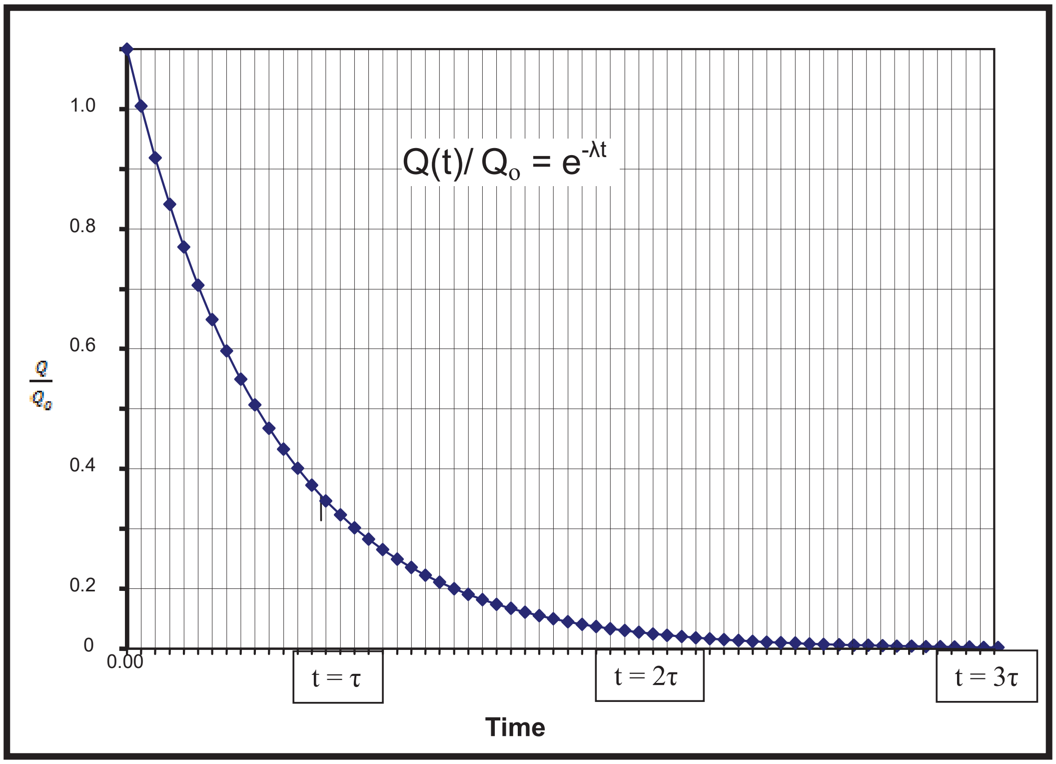
If uniform mixing is assumed, the resulting dilution at any time, t, from an initial quantity (Q_0) of tritium due to operation of an air cleanup system can be estimated with standard exponential methodology. The return air from the cleanup system is also assumed to be completely detriated.

$$Q(t) = Q_o \, e^{-\lambda t}$$

Where $Q_{\ o} = Q(t = o)$

where λ is the decay constant associated with the ventilation system equal to F/V, where F is the flowrate of the tritium removal system and V is the volume of the room cleaned. The graph of an exponential decay process is illustrated in Figure 4.5.

The quantity of tritium vs time is depicted in terms of time constants τ or 1/λ. When t = one time constant, roughly 63% of the tritium in the room as been removed; at 3 time constant, 95% removal has been achieved.



12) Figure 4-5: graph of an exponential decay process

λ is a combination of three terms: λ_R (removal system) λ_L (leakage) to account for leakage from the room to the environment and λ_D (radioactive decay) of tritium in the room, yielding

$$Q_t \;=\; Q_o\, e^{-t(\lambda_R+\lambda_L+\lambda_D)}$$

Since tritium's radioactive half-life is 12.3 years, λ_D is normally ignored; however, as discussed in section 2.3.5, Outgassing, there could be some situations where outgassing can occur over a number of months or even years (the Pantex release of 1989 discussed in Section 4.7.7 is an example of this situation), in which λ_D can affect the calculated results; however, since ignoring the term is conservative, there is little need for further consideration here; yielding

$$Q_t \;=\; Q_o\, e^{-t(\lambda_R+\lambda_L)}$$

Note that the terms λ_R and λ_L both reduce the quantity of tritium in the room. In order to reduce the amount of tritium leaking to the environment for any given room configuration with an associated leak path factor, the tritium removal system flowrate (F) must be increased. Design of room and cleanup systems is an iterative process with a higher flow rate cleanup system compensating for a leaky room and vice-versa.

4.1.3 Future Directions in Tritium Removal and Cleanup

Future tritium facilities should analyze the applicability of confinement systems in their facilities. For example, DOE O 420.1C states,

Confinement design must include the following:

a. For a specific nuclear facility, the number, arrangement, and characteristics of confinement barriers as determined on a case-by case basis.

b. The type, quantity, form, and conditions for dispersing the radioactive material in the confinement system design.

c. An active confinement ventilation system as the preferred design approach for nuclear facilities with potential for radiological release [footnote omitted]. Alternate confinement approaches may be acceptable if a technical evaluation demonstrates that the alternate confinement approach results in very high assurance of the confinement of radioactive materials.

The guidance for confinement ventilation systems and evaluation of the alternatives, is provided in DOE Guide (G) 420.1-1A, *Nonreactor Nuclear Safety Design Guide for Use with DOE O 420.1C,* Facility Safety.

As regulatory release criteria and ALARA concerns are strengthened, the desirability of barriers increases. Many past confinement systems were of the glovebox and room or building design. One disadvantage of these systems is that tritium is converted to oxide, which must eventually be handled and disposed (with the attendant risks). Room- or building-type confinement systems were used in some tritium facilities such as in the T-building at Mound, in TFTR at Princeton Plasma Physics Laboratory (PPPL), and in TSTA and Weapons Engineering Tritium Facility (WETF) at Los Alamos. If the system is not very close to 100 percent efficient, the released oxide (which is about 10,000 times more toxic than the gas) could give an overall detriment (e.g., if 10,000 Ci of gas are collected, but only one Ci of oxide is released by the system, the whole operation is just a draw). Some current designs, notably the cleanup systems at the SRS H-Area New Manufacturing Facility, make use of gettering without oxidation. The primary advantage of these getter systems is that tritium is removed and stored in elemental form and is not converted by the tritium removal system to the more radiotoxic tritiated water. The application for glovebox atmospheres is good; however, severe challenges for removing tritium from room atmospheres with the required flow rates, while not fouling the getters, appear less promising. Research into other than metal (e.g., fullerene) gettering material has been sponsored by DOE in the past; subsequently, fullerenes have been examined for hydrogen storage for the Hydrogen Economy[30]. Results indicate that stored tritium is stable for almost 10 years. The Loutfy et al. reference in section 1.4 presents an excellent overview on fullerenes.

It is possible that advances in design will make room or building-type confinement systems desirable, or it may turn out that these cleanup systems in general are found not to be cost-effective, and that a better use of resources to decrease environmental releases could be made by upgrading the existing primary and secondary systems. Additionally, there may be future glovebox decisions in which the oxidation process is still chosen over the gettering process due to programmatic reasons. There is no compelling agreement at this time to use room or building cleanup systems.

4.1.4 Inspection and Surveillance Requirements

Instrumentation should be provided to monitor the leak-tight integrity of process piping, tanks, and other equipment. The surveillance measurements may include those of pressure differentials or flow rates that relate to the design leak rate. Radiation monitoring instrumentation may be used to qualitatively assess changes in leak rate.

4.2 Separation and Purification of Tritium

Isotopes have very similar chemical properties; therefore, their separation is, in general, more difficult than chemical separation. Thermal diffusion, fractional absorption, and cryogenic distillation have been used to separate tritium from protium and deuterium. Cryogenic distillation is the most proven separation technology; it is currently used in various locations including the Darlington Tritium Removal System and is the baseline for ITER. Cryogenic distillation is truly continuous and easily scalable. Gaseous thermal diffusion, especially for smaller scale operations, has been used successfully in several countries. Other separation techniques are also in use. Materials such as palladium were known having hydrogen isotopic effects shortly after the discovery of deuterium in 1931 and tritium in 1934. Batch palladium chromatography for hydrogen isotope separation dates to the 1950s. However, a “continuous” process using Pd chromatographic separation did not appear for half a century until Dr. Myung W. Lee of Savannah River National Laboratory (SRNL, then called SRL) originated the breakthrough concept of a “continuous” chromatographic separation process in 1979. The process was successfully developed into a working unit in the 1980s and subsequently named the Thermal Cycling Absorption Process (TCAP). Because TCAP demonstrated unequivocal advantages, it has gradually replaced all other separation processes for hydrogen isotope production and is currently the sole process of purifying tritium at SRS.

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[30] R.O. Loutfy, E. M. Wexler. *Feasibility of Fullerene Hydride as a High Capacity Hydrogen Storage Material*, Proceedings of the 2001 DOE Hydrogen Program Review, DOE 2001.

TCAP is a gas chromatograph in principle using palladium in the column packing, but it is unique in the fact that the carrier gas—hydrogen—is being isotopically separated and the system is operated in a semi-continuous manner. The aspects of hydride technology that provide interim hydrogen storage, generate pressure, and create a vacuum have been innovatively applied to create the TCAP process. During operation, thermal cycling moves gas back and forth in the Pd/k column, achieving efficient isotope separation at each of the column's separation stages.

During each cycle, a gas mixture is fed into the Pd/k column, and pure product and raffinate are withdrawn from the two ends of the column. The TCAP invention is an engineering achievement with simple design and advanced control logic and is suitable for radioactive confinement. A simple on/off valve is the only mechanical moving part. The robust design presents features of inherent safety and extra-long life. It can be scaled from very small to large with versatility depending on throughput. Testing has demonstrated that TCAP can separate the naturally occurring 150 ppm D_2 in a standard hydrogen cylinder, thus producing pure protium.

4.3 Building Ventilation System

If tritium does penetrate its barriers, it can be released into the worker breathing space. In this scenario, the ventilation system should be designed to meet the following objectives:

- Move released tritium from the worker breathing space as soon as possible;
- Minimize the contamination of other areas while moving released tritium;
- Release the tritium-contaminated gases at an elevation and velocity that will result in massive dilution and mixing with outside air before the tritium reaches ground level.

The ventilation system may be designed using the following guidelines:

- The ventilation system can be a single-pass ventilation system. Outside air is brought in through a supply fan, conditioned for the comfort of the workers, passes through the ductwork to the ventilated spaces one time, goes through the exhaust ductwork to an exhaust fan, and is released to the environment through the facility stack. Room recirculation designs can also be used in conjunction with single-pass ventilation systems, the design and desirability of which is discussed in Sections 4.1.2 and 4.1.3;
- The air supply and exhaust systems should be designed to eliminate dead air spaces where tritium-contaminated gases may accumulate;
- The ventilation system ductwork should not be shared with non-tritium operations;

- To minimize cross-contamination from one room to another, the exhaust gas from each room should dump into a central exhaust duct. Exhaust gases from several rooms should not be combined before being dumped into the central exhaust duct;

- To minimize cross-contamination from one ventilation function to another, the gases for each type of function, such as room ventilation, high velocity air hood ventilation, and glovebox ventilation, from a single room should not be combined until they reach the central exhaust duct;

- To ensure good mixing and dilution by outside air, the exhaust gases should be released to the environment through an elevated stack at high velocity;

- The ventilation system should be designed to use pressure zone control to minimize cross-contamination. The ventilation control system is designed to hold the spaces occupied by the tritium operations at a negative pressure relative to the spaces surrounding the facility. If the whole building is a tritium facility, then the building is at a slightly negative pressure relative to the environment. If the tritium is in a single room in a building, the room is at a negative pressure relative to the adjacent rooms. Air continuously leaks into the tritium operating areas from the surrounding environment. The ventilation zones of tritium processing areas should be maintained under controlled temperature and humidity conditions at all times to reduce the in-leakage of moisture to inert atmosphere gloveboxes. (Further reductions of in-leakage of moisture into inert gloveboxes can be achieved by selecting a glove material that has a low permeation rate for moisture and by reducing the exposure time (e.g., glove port covers) and/or total glove surface area);

- The walls separating adjacent rooms should be reasonably sealed to minimize tritium released from contaminating an adjacent room. Administrative controls need to be in place to require caulking and sealing around wall penetrations such as conduit and piping;

- For the ventilation system to work as designed, the inside and outside doors and airlocks need to be used properly. Propping an outside or inside door open will upset the pressure zone control system. An outside door bypasses the stack and provides a path for a ground level tritium release. The doors should be equipped with automatic door closures, and all personnel should be instructed on the proper use of doors;

- For facilities handling gram quantities of tritium, a rule of thumb is 6 to 10 air changes per hour as the standard of performance;

- The ventilation rate should be based on analysis of the hazards of the operations. A facility that has the potential to release only a few curies of gaseous tritium into the breathing space does not need to operate at the same ventilation rate as a facility that has the potential to release larger quantities of tritium into the breathing space;

- Single-pass ventilation systems are expensive to operate because all air must be conditioned to make its single pass through the facility, and this conditioned air is not reused. Additionally, a high flow rate is desired in order to remove any released tritium from the facility as soon as possible to protect the workers;

- Special precautions such as personal protective equipment (PPE)—including respiratory protection and passing exhaust through particulate filters—may be needed when working with SMTs. D&D work with SMTs should be confined to occur within the workspace, if possible.

It is feasible to design future ventilation systems to operate at a variable flow rate that is a function of the time of day and the measured tritium concentration in the rooms. This would entail a higher initial cost, but would decrease the long-term operating costs without significant impact on facility safety. Additionally, the ventillation rate should be based on analysis of the hazards of the operations. For example, a facility that has the potential to release only a few curies of elemental tritium into the breathing space does not need to operate at the same ventillation rate as a facility that has the potential to release larger quantities of elemental or tritium oxide into the breathing space.

4.4 Chilled Water System

The chilled water system that is used to cool the tritium-related activities in a facility should be carefully designed to minimize the volume and tritium concentration of the contaminated water generated. The use of single-wall, water-to-gas heat exchangers in tritium removal systems and vacuum furnaces for example, will result in tritium contamination of the chilled water system. In some facilities, the same chilled water system is used to cool non-tritium activities in the same building, and, at some sites, the same chilled water system is also used in non-tritium related activities in adjacent buildings. As a result, tritium-contaminated wastewater is generated, and tritium contamination is spread from one piece of equipment to another, from one room to another, and from one building to another through the chilled water supply. This can lead to loss of control of a radioactive material.

The chilled water system should be designed to minimize the volume of water that can be contaminated with tritium. One technique is to use water-to-water heat exchangers or double-walled, gas-flushed water-to-gas heat exchangers to isolate the high volume central system from the tritium-related activities. The volume of water in the secondary loop is much smaller than that in the primary loop.

The primary reason for using chilled water for cooling equipment is cost. The systems are reliable, low in cost, small in size, readily available in many different sizes from many different manufacturers, and easy to maintain. Air-to-air heat exchangers can, in some cases, be substituted for chilled water cooling, but are larger in size and will not work in some applications. Refrigerated cooling systems are more expensive, but operationally eliminate the need for disposing of tritium-contaminated water generated by the chilled water systems. The coolant may also become contaminated with tritium under some circumstances.

4.5 Seismic and Other Natural Phenomena Design and Evaluation of Structures and Facilities

A discussion of the DOE NPH requirements is included in Section 5.7.1. These requirements apply to the facility Safety Structures, Systems and Components (SSCs). Section 5.7.2 and its references should be reviewed as part of the overall NPH design of a new, or evaluation of an existing, tritium facility.

An understanding of the types of loading produced by earthquakes, extreme winds, and other NPH is useful in planning mitigating approaches. An earthquake produces vibratory ground motion, which will affect the entire facility and its contents. Attention to anchorage and connection details to all structures, systems, and components is essential. Earthquakes may also cause ground displacement if the facility is near a fault. Loss of ground stability may also occur due to settlement or liquefaction and will depend on soil types and location of the ground water table.

Extreme straight-line winds or tornadoes generally affect the structural shell and systems and components outside of the structure. Extreme winds will produce direct pressure and suction on walls and roofs with increased loading at corners, eaves, and ridges. Extreme winds can produce missiles from debris near the facility or from nearby facilities. These missiles can strike walls and roofs and their effects should be evaluated. During tornadoes, in addition to pressure effects and windborne missiles, there will be an atmospheric pressure change or a pressure drop below ambient pressure as the tornado passes over a facility. This can affect wall and roof openings and ventilation system filters. Flooding or extreme precipitation events can also have a deleterious effect on structures and safety SSCs, as discussed in DOE-STD-1020-2012.

An approach used at SRS is to store resources in a Highly Invulnerable Encased Safe (HIVES). HIVES's are metal cabinets designed to store tritium reservoirs and Hydride Storage Vessels (HSVs). The HIVES is designed to be capable of protecting the pressure boundary integrity of stored reservoirs or HSVs against impact of structural elements freefalling from the collapse of the H-Area Old Manufacturing Facility and the adjacent stack caused by design basis NPH events. A HIVES is constructed of hardened T-1 steel, primarily from 1-inch plate. The overall dimensions are approximately 22" W x 39" D x 68" H, which includes both a cabinet and a matching bonnet assembly bolted to the cabinet top plate. This bonnet contains an aluminum hexagonal cell honeycomb material designed to absorb the impact loading stated above.

The process for determining the NPH Design Category (NDC) for facilities and individual SSCs is described in Section 2.2 of DOE-STD-1020-2012. Not all portions of a facility, or SSCs in a facility, must have the same NDC. NDC is determined by the radiological consequences of individual SSC failure. Once NDC is determined, Chapters 3-8 of DOE-STD-1020-2012 provide direction for determining the appropriate design basis loads for the various NPH. These chapters also provide design criteria, or reference national consensus standards for the criteria, to withstand the design basis loads.

Employing mitigation efforts before natural hazards occur is a very cost-effective means to provide life safety, to minimize damage and losses, and to reduce the impact on the facility and operations. It is extremely important to pay attention to all details because natural phenomena will find the weak links and cause damage.

4.6 Other Design Considerations

The following items should be considered in the design of a facility with tritium operations.

- Designated safety-class systems should employ the concepts of redundancy, separation, and diversity. Some designs in the past, notably at the SRS H-Area New Manufacturing Facility, employed redundant signals to a single actuation device. An improvement to this design concept would be to provide these redundant signals to two independent, separate (by distance) actuation devices.

- The maintainability and accessibility of components is sometimes overlooked in the original design. Inaccessibility of glovebox components can result in serious ergonomic injuries to workers. Most current glovebox designs are not designed for ease of large equipment replacement, resulting in long and usually open-box maintenance. SRS is developing a modularization of tritium component concept that allows for reduced maintenance costs and schedule. Additionally, this modularization concept results in lower glovebox volume-to-surface-area, reducing glass, gaskets, and gloves; thereby producing the added benefit of reduced hydrogen loading into the glovebox, as the majority of hydrogen loading is from the permeation of humidity through gloves and gaskets[31].

- A design requirement should be included to prevent the formation of explosive mixtures during the handling, processing, and storage of tritium gas and other hydrogen isotopes.

- Calibrated tanks and associated piping that are used for pressure-volume-temperature measurements should have surface treatment of their interiors (e.g., electropolished and passivated) to allow accurate volume measurements.

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[31] Randy Davis, *A Plan for Modularization of Tritium Components*. presented at the 34th Tritium Focus Group Meeting, Aiken, SC, April 22, 2014

- Tritium process and handling systems should use, wherever possible, nonflammable hydraulic, lubricating, and cooling fluids.

- The designer should consider not using hydrogenous fluids where they might become contaminated with tritium.

- The designer should consider providing for the retention of firewater with subsequent monitoring prior to disposal for all buildings where there is tritium.

- Barriers should be provided to prevent damage to equipment and injury to personnel while performing testing operations that could produce missiles or blast pressures. These barriers should be designed using conservative and proven design principles, such as those of DOE/TIC 11268, *A Manual for the Prediction of Blast and Fragment Loading of Structures*.

- An independent air system should be provided for breathing air. It should have dedicated, oil-free compressors or pressurized cylinders of breathing air, and provide breathing air in all areas of the tritium facility where it may be needed for maintenance operations and/or personnel safety. Contamination of the air supply (e.g., from refrigerant leaks or air intakes) should be detectable at levels low enough so as not to pose health concerns.

- The design considerations of the Fire Protection System should include the following:

    - determining safety classification of structures, systems, and components (SSCs) of both fire detection and fire suppression systems;
    - consideration of unique fire sources (e.g., uranium beds used for tritium storage);
    - compatibility of fire extinguishing agents with the fire sources in a tritium facility; and
    - containing and handling requirements for expended firefighting agents (including water inventory) that may become contaminated with tritium.

4.7 Lessons Learned

SRS Building 232-H, the H-Area New and Old Manufacturing Facilities, and the Tritium Extraction Facility; the WETF, located at Los Alamos National Laboratory (LANL); and the Tritium Research Laboratory (TRL) at SNLL are examples of facilities that were initially designed to perform tritium handling. Other facilities in use now were originally designed to do other work and have been retrofitted to perform tritium handling.

One of the major problems with the facility retrofit process is that the existing utilities such as ventilation, floor drains, gas supplies, and chilled water systems are shared with the adjacent non-tritium areas. The same ductwork, which is used to sweep released tritium from tritium operating areas, is used to provide ventilation for offices and non-tritium areas, and, as a result, tritium back diffuses into the non-tritium areas through the shared ductwork. The same chilled water system used to cool the tritium-related equipment is used to cool the non-tritiated office spaces, and leaks of tritium-contaminated chilled water result in contamination of clean areas. The floor drains from the non-contaminated areas drain into the same system as the floor drains from the tritiated areas, and, through the drains, gases flow from one area to another.

Tritium facility utilities such as chilled water, compressed air, gas supplies, ventilation, floor drains, sink drains, storm drains, and stacks should not be shared with other non-tritium areas. Sharing these systems spreads tritium contamination to other areas, complicates day-to-day management of the facility, and impacts the transition process if the facility is transitioned to other use. The use of hazardous materials should be reduced or eliminated in the initial design stages of the facility, as it will likely lead to the generation of mixed wastes and increased decontamination and decommissioning (D&D) costs.

4.7.1 SNLL Tritium Research Laboratory

Tritium operations have been terminated at the TRL, and the facility has been transitioned to other use. The problems encountered during transition of the TRL can serve as an example for facility designers of the future. The TRL was designed beginning in 1972 specifically to handle tritium, and a few changes to the initial design would have resulted in significant cost and timesaving during the transition process.

- The tritium removal system in the central glovebox had a supply and return manifold fabricated of six-inch diameter stainless steel pipe. The associated vacuum pump effluent manifold consisted of an all welded two-inch diameter pipe. These manifolds were approximately 200 feet long and extended down the central corridor of the building. The manifolds extended into each room from the corridor and were of all welded construction. During the transition process, special tooling had to be purchased to cut the six-inch and two-inch diameter pipe into sections small enough to be disposed of as solid waste.

  Designing the system to include flanges and isolation valves at specific locations would have resulted in some cost savings during facility dismantlement and transition. However, since too many valves and flanges can increase the potential for leaks during operation, the installation points should be placed only at strategic locations.

- The floor covering installed in the TRL consisted of 12-inch tiles glued to the concrete floor. Both the floor tile and the adhesive contained asbestos that had to be removed during transition of the facility. Additionally, tritium-contaminated liquids were spilled on the floor during operation and leaked through the tile into the concrete below. The use of adhesives made of non-hazardous materials would have resulted in significant cost savings.

PPPL has suggested that sealing the concrete with a thick, hard finish epoxy paint prior to installation of any floor covering would mitigate the impact of tritiated liquid spills, although even epoxy will experience tritium permeation.

As untreated concrete can hold a large amount of tritium—and it is difficult to decontaminate once it is absorbed—the selection and application of an appropriate coating or paint is critical to eventual decontamination efforts. Studies are underway in Japan that detail the effectiveness of various coatings[32] SRNL has deployed a very effective decontamination process for contaminated concrete (and soil) similar to a barbeque pit-type arrangement[33].

- The TRL was equipped with a recirculating chilled water system. The water-to-gas heat exchangers used the chilled water to cool the glovebox and vacuum effluent tritium removal system gases, vacuum furnace heat exchangers, glovebox temperature control systems, and in a variety of other tritium-related tasks. After a few months of operation, tritium contamination was found in the chilled water. In order to control the buildup of tritium in this system, the contaminated water was periodically drained from the system and replaced with uncontaminated water.

  If the initial design had used water-to-water heat exchangers as barriers, the volume of contaminated water would have been minimized. If double-walled, gas-flushed, water-to-gas heat exchangers had been used; the contamination would have been significantly lower.

- The TRL used approximately 100 oil-type vacuum pumps. Oil-free vacuum pumps were not as common when the TRL was designed. The vacuum pump oil becomes contaminated with tritium during use. At some DOE locations, tritium-contaminated oil is regulated as a mixed waste. Handling the tritium-contaminated oil is a significant safety hazard to operating personnel and should be eliminated where practical.

  If oil-free vacuum pumps had been used, the generation of mixed waste in the form of tritium-contaminated vacuum pump oil would have been eliminated as would the hazard associated with changing vacuum pump oil.

- The TRL was equipped with a ten-air-change-per-hour, single-pass, pressure-zone-controlled ventilation system. Some of the ductwork became contaminated and was removed and disposed of as solid low-level (radioactive) waste during the transition process.

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[32] Yuki Edao, *Measurement of Tritium Penetration through Concrete Material with Various Paints Coating.* Presented at the Tritium 2013 Conference, Nice, France

[33] Dennis Jackson et. al., Thermal Removal of Tritium from Concrete and Soil to Reduce Groundwater Impacts. Presented at the 34th Tritium Focus Group Meeting, April 22, 2014

Experience from SNLL has suggested that if the ventilation ductwork for room air ventilation had been separated from the ductwork used for high velocity air hood and glovebox ventilation, the contamination would have been easier to control during the transition process.

- The presence of hazardous materials in the form of asbestos and oil complicated the transition process. The materials used in fabrication and during facility operation should be reviewed, and, if possible, all hazardous materials should be eliminated.

- The initial design of the TRL included collection of wastewater from floor drains and sinks in two underground holding tanks so that the water could be checked for contamination before it was sent for disposal. The holding tanks were buried and could not be inspected. The wastewater drain system consisted of several hundred feet of buried drainpipe, which drained into the holding tanks. After a few years of operation, the underground tanks were replaced with holding tanks enclosed in a below-ground-level open concrete pit. If the tanks leaked, the concrete pit would contain the leak. This design also provided for inspection of the tanks for leaks. The buried, underground drain lines remained in place throughout the life of the facility.

  Wastewater holding tanks and collection systems should be designed so that potential leaks can be contained and the holding tanks and drain system can be inspected periodically.

4.7.2 SRS Old Tritium Extraction Facility

Between 1995 and 1997, a team of specialists performed D&D work on the Old Tritium Extraction Facility (232-F) at the SRS[34]. The work was conducted in six phases. Each phase presented challenges regarding the behavior of tritium, particularly tritium in equipment and structures of metal and concrete. A description of each phase is as follows:

- Phase I, Pre-Characterization/Isolation, was designed to identify the specific buildings, ancillary equipment, structures and premises to be cleaned, and isolate them in a way that avoided inadvertent contamination elsewhere in the complex;

- Phase II, Detailed Facility Characterization, included determination of the type and amount of actual contaminants and determined the location and type of disposal operations required. This phase was the most complex in terms of characterizing tritium contamination;

- Phase III, Decontamination and Dismantlement, involved removing hazardous materials such as PCBs, mercury, and lead, as well as radioactive constituents such as tritium and fission products. It also involved dismantling equipment and some structures;

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[34] DOE/SR-5000-510, “Tritium Facility Decommissioning, Pioneering Success at the Savannah River Plant,” October 1997

- Phase IV, Demolition with Explosives, involved the removal and destruction of the facility and stack;

- Phase V, Waste Management, was a continuing process throughout all six phases. It involved the determination of how much of each contaminant was present at the site and where the various site contaminants would be disposed;

- Phase VI, Green Grass Restoration, restored the site to a usable, visually aesthetic entity.

Detailed characterization was one of the most challenging phases in dealing with tritium, and required an expert understanding of the behavior of tritium in porous and nonporous materials. For example, because tritium migrates to the subsurface of porous materials, volumetric characterization required core boring. This was not well understood initially and early characterization activities, which involved smear and scabbling techniques or wiping and scraping the surface of concrete, gave false (low) tritium contamination levels. In some instances, areas that were smeared clean began to reveal elevated readings following rain in-leakage. In addition, destructive testing for nonporous material, such as metals, was required to fully characterize volumetric tritium contamination.

DOE-STD-1120-2005, *Integration of Environment, Safety, and Health into Facility Disposition Activities*, provides guidance for integrating and enhancing worker, public, and environmental protection during facility disposition activities.

4.7.3 Component Plugging at SRS Tritium Processing Facility

One of the tritium processing facilities at SRS experienced plugging of multiple components. The first indication was that the let-down filter on the compressor plugged; upon further inspection, fine particulate was found widespread in the facility, requiring mitigation and downtime. Sources of particulate contaminants included polymers used in compressor poppets, valve packing, valve seats, other components, and material introduced during maintenance such as lubricants (Krytox® and SWAK®), cleaning agents (Freon® and Vertrel®), along with dust and dirt from line break activities.

The general lessons learned captured by the SRNL investigators are germane to most tritium facilities and worth bearing in mind:

- wear and tear of components happens;
- tritium degradation of materials happens, and critical systems should be protected to the maximum extent possible;
- appropriate protective measures should be implemented (e.g., filters);
- particulate generation probably cannot be eliminated, but can be minimized; and

- particulate flow, migration, and accumulation should be considered in facility design[35]

4.7.4 Tritium Compatibility Lessons

4.7.4.1 Unanticipated Material Interactions

The damage done to organic materials by the presence of tritium in the internal structure of the material is not limited to the more obvious radiation damage effects. Tritium, particularly in the form of T+, has the insidious ability to leach impurities (and nonimpurities) out of the body of the parent material. In many cases, particularly where halogens are involved, the damage done by secondary effects such as leaching can be more destructive than the immediate effects caused by the radiation damage. In one such case, the tritium contamination normally present in heavy water up to several curies per liter was able to leach substantial amounts of chlorides out of the bodies of neoprene O-rings that were used for the seals. The chlorides leached out of the O-rings were subsequently deposited into the stainless steel sealing surfaces above and below the trapped O-rings, which led directly to the introduction of chloride-induced stress-crack corrosion in the stainless steel. The operational conditions that set up the introduction of the stress-crack corrosion were moderately elevated temperatures (i.e., less than 100 °C), low pressures (i.e., less than 3 atmospheres), and exposure times of 3-5 years. Fortunately, the damage was discovered before any failures occurred, the neoprene O-rings were removed, and the seal design was changed to a non-O-ring type of seal.[36]

In a second case, six failures out of six tests occurred when high-quality Type 316 stainless steel was exposed to tritium gas in the presence of Teflon TM shavings and 500 ppm moisture. All of the failures were catastrophic, and all were the result of massively induced stress-crack corrosion. The conditions that set up the introduction of the massively induced stress-crack corrosion in this case were moderately elevated temperatures (i.e., 104 °C), high pressures (i.e., 10,000 to 20,000 psi), and exposure times that ranged from 11 to 36 hours. Since the time to failure for all the tests was directly proportional to the pressure (i.e., the higher pressure tests failed more quickly than the lower pressure tests), since identical control tests with deuterium produced no failures, and since comparable testing without the Teflon™ shavings indicated no failures after 3,200 hours, it was concluded that fluorides were being leached out of the Teflon™ and deposited directly into the bodies of the stainless steel test vessels.[37] An interesting sideline to this test is that, after the tests, the Teflon™ shavings showed no obvious signs of radiation damage (i.e., no apparent discoloration or other change from the original condition.

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[35] P. Cloessner, SRNL, *Particulate Generation in a Tritium Facility*, presented at the 33rd Tritium Focus Group Meeting, Aiken SC, April 22-24, 2014.

[36] DOE HDBK 1132-99, Reaffirmed 2014, Design Considerations

[37] DOE HDBK 1132-99, Reaffirmed 2014, Design Considerations

4.7.4.2 Valves with incompatible valve material procured, installed, and tested ready for use at a new SRS Facility

The author of this Standard was a member of a DOE/NNSA Operational Readiness Review (ORR) team pre-visit for a new SRS tritium facility in 2006, not because of his tritium expertise, but rather due to his responsibilities as the POC for the Office of Primary Interest (OPI) for DOE O 425.1 in the area of nuclear facility startup and restart; there were in fact no other tritium personnel on the ORR Team. The author, in 1993 was the lead for a review team for initial startup of a SRS tritium facility and not being convinced by WSRC's assentation's of the capabilities of Dupont's ultra-high molecular weight polyethylene UHMWPE to withstand tritium environments without degradation, had negotiated with WSRC a periodic surveillance program and accelerated bench testing of the UHMWPE valves as one of the conditions for his team's recommendation for initial startup of this facility. Partly based on this experience, the author saw a need for DOE-wide guidance in the area of overall criteria for selection of valves in tritium service. He researched and published a number of articles[38] [39]and a DOE guidance document[40] for selection of proper types and materials of valves for use in differing applications and varying degrees of tritium service. These publications include discussions of significant historical DOE tritium release events.[41] [42] [43] [44]. The number-one recommended practice in references 40 through 42 was "do not use halogenated elastomers or polymers such as Kel-F™ Teflon®, or Viton® for wetted parts in valves used for extensive tritium service." The author, being fluent in the language of the part numbers stamped on the bodies of the tritium valves was able, during the pre-ORR visit walking tour of the SRS facility in 2006, to identify inappropriate (e.g., Kel-F™) valves in the tritium systems. The ½" BK series Nupro® valves contained PCTFE (polychlorotrifluoroethylene) stem tips and a PTFE (Teflon®) coated bellows gasket, which are incompatible with tritium service, and were being used throughout the facility violating the number-one recommendation above.

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[38] W. Weaver, *Guidelines for Valves in Tritium Service*. Fusion Technology July 1994.

[39] W. Weaver. Selection Criteria for Valves used in Tritium Service. Presented at the International Workshop on Physics and Technology of Tritium for Fusion Reactors, Varenna, Italy, 6-14 September 1993.

[40] W. Weaver. EH Technical Notice 94-01, *Guidelines for Valves in Tritium Service, 1994.*

[41] *Tritium Release During Downsizing Activity (B-331)*, Occurrence Report SAN-LLNL-LLNL-1991-1002, U.S. Department of Energy, April 1991.

[42] *Tritium Release – Equipment Failure,* Occurrence Report ALO-KO-SNL-TRL-1992-0002, U.S. Department of Energy, June 1992.

[43] *Enviroment Safety & Health Safety Note,* 91-3, U.S. Department of Energy, March 1992.

[44] *Tritium Release Not Part of Normal Operation*, Occurrence Report ALO-LA-LANL-TSTA-1991-0065, U.S. Department of Energy, March 1991.

The subsequent investigative review[45] performed by the Contractor also identified that these same valves were procured and were being used as spares for the operating facilities. Fortunately none of the 105 spares had yet been placed in service in the operating facilities and tritium had not yet been introduced into the new facility with its 195 installed valves. A thank-you call[46] from the Federal Project Director of the new facility to the author indicated that he had averted a major disaster by identifying the issue before tritium was introduced.

This event highlights the need for technically competent independent reviews, which is a recurring theme across the DOE complex. Both aspects, technical competance and independence are required for success. Too often, management (DOE and Contractor) views safety or independent technical reviews as an impediment to progress and press on without adequate independent technical reviews. It is rare that these reviews are detrimental, as they are neither costly nor time consuming, and are usually able to identify issues if they exist, before they become problems. One need only to look at the lack of early identification of the multiple design issues at the Waste Treatment and ImmobilizationPlant (WTP), the technical design problems delaying startup at the Integerated Waste Treatment Unit (IWTU,) the undiscovered design errors at Uranium Processing Facility (UPF) and the LANL/WIPP waste packaging issue to illustrate the need for better independent technical reviews. For example, during the packaging of waste containers at LANL for disposal at WIPP, the procedure called for adding chemicals that were not compatible within the same container, which might result in a potentially reactive mixture. During the interview process at LANL, organic chemists stated to this author, who was part of the accident investigation of the WIPP radiological release event of February 14, 2014[47], that if they had been involved in the review process of the packaging process, they would have identified this incompatibility issue in the procedure as a poor technical practice, independent of any potential consequences. The cost associated with this type of review would have been minimal. Additionally, the Inspector General (IG) Management Alert of September 2014[48] further identified a lack of adequate technical review as a weakness in the LANL process. Likewise, the Tritium Focus Group discussed that the author's pre-ORR walkdown was probably the least amount of time expended for the most amount of money saved in the history of the DOE complex. While this may indeed be true, the technically competent reviewer has invested much time to achieve the expertise and should be used to the fullest extent possible. However, a recurring theme seen in these major mishaps is management's lack of awareness and use of technical personnel available. Even in the example of the success of the author's pre-ORR identification, it was due to his OPI function for the ORR and not tritium expertise that he was involved with the ORR team. Shortly after returning to Headquarters from the pre-ORR visit, his organization was dissolved, and as such he was no longer involved with OPI activities for the

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[45] ORPS Report NA--SRSO-WSRC-TRIT-2006-0002, *Incompatible Nupro Valves Stem Tip Material*, finalized 09/28/2006.

[46] Personal communication from Kevin Hall to Bill Weaver, August 15, 2006.

[47] DOE EM Accident Investigation Phase II Report, Radiological Release at the Waste Isolation Pilot Plant, April 2015.

[48] Management Alert, US DOE Office of the Inspector General, DOE/IG-0922, September 2014.

ORR. Although he was the only Federal employee in the NNSA/DOE complex with tritium in his job title, he was slated for transfer to an organization that had little to no involvement in tritium activities, and was removed from the ORR team.

4.7.5 Oxygen monitor failures

Older oxygen monitors have been observed to be failing at a SRS Tritium facility and spare board are difficult to obtain. These boards are the essential components for the Delta F analyzers. The Tritium facilities currently have 24 of these analyzers that are used to monitor the Oxygen levels in the glovebox atmosphere. These analyzers are credited Safety Significant controls and their purpose is to ensure the glovebox oxygen concentration is below the minimum concentration required to support combustion inside the glovebox.

Since the Delta F analyzers are over twenty years old, the manufacturer no longer supports this analyzer and spare parts are no longer available. Until the upgrading of the facility's oxygen monitor systems is complete, the Delta F system must remain operational. Hence, the option to repair the circuit boards was pursued.

During the repairs, it was discovered that the opamps in the cell amplifier and x100 amplifier were the most common failures. A pattern developed indicating that LTC1500 and MAX430 opamps were failing. Due to the location of these opamps, it is suspected that the age of the chips is at fault. Since these chips are not in production, surplus parts were used to replace failed chips. The replacement opamps also fail on a regular basis.

To bolster the spare parts program, a project was funded to reverse engineer and redesign the Delta F main board. The redesigned board will significantly reduce the failure rate and allow the normal process for instrument replacement to continue.

4.7.6 Tritium flammability limits in glovebox atmospheres.

Los Alamos National Laboratory declared a Potential Inadequacy of the Safety Analysis (PISA) documented in the September 17, 2014, Occurrence Reporting and Processing System report, NA-LASO-LANL-TRITFACILS-2014-0006[49]. This PISA was due in part by the fact that hydrogen species (including tritium) have a lower flammability limit in argon/air mixtures compared to air alone.

The presence of inerts (argon, nitrogen) changes the hydrogen lower flammability limit (LFL) and changes the limiting oxidant concentration (LOC). The Combustion Research Center (CRC) 2006 publication "Flammability Limits of Hydrogen – Air – Argon and Hydrogen – Air – Nitrogen Mixtures" reported a hydrogen LFL of 3.68% (~3.7 vol. % hydrogen, 26.8 vol. % air, and 69.5 vol. % argon) and

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[49] Occurrence Reporting and Processing System report, NA--LASO-LANL-TRITFACILS-2014-0006, September 17, 2014

an oxygen LOC of 3.33% (~6.5 vol. % hydrogen, 16 vol. % air, and 77.5 vol. % argon)" [50]. Per the CRC 2006 publication, nitrogen environments yielded a higher LFL (4.44% hydrogen) and higher LOC (4.87%) compared to argon environments. As discussed in section 4.1.1.a(2)(b), *Medium-Quality Secondary Container*, tritium gloveboxes in the DOE complex are operated at a slight negative pressure to the room and could experience air in-leakage resulting in the Hydrogen-Air-Argon or Hydrogen-Air-Nitrogen atmospheres such as those examined in the CRC.

Other glovebox atmospheres such as argon-hydrogen-oxygen, which are atypical for DOE, have been shown to have LFLs that are lower than for argon-hydrogen-air environments[51]. An experimental data point [52] indicates a lean flammability limit below the 3.68% CRC 2006 value although details are lacking at this time to correlate these findings with LFL and the lean flammability results are not in agreement with published LFL data. For example, a report from Fauske & Associates [53] states: *"*for the case of air ingression into an existing H2-Ar mixture, a lower oxygen limit between 4% and 5% is recommended by using a linear fit: $x_{O2} = 5 - (x_{Ar}-50)/20$ where x is mole percent and $50\% < x_{Ar} < 70\%$. Use 5% O_2 when $x_{Ar} < 50\%$ and use 4% O_2 when $x_{Ar} > 70\%$. However, the mixture is completely inert when $x_{Ar} > 76\%$."

Other publications [54] test mixture flammability inside a long tube instead of inside a pressure vessel as specified by ASTM standards. Flame propagation inside a long vertical tube ignited from the bottom is aided by natural convection. These experiments therefore produce more conservative flammability results (i.e., broader lean and rich flammability limits). The applicability of these results vs. pressure vessel flammability results for gloveboxes needs to be determined.

Not all reported results in this area are of the same quality or applicability. The CRC data has the highest pedigree and is the best to reference. It has the advantage of using a standardized accepted test method (ASTM) and better instrumentation compared to early tests on flammability. Tests were reproduced and thus the accuracy of the data is available. The Fauske report uses data from the Van Heiningen report which was issued in 1936. The apparatus used for the 1936 study involves vertical tubes and potential for hydrogen buoyancy as discussed above. Additionally, the Van Heiningen report uses several different references for work done even earlier. Data is reported to one decimal point; no indication is provided on reproducibility. The CRC study reported values measured to two decimal points and generated many more data points to determine the flammability curve. The early studies used oxygen instead of air whereas the CRC studies used air.

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[50] Combustion Research Center, Flammability Limits of Hydrogen – Air – Argon and Hydrogen – Air – Nitrogen Mixtures, Report No. CRC-2709 Rev. 3, Issued May, 9 2007

[51] Shebeko, Y. N., *et al.* "The Influence of Inert Retardants on the Combustion of Hydrogen-Oxygen Mixtures under Elevated Temperatures and Pressures". *Combustion, Explosion, and Shock Waves*. Plenum Publishing Corp., Vol. 30, No. 2, 1994

[52] Terpstra, M. A. "Flammability Limits of Hydrogen-Diluent Mixtures in Air", University of Calgary, 2012

[53] Flammability and Combustion of H2, Air or O2 and Ar Gas Mixtures, M.G. Plys, Attachment 11 to SNF-22059, Rev 0 July 2004

[54] Coward, HF and Jones GW, Limits of Flammability of Gases and Vapors, Bulletin 503, US Bureau of Mines, 1952

Because of the questions raised in citing old data from questionable test apparatus, instrumentation that is not as precise as what is available today, and use of oxygen instead of air, these results have less credibility than the CRC values.

Safety analyst and operations personnel associated with tritium gloveboxes using argon or nitrogen should be aware of the potential adverse impact on LFLs and LOCs based on their specific conditions and configurations.

4.7.7 Pantex Tritium Release from Gas Reservoir

Pantex experienced a large tritium release in 1989. This and other releases across the DOE complex concerned Secretary Watkins to the point that he commissioned an eight member Tritium Task Group (TTG) in 1991, to visit every DOE tritium facility and review not only the releases but also to provide him with insight on how to improve operations. This author was one of the eight TTG members, three of which are still active DOE employees. There was also a thirteen member Working Group available for support to the TTG. The current Tritium Focus Group (TFG), which now meets semi-annually and whose Charter is Appendix H, was initially formed to respond to the findings and recommendations of the TTG. Only one unintentional tritium release has occurred at the Pantex Plant in its 40-year operating history. The incident occurred on May 17, 1989, during a normal weapons disassembly and retirement operation. An electroexplosive-actuated valve operated accidentally, resulted in discharge of tritium from the gas reservoir into the disassembly cell, and ultimately to the environment. The incident resulted when the disassembly technician, who did not realize the valve had malfunctioned, loosened a gland nut on the reservoir. The equipment contaminated by this event continued to offgas tritium, which accounted for the majority of the tritium release from the site in fiscal year 1990.

When the tritium was released into the cell, portable and fixed tritium monitors alarmed and all workers immediately exited from the cell. Of the five workers in the area at the time of the incident, four received negligible exposures while the fifth worker received a significant dose.

As a result of this incident, three investigation teams were assigned to examine various aspects of the event. The first team evaluated the cause of the accident, assessed the nature and extent of the tritium release, personnel exposures, and Plant contamination, and determined the adequacy of the Plant's emergency response at the time of the incident. A second team evaluated the design characteristics of all electroexplosive devices in the weapons stockpile and assessed weapons handling procedures at the Pantex Plant. The third group reviewed tritium operations at the Pantex Plant and the associated airborne effluent monitoring, filtering, and confinement systems.

The purpose of the third team's review was to identify options for improving the safety of tritium operations and to determine associated costs and implementation schedules. The review focused on what actions could be taken at the Pantex Plant to eliminate or mitigate the consequences and effects of tritium release on assembly and disassembly workers, other Pantex Plant employees, the general public, and the environment.

As a result of the first team's investigation and recommendations, changes were made in the Pantex Plant disassembly operations procedures. The operators now monitor for tritium prior to and during the tritium reservoir removal. Additionally, improvements were made in operator radiological training and in Plant operating and emergency procedures, and the health physics staff was increased.

The second investigation team recommended modifications to the design characteristics of electroexplosive devices in the weapons stockpile and revisions to the Pantex Plant weapons handling procedures.

The third investigation team identified four improvement options in equipment and facilities, and five improvement options in environmental monitoring and protection.

5.0 DESIGN OF EQUIPMENT

The design requirements for tritium are a function of the tritium form, quantity, concentration, pressure, and period of storage. High concentrations of tritium gas stored at high pressure (> 2,000 psia) are difficult to contain due to tritium and helium embrittlement of the container materials. Design of these systems requires careful selection of the materials of construction and should be designed using expertise in high-pressure tritium containment.

Low concentrations of tritium in gaseous form mixed with other gases at low (< 600 psia) to medium (600 to 2,000 psia) pressure, regardless of the quantity, do not significantly impact the strength of the materials they are stored in. As a result, standard designs can be used.

Tritiated water in the form of T_2O is somewhat corrosive unless properly stored with an overpressure of T_2 gas. This is due to the suppression of the formation of oxygen in the cover gas and peroxides in solution[55]. Tritium systems should be designed by persons with tritium experience.

Low concentrations of HTO (mCi/mL to Ci/mL) recovered from tritium removal systems have proven to be corrosive when stored in liquid form in metallic containers and have resulted in the development of significant leaks in containers within days or weeks. Storage of this same water solidified on clay or on molecular sieve material, regardless of the quantity, is stable and noncorrosive and may be stored for many years in the container.

5.1 Material Compatibility

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[55] John Gill, Babcock & Wilcox of Ohio, Personal Communication to W. Weaver, July 1998.

Proper materials selection and rigorous design have led to tritium-handling systems that are extremely safe for long periods of time. Materials exposed to tritium under certain conditions can be susceptible to hydrogen embrittlement. The chances of embrittlement are significantly reduced by proper material selection.

Tritium can permeate vessel barriers, especially in components operating at elevated temperature. Currently available tritium permeation data are normally sufficient to estimate the order of magnitude of tritium permeation through barriers. These estimates can be used during system design or to determine whether additional purging and stripping systems are required to “clean up” permeated tritium or whether other design changes (such as wall thickness, material, or coating) are required to reduce permeation. Tritium permeation can lead not only to contamination outside the barrier, but it can also result in significant quantities of tritium being dissolved in parts, which can lead to hydrogen embrittlement. Over time, this tritium decays to ^{3}He, which has been found to accentuate hydrogen embrittlement and to cause weld cracking (see below).

5.1.1 System Design

High quality tritium primary containers should have a low leak rate and probability of failure or leaking. The consequences of tritium leaks can include personnel uptake, release to the environment, ignition (if mixed with oxygen), and violation of operating permits. Pressure and vacuum vessels used in tritium systems are generally designed and constructed using codes and standards applicable to boiler and pressure vessels. In the United States, use of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code is recommended but not required. Pressure and vacuum vessels constructed using this code have an accepted rigor in design, construction, and inspection that facilitates approval and acceptance by regulators. The ASME Boiler and Pressure Vessel Code primarily cover the design of vessels, but does not cover all of the aspects of a tritium system design. No real tritium compatibility specifications exist for tritium relief devices, and ASME leak rate standards are only in the 10^{-3} leak rate region. Other design standards, such as resistance to seismic events, also are to be followed, depending on the location and regulators of the facility. See Sections 4.5 and 5.7 for more discussion on seismic design.

5.1.1.a Leak Testing

After fabrication, thoroughly leak-testing tritium systems are extremely important. Normally, leak testing employs commercial helium-mass-spectrometer-based systems and is performed after any other required proof, pressure, or vacuum performance tests. Other leak detection methods, such as rate-of-rise, can also be employed in addition to helium mass spectrometry. A dilute solution of tritium in an inert gas can also be used to detect small leaks. Tritium is a highly effective leak detection species, since it travels rapidly through cracks and can be easily detected at very low levels.

5.1.1.b Joining

Components of tritium systems are commonly joined by welding. Welds are normally designed so they can be non-destructively inspected by a suitable method such as radiography or ultrasonic testing. Also, the weld design should minimize so-called “virtual leaks” on the interior of tritium containing volumes. Examples of weld practices to minimize virtual leaks include 1) using full penetration welds where possible, and 2) welding feed-through to the interior wall surface (not on the outside, which would leave a gap on the inside around the feed-through that is difficult to outgas). Standard weld rod filler materials are chosen, depending on the base alloy. Every effort should be made to reduce or eliminate residual welding stresses. Oxide-free welds are also preferred to decrease tritium holdup due to tritium-protium exchange on the oxide. The welds can be produced by inert gas flow through tubing and piping or by wire brushing using a suitable alloy wire brush.

There is less experience using other joining methods such as brazing or high temperature soldering in tritium systems. Dissimilar materials may have to be joined by a transition junction, using an intermediate material to enable proper welding and accommodate differences in thermal expansion and other properties.

Tritium gas permeates austenitic stainless steels, and, over time, ^{3}He is created by beta decay of tritium in solution in the material. Welding stainless steel containing solute helium is difficult because intergranular cracking can occur. During welding, the solute helium agglomerates at grain boundaries and forms both intergranular cracks in the heat-affected zone and pores in the fusion zone. Low-heat-input weld techniques have been shown to mitigate this problem to some degree; however, weld repair of helium-containing stainless steel is normally difficult to perform without some cracking.

All-metal mechanical joints are also a sound way to join components in tritium systems. Typically, copper, silver-plated nickel, or silver-plated stainless steel have been used as gaskets. Commercial high and ultrahigh vacuum fittings are normally compatible with tritium.

5.1.1.c Surface Coatings and Treatments

Aluminum and aluminide coatings have been successfully employed on stainless steel to reduce permeation into and through the steel. These coatings can be applied on large items using a proprietary fluidized bed furnace, having a controlled atmosphere (in the so-called "calorization" process), using a suitable aluminum-rich slurry coating, employing chemical vapor deposition, or a "pack bed" process. Gold has also been used as a permeation barrier in some applications and is often applied over a thin nickel buffer layer ("strike") that has been applied to the bulk metal (e.g., stainless steel) after proper surface preparation.

Several companies treat stainless steel surfaces using various proprietary electrochemical processes to "passivate" the surface. These processes probably enrich the chrome content of the surface oxide, polish the surface (thereby reducing the effective surface area), and remove carbon and hydrogen from near the surface. All of these microstructural changes may be desirable for tritium systems in which the process gas must remain at high purity. Capillary lines that route gas to mass spectrometers are commonly passivated to reduce changes of gas composition by isotope exchange, while the gas flows from the sample location to the mass spectrometer. Passivated surfaces reduce the rate of isotope exchange in hydrogen isotope mixtures, probably by reducing the catalytic effect of the surface decomposing hydrogen isotope molecules to atoms, which enables isotope exchange. Vacuum systems having surfaces treated in this way evacuate faster. This type of surface passivation can be expensive; so many parts of tritium systems are not passivated; normally only parts requiring the special properties of passivated surfaces are treated.

5.1.2 Structural Metals

Exposure of metals to high pressure (> 2,000 psia) hydrogen, deuterium, or tritium may result in hydrogen embrittlement of the material. This could eventually result in material failure. The time until failure is a function of the container material, the pressure, and temperature. Additionally, materials exposed to high-pressure tritium are also subject to helium embrittlement. Tritium at high pressure enters the metal and decays to ^{3}He. The buildup of helium in the metal results in helium embrittlement, which, depending upon the pressure, temperature, and type of material, will eventually result in failure of the material. Exposure of metals to low- to medium-pressure tritium at normal temperatures does not generally result in material failure within any reasonable period of time.

Some metals are more resistant to embrittlement than others and, therefore, are more compatible with tritium. Depending upon the specific application, Types 304L and 316L stainless steel are generally considered to be the most hydrogen compatible and readily available stainless steels for tritium service. High-pressure vessels, valves, and tubing designed of these materials when used at their rated pressure and temperature will provide many years of service without material failure. When equipment is designed for tritium operations, a materials expert should be consulted to ensure that the materials selected are compatible with their intended service.

5.1.2.a Austenitic Stainless Steels

The recommended materials of construction for tritium-handling systems are from the class of wrought 3XX series of austenitic (face-centered-cubic) stainless steels, including Types 304L, 316L, and 347. Types 304L and 316L stainless steel are most often used in tritium processing systems, but are limited to temperatures less than 425 C. These steels provide good strength, weldability, and resistance to hydrogen embrittlement. Components fabricated from these materials are procured routinely. Many commercially available vacuum system components that are used in tritium systems, such as valves, piping, pumps, and analytical instrument sensors, are fabricated from these types of austenitic stainless steel. Wrought materials are preferred to cast because wrought materials normally have a more homogenous microstructure. In the past, tritium has leaked through parts having poorly oriented stringers and inclusions, including seamless tubing. The forging direction of some wrought components has been specified so that the orientation of inclusions is not in a direction that could result in a tritium leak path. Low carbon grades (such as 304L and 316L) are preferred to avoid weld sensitization and to reduce the number of inclusions (impurity particles such as oxides and carbides). Modern vacuum-arc-remelted steels are a good choice because they have lower impurity levels, thereby resulting in fewer inclusions that could aid hydrogen-induced cracking or provide leak paths. Typically, tritium system components employ seamless pipe and tube where practical.

Stabilized grades, such as Type 347, have been employed in applications where post-weld heat treatment is not possible. This usually occurs when a process vessel contains a working material (such as hydride or getter) that will degrade when exposed to the post-weld heat treatment, which is typically performed at about 1,100°C for austenitic stainless steels.

High carbon grades, such as Type 347H or Type 316H (having 0.04 percent carbon minimum), have been successfully employed for tritium service if high temperature strength is required. Type 347H is employed in the Hydride Transport Vessel (see section 6.2.2), and Type 316H was employed for the SRS Building 232-H Extraction Furnace retorts. Type 310 stainless steel has good oxidation resistance and can be considered for elevated temperature applications if oxidation is a concern. Type 316 stainless has superior creep resistance but inferior oxidation resistance to Type 310.

Some types of higher-strength austenitic stainless steels not generally employed for tritium service may be required for fasteners (such as nuts and bolts) in mechanical joints in high-temperature regions. This may be acceptable if the bolts are exposed to only residual amounts of tritium. These materials also may be used to contact 3XX stainless steels to avoid galling of mating screw contact surfaces. Typical materials used in these applications include Nitronic 60, Nitronic 50 (also called 21-13-9) and Nitronic 40 (also called 21-6-9); these are all nitrogen-strengthened austenitic stainless steels.

5.1.2.b Copper and Copper Alloys

In principle, copper is a suitable material in tritium systems. Copper has several tritium-compatible properties. Tritium has a low permeability in copper, and copper is a ductile, stable, face-centered cubic metal and so is resistant to hydrogen embrittlement. The high thermal conductivity of copper is a desirable property for process vessels requiring heat flow or constant temperature. Copper can be easily joined in a number of ways (e.g., soldering, brazing, and welding). In spite of these advantageous properties, copper and copper alloys are not commonly used in tritium systems. Several factors may account for this. The ASME allowable design strength of copper falls rapidly at temperatures above 200°C, making it difficult to use copper at elevated temperature. Also, hydrogen isotopes can react at elevated temperature with oxygen in copper, whether the oxygen is in solid solution or in copper oxide precipitates. In either case, water is formed and over time water vapor agglomerates at grain boundaries, which eventually results in intergranular cavitation, cracking, and failure. This failure mechanism is sometimes termed “steam embrittlement.” Also, a transition junction (normally nickel) is required to join copper and the stainless steel components of the remainder of the system.

5.1.2.c Aluminum and Aluminum Alloys

Aluminum has properties making it potentially desirable for tritium systems. It has a low density and a high thermal conductivity. Hydrogen isotope permeability is very low in aluminum compared to stainless steel. Aluminum is used in applications where light weight is important, such as in containers that must be lifted by personnel in gloveboxes. Aluminum can be used for the construction of medium-quality static containment vessels. However, aluminum is not commonly used in tritium systems. Stainless steel has much higher strength, at room and elevated temperature. Welding aluminum requires more precautions because aluminum reacts with atmospheric water vapor, which can cause porosity due to hydrogen in the weld fusion zone.

5.1.2.d Materials to Avoid

Plain carbon steels and alloy steels cannot be used for tritium service. These steels have high strength and (normally) a body-centered-cubic crystal structure, both of which make the material less ductile and much more susceptible to hydrogen embrittlement. Ferritic stainless steels (such as Type 430), martensitic stainless steels (both quench-and-tempered (such as Type 410) and precipitation hardening (such as 17-4 PH and PH 13-8 MO)) and precipitation hardened austenitic stainless steels (such as AM-350) should not be used for general tritium service; they are all more susceptible to hydrogen embrittlement than the austenitic stainless steels. Additionally, free-machining grades of austenitic stainless steel (such as Type 303) should not be used.

Other materials that cannot be used for tritium service are any material that forms hydride near room temperature and atmospheric pressure. Examples include zirconium, tantalum, niobium, and many alloys of these materials.

Powders should not be used in or on containment and confinement systems. Materials in powder form and materials that are prone to form powders, such as insulation cloth, fabrics, polymers, and ceramics should be carefully considered and avoided if possible.

5.1.3 Polymers

All polymers degrade when exposed to radiation. Both tritium and tritiated water permeate all polymers, and permeated tritium deposits the beta decay energy throughout the polymer bulk. (Although the tritium beta energy is very low and has a small penetration depth in matter, permeation allows tritium atoms to be near enough to polymer chains throughout the bulk to cause changes in the polymer by radiation.) Types of radiation-induced changes in polymer properties include softening (degradation) or hardening, ductility loss, color change, dimensional change, and gas evolution. Because of these effects, polymers should only be used in tritium systems where no metal alternatives exist. Normally, only polymers that harden during radiation exposure are employed, and are replaced before they begin to deteriorate. In addition to polymer breakdown itself, products of degradation can form corrosive liquids or acids such as HF and HCl. Polymer parts have to be easily replaceable as a part of normal operations, and a program of regular inspection and replacement should be established. The system should be designed to expose any polymers to as little tritium as possible. Typical uses of polymers in gas-handling systems include gaskets, O-rings, electrical cable insulation and valve parts, including seats, stem tips, and packing.

Polymers relatively resistant to radiation can typically withstand up to about 1 million rad (1 rad = 100 erg energy deposited per gram of material). By knowing the solubility of tritium in a polymer at a given temperature and tritium partial pressure and the decay rate of tritium, the approximate dose can be calculated, assuming the tritium concentration has reached equilibrium.

Many effects of radiation on polymers are accentuated by oxygen. Protecting polymers from oxygen or air will likely lengthen the lifetime of polymers exposed to tritium. Also, temperatures above about 120°C accelerate radiation effects in polymers, so the temperature of any polymer parts should be kept as low as possible. Inert additives such as glass or graphite generally enhance the resistance of polymers to radiation. Addition of antioxidants may also enhance radiation resistance.

5.1.3.a Plastics

Vespel®, a polyamide, has been successfully used for valve stem tips in some tritium laboratories. Ultra-High-Molecular-Weight Polyethylene (UHMWPE) and High-Density Polyethylene (HDPE) have been used for valve stem tips in automatic valves at SRS. Although the latter two materials were designed by DuPont™ specifically for tritium service at SRS, their performance has not reached the desired level. Use of HDPE has been discontinued. Valves that use UHMWPE are selectively transitioned to Vespel if the service conditions warrant the change and as the need arises to repair or replace. EH Technical Notice 94-01, *Guidelines for Valves in Tritium Service,* describes in detail differing materials and actuators for varying applications. Low-Density Polyethylene (LDPE) is very permeable by tritium and tritiated water and should not be considered for use in tritium systems. Polytetrafluoroethylene (PTFE), trade-named Teflon®, degrades and decomposes in tritium, resulting in the formation of HF. In humid air, HCl and HF are formed, which are both highly corrosive. Generally, chlorofluorocarbon polymers should not be used in tritium service. Polyvinyl chloride (PVC) and vinylidene chloride (Saran™) are among several polymers used in tritium protective clothing, but should not be used for process equipment because they contain chlorine.

5.1.3.b Elastomers

Tritium readily permeates into and diffuses through elastomeric materials and, depending upon thickness, begins appearing on the outside of the elastomeric seal within hours after exposure. Elastomers are subject to radiation damage. They harden and lose their sealing ability due to exposure to high concentrations of tritium.

Ethylene propylene diene monomer (EPDM) elastomers are employed for low-pressure process flange gaskets because of EPDM's relatively good performance in tritium service. In some cases, Buna-N process flange gaskets are being replaced by EPDM when the gaskets are changed; however, in other applications, Buna-N remains in use. Viton®, a common O-ring material, is used, but can embrittle in months in tritium service. Butyl rubber has low permeation for both tritium and tritiated water, but is not as resistant to radiation damage as EPDM. Butyl rubber is used for glovebox gloves. Silicon rubber can be used in glovebox construction and has been found to be effective at Mound and other sites. Water vapor in the air outside gloveboxes permeates gloves and can lead to a significant portion of the residual tritiated water vapor in tritium gloveboxes.

Kel-F®, and its successor Neoflon™, are common chlorofluorocarbon polymers (Polychlorotrifluoroethylene (PCTFE)) and are incompatible with tritium. They, like Teflon®, degrade in tritium gas and should not be used.

5.1.4 Fabrication Cleaning and Testing

Materials used in tritium should be fabricated in a manner that produces little contamination. Methods that are consistent with high vacuum processes are often useful. When joining is required, gas tungsten arc welding (GTAW) is the preferred welding method. Other joining processes; e.g., gas metal arc welding (GMAW), tend to be prone to pore formation and thus are not suitable for storage vessels.

As indicated previously, forged materials that are designed to orient stringers, inclusions, and properties favorably are preferred to cast structures. Also, thin wall tubing may require additional testing, such as pressurized proof testing, to ensure suitability for service. SRS has experienced tubing failures due to wall stringers oriented parallel to the tubing. These leak sites can be weld-repaired.

Welds should be purged to prevent oxide formation or should be cleaned with a wire brush to remove any oxide film present. If stainless steel materials are wire brushed, the brush should be made from stainless steel and should not be used on carbon steel or stored in an area where inadvertent contact can be made with stainless steel products.

Any surfaces that require cleaning should be cleaned with compounds that contain little to no halide compounds. Halides are to be avoided since they can promote stress corrosion, cracking, and leaks. Solvents used in gloveboxes should clean effectively, but not contribute particulate, ions, or residues. They should be low vapor pressure, high auto-ignition, high flashpoint, high lower flammability limit (LFL) materials that are non-toxic and do not contribute chemical species that can coke or poison detritiation systems. Manufacturing solvents and cleaning requirements should be carefully considered during the procurement specification to minimize the impact of tritium introduction to the system (e.g. halogens, hydrocarbons, and particulates).

Tritium compatibility testing is conducted in a variety of ways. SRS has been conducting compatibility testing for many years[56] [57]. Stainless steel forgings, welds, and bar stock have been charged with tritium, aged, and tested to determine the effects of hydrogen, helium, and hydrogen and helium combined. In general, these tests are accomplished by preparing samples of the appropriate geometry, acid-cleaning them, placing them in charging vessels and loading them with the desired pressure of tritium at temperatures that do not exceed 350°C. They are then held until they are saturated, quenched to retain the tritium, and aged at -70°C for the required duration to develop the target amount of helium, and then tested.

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[56] Morgan, Michael J. SRNL. *Tritium Aging Effects on the Fracture Toughness Properties of Stainless Steel Base Metal and Welds*. SRNL-STI-2009-00475.

[57] Morgan, M.J., M.H. Tosten, and G.K. Chapman, SRNL. *The Effects of Hydrogen, Tritium, and Heat Treatment on the Deformation and Fracture Toughness Properties of Stainless Steel,* SRNL-STI-2013-00534.

Polymer samples are tested in much the same manner. Samples are loaded into suitable pressure vessels and charged with tritium for the desired duration. The gases generated and mechanical properties are determined. Experiments may be staged with multiple samples to obtain the desired properties as a function of time or dose as needed to support the program. SRNL has extensively used dynamic mechanical analysis to determine the effects of radiation and tritium exposure on the loss modulus of polymers[58] [59] [60] [61] [62] [63].

5.2 First Wall Design

5.2.1 High-Pressure Tritium

For high-pressure tritium, it is generally recommended that the first wall be of all metal construction and hydrogen compatible materials including valves, valve seats, and tubing. The use of non-hydrogen-compatible materials results in material failure and the release of the contained tritium. Elastomers are not tritium-compatible, and, as a result, elastomeric seals and valve seats are not recommended for use in the containment of high-pressure tritium. There are exceptions to this general case, and other criteria may be used when justified by analysis. Additionally, the orientation of high pressure valves with respect to the high pressure tritium is also important. Historically high pressure valves with teflon packed valve stems have been used for long term component storage. This practice was acceptable if the valve was properly oriented where the pressure only wets the metal valve tip. If incorrectly oriented the high pressure tritium is in contact with the teflon packing and the valve will fail.

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[58] Clark, Elliot A. *Radiolytic Gas Production Rates of Polymers Exposed to Tritium Gas*. SRNL, 2013. SRNL-STI-2013-00506.

[59] Clark, Elliot A. *Effects of Gamma Irradiation on EPDM Elastomers*. SRNL, 2013. SRNL-STI-2011-00580, Revision 1.

[60] Clark, Elliot A. and Shanahan, Kirk L. *Effects of Tritium Exposure on UHMU-PE, PTFE, and Vespel* SRNL, 2006.WSRC-STI-2006-00049.

[61] Clark, Elliot A, et al., *Effects of Tritium Gas Exposure on Polymers*. SRNL, 2010. SRNL-STI-2010-00111.

[62] Clark, Elliot A. and Shanahan, Kirk L *Effects of Tritium on UHMW-PE, PTFE, and Vespel Polyimide* SRNL, 2008 WSRC-STI-2008-00281.

[63] Clark, Elliot A. *Tritium Effects on Dynamic Mechanical Properties of Polymeric Materials*. SRNL, 2008. WSRC-STI-2008-00077.

5.2.2 Low- and Medium-Pressure Tritium

For the containment of low- and medium-pressure tritium, it is generally recommended that the first containment wall be of all metal construction of hydrogen-compatible materials where possible, including valves, valve seats, and tubing. Hydrogen and helium embrittlement of the materials of construction is not usually significant at low and medium pressures. As a result, non-hydrogen-compatible materials may be used if required by the design or if the required component is not available in other materials.

It is difficult to design a vacuum system, which, in some cases is the first wall, without including some non-hydrogen-compatible materials and elastomers. However, embrittlement of the materials is not an issue because the tritium exposure is transient and the pressure is low. Under these conditions, the elastomers are not exposed to tritium continuously, and most can be used in tritium operations under this condition. Surveillance and/or preventive maintenance schedules should be selected in order to maintain elastomer functionality.

5.3 Secondary Wall Design

5.3.1 High-Quality Secondary

The design requirements for a high-quality secondary wall are the same as the primary wall. If the secondary wall is required to provide long-term containment of high concentrations of tritium, it should meet the same requirements as the primary or first-wall container.

5.3.2 Medium-Quality Secondary

If the secondary wall is a glovebox and only contains tritium that has been diluted by the glovebox gases for a short duration (e.g., a few hours) while the glovebox is cleaned up by the tritium removal system, the requirements can be relaxed. Although the quantity of tritium contained may be quite large, the low pressure and concentration of tritium will not result in material failure due to tritium exposure.

5.3.3 Low-Quality Secondary

If the secondary wall is a room or building and only contains tritium that has been diluted by the air contained in the room or building while the room or building is cleaned up by the tritium removal system, the construction requirements can be relaxed. Although the quantity of tritium contained may be quite large, the low pressure and concentration of tritium will not result in material failure.

5.4 Cleanup System Design

Most of the components of the tritium removal and cleanup system are only exposed to tritium at low concentrations and pressure for short time periods. The only long-term exposure is in the water collection system where the water is collected at low pressure on a molecular sieve. All-metal construction is recommended, but the materials of construction are not required to be hydrogen-compatible materials. Appropriate, elastomeric sealing materials have been used successfully in these systems for many years without significant problems. When possible, metal seals should be used because they are more durable and reliable and require less maintenance than elastomeric seals.

To minimize the potential for the generation of mixed waste and to decrease radiation exposure of the workers, oil-free pumps should be used where possible.

5.5 Storage System Design

Storage systems should consider the total cost of the storage cycle and the purpose for the storage. Storage techniques that increase the complexity of the handling process without adding beneficial features should not be used. The barrier concept discussed in Sections 4.1.1 through 4.1.3, in addition to the wall design considerations discussed above, should be incorporated into all storage system designs.

5.5.1 Short-Term Storage

Tritium used to support the day-to-day activities in a facility has to be readily available to the facility customers. If the facility uses tritium in gaseous form and its decay to helium does not impact the process, then, to simplify the operation and the equipment, the tritium can be stored in gaseous form. The storage container should be fabricated of all metal, hydrogen-compatible materials including valves, valve seats, and seals.

5.5.2 Medium-Term Storage

If tritium is only used in periods of two years or less, the requirements do not change significantly from those of short-term storage. Experience has shown that tritium can be stored safely at near atmospheric pressure for long periods of time. If the buildup of helium in the supply does not impact the use, then storage as a gas is an acceptable alternative. There are, however, advantages of tritide bed storage for medium-term use. Impurities such as nitrogen and oxygen form uranium nitride and uranium oxide and are removed from the gas stream as the bed is heated and cooled. Helium, which accumulates due to the decay of tritium, and any other impurities remain in the overpressure gas above the bed and may be pumped off after the bed had been cooled down and the tritium has been gettered by the uranium. As a result, the uranium bed not only provides for tritium storage but also provides a means of maintaining a reasonably pure and stable tritium supply.

5.5.3 Long-Term Storage

Due to its short half-life, storing tritium for several years implies that it is not readily needed. It should be placed in a safe and stable condition while it decays.

5.5.3.a Storage as a Gas

Metal tritides have the advantage of significantly decreasing the volume required to store tritium without increasing the pressure of the gas during storage. Compared to the fabrication and preparation of metallic storage beds, regardless of what metal is used, the cost of storage of tritium as a gas at near atmospheric pressure is economical. ASME code-designed stainless steel tanks are available or can be designed and fabricated at a reasonable cost. A tank at atmospheric pressure would have a final pressure of 15 pounds per square inch after all of the tritium had decayed, so embrittlement is not an issue. The long-term storage of hydrogen and tritium in containers is well understood in comparison to the understanding of long-term storage of metal tritides.

5.5.3.b Storage as a Metal Tritide

Uranium beds designed at Sandia in the late 1970s for laboratory use were about the size of two one-gallon paint cans. This included the secondary containment system and electric heaters used to drive off the tritium during removal. These beds were designed to store 50 grams of tritium as uranium tritide that was easily recoverable in a matter of less than an hour. Long-term storage in this type of container is expensive, but the tritium can be easily and quickly recovered for use. Additionally, large uranium beds capable of storing 1,000 liters of hydrogen were used at Mound. Also, impurities such as nitrogen and oxygen form uranium nitride and uranium oxide, and are removed from the gas stream as the bed is heated and cooled. Helium, which accumulates due to the decay of tritium, and other non-reacting impurities remain in the overpressure gas above the bed.

To help resolve unknowns regarding consequences of air-ingress accidents in uranium beds, a series of air-ingress experiments was conducted at Ontario Hydro Research Division, with the participation of PPPL and the Idaho National Laboratory. The experiments indicated that the resulting reaction was restrained with only modest temperature excursions. This leads to the conclusion that the hazards associated with an air-ingress accident involving a uranium bed is smaller than previously anticipated. Additionally, tests conducted by WSRC indicate that, except for catastrophic container failure, the tritium release due to air in-leakage into a uranium tritide bed is limited by diffusion.

Titanium hydride is not pyrophoric at room temperatures, is a stable material, and has been studied for use in the long-term storage of tritium. It is reported to be less prone to spontaneous ignition in air than the parent metal. Following the hydrating process, if the titanium hydride is exposed to air under controlled conditions, a small quantity of hydrogen is released from the material as the oxide layer forms on the surface of the material. Following formation of the oxide layer, titanium hydride is stable in air. Hydrogen will not be released unless the material temperature is significantly increased.

Palladium tritide is not pyrophoric at room temperature, is a stable material, and the overpressure of tritium over the bed at room temperature is approximately 50 torr. Metal tritides have the advantage of significantly decreasing the volume required to store tritium without increasing the pressure of the gas during storage.

5.6 Surveillance and Maintenance

The level of equipment surveillance (including radiological monitoring) and maintenance required is based on the hazard class of a facility; i.e., Hazard Category 1 through 3 or Radiological. The specific requirements for the different classes of facility equipment are a function of the safety issues associated with the equipment. These are specified in the facility safety analysis report or other facility safety documentation and in the facility maintenance plan.

5.7 Seismic Considerations

This section describes 1) DOE NPH requirements for protection against natural phenomena such as earthquakes, and 2) seismic design and evaluation of equipment and distribution systems. Seismic and wind design and evaluation of structures and facilities are discussed in Section 4.4.

5.7.1 DOE Natural Phenomena Hazards Requirements

DOE has requirements for the mitigation of natural phenomena (such as earthquakes, extreme winds, and floods) on its facilities. These requirements are in DOE O 420.1C, *Facility Safety*, and DOE-STD-1020-2012, *Natural Phenomena Hazards Analysis and Design Criteria for DOE Facilities*, which is invoked as a requirement by O 420.1C. DOE O 420.1C provides the overall requirements for mitigation of the effects of natural phenomena. DOE-STD-1020 defines the basic performance requirements for SSCs subjected to NPH loads.

Section 2.2 in DOE-STD-1020-2012 describes the process of determining an NPH design category for each SSC based on the radiological consequences of its failure. It references a categorization scheme described in Appendix A of DOE-STD-1189-2008, *Integration of Safety Into the Design Process*, as well as ANSI/ANS-2.26-2004, *Categorization of Nuclear Facility Structures, Systems, and Components For Seismic Design*. An equipment list should be developed to show the NDC for each SSC identified.

Chapters 3-8 of DOE-STD-1020-2012 contain requirements for characterizing various natural hazards (seismic, wind, flooding, lightning, precipitation, and volcanic eruption) and determining appropriate performance goals and design basis loads to withstand these hazards. These chapters rely extensively on national consensus standards for the hazard characterization and design requirements. The rigor of required hazard characterization increases for SSCs categorized as NDC-3 or higher. Site-specific hazard characterization is not required if all SSCs are NDC-2 or lower; hazard characterization and design criteria need only follow ASCE/SEI 7-10, *Minimum Design Loads for Buildings and Other Structures*. However, if valid, site-specific hazard values are available, even if a facility is NDC-2 or below, such values are always preferable for use in design.

5.7.2 Seismic Design and Evaluation of Equipment and Distribution Systems

In the event of an earthquake, DOE facilities need to have adequate measures for the protection of the public, workers, environment, and investment. Due to the evolutionary nature of design and operating requirements as well as developments in engineering technology, existing DOE facilities embody a broad spectrum of design features for earthquake resistance. These features depend on factors such as vintage of the facility design and construction and hardware supplier practices at the time of design and construction. The earliest vintage facilities often have the least design consideration for seismic-induced forces and displacements and exhibit the greatest difference between their design basis and current requirements for seismic design criteria for new facilities. Chapter 9 of DOE-STD-1020-2012 discusses periodic reviews of NPH assessments that are mandated by DOE O 420.1C. If a periodic review leads to a new seismic hazard assessment for a site, and the hazard increases, Chapter 9 also describes steps for evaluating seismic design of existing facilities. DOE has developed a Seismic Evaluation Procedure (SEP)[64] that can assist with such evaluations. It provides a technical approach and generic procedures and documentation requirements, which can be used to evaluate the seismic adequacy of existing equipment and distribution systems.

The SEP is intended to provide DOE facility managers, safety professionals, and engineers with a practical procedure for evaluating the seismic adequacy of equipment. Often the approach used to review the seismic capacity of equipment in facilities is to conduct sophisticated evaluations, which can be very time-consuming, complex, and costly. Much of the available funding and time can be spent on analysis rather than on the real objective of increasing the seismic capacity of the equipment. The SEP is designed to be an extremely cost-effective method of enhancing the seismic safety of existing facilities and reducing the potential for major economic loss that can result from equipment damaged by an earthquake.

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[64] DOE/EH-0545, *Seismic Evaluation Procedure for Equipment in DOE Facilities*, March 1997.

The following is a suggested list of topics to consider in the design of new systems and equipment or the evaluation of existing systems and equipment. Representative equipment that may be found in tritium facilities is listed in Table 5-1.

ELECTRICAL EQUIPMENT

- Batteries on Racks
- Motor Control Centers
- Low- and Medium-Voltage Switchgear
- Distribution Panels
- Transformers
- Battery Chargers and Inverters
- Instrumentation and Control Panels
- Instruments on Racks
- Temperature Sensors
- Computer Data / Storage Systems
- Alarm Instrumentation
- Communications Equipment
- Tritium Monitors

PIPING AND RACEWAY SYSTEMS

- Cable and Conduit Raceway Systems
- Piping
- HVAC Ducts
- Underground Piping
- Underground Raceways
- Stacks
- Conveyors of Material

TANKS

- Vertical Tanks
- Horizontal Tanks and Heat Exchangers
- Underground Tanks
- Canisters and Gas Cylinders
- Miscellaneous Tanks

MECHANICAL EQUIPMENT

- Fluid-Operated / Air-Operated Valves
- Motor/Solenoid-Operated Valves
- Horizontal Pumps
- Vertical Pumps
- Chillers
- Air Compressors
- Motor-Generators
- Engine-Generators
- Air Handlers
- Fans
- HEPA Filters
- Gloveboxes
- Miscellaneous Machinery

ARCHITECTURAL FEATURES

- Unreinforced Masonry (URM) Walls
- Hollow Clay Tile Walls
- Suspended Ceilings
- Raised Floors
- Storage Racks
- Cranes
- Elevators

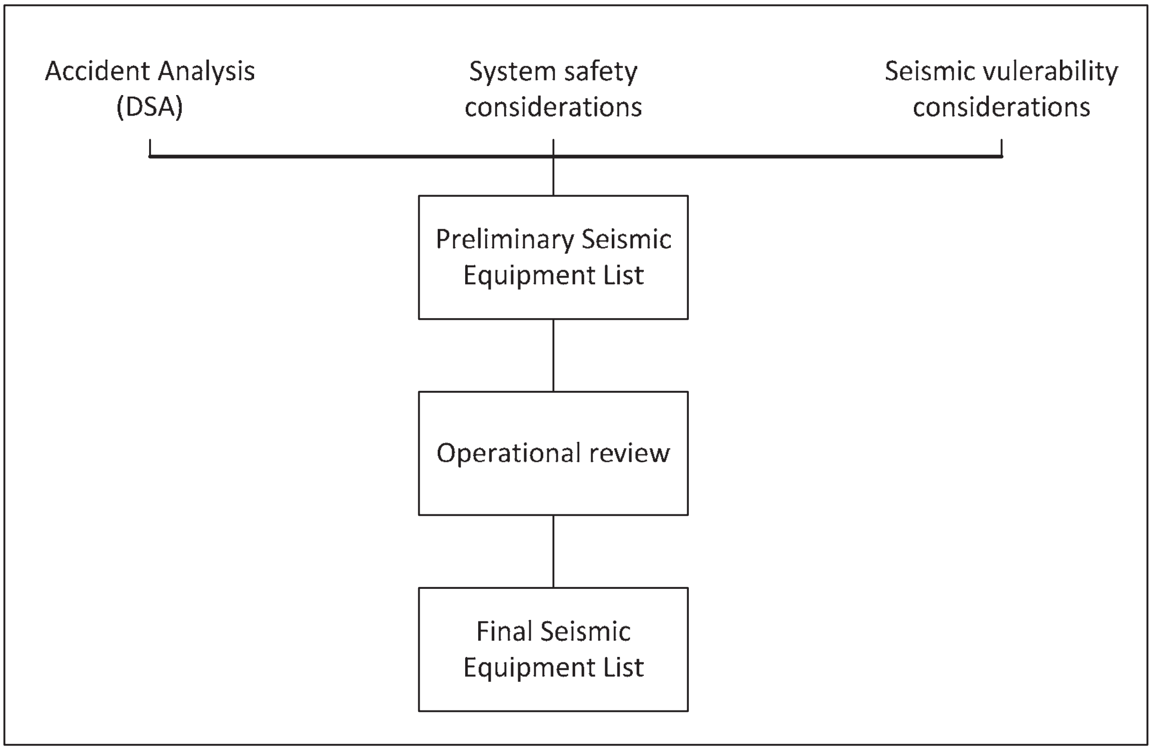
SWITCHYARD AND SUBSTATION EQUIPMENT

- Power Transformers
- Miscellaneous Equipment

4) Table 5-1: Representative equipment found in tritium facilities

The seismic design category (SDC) for each system or component to be reviewed must be defined (SDC 1, 2, 3, 4, or 5) in the Seismic Equipment List (SEL). The methodology and procedures described in the SEP for evaluating the seismic adequacy of equipment are based on the observed performance, failure, and response of various types of SSCs during and after they were subjected to either actual or simulated earthquake motion. An SSC in a DOE facility can be evaluated for seismic adequacy provided that the associated guidelines, limitations, requirements, and caveats described in the SEP are satisfied.

The general approach for the development of the SEL is envisioned to be a three-step process as depicted in Figure 5-1. After a SEL Team is selected, the first step of the process is the development of the preliminary SEL from a list of the facility SSCs. The SEL Team consists primarily of safety professionals and systems engineers with assistance from seismic engineers and facility operators. Only a portion of the facility SSCs will be contained in the SEL and, in many cases, the SEL will contain only safety-related SSCs that must function during or after a seismic event. The selection of the SSCs belonging on the SEL should be based on the results of accident analyses. These accident analyses should consider the appropriate facility hazards as required by DOE Orders and Standards.



13) Figure 5-1: Development of the seismic equipment list

For the DOE facility being seismically evaluated, accident analyses and their results are typically provided in a DSA. The preliminary SEL should be based on information provided in this DSA. For a nonreactor nuclear facility, DOE-STD-3009-94, Change Notice 3, provides guidance on the preparation of a DSA. Using the guidance in DOE-STD-3009-94, Change Notice 3, and the appropriate accident analyses in the DSA, SSCs can be differentiated into safety-class or safety-significant, and the preliminary SEL can focus on those facility SSCs. For facilities without a DSA, accident analyses comparable to those required for a DSA should be performed.

The preliminary SEL should focus on those SSCs that are classified above as SDC-2 or higher. In addition to selecting SSCs based on an accident analysis or on SDC level, there are system safety considerations and seismic vulnerability considerations that should be addressed when developing the preliminary SEL.

The design or evaluation of SSCs on the SEL should follow the requirements in DOE-STD-1020-2012. SSCs should be assigned an SDC (or NDC for the purpose of broader NPH mitigation) and Limit State per the scheme described in ANSI/ANS-2.26-2004, as modified by Appendix A in DOE-STD-1189-2008. SSCs categorized as SDC-2 or lower should be designed or evaluated using ASCE/SEI 7-10. SSCs categorized as SDC-3 or higher should be designed or evaluated using ASCE/SEI 43-05, *Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities*.

Evaluation of existing systems or components must be conducted based on the "Actual" condition of the item. This may be different than the design or "As-Built" condition due to field modifications or deterioration of the item during its service. An examination of the item, its installation, and current condition should be made during a walkdown by seismic engineers as defined in the SEP. Existing systems and components may also be evaluated by use of experience data. This is an alternative approach that can be used if the systems or components are installed in an acceptable manner and meet the rules specified in the SEP to ensure that the item is similar to items in the experience database. DOE has developed references for the evaluation of existing systems and components, has conducted training on their application, and has implemented the seismic evaluation guidelines for systems and components at DOE facilities.

5.8 Fire Scenarios

There have been investigations that point to the conclusion that fire scenarios are the dominant risk at most tritium facilities. These fire scenarios are not restricted to seismic-induced fires, but include fires from all sources. These analyses show that the slower burning fires are limiting, and that in these cases, fires result in oxidation rates that are high (in excess of 90 percent). Additionally, compliance with National Fire Protection Association (NFPA) requirements, while limiting the potential for fire spread, does not ensure that a full-facility fire scenario need not be analyzed. LA-UR-02-3803, *Oxidation of Tritium Gas under Accident and Transport Conditions*[65], describes the oxidation rates that are used in safety analysis at LANL. The bounding value given a fire defaults to 100 percent oxidation for use in safety analyses, as a realistic, less-conservative value has not been able to be identified.

It is possible that fires of sufficient magnitude and frequency may occur at some tritium facilities so as to warrant safety system classifications of both the fire suppression and fire detection systems. This importance that DOE places on fire protection at tritium facilities was highlighted as early as September 1991, during a review of the then-under-construction RTF. The exemption request from some requirements of DOE Orders 5480.7 and 6430.1A for RTF was not approved by HSS/NS until physical modifications were made to the tritium facility[66]. The definition of tritium that was Material at Risk in selected fire (and seismic) scenarios was also developed prior to startup at RTF, which eventually led to decisions on upgraded selected storage containers and inventory limits[67].

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[65] Jofu Mishima and Chris Steele, "Oxidation of Tritium Gas under Accident and Transport Conditions", LA-UR-02-3803, July 2002.

[66] DOE Memorandum from S. Blush, NS-1, to R. Claytor, DP-1 and P. Ziemer, EH-1, *RTF Assessment of Fire Barriers*, October 28, 1991.

[67] DOE Memorandum from T. Rollow, Acting Director, Office of Nuclear Safety, to N. Goldenberg, NE-70, *NS Response to Request for Deviation for Replacement Tritium Facility (RTF)*, May 7, 1993.

5.9 Instrumentation

This section provides information on several instruments used to detect and monitor tritium. The references are provided for information only. DOE does not certify that a particular product is superior to any other product from another vendor. References in this Standard do not imply endorsement by DOE.

5.9.1 Tritium Monitoring Systems

Several different types of instruments may be used to detect and measure tritium in the operation of a facility. Examples and a discussion of such instrumentation follow:

- Ionization Chambers — Tritium decays to ^{3}He by the ejection of a beta particle. The beta particle generated by the decay of tritium ionizes the surrounding gas. The number of ions produced due to the loss of energy of the beta particle is a function of the type of gas. A sample of gas is collected in the ionization chamber and the ionization current is measured. The resulting chamber ionization current is proportional to the quantity of tritium in the gas. The larger the measuring chambers volume, the higher the output current and the easier it is to measure. However, as the volume of the chamber increases, the longer it will take to get an accurate measurement. Modern electronic systems have solved most of the problems associated with measuring small ionization currents in small volumes and as a result, the volume of the ionization chambers has been reduced over the years from 50 L down to 1 or 2 L. Most tritium measuring instruments have an ionization chamber.

- Proportional Counters — Gas proportional counters are also used to measure the amount of tritium contained in a gas. A sample of the gas to be monitored is mixed with a counting gas and passed through a proportional counter tube where the pulses caused by the decay of tritium are counted. Proportional counter monitors can be used for most gas monitoring applications and are also available to measure surface contamination.

- Scintillation Crystal Detectors — Scintillation detector systems are used to measure the total mole percent of tritium in a sample of gas independent of the chemical composition of the tritium in the gas (HT, DT, T_2, and CH_xT_y). A sample of the gas is introduced into a measurement chamber at low pressure, generally less than a few torr. The chamber contains a scintillation crystal, which is exposed to the tritium as it decays. The light pulse produced in the scintillation crystal is either counted or is used to produce a current, which is proportional to the mole percent tritium contained in the gas sample. See reference 66 for expanded discussion[68] Crystal

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[68] Ellefson, R.E.; Ellefson Anal. Services, Centerville, OH: Price, B.R. LANL and West, D.S. LANL*, Trirtium inventory measurement by beta scintillation detection,* Fusion Engineering 1995, SOFT '95. Seeking a New Energy Era., 16th IEEE/NPSS Symposium (Volume: 2).

scintillation detection is generally used to measure the mole percent of tritium in gases containing high concentrations of tritium.

- Mass Spectrometer — Magnetic sector, quadrupole, and drift tube mass spectrometers are used as analytical tools to measure the individual components that make up the gas being measured. Mass spectrometers are generally used for the purposes of assay and accountability or for scientific purposes. A sample of the gas to be measured is introduced at low pressure (a few microns) into a chamber and ionized. The ions produced are then measured by a means that discriminates on mass. The number of ions produced at each mass is measured and is proportional to the partial pressure of the component in the gas sample.

  The sophistication of the measurement systems varies greatly from facility to facility throughout DOE. Light isotope, drift tube, mass spectrometers require a large capital investment and require a skilled staff to operate, and, in some cases, may not be cost-effective. All DOE tritium facilities do not require a light isotope drift tube mass spectrometer. Quadrupole mass spectrometers and crystal scintillation detectors are much less expensive, but still require operation by knowledgeable well-trained personnel. The DOE assay and accountability requirements and regulations do not currently reflect this difference in sophistication and cost and currently place the same requirements on small as well as large-scale operations.

- Liquid Scintillation Counters — Due to the need to measure the removable tritium on surfaces and in the body water of workers, almost all tritium facilities are equipped with or have access to a liquid scintillation counter. If the scintillation counter is not available on site the service can generally be purchased from a local firm. Liquid scintillation counters are used to measure the quantity of tritium on surfaces, in liquids, and in dissolved samples. For removable surface contamination measurements, a wipe of the surface to be measured is taken using dry paper or a Q-Tip. The filter paper or Q-Tip is then placed in a scintillation cocktail, and the quantity of tritium is measured by counting the light flashes that occur in the scintillation cocktail as the tritium decays. The surface contamination is then calculated in units of dpm/100 cm^2.

  For liquid measurement, a sample of the liquid to be measured is placed in the liquid scintillation cocktail and measured. The tritium concentration of the liquid is calculated in Ci/mL. For solid measurement, a known weight of a material is dissolved to produce a liquid and then the liquid is sampled and measured in the scintillation counter. The quantity of tritium is then calculated in units of Ci/g of the original solid.

- Raman spectroscopy has emerged as a reliable method for near real-time analysis of tritium-bearing gas mixtures with potential applications in process control, accountability, and environmental release calculations. Karlsruhe has reported good results being deployed in a number of applications there[69].

- Gas Samplers — Many different types of gas samplers have been developed and used for measuring very small quantities of tritium in very large volumes of gas. These samplers are used to measure quantities of tritium released through a facility stack and for environmental monitoring at a site. Stack exhaust gas monitoring systems generally use an ionization chamber to measure the tritium in the stack gases and a gas sampler to measure the extremely low levels of tritium that cannot be measured by an ionization chamber.

  Most of the stack gas samplers are patterned after the ethylene glycol sampling system developed at Mound Laboratories. A commercial version of this system is now available. In this system, a sample of gas from the stack is circulated through six ethylene glycol bubblers in series. The first three bubblers remove tritium in the form of HTO, DTO, and T_2O. The gas stream is passed through a heated catalytic reactor where tritium in the form of HT, DT, T_2, and CH_xT_y, is cracked and oxidized to form water. This sample is then passed through three more ethylene glycol bubblers to remove the tritium gas, which is now in the form of water. After a period from a few hours to days, a sample of the ethylene glycol from each bubbler is removed and counted using a scintillation counter to determine the quantity of tritium in each bubbler. Tritium recovered from the first three bubblers is proportional to the tritium in oxide form contained in the stack gases and the tritium recovered in the last three bubblers is proportional to the quantity of tritium in elemental form contained in the exhaust gases.

  Due to the extremely small quantity of tritium contained in the atmospheric gases surrounding a tritium facility, the environmental gas samplers use higher flow rate sampler systems than those required for stack monitoring, and, in general, collect the water on molecular sieve traps. The water collected on the molecular sieve traps is then recovered from the trap and the tritium concentration of the gas passing through the trap is calculated from the tritium concentration of the collected water, the gas flow rate through the trap, and the sampling time.

- Portable Room Air Monitors — There are several handheld portable room air monitors on the market, and their capabilities and ranges vary as a function of the different manufacturer and the purpose for which they were designed. It is convenient in some activities to have the capability to connect a small hose to the monitor so that it may be used to detect tritium leaks around equipment.

- Fixed Station Room Air Monitors — Fixed station monitors are designed to be installed in fixed locations and to be used to monitor the room air tritium concentrations. Depending upon the

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[69] M. Schlösser, *How Can Raman-Inactive Helium be Made Visible in Raman Spectra of Tritium-Helium Gas Mixtures?* Presented at the Tritium 2013 Conference, Nice, France

manufacturer they may have several ranges and are equipped with one or two alarm set points and audible as well as visual alarms.

- Glovebox Atmosphere Monitors — Glovebox monitors may be open mesh or closed ionization chambers and are designed to monitor the higher levels of tritium inside the glovebox containment systems.

- Hood and Exhaust Duct, Air Monitors — Hood and exhaust duct air monitors are similar to fixed station monitors in range and characteristics.

- Exhaust Stacks, Air Monitors — Exhaust stack monitors are similar to fixed station air monitors except that they generally have larger ionization chambers to increase the sensitivity of the monitor.

- Personnel Friskers and Breath Analyzers — There has been some interest in instruments which can be used to frisk personnel as they enter and exit tritium contaminated areas. One DOE facility implemented a process of personnel frisking consisting of the use of skin surface wipes counted in a liquid scintillation counter upon entry and exit from tritium contaminated areas. In another facility, a hand station based on counting the associated gas flow across the hands was used. To date, the development work required relating measurements made by these techniques to dose or worker exposure has not been completed. It is expected that differences in the body chemistry of personnel and differences in the time delay between tritium exposure and equilibration of tritium in the body will continue to make the results of skin surface contamination measurement and breathe analysis monitoring inconsistent. The impact of false alarms and inconsistent results on worker confidence will probably continue to make these systems unsatisfactory for worker monitoring.

5.9.2 Specialized Instrumentation

There are many other types of specialized devices and/or instrumentation vendors, and some may be superior to those discussed here. No endorsement of these devices should be inferred by the reader.

5.9.2.a Remote Field Tritium Analysis System

A system for the remote, *in-situ* analysis of tritium in surface and ground waters has been developed at SRS. Using automated liquid scintillation counting techniques, the Field Deployable Tritium Analysis System (FDTAS) has been shown to have sufficient sensitivity to measure tritium in water samples at environmental levels (10 becquerels (Bq)/L [~270 pCi/L] for a 100-minute count) on a near-real-time basis.

The FDTAS consists of several major components: a multi-port, fixed-volume sampler, an on-line water purification system using single-use “tritium columns,” a tritium detector employing liquid scintillation counting techniques, and the serial communications devices. The sampling and water purification system, referred to as the “autosampler”, is controlled by a programmable logic controller pre-programmed to perform a well-defined sampling, purification, and flushing protocol. The tritium analyzer contains custom software in the local computer for controlling the mixing of the purified sample with liquid scintillation cocktail, counting, and flushing the cell. An external standard is used to verify system performance and for quench correction. All operations are initiated and monitored at the remote computer through standard telephone line modem communications[70]. This system has been commercially licensed by SRS to a firm that has a line of other sampling monitors.

5.9.2.b Surface Activity Monitor

A surface activity monitor (SAM) for measuring tritium on metal (electrically conducting) and nonmetal (electrically nonconducting) surfaces was developed at Ontario Hydro Technologies (now known as Kinectrics, Inc.).[71] [72] [73]

The monitor detects tritium on the surface and in the near-surface regions by virtue of primary ionization in air due to the outward electron flux from the contaminated surface. The resulting ion pairs are measured by imposing an electric field between the contaminated surface and a collector plate. A simple theoretical model relates the total tritium concentration on a surface to the measured current.

Experiments benchmarking the application of the surface activity monitor on metal surfaces against independent measurement techniques of aqueous dissolution and thermal desorption show equivalence in the total tritium activities measured. Comparison of surface activity monitor measurements with the dry polystyrene smear protocol has shown that the two methods are complementary. Smearing measures the activity removed by the smear action, which can be used to infer the total activity on the surface. Surface activity monitor measurements determine the total activity on the surface, which can be used to infer removable activity. Ontario Hydro has stated that this device is the only surface monitor for tritium that provides an absolute measurement of the total activity on metallic surfaces.

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[70] K.J. Hofstetter, et al., “Field Deployable Tritium Analysis System for Ground and Surface Water Measurements,” accepted for publication in the *Journal of Radioanalytical and Nuclear Chemistry*.

[71] N.P. Kherani, W.T. Shmayda, “Ionization Surface Activity Monitor for Tritium,” Fusion Technology 28(3) (1995) 893

[72] W.T. Shmayda, N.P. Kherani, D. Stodilka, “Evaluation of the Tritium Surface Activity Monitor,” Proceedings of the Symposium on Fusion Technology, Lisbon, 1996.

[73] N.P. Kherani, W.T. Shmayda, “Monitor for Measuring the Radioactivity of a Surface,” US and European patents pending.

Experiments demonstrating the application of the surface activity monitor on a variety of non-conducting surfaces have been conducted. Some of the non-conducting surfaces examined include paper, concrete, granite, and wood. It has been determined that the best approach to measuring the tritium on non-conducting surfaces is indirectly; that is, a surface smear is taken and the activity on the smear is done by the SAM. Although this can be done with a liquid scintillation counter by using a SAM, no liquid scintillation cocktail is used, and the measurement is immediate.

Currently, the SAM is commercially available through Tyne Engineering (model 7001), which has measurement ranges of 10-1600 nCi/dm^2. A summary of the technical specifications of the instrument is given in Table 5-2.

5.9.3.c Breathalyzer

A Tritium-in-Breath Monitor was at one time being developed in Canada by Scintrex. It was to be an automatic monitor dedicated towards health physics and radiation biology applications. The Tritium-in-Breath Monitor measured levels of exhaled tritium within 5 minutes of sampling, thus saving considerable time and effort in the monitoring process. This rapid assessment has a sensitivity level of 5 µCi/L-urine equivalent, which may be sufficient for alarming the cautionary levels of in-body tritium. Preliminary development of this equipment had been done at the Atomic Energy of Canada Ltd. research laboratory in Chalk River, Canada. At present, no commercial unit is available.

TABLE 5-2: Model 7001 Surface Activity Monitor technical specifications	
Range	0-3552 kDPm/dm^2 (10-1600 nci/dm^2
Detection Sensitivity	+/-2.22 KDPM/dm^2 (+/- 1nCi/dm^2)
Response Time	<2 minutes
Measurement Area	100 cm^2
Display	128x64 Graphic LCD Display
Accuracy	+/- 4.44 kDPm/dm^2 (0-100 kDPm/dm^2), +/-10% above 100kDPm/dm^2
Process/Ambient Temperature	15°C to 50°C
Relative Humidity	5 to 65%
Air Pressure	80-120kPa
Tritium Wetted Parts	316L Stainless Steel, Teflon
Dimensions	150 mm dia x 178 mm high x 254 mm wide
Weight	2.7 kg

5) Table 5-2: Model 7001 Surface Activity Monitor technical specifications

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^{o}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:

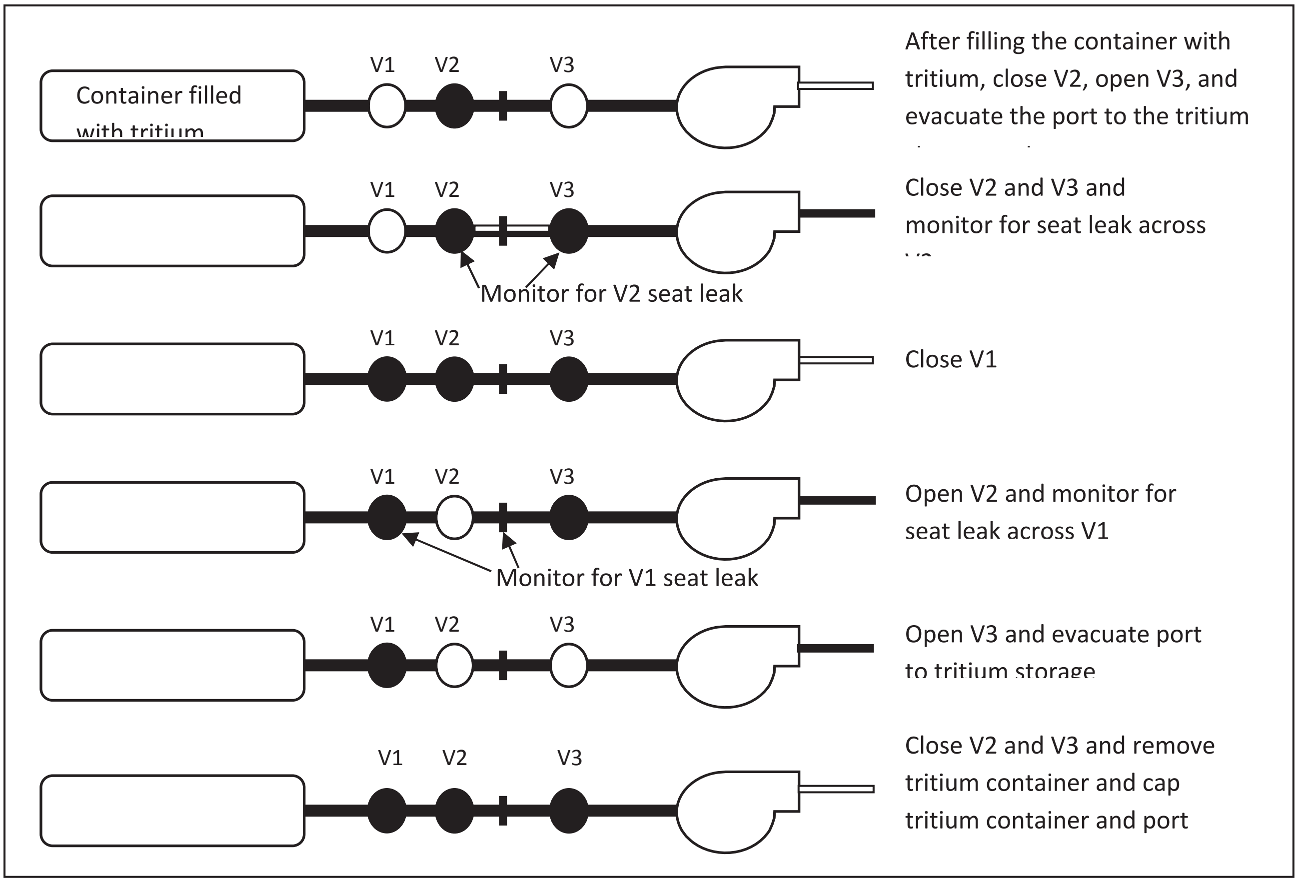
The tritium is trapped inside the container by the first valve, and the valve seat of the first valve is exposed to tritium during storage or transport.

- 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



After filling the container with tritium, close V2, open V3, and evacuate the port to the tritium

Close V2 and V3 and monitor for seat leak across

Close V1

Open V2 and monitor for seat leak across V1

Open V3 and evacuate port to tritium storage

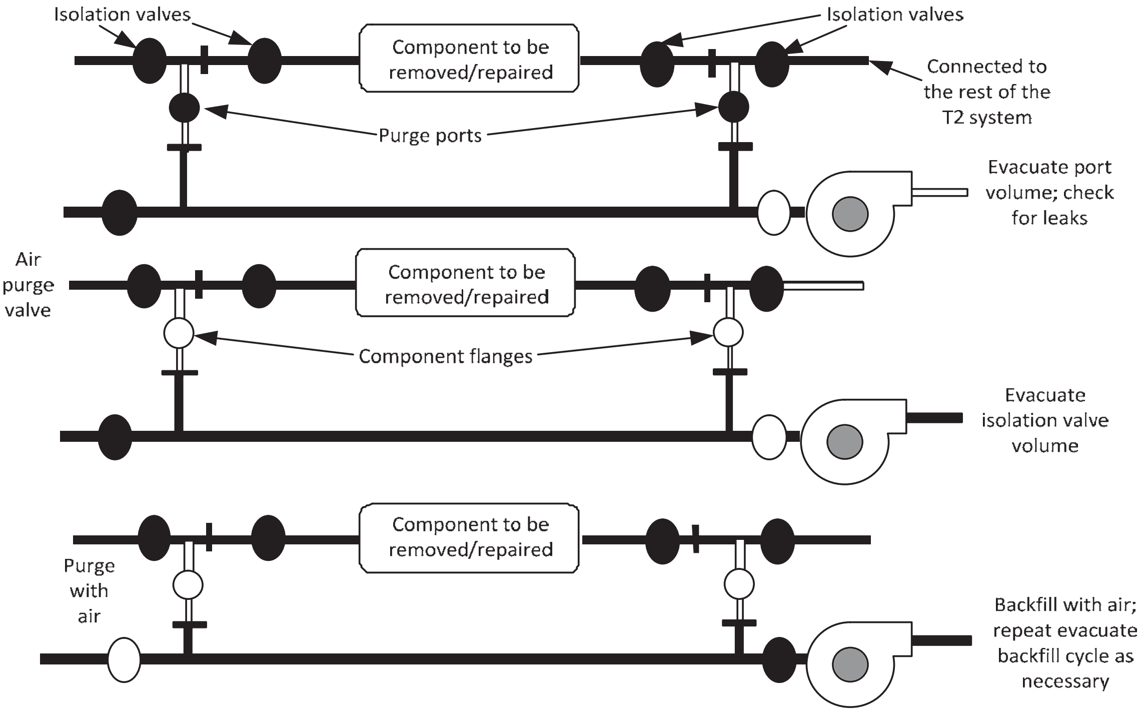
Close V2 and V3 and remove tritium container and cap tritium container and port

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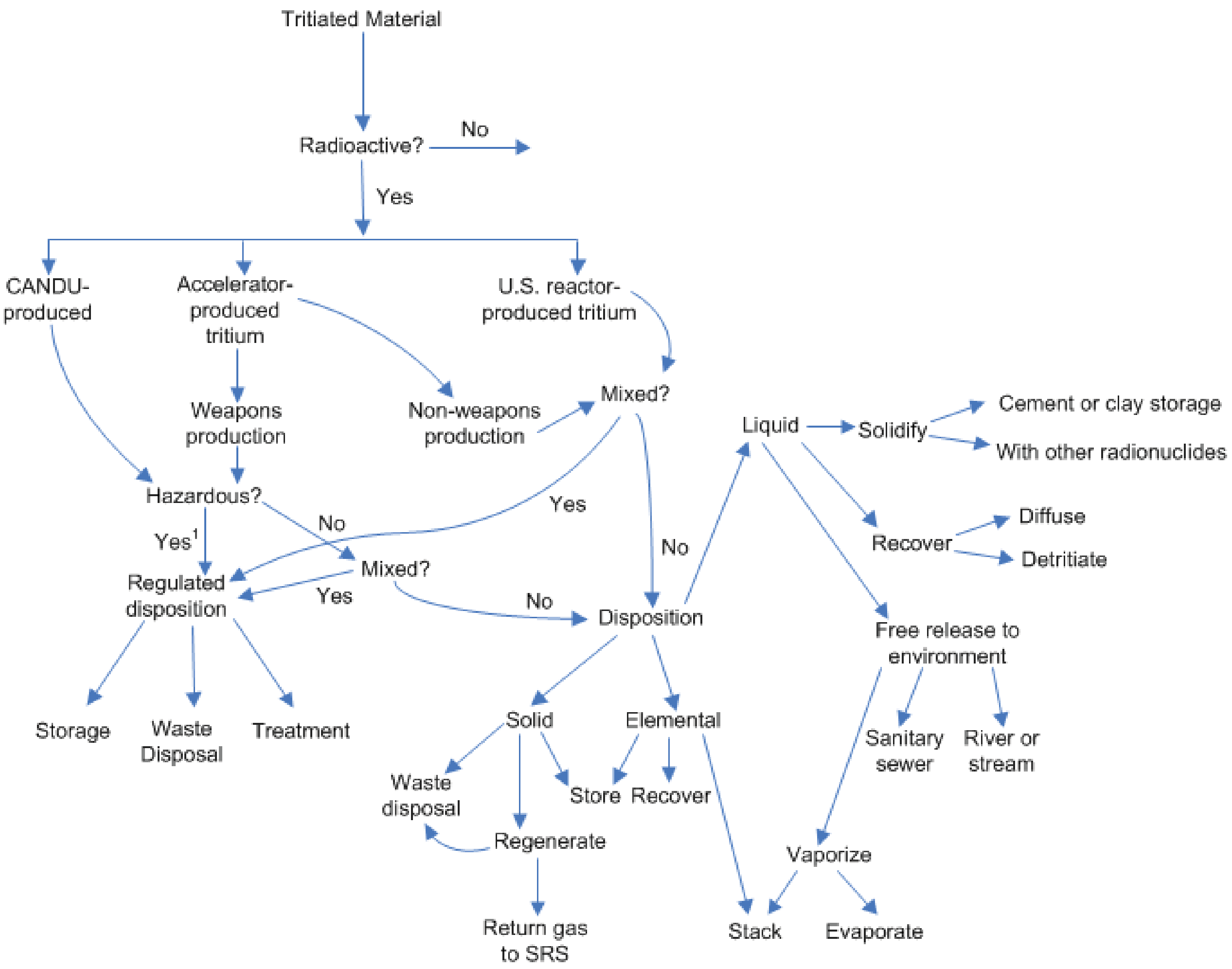
[83] Preamble of the First Third Land Disposal Restrictions Final Rule, 53 FR 31149, August 17, 1988.

[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

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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.

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.,

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^oC. 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 x 10^{-11} Ci
- 1 Ci = 3.7 x 10^{10} disintegrations/second = 3.7 x 10^{10} Bq = 37 GBq
- Avogadro's Number = 6.023 x 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 < 4×10^4 Bq of tritium/cm^2 (1 µCi/cm^2)
    - Nonfixed plus fixed contamination on the inaccessible surface averaged over 300 cm^2 is < 8×10^5 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 < 8×10^5 Bq of tritium/cm^2 (20 µCi/cm^2).
    - Nonfixed plus fixed contamination of accessible surface averaged over 300 cm^2 is < 8×10^5 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 = 2×10^{-5} µCi/mL = 7×10^5 Bq/m^3
  - DAC for HT = 200,000 µCi/m^3 = 2×10^{-1} µCi/mL = 9×10^9 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^{o}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 x 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 x 10^{-7} cm^3 /s $^{/}$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 ^{3}He, 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)*

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

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)
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

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

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

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

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

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

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.

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

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

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

Rearranging the equation becomes

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

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^{o}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 ^3He 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.

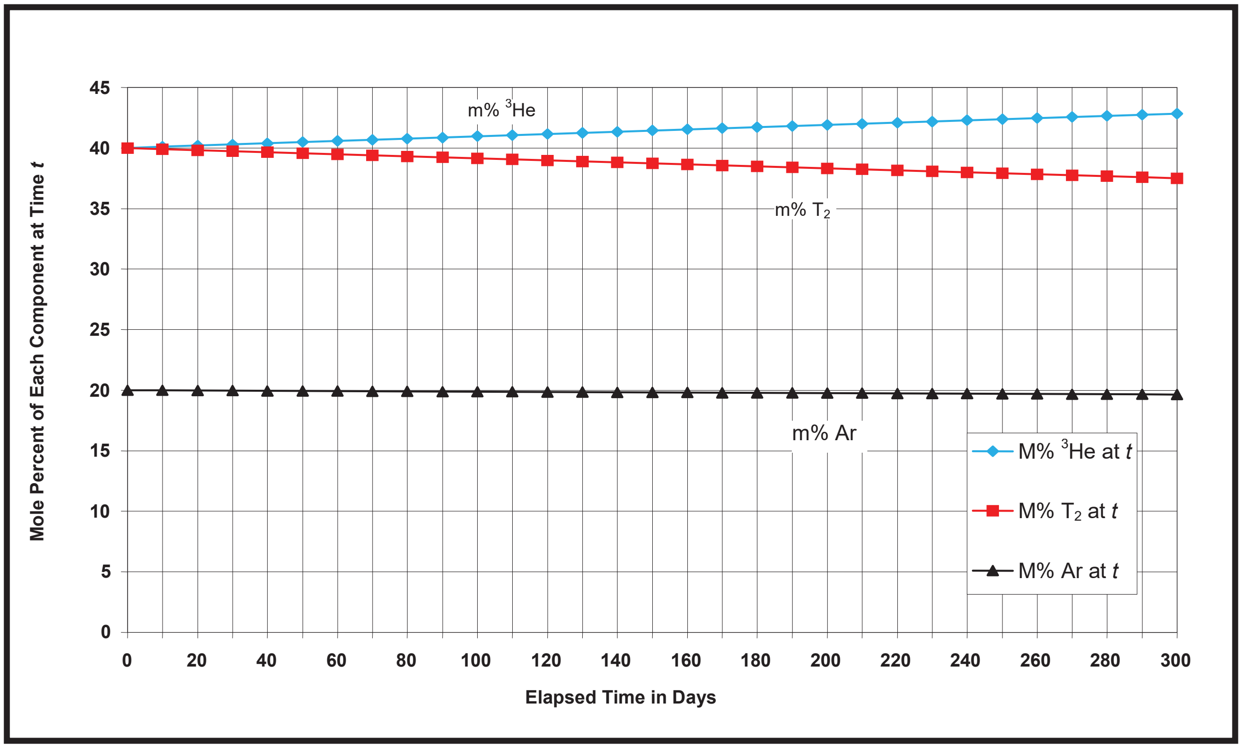

FIGURE C-1. Change in mole percent with time in a mixture of 40% T_2, 40% ^3He, and 20% Ar

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

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

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

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

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

where ($n_{(\mathrm{T2CSmplDa})} \times e^{((t \times \ln(0.5))/4500.88)}$) = number of decayed moles of T
($n_{(\mathrm{TotCAnlDa})}$) = 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

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

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

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

Rearranging the equation, it becomes

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

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

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

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

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

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.

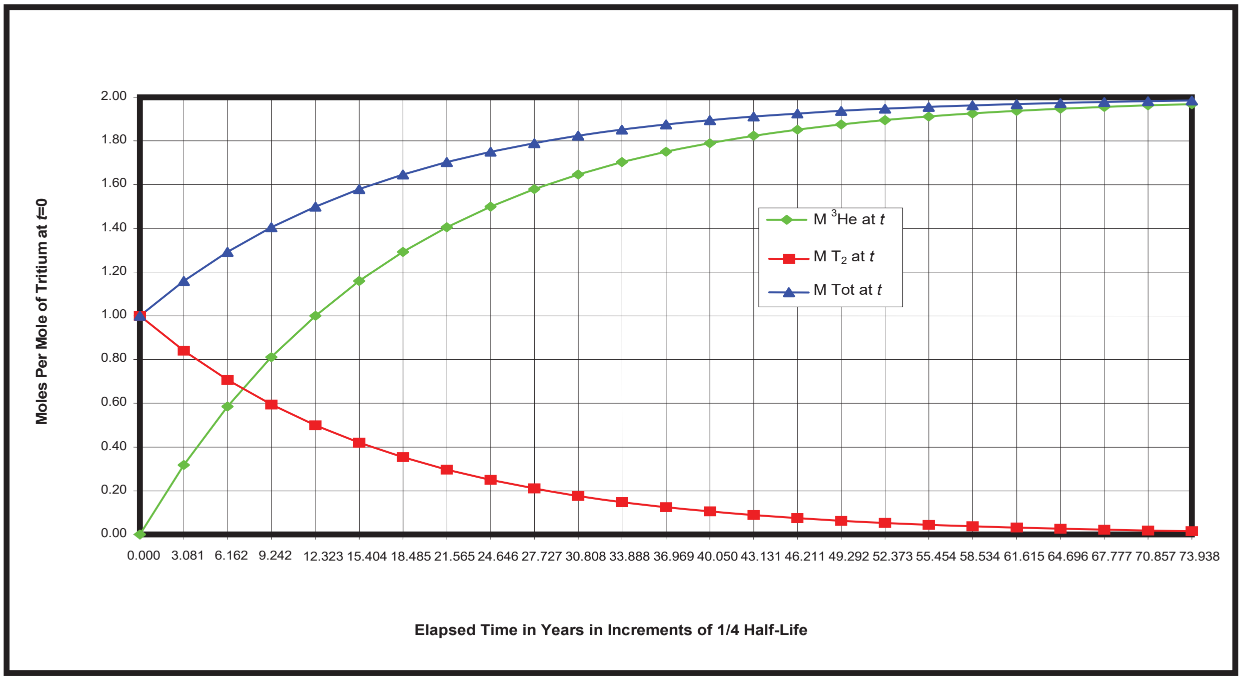

FIGURE C-2. Moles of T_2 and ^{3}He versus time

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.

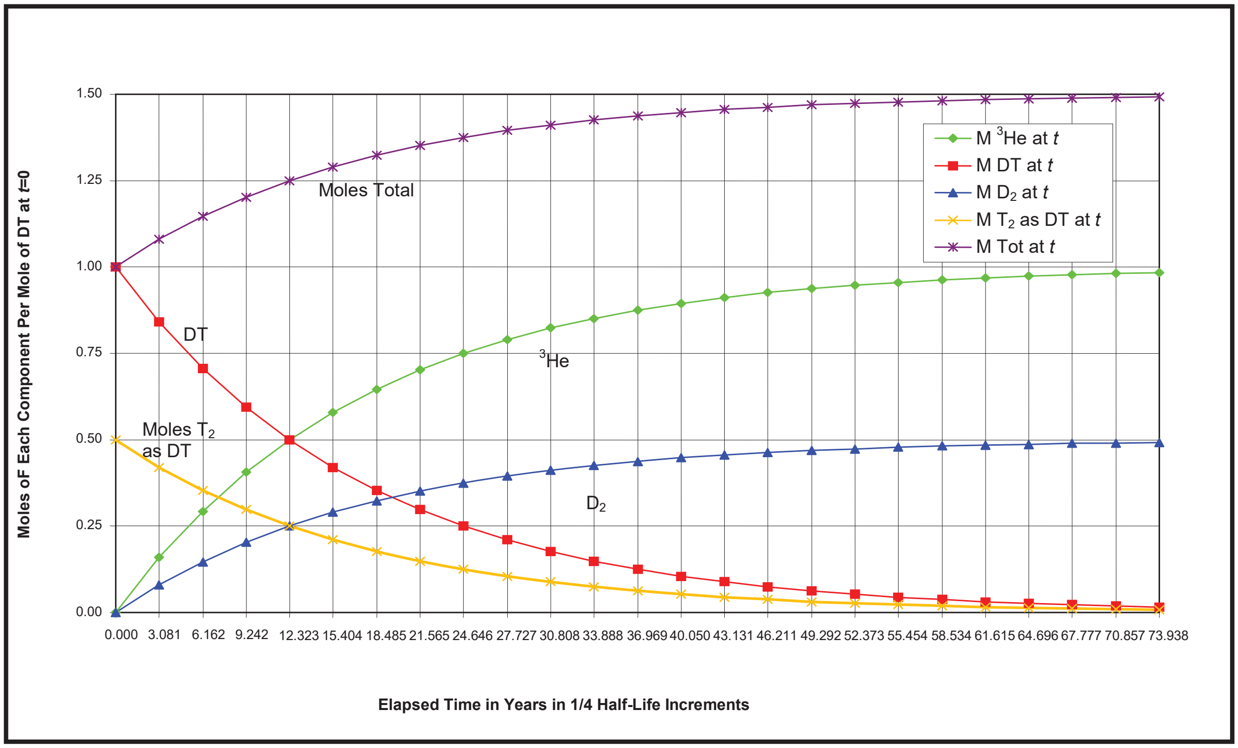

FIGURE C-3: Moles of DT, ^{3}He, and D_2 versus time

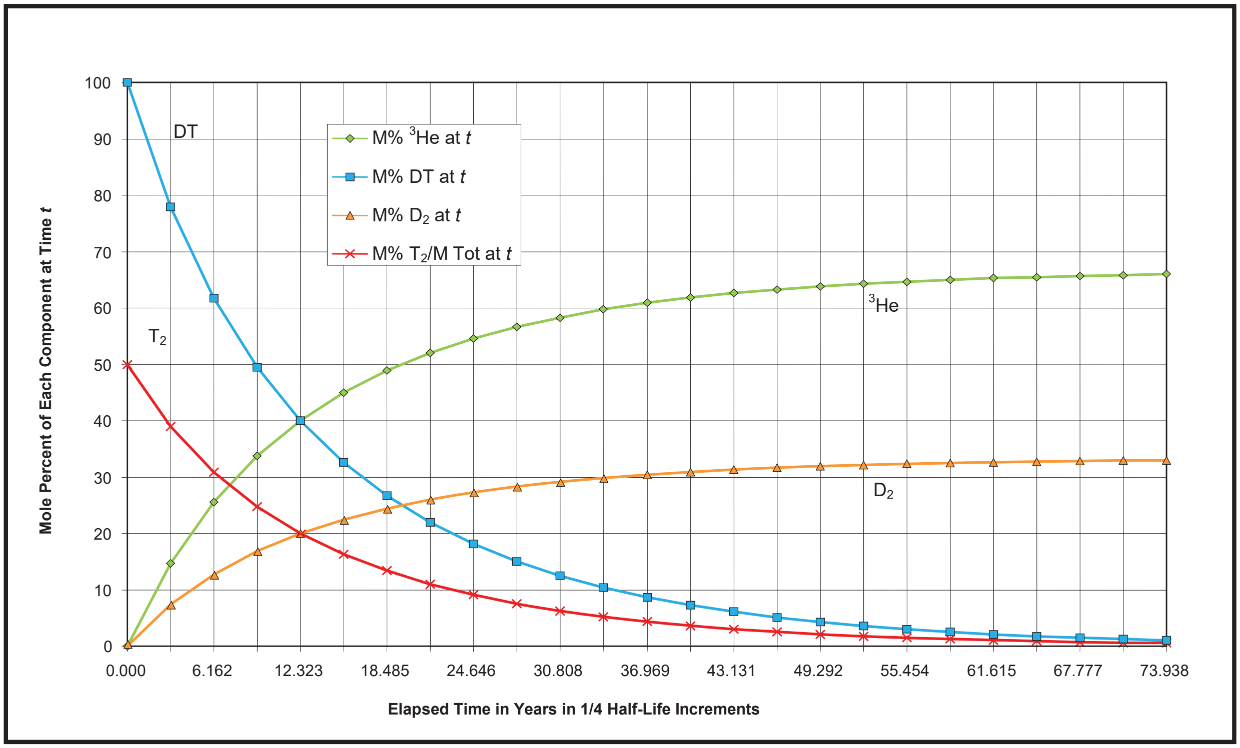

FIGURE C-4: Mole percent of T_2, D_2, DT, and ^{3}He versus time

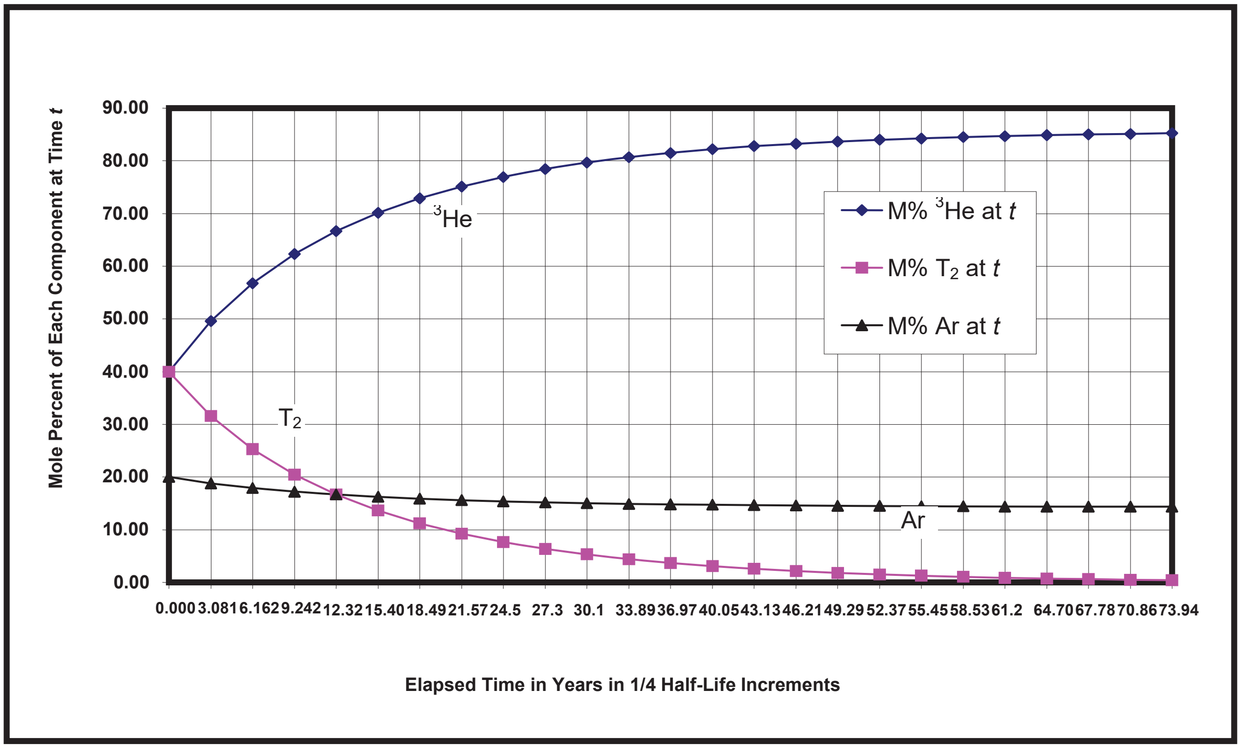

FIGURE C-5: 40% T_2, 40% ^{3}He, and 20% Ar - Changes in mole percent of components versus time

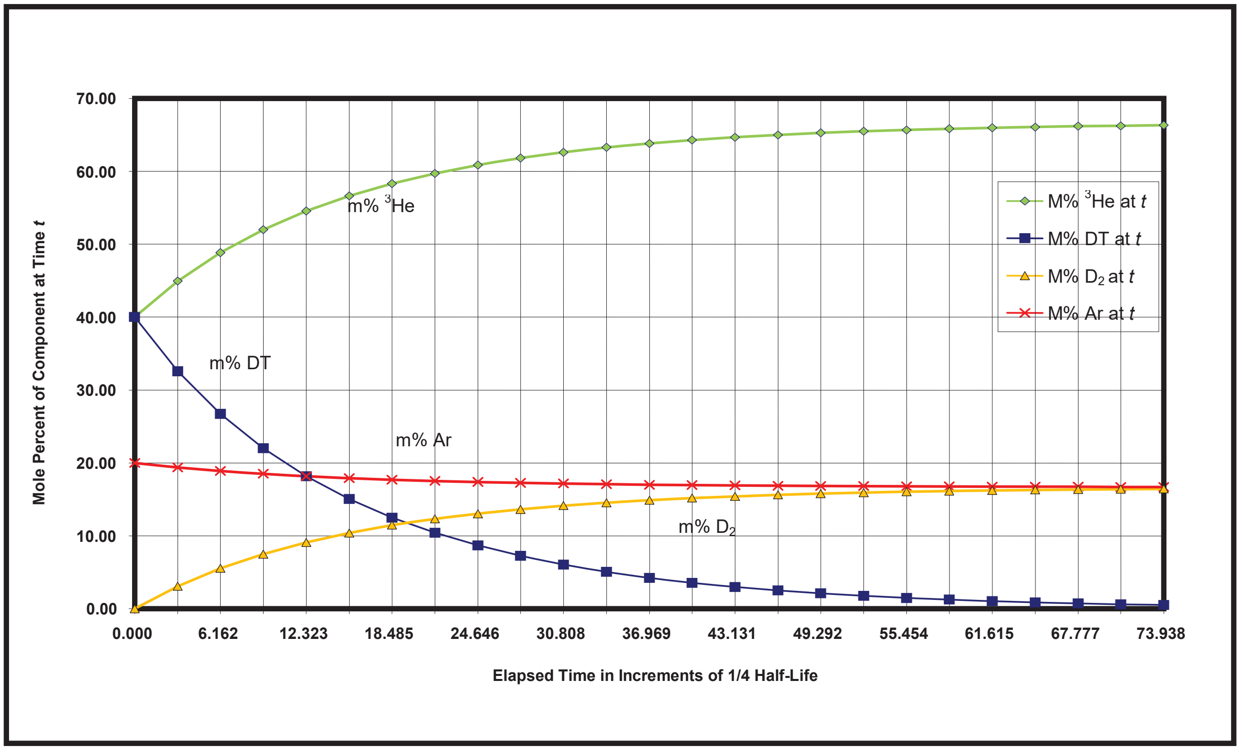

FIGURE C-6: 40% DT, 40% ^{3}He, and 20% Ar, Change in mole percent of each component versus time

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^{o}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.

APPENDIX E: RADIOLOGICAL CONTROL PROGRAMS FOR SPECIAL TRITIUM COMPOUNDS

E.1 - INTRODUCTION

Over the past several decades, DOE and its predecessor agencies have undertaken a variety of missions related to basic scientific research and national defense. A number of these missions have involved development, processing, storage, and disposal of a wide variety of exotic and hazardous materials, now requiring comprehensive health and safety programs to ensure protection of affected workers, the public, and the environment. DOE's programs for protection of individuals from exposure to these materials have evolved over the years in response to changes in scientific knowledge and societal perceptions and impacts. In some cases, the physical and hazardous properties of the materials were not well known at the time of their development or use. In other cases, the hazards were fairly well known and characterized, but changes in scientific knowledge and public risk perceptions have required improvements in associated protection programs.

E.1.1 Special Tritium Compounds

The detection and control of tritium are complicated when it is produced, used, and stored in some chemical and physical forms. Hydrogen atoms tend to migrate through surfaces with which they come into contact, and form chemical bonds with the host material. The hydrogen atoms may be released at a later time (commonly referred to as "outgassing"). While this property of hydrogen atoms may complicate measures implemented to contain tritium, it has been well characterized, and appropriate materials can be specified to minimize adverse effects. This property can also be used to practical advantage, as hydrogen atoms (including tritium) can be diffused through other materials, particularly metals, such as uranium, and stored within the matrix for subsequent extraction and use.

Tritium atoms may also diffuse into, and form chemical bonds with, organic materials with which they come into contact. The most common of these host materials are lubricating oils, solvents, and plastics.

The term "special tritium compound" (STC) is used to describe compounds of tritium with both host metals and organic materials.

The diffusion of tritium atoms into host metals and organic materials can complicate efforts to detect, and quantify levels of, tritium contamination in the workplace, environment, and individuals. These complications generally fall into three categories:

- Detection – Once the tritium atom has migrated into the matrix of the host material, some fraction of its emissions may not be able to escape that matrix.

- Physical and Chemical Behavior – If a tritiated material exists in a particulate state, or is subjected to any forces (e.g., heat, abrasion, cutting) that reduce it into a particulate state, the tritiated particles will behave physically and chemically like the host material particles, not like hydrogen. Therefore, the characteristics that are important variables in the internal dose assessment process (e.g., particle size, solubility, disassociation, inter-compartment transport, etc.) differ from those of tritium in elemental or oxide form.

- Emitted Radiation – Although only a small percentage of the beta particles emitted by the decaying tritium atoms may be detected, they may interact with surrounding host material atoms, resulting in emission of bremsstrahlung radiation and internal bremsstrahlung from the nucleus.

The behavior of STCs can present a variety of challenges to a facility's radiation protection program. For example:

- It may be difficult to differentiate between STCs and the more common forms of tritium using routine workplace monitoring techniques.

- The physical and chemical behavior of STCs may render commonly used tritium bioassay and available internal dose models ineffective.

- Difficulties in identifying and quantifying STC contamination can cause significant delays between performance of workplace monitoring and completion of analyses.

In the absence of effective detection methods, workplace hazards cannot be accurately assessed. Without accurate hazard assessments, associated activities such as area posting, material labeling, job planning, access control, and material release, are made difficult.

The risk associated with STCs can be put into perspective by comparing the effective dose equivalent per unit uptake (DCF) of various STCs with the effective dose equivalent per unit uptake of other representative radionuclides. As shown in Table E.1-1, the DCF for Hf-T, a very insoluble tritiated particle is about 24 times that of HTO, while it is one twentieth that of ^{137}Cs, and one 270,000th that of ^{239}Pu. Thus, while the risk associated with STC that results in the greatest dose per unit uptake is greater than that from tritium oxide, it is less than the risk associated from fission products and actinides. Note that when assessing the hazard from STCs and other forms of radioactive material, it is necessary to know the quantities of material available for uptake, as well as the opportunities for uptake in addition to the inherent risk associated with the radioactive material.

Table E.1-1 Ratio of DCF for Selected STC to DCF for Selected Radioactive Isotope or Compound

STC/Isotope	Ratio of DCF for selected STC to DCF for selected radioactive isotope or compound
$\frac{H_f - T}{HTO}$	24
$\frac{OBT}{HTO}$	2
$\frac{H_f - T}{^{137}Cs}$	1/20
$\frac{H_f - T}{^{235}U}$	1/77,000
$\frac{H_f - T}{^{239}Pu}$	1/270,000

E.1.2 Prevalence of STCs in the DOE Complex

STCs typically occur in much smaller quantities than other more common types of radioactive materials such as plutonium, uranium and HTO. However, STCs are found at many DOE sites as a result of past and present DOE operations. Table E.1.1 lists examples of the types of processes involving the production or handling of STCs as well as various types of structures that could contain or be contaminated with STCs. The important consideration is that there is a wide variety of processes and locations where STCs may exist. Accordingly, it is prudent to consider the need for radiological protection measures specific to STCs when planning work in areas where any form of tritium has been handled or stored.

Table E.1-2 Examples of Processes and Locations where STCs Could Exist

Processes	**Locations**
Tritium targets for neutron generators	Gloveboxes
Reactor operations	Fumehoods
Fusion Experiments	Ventilation systems
Extrication of tritium from fuel elements	Waste containers
Isotope separation	Sampling equipment
Storage of tritium	Ground water
Stripping tritium from non-hydrogen gas streams	Uranium beds
Operations involving tritium labeled compounds	Spent fuel
Operations producing tritium contaminated oils and solvents	Alcohol wash systems Weapons components
Waste treatment and storage	Molecular sieves and getters
D&D Operations	Kiln piping
Weapons testing	Be reflector blocks
	Neutron Generators
	Heavy water purification evaporators
	Fuel storage basins

Although the DOE radiological protection community has been aware of STCs for many years, to date their impact has been limited by the design features that are incorporated into DOE facilities that handle significant quantities of tritium. These design features include various forms of material containment and control, such as gloveboxes and HEPA-filtered ventilation systems that effectively prevented significant releases of STCs to occupied areas of the workplace or the environment. Sensitivity to STC contamination has been increased as a result of the emphasis in DOE activities involving decontamination and decommissioning. These activities compromise the effectiveness of the installed design features and allow releases of STC contamination to the surrounding areas if proper controls are not developed. Such releases may cause exposures to individuals in the area and releases of STCs to the environment, both on- and off-site. In light of such experiences, DOE suggests that individuals setting up programs for radiological control of STCs contact sites that either work, with or have worked, with tritium to learn of their experiences with STCs[90].

E.1.3 Appendix Overview

This appendix examines the difficulties associated with STC detection, assessment, and control, and provides guidance for incorporating effective STC monitoring and control regimens into the facility's radiation protection program. It has been developed based on currently available technology for detecting, identifying, characterizing and monitoring STCs. Accordingly, in some cases a more conservative approach to radiological controls is recommended when compared to oxide and gaseous forms of tritium. As technology in this area changes, more efficient approaches to radiological protection may be instituted.

This appendix should be used in conjunction with DOE's existing requirements and guidance documents, including:

- 10 CFR Part 835, *Occupational Radiation Protection*;
- DOE G 441.1-1C, *Radiation Protection Programs Guide for Use with Title 10, Code of Federal Regulations, Part 835, Occupational Radiation Protection*;
- DOE-STD-1098-2008, *Radiological Control Standard* (use the revised version, reflecting the 2007 amendment to 10 CFR Part 835, when available);
- DOE-STD-1121-2008, *Internal Dosimetry Standard* (use the revised version, reflecting the 2007 amendment to 10 CFR Part 835, when available);
- DOE-STD-1111-2013, *DOE Laboratory Accreditation Program Administration*, and its supporting standards (use the revised version, reflecting the 2007 amendment to 10 CFR Part 835, when available); and
- DOE's Radiological Control Technical Positions.

Should any conflict exist between the guidance provided in this appendix and DOE's requirements for radiation protection, DOE's requirements take precedence. Such conflicts should be brought to the attention of the DOE office responsible for worker protection policy.

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[90] The Mound Site performed decontamination and decommissioning (D&D) operations involving STCs during the period this document was developed.

E.2 - nature of special tritium compounds

E.2.1 General

Hydrogen atoms (including those that exist in the form of tritium) tend to diffuse through materials with which they come into contact, and may be released at a later time (commonly referred to as “outgassing”). While this property of hydrogen atoms may complicate measures implemented to contain tritium, it has been well characterized, and appropriate materials can be specified to minimize unintended adverse effects. This property can also be used to practical advantage, as hydrogen atoms can be diffused through other materials and stored within the matrix for subsequent storage, transportation, release, and use. The diffusion and outgassing processes may be accelerated through application of heat.

E.2.2 STC Host Materials

STCs can be created by intentional combination of tritium with the desired materials or by inadvertent contamination of a material that has been subjected to the presence of tritium for a period of time. DOE facilities have used a wide variety of metals (e.g., Ti, Zr, U, and Hf) in tritium research, purification, and storage, creating an equally wide variety of materials referred to as “metal tritides.” Metal oxides (e.g., rust), siliceous materials (environmental dust), and carbonaceous materials (e.g., polymers or environmental dust) can also become tritium contaminated and exhibit particulate properties. Tritium may also react with organic materials, resulting in the formation of organically-bound tritium (OBT). The main types of OBT encountered in the DOE complex are solvents, oils, and solid particulates (e.g., plastics, nylon, and organic dust forms). OBT can be particulate or non-particulate in nature.

E.2.3 STC categorization

In this appendix, STCs will be categorized according to (1) the ease with which the tritium (either unbound or bound to an STC) can be taken into the blood stream and (2) the physical form of the STC (e.g. particle, vapor, large solid, and liquid). Table E.2-1 illustrates this classification scheme.

Table E.2-1: Categorization of Special Tritium Compounds

Ease of Uptake	**Insoluble**			**Soluble**	
Physical Form	**Liquid/vapor**	**Particle**	**Large Solid Form**	**Liquid/vapor**	**Particle**
Examples of STCs	Tritiated pump Oil	Hf-T Ti-T Eu-T Zr-T Dust Rust Pump oil droplets Tritiated flyash	Tritiated Ti source for neutron generator, Tritiated nylon	Tritated methane Tritiated solvents (acetone, octane)	U-T Pd-T

E.2.3.1 Ease of Uptake

Ease of uptake refers to the rate at which radioactive material can be taken into the bloodstream. For STCs this process includes both uptake of tritium that has dissociated from a host or carrier material and uptake of an entire STC molecule. The International Commission on Radiological Protection (ICRP) (ICRP 1994) categorizes most radioactive materials in terms of the rate of absorption from the respiratory tract to the bloodstream. Materials absorbed on the order of days are classified as absorption type F (fast), those absorbed on the order of weeks are classified as absorption type M (moderate) and those absorbed on the order of years are classified as absorption type S (slow). (In addition to the three absorption classes, ICRP (ICRP 1998b) categorizes certain vaporous substances that are very rapidly absorbed by the body as absorption type V.) For purposes of understanding some of the radiation protection elements expounded in this appendix, absorption type S and M materials can be considered insoluble and absorption type F material can be considered soluble. Note that the ICRP classification forms the basis of the dosimetric calculations contained in Chapter 5.

Note: The Mound Technical Basis Document (Mound 2000) was the document from which much of the material in this appendix was derived. In that document, the term stable was used instead of insoluble to describe particles from which the tritium dissociates slowly, and the term unstable was used instead of soluble to describe particles from which the tritium dissociates more rapidly. The terms soluble and insoluble are selected for this appendix in order to apply the same terminology used to describe the ease of tritium uptake by the body as the primary means of categorizing STCs.

Particles or particulates can be formed from all types of STCs. To explain the uptake of these particles, the ICRP in publication 71 (ICRP 1995) assumes that the tritium dissociates from the rest of the particle and is then taken up by the body. The tritium that is not dissociated from the particle irradiates those tissues it comes in contact with while it remains in the body. Tritiated particles from which the tritium dissociates slowly may be considered insoluble. Tritiated particles from which the tritium dissociates rapidly may be considered soluble. In this appendix the operational distinction between soluble and insoluble particles is that insoluble particles do not readily release the bound tritium to either aqueous solutions or air during the interval between sample collection and analysis.

Another class of STCs is organically bound tritium (OBT) compounds. For these types of STCs, the tritium is not readily released from the carbon-tritium bonds to air or aqueous solution (Hill, 1993), therefore, the classification as either soluble or insoluble is based on the ease by which the entire molecule is taken up by the body.

Soluble OBT is rapidly taken up by the body while insoluble OBT is more slowly taken up by the body.

E.2.3.2 Physical Form

The manner in which in the body takes up STCs, and hence the radiological controls for STCs, is determined by the physical form of the STC. In this appendix, the primary physical forms considered are particles, liquids/vapors and large solid forms.

Radiological controls for STC particles that are considered to be insoluble should be based on the controls used for radioactive particles rather than the controls used for HT, HTO, or vaporous forms of OBTs. Such tritiated particles are called insoluble tritiated particulates (ITPs). Examples of ITPs are tritiated metals, and metal oxides; the large molecule component of OBT oil; and solid OBTs such as flyash, nylon, and organic dust.

Soluble types of particulates rapidly release their bound tritium. For these types of triated particles, radiation protection measures should be based on those for HT or HTO. In addition, the detection problems that arise from the binding of tritium within the host particle are not an issue (see section E.3.1). Examples of such particles are metal tritides of palladium and uranium.

STCs that are liquids and vapors are primarily comprised of OBTs. Soluble OBT can be incorporated into the body by inhalation, ingestion, or absorption through skin. Soluble OBT distributes throughout the body causing a whole body dose. ICRP 78 (ICRP 1998b) categorizes the dissolution rate for soluble vaporous OBT as Type V (very fast dissolution). Solvents are considered to be soluble. Radiological protection measures for vaporous OBTs are identical to those used for HT and HTO. When establishing controls for liquid forms of OBTs, it is important to consider absorption of the OBT through the skin, the vapor pressure of the liquid, the possibility that the liquid is composed of more than one OBT, and dispersal mechanisms that could aerosolize the liquid. An example of a liquid OBT is tritiated oil that has been used in pumps and air compressors.

Tritiated oils are a type of OBT that contains both soluble and insoluble components. Oils can be taken into the body by inhalation when they are in particulate form or can be absorbed through the skin. Because tritiated oils primarily consist of insoluble components (see section E.4.4.3) tritiated oils will be treated in this appendix as insoluble particulates when they become airborne droplets.

The term “large solid forms” denotes the types of STCs that cannot be easily taken into the body, such as a piece of tritiated metal. Radiological controls for these types of tritides consist primarily of planning for situations in which portions these materials could be converted to airborne particles. Otherwise, standard radiation protection measures for the control of radioactive materials should be applied to these types of STCs. See section E.3.2.1.2 for additional guidance that may be applicable to large solid forms of STCs.

E.2.4 Sources of STCs

As discussed previously, tritium, including that contained in STCs, is not considered an external exposure hazard. However, particulate tritiated materials, both metals and organic materials, can be ingested or inhaled, creating an internal exposure hazard. Solid particulate OBTs, such as plastic, nylon, organic dust, or the large molecule component of OBT oil, can become airborne by dispersal mechanisms. Equipment that can impart energy to oil through motion or release of pressure (e.g. pumps and air compressors) can cause an oil mist aerosol to be generated.

E.2.5 Hazards Associated with STCs

E.2.5.1 Ingestion of Tritiated Particulates

Following ingestion, STCs lose some fraction of their tritium as HTO or soluble OBT in digestive fluids. The fraction of the infused tritium lost via this mechanism varies depending on the stability of the specific material in the gastrointestinal (GI) tract; some particles lose essentially all of the tritium and others lose very little. In any case, the only significant dose following ingestion of tritiated particles is due to the resulting HTO or soluble OBT that is released. The remaining intact particles will undergo fairly rapid elimination from the body, resulting in negligible internal doses. Since HTO or soluble OBT is rapidly assimilated physiologically, this dose component is readily assessed via urine bioassay.

E.2.5.2 Inhalation of Tritiated Particles

Following inhalation, tritiated particles may be deposited in the lung. Once deposited in the lung, the infused tritium is removed from the body via two mechanisms:

1 Dissolution of tritium from the particulate and absorption into the body as HTO or soluble OBT, and removal through urine, expired air, and perspiration.

2 Mechanical transport of the particulate itself to the GI tract and removal through feces. (Inhaled tritiated particles removed to the GI tract will cause dose in the same manner as ingested triatiated particles, as described above.)

Removal of ITPs from the lung through mechanical transport can be slow. ITPs can, therefore, reside in the lung for a considerable amount of time. Accordingly, a significant portion of the entire committed effective dose equivalent resulting from the decay of the associated tritium atoms results from deposition in the lungs. The lung receives 85% of the committed effective dose equivalent (E_{50}) from Type M ITPs and 96% of the E_{50} from absorption type S ITPs (ICRP 71).

E.2.5.3 Skin Absorption of OBT

Solvent OBT

Solvent OBT can, if spilled as liquid on the skin, produce a skin absorption pathway. The potential hazard of this pathway should be assessed using known or analyzed activity of the liquid. Solvent OBT absorbed through the skin can be assessed using urine bioassay.

OBT Oils

Tritiated oils can be found in pumps and compressors located in areas where work with tritium has been conducted. Trivedi (1995), in experiments on rats, has shown that tritiated oils can be absorbed by mammalian skin. Accordingly, skin absorption is a valid intake pathway for oil components. See section E.4.4.3 for additional information on skin absorption of OBTs.

E.2.6 Challenges to Radiological Control Programs

Based on the previous discussion, STCs may present the following challenges to a facility's radiological control program (above and beyond the challenges presented by more common types and forms of radioactive material):

- Because some of the tritium emissions may be shielded by the particle, routine area monitoring efforts may not yield an accurate assessment of radionuclide levels on surfaces and dispersed in workplace atmospheres.

- The difficulties in area monitoring may cause corresponding difficulties in related activities, such as posting, labeling, work planning, access control, decontamination, personal protective equipment (PPE) handling, and material control and release.

- Retention of insoluble tritiated particles in the body may render conventional tritium bioassay methods (urinalysis) ineffective. While ITPs behave as other particulates that are inhaled, an acceptable bioassay procedure is not currently available for some types of ITPs.

E.3 – WORKPLACE MONITORING FOR STCs

E.3.1 Observed Versus Actual Activity

Because of the difficulties associated with detection and quantification of STC levels on surfaces and in sample media, two terms are used in this appendix to describe analysis results. The observed activity is that activity that is detected by the prescribed analytical method. Because of the self-shielding attribute of particulate STCs, the observed activity (the activity detected by the sample analysis equipment, corrected for efficiency and detector-to-sample geometric considerations) may not reflect the total quantity of radioactive material present. The term actual activity is used to describe the total activity present in STCs. In practice, the actual activity may be estimated from the observed activity by use of appropriate conversions to adjust for particle self-shielding and other factors that may interfere with accurate sample analysis. However, in many cases, characterization of metal tritide species make it difficult to quantify and correct observed activity to actual activity. As discussed in this document, the use of observed activity is particularly relevant for internal dose assignment.

Current regulations require that certain radiation protection measures (e.g., contamination area posting and access control, radioactive material labeling and control) be implemented based on assessments of the quantities or concentrations of radioactive material present. This appendix adopts the concept of observed activity (discussed above) as an appropriate surrogate for the actual activity for STCs. 10 CFR Part 835 Appendix D, Footnote 2 provides surface contamination values in disintegrations per minute as meaning “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.” The use of observed activity is consistent with this provision. This appendix will demonstrate that certain other measures, such as individual dose estimates, may be based on assessments of observed activity.

E.3.1.1 Conversion from Observed to Actual Activity

The conversion of analysis results from observed to actual activity is relatively straightforward, once the various factors that introduce analytical uncertainties are understood and quantified. First, corrections are applied to convert the analysis results presented by the instrument, usually counts per unit time, to the desired activity units and to eliminate the more common analytical uncertainties (e.g., uncertainties arising from detector efficiencies and counting geometry considerations). Further discussion on the use of observed activity is discussed in Section E.5.2, especially considerations given to self-absorption of emissions that do not escape the STC particle.

In more conventional sample counting applications, application of these corrections will yield the actual activity present in the sample, usually in units of disintegrations per minute per 100 square centimeters (dpm/100cm^2) for surface contamination monitoring or microcuries per cubic centimeter (µCi/cm3) for airborne radioactivity monitoring. (These are the units used for surface contamination and airborne radioactivity control criteria, respectively, in 10 CFR Part 835). However, due to the physical characteristics of STCs, application of these corrections yields only the observed activity. Further corrections may be made to account for that fraction of the tritium decays that have escaped detection and accurate quantification due to the physical characteristics of the STC.

The corrections needed for conversion from observed to actual STC activity typically include the following:

- A correction to increase the observed activity to account for the number of tritium disintegrations that have escaped detection due to shielding effects of the host particle;
- A correction to reduce the observed activity to account for detected events resulting from bremsstrahlung radiation created by tritium beta interactions with the host particles;
- A correction to reduce the observed activity to account for detected events resulting from tritium that is present in non-STC form.

Application of these corrections will require knowledge of, or conservative assumptions regarding, the host particle characteristics, including atomic number, size, and density. However, using currently available technology, it is difficult to determine the actual activity in a manner that will make it a better indicator of the hazard to a worker than observed activity.

E.3.2 Area Monitoring

Because tritium emits only low energy beta particles, it is not considered an external radiation exposure hazard. Therefore, area monitoring efforts are restricted to surface contamination monitoring and airborne radioactivity monitoring.

Tritiated metals, metal oxides, dust, and oil can occur in nearly any tritium area. Any tritium contamination collected on a swipe survey or a particulate filter should be viewed as a potential combination of these. This is particularly the case for residual concentrations of tritium, i.e., where elemental (HT), aqueous (HTO), or solvent (soluble OBT) tritium sources have been removed or evaporated. However, it is the insoluble tritiated particulates with very long biological retention times (i.e., ICRP absorption type S or M) that pose the greatest hazard to workers. As a result, workplace sampling/monitoring programs that can quantify specific, and distinguish between various types of STCs, will allow one to depart from the conservative approaches recommended in this Appendix.

E.3.2.1 Surface Contamination Monitoring

Chapter 11 of DOE G 441.1-1C, *Radiation Protection Programs Guide for Use with Title 10, Code of Federal Regulations, Part 835,* Occupational Radiation Protection, establishes appropriate guidance for developing and implementing a surface contamination monitoring program. This Guide addresses such issues as monitoring requirements, techniques, frequencies, and actions. This guidance is generally applicable to monitoring for STC contamination; however, the sample analysis and activity determination techniques may be different due to the difficulties in accurately assessing STC levels. Note that if the host material of the STC is radioactive, the conventional methods in the DOE guidance can be used to assess the extent and level of surface contamination.

E.3.2.1.1 Removable Contamination

The first step in establishing a surface contamination monitoring program for STCs is to identify the likelihood that STCs may exist in an area, and if they do, identify the most likely locations. The best sources of this type of information are historical records and workers' memories of facility usage. Facility management should consult these sources for information regarding previous usage of the facility and specific areas. Almost any area in which tritium was used, handled, or stored may be considered a candidate for possible STC contamination. For larger facilities, identification of all STC contaminated areas may be quite time-consuming. Priority should be placed on identifying and controlling those areas that are likely to have the highest STC contamination levels. In many cases, this will include areas where STCs were intentionally created, and areas that have (or had) the highest tritium contamination levels and the highest probability of host particle formation and distribution. Such a determination should be made by review of historical radiological monitoring records and facility processes.

Following identification of those areas most likely to be contaminated with STCs, detailed radiological monitoring should be performed to identify tritium contamination levels in the area(s). Such monitoring should be performed using routine tritium contamination monitoring techniques, commonly using either wet or dry smears and liquid scintillation counting (LSC). In the past, many facilities have used Styrofoam smears for collection of surface contamination samples for tritium analyses. Recent changes in the composition of scintillation cocktail fluid have largely eliminated this practice. The actual sample medium used is not particularly significant as long as its efficacy in collection of particulate material and suitability for LSC analysis can be demonstrated. Counting techniques should be capable of detecting contamination at or below the values provided in Appendix D of 10 CFR Part 835.

If the results of comprehensive smear surveys indicate that tritium surface contamination levels are less than one-tenth of the 10 CFR Part 835 Appendix D value, it may be appropriate to assume that there are no significant levels of STC contamination (i.e., levels requiring additional controls, such as posting, access control, or personnel monitoring). If the results of the smear surveys indicate that tritium contamination levels equal or exceed one-tenth of the 10 CFR Part 835 Appendix D values, further surveys may be conducted to identify sources of STCs in the area. It may also be necessary to identify the host material(s) so that appropriate assumptions may be made regarding assimilation and elimination.

E.3.2.1.2 Total Contamination

DOE has not established a total (fixed plus removable) surface contamination value for tritium and tritium compounds. The reason is that tritium can diffuse into the volume of any material to which it is exposed – at least in the gaseous forms normally encountered. After the exposure, the tritium migrates to the surface, making a definitive distinction between fixed and removable contamination difficult to ascertain.

The external hazard from tritium radiation is not considered to be a significant source of radiation exposure. Accordingly, compliance with the tritium value for removable contamination will provide an acceptable level of protection to individuals in controlled areas. This approach is considered acceptable for most tritium contamination.

However, there may be cases where tritium binds tightly to the matrix into which it has diffused, and removable contamination levels are below the values in 10 CFR Part 835. Such cases could occur when insoluble tritiated particles are fixed to a surface or from tritium exposure to bulk quantities of metals of the types from which insoluble metal tritides are formed. Footnote 6 to appendix D of 10 CFR Part 835 states that a total surface contamination value of 10,000 dpm/100 cm^2 may be applicable to these types of STCs. Accordingly, the provisions in 10 CFR Part 835 subpart L pertaining to total surface contamination values may be applied to these types of STCs when the total surface contamination level exceeds 10,000 dpm/100 cm^2. Note that when performing operations on these types of tritiated materials, it is important to assess the potential for producing airborne concentrations of respirable ITPs, and if needed, institute radiological controls appropriate to this hazard.

E.3.2.2 Air Monitoring

Chapter 10 of DOE G 441.1-1C, *Radiation Protection Programs Guide for Use with Title 10, Code of Federal Regulations, Part 835,* Occupational Radiation Protection, provides detailed guidance for developing and implementing an air monitoring program. This guidance is supplemented by that provided in the RCS (DOE 1999g) and in DOE-STD-1121-98 (DOE 2008), Internal Dosimetry. This guidance, combined with that provided above for identifying areas possibly affected by STC contamination, is applicable to the air monitoring program for STCs. Experience from DOE sites have indicated the need to understand that particulate STCs are generally not detected by Tritium CAMs due to the presence of an inlet filter. The practice of running two tritium CAMs side by side with one having this inlet filter removed may demonstrate whether particulate STCs are present. The presence and effect of this inlet filter may not be well understood by Health Physics staff. Likewise, the guidance that follows regarding identification of STCs and analysis of samples is applicable to the air monitoring program.

E.3.2.3 Identification of STCs

After collecting samples of surface contamination or airborne radioactive materials, it is necessary to conduct radioanalyses to identify the levels of tritium (and other radionuclides) present in the sample. Tritium analyses are most commonly performed by liquid scintillation analysis of surface smears and air samples. Although liquid scintillation analysis is very effective in analyzing for the presence of tritium, this technique does not differentiate between the various forms of tritium (HT, HTO, STCs). Thus, non-standard techniques are used to distinguish STCs from HT and HTO, and to distinguish among the various STC species. To date, a rapid, simple and reliable method for identifying STCs has not yet been developed.

In the following paragraphs, several techniques for identifying ITP species are described. However, all of these approaches have some drawbacks. In practice, some combination of the suggested analyses, or an entirely new approach, may be needed to evaluate the ITP contamination levels present in the samples. Such efforts will probably not be straightforward, and conservatism should be used to ensure that the hazards associated with ITPs are adequately addressed to provide an adequate level of protection to DOE workers. If the analyses cannot definitively identify the ITP species, identity of the ITP species may be based on process or historical knowledge.

In all cases, a conservative approach should be used in identifying STC-contaminated samples. If the combination of factors identified below (facility history, radioanalytical identification of tritium, particle species identification) leads to a conclusion that the presence of STCs is likely, then the appropriate controls should be implemented consistent with this manual.

For ITPs, a scanning electron microscope (SEM) may be used to identify the presence of particles that are likely hosts for tritium atoms. In combination with conventional radioanalysis using LSC, an estimate of the gross ITP contamination levels may be obtained. This technique does have limitations however. While the LSC can definitively establish the levels of tritium emissions, and the SEM can definitively establish the presence and size of possible host particles, it is difficult to establish a direct correlation between the detected tritium emissions and the particulate levels. That is, it is not possible to state with any degree of certainty what fraction of the detected emissions arise from ITPs versus that fraction that arises from the more common tritium forms that may also be present. The presence of likely host particles does not definitely prove the presence of tritium within those particles.

The presence, of but not the amount of, ITPs may also be inferred by performance of multiple LSC analyses of individual samples, and trending the results to evaluate the change in observed tritium activity as the tritium atoms are released from the host material in the sample. (The observed tritium activity increases because the betas emitted by the tritium molecules that have been released from the metal particle are not absorbed by the metal particle.) This method may only be effective for those ITPs where the tritium is released from the host particle over a time period ranging from a few hours to a few weeks. If the tritium binds to the host material very tightly, the growth in observed activity will not be apparent over any reasonable time interval between repetitive sample analyses. If the ITP species is soluble, then all of the captured tritium may be released into the counting solution before the sample is counted. Because the lack of an effect does not mean ITPs are not present, the use of this method depends on reliable knowledge of the strength of the bond between the tritium and the host material in the ITPs expected to be present in the work area. Information on dissolution is contained in Appendix D. If tritium is likely to be present in both ITP and conventional forms, efforts should be undertaken to differentiate between the different species so that the appropriate controls (e.g., based on applicable limiting values) may be applied. One approach is to count a survey or air filter sample in LSC cocktail. This measurement gives a value for ITP plus HTO content. Addition of fresh water to the cocktail followed by evaporation (thus removing HTO from the sample) and then recounting the sample, provides a value for ITP alone. Other approaches at differentiating between the species may include attempts at ITP dissolution, following by sample counting to detect changes in analysis results due to particle dissolution and subsequent increase in detected tritium atom disintegrations. The accuracy of this approach is dependent on knowledge of the specific species present, the fraction of tritium intra-particle disintegrations that may be detected, and the effectiveness of the dissolution technique for that species.

A useful characterization method, which could also be considered, is solubility testing: workplace survey samples tested for tritium released from particulates into simulated lung fluid. The results of this testing directly affect dose modeling via use of the critical parameter of ICRP solubility Type (F, M, S) for tritiated particulates in the lung.

In the absence of acceptable information with regard to differentiation of species, it may be necessary to base protective actions on conservative assumptions. Specifically, base protective actions on the most limiting values for dose conversion factors, airborne concentration and surface radioactivity levels associated with species likely to be present.

E.3.3 Individual and Workplace Controls for STC-Contaminated Areas

Following identification and quantification of STC contamination in an area, certain protective actions may be warranted and necessary to ensure compliance with DOE requirements for protection of individuals in or near the area. These protective actions and their regulatory basis include:

- Dose limits in accordance with 10 CFR Part 835, Subpart C;
- Area posting in accordance with 10 CFR Part 835, Subpart G;
- Material labeling in accordance with 10 CFR Part 835, Subpart G;

- Material controls in accordance with 10 CFR Part 835, Subpart L;
- Area control in accordance with 10 CFR Part 835, Subpart F;
- Individual monitoring in accordance with 10 CFR Part 835, Subpart E; and
- Area monitoring in accordance with 10 CFR Part 835, Subpart E.

Guidance needed to implement these requirements is contained in the following sections of this Appendix.

- Guidance on individual monitoring is in Chapter 4;
- Guidance on intake and dose assessment is in Chapter 5;
- Guidance on dose conversion factors for insoluble tritiated particles (Sv per unit of observed activity) Table 5-14 and Appendix A; and
- Guidance on workplace controls is in Chapter 6.

NOTE: Both individual and workplace monitoring may require air monitoring, such as breathing zone air monitoring, grab sampling, and continuous air monitoring.

E.4 – INDIVIDUAL MONITORING

As noted previously, 10 CFR Part 835 requires monitoring of certain individuals' internal and external doses. Except for rare situations, exposure to STCs will not result in a significant external dose; therefore, this appendix does not address external monitoring program requirements. STCs may contribute to an individual's internal dose. DOE has provided internal dose monitoring program guidance in Chapter 5 of DOE G 441.1-1C (DOE 2008), *Radiation Protection Programs Guide*, and detailed internal dose monitoring guidance in DOE-STD-1121-98 (DOE 2008)*, Internal Dosimetry Standard*. The guidance provided herein supplements that provided in those two standards and addresses issues that are specifically related to STCs.

The first step in determining an individual's internal dose as a result of a radioactive material intake is to determine the magnitude of the intake (in units of activity). Determinations of individual radioactive material intakes generally fall under three different methods: 1) in-vivo analyses, such as whole body or organ counting; 2) in-vitro analyses, using analyses of excreta (urine or fecal analyses); and 3) air monitoring, using results of airborne contamination monitoring programs. The actual method used depends on a number of factors, including the characteristics of the material to be analyzed, results of prior scientific analyses to develop applicable protocols, and the analytical equipment available. This chapter discusses evaluations of the need for individual monitoring and the applicability of these various methods to evaluating intakes of STCs.

E.4.1 Identifying Individuals to be monitored

Compliance with the internal dose monitoring requirements requires consideration of several factors. One of the first steps in implementing the internal dose monitoring program is identifying those individuals for whom internal dose monitoring will be required. This determination depends on the individual's classification (i.e., radiological worker, declared pregnant worker, occupationally-exposed minor, or member of the public) and likely dose. An estimate of an individual's likely dose requires knowledge of multiple factors, including:

- Areas to which the individual has access;
- Radiological conditions in those areas;
- Amount of time to be spent in affected areas; and
- Activities that the individual will be performing.

Consistent with Chapter 5 of DOE G 441.1-1C, *Radiation Protection Programs Guide for Use with Title 10, Code of Federal Regulations, Part 835,* Occupational Radiation Protection, this type of assessment is typically conducted on a work-group, rather than on an individual, basis. However, individual assessments may be required occasionally to address special or unique activities (e.g., declared pregnant workers, short term specialist assignments). The assessment should include all of the individual's internal exposures at the DOE activity, including those not related to STC exposure.

If estimates indicate that the individual's dose is unlikely to exceed the applicable mandatory monitoring threshold provided in 10 CFR 835.402, then no individual monitoring is required. However, the area monitoring program should provide sufficient data to support decisions regarding individual participation in the individual monitoring program. See Chapter 3 of this Appendix for guidance on the area monitoring program for STCs.

E.4.2 Air Monitoring for STCs

Should an individual monitoring program be required, a decision must be made regarding the relative merits of radiobioassay versus application of air monitoring results to determine individual doses. Generally, radiobioassay is the preferred monitoring method for assessment of individual internal dose. However, 10 CFR 835.209 allows the use of air monitoring data under certain conditions, including when air monitoring data will provide more reliable and accurate results. In many cases (as discussed in the remainder of this chapter), this condition may exist with exposure to certain types of STCs. For more information on development and implementation of an air monitoring program for purposes of individual dose assessment, see Chapter 10 of DOE G 441.1-1C, *Radiation Protection Programs Guide for Use with Title 10, Code of Federal Regulations, Part 835,* Occupational Radiation Protection, and DOE-STD-1121-98 (DOE 2008), *Internal Dosimetry*.

Historically, insufficient information has been available to facilitate development and implementation of effective radiobioassay programs for certain insoluble STCs with long biological retention periods. However, there has been a significant amount of recent research into such programs, the practical results of which are summarized in this Standard.

E.4.2.1 Air Monitoring - Area Monitoring vs. Individual Monitoring

It is important to differentiate between the purposes and scope of the various facets of the air monitoring program. Monitoring of the air in any specified area is usually performed using portable air samplers, fixed head air samplers, or continuous air monitors. As a facet of the area monitoring program, such monitoring is subject to the regulatory requirements found in 10 CFR 835.401 and 835.403. Air monitoring for the purposes of individual dose determination is usually performed using a lapel (personal) sampler worn by each affected individual. Because a lapel sampler is worn by, and monitors the breathing zone of a specified individual, the results of that sample analysis are applicable to that individual. If such monitoring meets the conditions of 10 CFR 835.209(b) it may be used to monitor individuals for exposure to internal radiation. The results of such monitoring may be used to meet the regulatory requirements in 10 CFR 835.401, 835.402(c), and 835.403. Because air sampling is distinct from bioassay, the requirements in 10 CFR 835.402(d) for performance accreditation of radiobioassay analysis does not apply to air sampling programs.

Note however, that the results of area monitoring (using portable, fixed head, or continuous air monitors) may be used for individual dose assessment under certain conditions (given that assurance can be provided that the sample is representative of the air actually breathed by the affected individual(s)). See section 5.2.4.1c for guidance on calculating dose from air sampling measurements.

E.4.3 Radiobioassay

E.4.3.1 Urinalysis

Soluble materials ingested or inhaled into the body will be excreted via urine. Given a known concentration of the contaminant in the urine and knowledge of the time of exposure and rate of dissolution of the material in body fluids, one can derive a total intake quantity. While determinations of the exposure time and urine concentration may be relatively straightforward for STCs, determination of dissolution rates is more difficult. Chapter E.5 discusses the use of urinalysis in performing internal dose assessments for intakes of STCs.

E.4.3.2 Fecal Analysis

Insoluble materials that are inhaled into the body will be excreted via feces. Retention factors and total intake quantity can be calculated based on the models described in Section E.5.1, given a known concentration of the contaminant in the feces and knowledge of the time of exposure, radioactive decay rate and solubility. Determination of exposure time is straight forward, but no reliable protocol currently exists for determining concentrations of STCs in feces[91].

E.4.3.3 In-Vivo Analyses

In-vivo analyses depend on the detection of radiation emitted by radioactive materials within the body by radiation detectors external to the body. In-vivo analysis is not considered to be a viable means of assessing tritium intakes, including STC intakes. The are two reasons for this: 1) Tritium decays by emission of low energy beta particles that will not penetrate through the body to the external detector, and 2) A significant number of bremastrahlung x-rays resulting from the interaction of the tritium beta with the metal atoms of the host particle are not expected to escape to an external detector unless the STC uptake is massive.

E.4.4 Organically Bound Tritium (OBT)

E.4.4.1 Soluble OBT

Soluble OBT migrates through the skin or lung into the bloodstream by the physical processes of dissolution and diffusion. The two processes are inseparably linked and often simply called "absorption.” Soluble OBT is also readily absorbed through the GI tract following ingestion. Rapid dispersion minimizes organ-specific (e.g., lung) differential doses. Following absorption into the body, soluble OBT is excreted via urine. A biokinetic model is available which relates intakes of soluble OBT to urine excretion rates, and a dose conversion factor is available for soluble OBT intakes. Urine bioassay is therefore considered to be a viable approach to estimating intake and dose from soluble OBT. More information on dose assessment for soluble OBT is found in Section E.5.2.5.1.

Solvents such as octane, cyclohexane, or acetone have been used in pump cleaning and material dispersion processes. They can become tritium-labeled from extended exposure to tritiated materials. Since these solvents are volatile, they are considered soluble. Individual intake of tritiated solvents can occur through inhalation, ingestion, or by diffusion through the skin. Evaluation of solvent OBT intakes should be performed via urine bioassay.

The potential hazard of the skin absorption pathway should be assessed using known or analyzed activity of the liquid. Volatile OBT also produces vapors that can become dispersed in air, resulting in a second potential intake pathway of vapor inhalation and skin absorption. Therefore, a derived air concentration (DAC) will be derived in the next section to relate volatile OBT air concentration to

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[91] For intakes expected to result in doses greater than 2 rem CED, one may want to collect and store fecal samples pending development of a fecal biokinetic model and assay technique.

the annual E_{50}. Volatile OBT vapors are readily detected by the tritium gas air monitoring method (e.g. ion chamber).

If a solid particle OBT that has been determined to be soluble in lung fluids is discovered, urine bioassay should be used to determine internal dose.

E.4.4.2 Insoluble OBT

For insoluble OBT, absorption through the skin is not an intake pathway. Following ingestion (or inhalation), some fraction of insoluble OBT will likely be digested in the stomach and converted to HTO or soluble OBT. Since HTO and soluble OBT are rapidly assimilated physiologically, these dose components are readily assessed via urine bioassay. In addition however, following inhalation, insoluble OBT may reside in the lung for a period of time, delivering dose to the lung. This material is largely formed by incidental contamination of environmental dust and is found in many tritium contamination areas. Solid particulate OBT is considered primarily insoluble, and intake by skin absorption is not expected, but inhalation is a possible intake mechanism. Inhalation of this material, including determination of intake through air monitoring, assignment of doses, and use of DCFs and DACs, should be treated the same as inhalation of ITPs. Because monitoring and intake/dose assessments are the same for solid particulate OBT and for general ITP, these two materials do not need to be distinguished during workplace characterizations when the dose conversion assumptions are applied.

E.4.4.3 OBT Oils

Trivedi (1995) has shown that tritiated oils can be absorbed by mammalian skin. In this section skin absorption and inhalation are both discussed as valid intake pathways for various components of tritiated oils. Tritiated oils can be found in pumps and compressors located in tritium areas. Tritiated pump oils are considered to be a mixture of three components: insoluble large molecule OBT (original molecules of oil labeled by tritium), soluble small molecule OBT (created by radiolytic and degradation processes and labeled by tritium), and HTO. As a reasonable hypothesis, the tritium in OBT oils should be expected to partition into the above three components in approximately an 80:10:10 ratio, respectively. This is consistent with a ~70:20:10 ratio of tritium components removable from oily surfaces (Johnson, 1988), where all three components were observed. The small molecule OBT is expected to be more concentrated on surfaces than in bulk oil due to absorption of small polar molecules created from degradation processes. Data on liquid tritiated oil (Trivedi, 1995) indicates that the 80:10:10 assumption should be conservative.

The latter two of the three expected OBT oil components can in fact diffuse through the skin. The HTO component can migrate quickly through the skin into body water and intake should be assessed via urine bioassay. Small molecule OBT can also migrate through skin and is considered soluble; intake of this component should also be assessed via urine bioassay. Estimation of potential absorption intakes of HTO and soluble OBT resulting from direct skin contact with liquid oil should be performed via analysis of oil samples for total tritium, and application of the above 80:10:10 ratio. Inhalation intakes of vapors liberated from volatile HTO and soluble OBT components of oil can also occur. Estimation of potential vapor intakes should be performed via monitoring of air above the oil by ion chamber.

Large molecule OBT is considered to be insoluble. Estimates of diffusivity of large organic molecules such as pump oils in polymer films, extrapolated from data for chemically similar but smaller molecules (Rogers, 1985), suggest that oils migrate extremely slowly (if at all) into and though polymeric materials. For this reason, the large molecule OBT component of oils is estimated not to absorb into (or through) skin and therefore not to impart skin (or other tissue) dose via this pathway.

The large molecule component of oil is not volatile. However, this oil component can be dispersed into air from the either operation of equipment or the sudden release of residual pressure, and therefore can be inhaled as a mist (and deliver a lung dose). Tritium from the insoluble large oil molecules does not dissolve into the body, due to the stability of the carbon-tritium bond in oils. Oil absorption in lung tissues is low, and clearance rate of oil from lung could be as slow as that of particulates. Therefore, an oil mist intake should be treated like an intake of solid particulate OBT (i.e., as an intake of ITP). Oil mist inhalation should be monitored by air sampling and assessed as ITP inhalation.

Estimation of potential inhalation intakes of OBT oil mists should be performed via analysis of oil samples for total tritium, and application of the above 80:10:10 ratio. Because OBT oils are expected to be a mixture of HTO, soluble OBT, and insoluble OBT as discussed above, radiological protection planning for activities involving OBT oils should consider potential intakes of, and air monitoring for, all three components. To avoid mist formation, energetic activities involving unencapsulated OBT oil should be discouraged.

E.5 – INTAKE AND DOSE ASSESSMENT

E.5.1 Biokinetic and Dosimetric Models

A biokinetic model describes in quantitative terms the retention and transport of a material in the body. A dosimetric model 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 biokinetic model used in ICRP 78 is recommended to evaluate intakes of STC. The dosimetric models for STC are described primarily in ICRP 67 and 71. The STC biokinetic and dosimetric models will be briefly reviewed in this section and a detailed benchmark calculation is provided in Appendix B. The relevant ICRP publications should be consulted for a more detailed discussion of the models and their application.

The STC biokinetic model may be broken down into three major components:

- The respiratory tract model from ICRP 66;
- The systemic retention and excretion model from ICRP 67 and 78; and
- The gastrointestinal (GI) tract model from ICRP 30 and 78.

An example of how these models are coupled is shown in Figure 5-1. Inhaled aerosols of insoluble metal tritides (IMTs) that are initially deposited in the respiratory tract are:

- mechanically cleared from the body (e.g., by blowing the nose);
- dissolved and absorbed into the bloodstream; or
- mechanically cleared to the GI tract

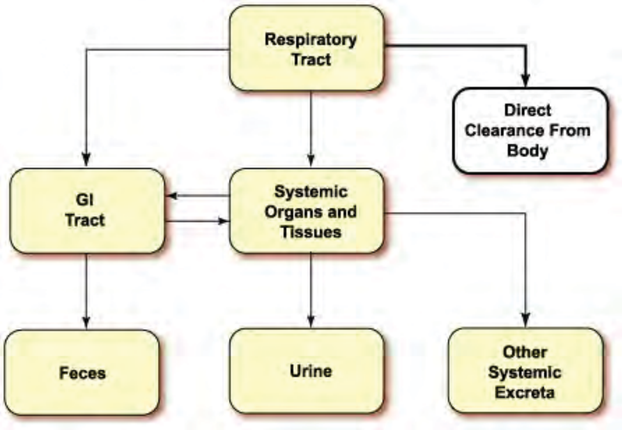

Figure E.5-1. Simplified Overview of How the Major Components of the ICRP 78 Biokinetic Model for Insoluble Metal Tritides are connected

Note: “Other systemic excreta” refers to ICRP Publication 23 and includes insensible losses, breath, sweat, diffusion, and transpiration.

Tritium absorbed from the respiratory and GI tracts is assumed to be in the form of tritiated water (HTO) and to behave exactly like HTO that was absorbed directly into the bloodstream. The systemic biokinetic model is discussed in more detail in Section 5.1.1.

Material cleared to the GI tract may be dissolved and absorbed into the bloodstream or cleared to the feces. The ICRP 30 GI tract biokinetic model is used to quantify these processes, but some modifications have been made to the dosimetric model. More details on the biokinetic and dosimetric models are provided in section E.5.1.2.

E.5.1.1 The Systemic Biokinetic Model

The systemic biokinetic model and relevant portions of the GI tract model are shown in Figure E.5-2. STC particles that dissolve in the respiratory tract release tritium that is converted to HTO and absorbed into the bloodstream. STC particles that dissolve in the GI tract are also converted into HTO and absorbed into the bloodstream from the small intestines (SI). The fraction of STC that enters the GI tract and is subsequently absorbed from the SI is referred to as f_1.

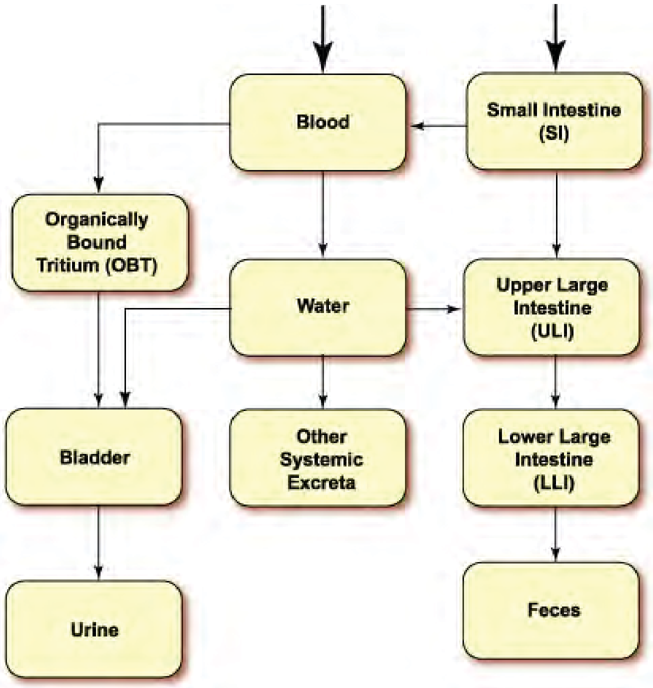

Figure E.5- 2. Systemic Biokinetic Model for Tritiated Water Coupled to the GI Tract Model

Of the HTO that is absorbed into the bloodstream, 3% is converted to organically bound tritium (OBT) that is retained in the body with a 40-day half-life and the remaining 97% remains as HTO and is retained with a 10-day half-life. All of the OBT is excreted into the urinary bladder and ultimately the urine. HTO is excreted from the systemic compartment in the following ratios:

- 1.4/3 fraction to the urinary bladder and ultimately the urine;
- 0.1/3 fraction to ULI and ultimately the feces; and
- 1.5/3 fraction to other systemic excreta (e.g., “insensible losses” and sweat).

These ratios are derived from the ICRP 23 (ICRP 1975) Reference Man water balance.

E.5.1.2 Urinary and Fecal Excretion Functions

Urinary and fecal excretions are expressed as 24-hour incremental quantities. For example, the 24-hour incremental urinary excretion ΔEu(t) of tritium (not accounting for radioactive decay) at t days after intake is given by

$$\Delta E_u(t) = Q_u(t) - Q_u(t-1) \qquad \text{(Eq. E.5-1)}$$

where $Q_u(t)$ is the content of the urine compartment at time t and $Q_u(t-1)$ is the content of the urine compartment one day earlier. The 24-hour incremental urinary excretion of tritium that does not account for radioactive decay is adjusted for radioactive decay to give the incremental excretion of tritium $\Delta e_u(t)$

$$\Delta eu(t) = \Delta Eu(t)e\text{-}\lambda t \qquad \text{(Eq. E.5-2)}$$

Urinary excretion of tritium following an intake of HTO is typically expressed as the concentration of radioactivity per unit volume of urine such as μCi/L, based on the assumption that the concentration of HTO is the same in all body fluids. The presence of the OBT compartment in the ICRP 78 model precludes this approach and warrants the use of 24-hour incremental urinary excretion functions. Incremental fecal excretion is calculated in an analogous fashion. A detailed example of how to calculate urinary and fecal excretion is given in Appendix B.

E.5.1.3 The Respiratory Tract Biokinetic Model

The ICRP respiratory tract model is rather complex and is discussed in great detail in ICRP Publication 66. A brief overview is provided here as an introduction to the model. The respiratory tract model is composed of five major components:

- anatomical model;
- deposition model;
- particle mechanical clearance model;
- particle dissolution model; and
- dosimetric model

The respiratory tract is divided into the thoracic and extrathoracic (ET) regions, which are divided further into compartments that are used in the deposition and clearance models. These compartments are listed in Table E.5-1. Comparing the regions of the ICRP 66 and ICRP 30 respiratory tract models, the ET region corresponds to the nasal passage (NP) region, the BB and bb regions correspond to the trachea and bronchial tree (TB) region, and the AI region corresponds to the pulmonary parenchyma (P) region.

The deposition model describes quantitatively where inhaled particles will be deposited in the different regions of the respiratory tract. The sites of initial particle deposition are all the compartments listed above except for the thoracic and extra-thoracic lymph nodes. Although it is possible to calculate deposition fractions from first principles, in practice we usually obtain the fractions from published tables like Table E.5-2 below. These deposition fractions for occupational exposure were taken from the ICRP-CD[92].

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[92] The ICRP Database of Dose Coefficients for Workers and Members of the General Public, Version 1.0 (Pergamon Press: New York) 1998.

Table E.5-1: Compartments of the Respiratory Tract Model

Extrathoracic

ET_1	Anterior nose
ET_2	Posterior nose, larynx, pharynx, mouth
ET_{seq}	sequestered particles
LN_{et}	extrathoracic lymph nodes

Thoracic

- BB - bronchial region
  - BB_1 - fast clearing compartment
  - BB_2 - slow clearing compartment
  - BB_{seq} - sequestered particles
- bb - bronchiolar region
  - bb_1 - fast clearing compartment
  - bb_2 - slow clearing compartment
  - bb_{seq} - sequestered particles
- AI - alveolar-interstitial region
  - AI_1- fast clearing compartment
  - AI_2 - moderate clearing compartment
  - AI_3 - slow clearing compartment
- LN_{th} - thoracic lymph nodes

Table E.5-2: Deposition Tractions for Different Compartments of the Respiratory Tract as a Function of Particle AMAD

	AMAD (:m)									
	0.001	0.003	0.01	0.03	0.1	0.3	1	3	5	10
AI1	1.359E-04	6.714E-03	8.229E-02	1.525E-01	8.688E-02	4.458E-02	3.198E-02	2.314E-02	1.596E-02	7.104E-03
AI2	2.718E-04	1.343E-02	1.646E-01	3.051E-01	1.738E-01	8.916E-02	6.396E-02	4.627E-02	3.191E-02	1.421E-02
AI3	4.530E-05	2.238E-03	2.743E-02	5.085E-02	2.896E-02	1.486E-02	1.066E-02	7.712E-03	5.319E-03	2.368E-03
bbf	2.262E-02	8.770E-02	1.291E-01	7.188E-02	3.328E-02	1.523E-02	8.327E-03	7.580E-03	6.569E-03	4.131E-03
bbs	2.294E-02	8.895E-02	1.309E-01	7.290E-02	3.376E-02	1.544E-02	8.087E-03	5.954E-03	4.384E-03	2.099E-03
bbseq	3.212E-04	1.245E-03	1.833E-03	1.021E-03	4.726E-04	2.162E-04	1.157E-04	9.541E-05	7.721E-05	4.392E-05
BBf	3.117E-02	4.420E-02	2.685E-02	1.035E-02	4.746E-03	3.260E-03	6.489E-03	1.127E-02	1.171E-02	9.436E-03
BBs	3.161E-02	4.483E-02	2.723E-02	1.049E-02	4.813E-03	3.293E-03	5.844E-03	7.290E-03	5.921E-03	3.116E-03
BBseq	4.425E-04	6.276E-04	3.812E-04	1.469E-04	6.738E-05	4.619E-05	8.694E-05	1.308E-04	1.243E-04	8.848E-05
ET2	4.390E-01	3.494E-01	1.735E-01	6.994E-02	3.218E-02	5.820E-02	2.111E-01	3.699E-01	3.989E-01	3.836E-01
ETseq	2.196E-04	1.748E-04	8.680E-05	3.499E-05	1.610E-05	2.912E-05	1.056E-04	1.850E-04	1.996E-04	1.919E-04
ET1	4.433E-01	3.327E-01	1.539E-01	6.248E-02	3.071E-02	5.217E-02	1.652E-01	3.001E-01	3.385E-01	3.471E-01

Note that an “f” in Table E.5-2 corresponds to a “1” or “fast clearance compartment” in Table E.5-1 and an “s” correspond to a “2” or “slow clearance compartment.” For example, bbf in Table E.5-2 is the same as bb1 in Table E.5-1.

The model for mechanical clearance of particles from the lung is shown in Figure E.5-3. While particles are being mechanically cleared, they are also dissolving and being absorbed into the bloodstream. Mechanical clearance and dissolution are therefore competitive processes.

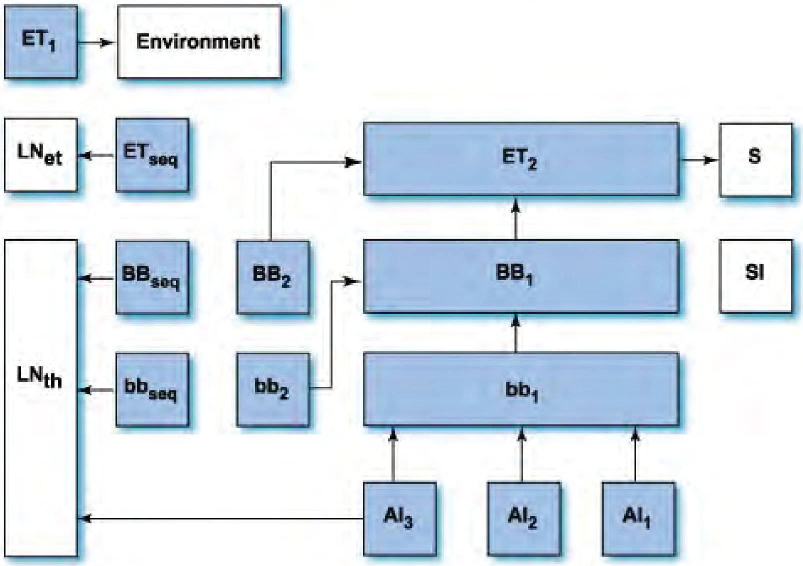

Figure E.5-3: Deposition of Aerosols in the Respiratory Tract and Their Subsequent Mechanical Clearance

NOTE: Shaded compartments are sites of initial deposition.

According to the model, the rate of mechanical clearance is dependent on where the particle is deposited, but is independent of the chemical form of the particle. On the other hand, the dissolution rate is dependent on the chemical form of the particle, but independent of where the particle is in the respiratory tract[93]. The dissolution model is shown in Figure E.5-4. To account for time-dependent dissolution a two-stage dissolution model is used. In this model, particles either dissolve directly (and are absorbed) or are transformed to an intermediate form before they dissolve.

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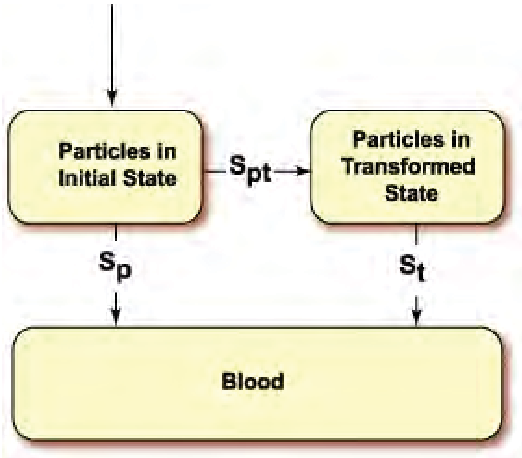
[93] Except for the ET1, where no dissolution or absorption is assumed to occur.

Figure E.5-4: Two-stage Dissolution of STC Particles in the Respiratory Tract



Default parameters are supplied in ICRP 66 for the mechanical clearance and dissolution models. The dissolution rate of particles may be classified as Type F (fast), M (moderate), or S (slow). These classes correlate roughly with the Class D/W/Y classes in the ICRP 30 respiratory tract model. However, note that D/W/Y refers to total clearance rates (mechanical plus dissolution) whereas F/M/S refers to dissolution rates only. The parameters for the time-dependent dissolution model can be experimentally determined by measuring the dissolution rate in-vitro over time, which is usually represented with two exponents[94]:

$$r(t) = f_r \cdot e^{-s_r t} + (1 - f_r) \cdot e^{-s_s t} \qquad \text{(Eq. E.5-3)}$$

where

- $r(t)$ is the fraction of material not dissolved at time t;
- f_r is the fraction that dissolves rapidly with rate constant s_r; and
- s_s is the rate constant for the fraction $(1 - f_r)$ that dissolves slowly.

These parameters may then be converted to the s_p, s_{pt}, and s_t in the dissolution model (Figure E.5-4).

E.5.1.4 The GI Tract Biokinetic Model

The GI tract biokinetic model, which has already been discussed to some extent in Section E.5.1.1, is essentially the same model used in ICRP 30. STC particles enter the GI tract though the stomach via clearance from the lungs and oral ingestion. The f1 is assumed to be 0.1 for Type M) STC particles and 0.01 for absorption type S STC particles. The tritium released from STC particles is absorbed

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[94] See ICRP 71, Annexe D, *Assignment of Compounds to Absorption Types from Experimental Data*.

into the bloodstream from the small intestines as HTO. Ingestion of HTO is treated as a direct injection into the bloodstream, bypassing the GI tract. A fraction of the HTO excreted from the body leaves via the feces. As mentioned before, this fraction is (0.1)/3 and it is assumed to enter the GI tract through the upper large intestine.

E.5.1.5 Dosimetric Models

Ultimately, we would like to calculate the equivalent dose to the organs and tissues listed in the first column of Table E.5-3. These doses are then weighted with the appropriate tissue weighting factor from the second column and summed to give effective dose H_e:

$$H_e = \sum_{i=gonads}^{remainder} w_i \, H_i \qquad \text{(Eq. E.5-4)}$$

Figure E.5-5: GI Tract Biokinetic Model

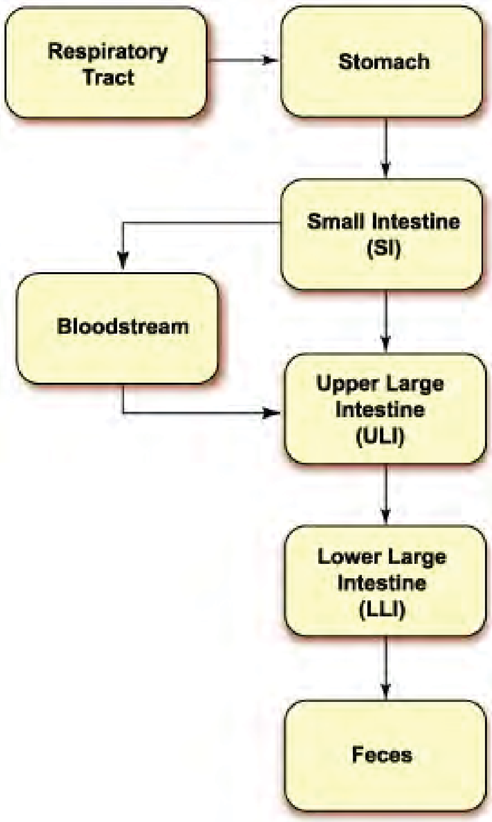

Table E.5-3: Organs and Tissues with Explicit ICRP 60 Tissue Weighting Factors

Tissue or Organ	Tissue Weighting Factor
gonads	0.20
bone marrow	0.12
colon	0.12
lung	0.12
stomach	0.12
bladder	0.05
breast	0.05
liver	0.05
esophagus	0.05
thyroid	0.05
skin	0.01
bone surface	0.01
remainder	0.05

The first step in calculating He is to calculate the individual organ doses. To do this we have to know:

- The source organs[95];
- The number of decays that take place in the source organs;
- The target organs[96]; and
- The fraction of energy released in the source organs that are absorbed in a target organ.

The source organs for STC are listed in Table E.5-4.

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[95] The organs in which tritium will decay and release radiation.

[96] The organs that absorb the radiation emitted by the source organs

Table E.5-4: Dosimetric source organs for intakes of STC

ET1sur
ET2sur
ET2seq
LNet
BBgel
BBsol
BBseq
bbsol
bbgel
bbseq
AI
LNth
UBcont
SI
ULI
LLI
tissue

The abbreviation "sur" refers to surface, "seq" refers to sequestered, "gel" refers to material in the fast-clearing mucus on top of the cilia, and "sol" refers to material in the slow-clearing solution between cilia. The source organ "tissue" is the soft tissues of the body.

A source organ may be associated with more than one biokinetic compartment. For example, the source organ ET1sur is composed of only the ET1 biokinetic compartment, whereas the ET2sur source organ is composed of the ET2 and transformed ET2 (TET2) biokinetic compartments. The "tissue" source organ is composed of the OBT, HTO, and blood biokinetic compartments (see Appendix B). The decays U that take place in the source organs are calculated by integrating the retention function for the biokinetic compartments that compose the source organ.

Once the source organs and their associated biokinetic compartments are specified, the target organs are specified. The target organs for STC are listed in Table E.5-5.

Table E.5-5: Dosimetric Target Organs

Adrenals
UB wall
BS
brain
breasts
S wall
SI wall
ULI wall
LLI wall
kidneys
liver
ET1bas
ET2bas
LNet
BBbas
BBsec
bbsec
AI
LNth
muscle
ovaries
pancreas
RM
skin
spleen
testes
thymus
thyroid
uterus

In this table “bas” refers to basal cells, “sec” to secretory cells, “UB” to urinary bladder, and BS to bone surface.

The specific effective energy (SEE) describes the dose delivered to a specific target organ *T* for each decay in a specific source organ *S*. The SEEs are calculated with the computer code SEECAL[97]. The SEEs calculated with SEECAL do not account for self-absorption of the tritium beta radiation in the tritide particle. If corrections for self-absorption are made, care should be taken not to adjust the dose received from HTO. The dose H_T to target organ *T* from the decays in source organ *S* is given by

$$H_T = U_S \, SEE(T \leftarrow S) \qquad \text{(Eq. E.5-5)}$$

The total dose to the target organ is the sum of the dose from all source organs.

$$H_T = \sum_i U_{S_i} \, SEE(T \leftarrow S_i) \qquad \text{(Eq. E.5-6)}$$

The doses to some target organs are combined in a prescribed fashion to calculate the dose to an organ composed of two or more other organs or tissues. For example, 57% of the dose to the upper large intestine is added to 43% of the dose to the lower large intestine to give the dose to the colon. The tissues of the respiratory tract are another example of this procedure. Two organs in Table E.5-3 deserve special note. First, the dose to the thymus is assumed to be the same as the dose to the esophagus. Second, the dose to the remainder is the mass weighted mean of the doses to the *i* organs not listed in Table E.5-3.

$$H_{remainder} = \frac{\sum_i H_i \, m_i}{\sum_i m_i} \qquad \text{(Eq. E.5-7)}$$

E.5.2 Internal Dose Determination

The first step in determining an individual’s dose resulting from a radionuclide intake is the intake assessment (i.e., determining the amount of radioactive material present in the body). Intake assessments for tritium exposure usually rely on radiobioassay. Common radiobiossay techniques for tritium (urinalysis) are rendered ineffective for some forms of STC intakes (especially insoluble particulates) because of the difficulties associated with relating the results of these analyses to specific intake levels. Therefore, representative air monitoring results are often used for assessment of doses resulting from exposure to particulate STCs. Variations in individual biological characteristics and statistical uncertainty associated with any measurement make dose assessment

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[97] M. Cristy and K. F. Eckerman, *SEECAL: Program to Calculate Age-Dependent Specific Effective Energies*, ORNL/TM-12351 (1993).

a process that should be approached with reasonable assumptions and documentation that support the calculation methodology.

Personal intake and dose assessments can be based on data from representative air monitoring using any appropriate technique, most commonly lapel sampling. Fixed air sample heads and portable air samplers may be used for individual dose assessment if one can ensure the sample is representative of the air inhaled. However, they are primarily used to verify the adequacy of radiological controls and postings and to document radiological conditions in the area of interest. See DOE-STD-1121-2008, *Internal Dosimetry*.

E.5.2.1 Intake Determination Methodology for Tritiated Particulates

Determinations of individual radioactive material intakes generally fall under three different methods: 1) in-vivo analyses, such as whole body or organ counting; 2) in-vitro analyses, using analyses of excreta (urine or fecal analyses); and 3) area monitoring, using results of area surface and air contamination monitoring programs. The actual method used depends on a number of factors, including the characteristics of the material to be analyzed, results of prior scientific analyses to develop applicable protocols, and the equipment available. This section discusses the applicability of these various methods to evaluating intakes of particulate STCs.

E.5.2.1.1 Air Monitoring

Air monitoring can be used to estimate intake directly, as opposed to indirect bioassay methods. Intake is considered to be proportional to the actual activity captured on the filter of an air-sampling device (assuming that the sample is representative of the breathing air for the individual in question). Particulates tend to shield their tritium beta activity by self-absorption of the beta radiation within the mass of the particle. Observed activity on a filter sample, measured by suspending particulates from the filter into liquid scintillation counting (LSC) solution, therefore under-represents the actual activity available for deposition to the lung, i.e., intake. Self-absorption factors (SAFs) vary, as a function of respirable (<10 μm AMAD) particulate size and material, by a factor of approximately 10. However, when tritiated particulate intake is defined in terms of observed activity (and when DCF is correspondingly defined in terms of observed activity intake), the uncertainty in the observed intake essentially disappears, since self-absorption is accounted for.

If large non-respirable tritiated particulates are captured on an air filter, intake amounts are somewhat overestimated. However, non-respirable particulates are not efficiently suspended in breathable air and are therefore discriminated against during air monitoring. Also, some dissolution can occur in the scintillation cocktail, which measures the activity captured on a filter. Dissolved tritium is not subject to self-absorption. Intake calculations are therefore overestimated if all activity observed by LSC is taken as “insoluble” and treated as absorption type S particulate. Since SAFs vary by a factor of ~10, dissolution in cocktail will not increase intake computations by more than a factor of ~10. Either non-respirable particulate collection or dissolution in cocktail will act to provide overestimation of intake (and ultimately dose) from air monitoring results.

Intake (and dose rate) in this case may be expressed as follows:

$$D(t) \propto I_O + \Delta I_O \qquad \text{(Eq. E.5-8)}$$

where: D(t) = dose rate to lung from observed intake at time t
I_o = observed intake to lung from air monitoring results at time zero

ΔI_o = incremental observed activity from particle dissolution or non-respirable particulates at time zero.

Shortfalls

The main shortfalls associated with this methodology for intake (and dose) estimation are: 1) actual activity intakes are uncertain by a factor of 10 because of self-absorption. 2) Activity collected and measured on filters may over-represent the intake because of capture of non-respirable particulates and dissolution in scintillation cocktail.

As previously discussed in this appendix, intakes in terms of actual activity do not need to be determined. Intakes of observed activity are adequate to provide dose estimates as long as dose conversion factors are available which are described in terms of observed activity. In Section 5.2.3, self-absorption is investigated in detail. The section shows that actual activity of an intake is underrepresented by the observed results from scintillation counting of an air filter sample, but that the actual activity is correspondingly not fully available to impart dose to lung tissue. In fact, the observed activity more nearly represents dose or dose potential, since only that beta radiation that is not self-absorbed and escapes the particulate (i.e., that beta radiation "observed" via LSC analysis) is available to impart dose.

In Section 5.2.4, the ICRP 66 lung model is used to tabulate dose conversion factors for a given actual intake of tritiated particulate of various materials and various particle size distributions, assuming an ICRP 78 biokinetic model and absorption type S (slow). ICRP 66 dose, which does not account for self-absorption, is corrected by a "self-absorption factor for energy" (SAFe) for the various particulates to determine the actual dose from the actual intake. SAFe is the fraction of the total tritium beta energy generated within a particulate which is released from that particulate. The tritium beta activity expected to be observed in LSC analysis from the actual intake is then determined using the "self-absorption factor for beta particles" (SAFβ) for the various particulates. The result of these calculations is a table of actual dose conversion factors for intakes, based on observed data, for a variety of materials and particulate sizes. Because actual activity is not needed to assign dose from air sampling data, uncertainty in actual activity (because of uncertainty in particle size and hence SAFs) is not a shortfall to the air monitoring method for dose estimation. In addition, uncertainty in observed activity is only a function of the LSC assay technique, which is expected to be small and comparable to the uncertainty when LSC is used to analyze in urine or fecal bioassay samples.

The remaining expected shortfall is an overrepresentation of tritiated particulate intake because of capture of non-respirable particulates or dissolution in scintillation cocktail. These factors are not expected to increase dose calculations significantly. Non-respirable particulates will tend to be less available to sampling than respirable particulates, due to rapid settling. Dissolution in scintillation cocktail would have to occur within the few-hour period prior to analysis during which samples are immersed in the cocktail. These factors would serve to exaggerate intake estimates. The extent of this overestimation is the only significant uncertainty in the intake determination from air sampling methodology.

Another uncertainty occurs in the air sampling method which affects all methods. For a given intake, an expected dissolution rate must be chosen for the determination of the DCF. If absorption type S is chosen, when in fact captured particulates on an air-sampling filter more rapidly dissolve in the lung, this constitutes dose overestimation. Dose can be overestimated by a factor of 50 for this reason. The typical overestimate is expected to be a factor of less than 10, because the fastest dissolving particulates lose tritium to air during sampling and prior to analysis.

For the case of materials with extremely slow dissolution rates, such as hafnium tritide, the assumption of absorption type S is reasonable (Cheng 1999a), and would not lead to significantly underestimated doses. For absorption type S and Type "Super S" materials, mechanical clearance is the predominant removal mechanism, as opposed to dissolution. ("Super S" is a term used to describe any material where the absorption rate for tritium is slower than that of absorption type S.) Section E.5.2.2 shows that dose conversion factors for Hf, as derived from absorption type S and "Super S" input data, differ by only a few percent.

Detectable intake (and dose) is small in the case of air sampling. Section 5.2.2 determines a derived air concentration (DAC) value for absorption type S tritiated particulate which, when inhaled for one hour, imparts an E_{50} of 2.5×10^{-5} Sv (2.5 mrem); this value is shown to be 4.8×10^4 Bq observed/m^3. Doses less than one millirem (10 µSv) are measurable and can be readily assigned in the case of air monitoring. Sensitivity is not a shortfall, but is in fact an advantage to air monitoring.

An additional shortfall for analyzing air samples using LSC is self-absorption of tritium beta radiation due to dust loading. An effective compensatory measure for this shortfall is the use of detergent in preparing air samples to remove dust from samples and suspend it in the counting solution.

The ability to obtain air samples that are representative of the monitored individual's breathing zone is another potential shortfall of air monitoring. However, according to U. S. Nuclear Regulatory Commission Regulatory Guide 8.25, samplers located within about 1 foot of the worker's head may be accepted as representative without further determination. Therefore, when lapel breathing zone air samplers are properly used, representative air sampling is not considered a shortfall.

E.5.2.1.2 Urinalysis

Note: *This section is reserved pending development of a relevant protocol* (nothing on horizon).

E.5.2.1.3 Fecal Analysis

Note: *This section is reserved pending development of a relevant protocol* (nothing on horizon).

E.5.2.1.4 In-Vivo Analyses

As noted in section E.4.3.3, in-vivo analyses depend on the detection of radiation emitted by radioactive materials within the body by radiation detectors external to the body. Because tritium decays by emission of low energy beta particles that will not penetrate through the body to the external detector, in-vivo analysis is not considered to be a viable means of assessing tritium intakes, including STC intakes.

E.5.2.2 Dose Conversion Factor for Insoluble Tritiated Particulates

Following the intake assessment, the intake amount may be converted to internal dose using a DCF. Determination of internal dose from a tritiated particulate is mainly dependent on two factors: particle size distribution, and dissolution rates of tritium from the particulate into lung fluid. The impact of these factors on DCF or intake determination, as derived from either bioassay or air concentration data, is discussed below.

E.5.2.2.1 Particle Size Distribution

For any given exposure scenario, the particle size distribution will affect both the dissolution rates (discussed below) and the deposition fractions (i.e., the fractions of the material that are deposited into each compartment of the lung). Smaller particle sizes tend to deposit deeper into the lung, causing a greater dose than particles that are physically larger, but otherwise similar (e.g., chemical characteristics, quantity of infused tritium). Uncertainty in the particle size distribution causes uncertainty in the DCF, regardless of the method used (e.g., bioassay or air monitoring) to determine activity intake. Therefore, the particle size distribution factor cannot be used to choose a preferred data collection method (urine bioassay vs. fecal bioassay vs. air concentration values).

Fortunately, errors in determining the particle size distribution tend to result in self-canceling errors in the dose assessment. If particle sizes are smaller than assumed, the particles will deposit deeper in the lungs, resulting in the use of a larger DCF and calculation of a larger dose. However, because the dissolution rate for the smaller particles will be higher than assumed, intake will be overestimated. Because intake is overestimated while DCF is under-estimated, the effects of particle size uncertainty on assigned dose tend to cancel. These effects are explored later in this chapter.

Deposition fractions affect clearance rates of particulates from the lung to the gastrointestinal (GI) tract. Since urine excretion of tritium dissolved from tritiated particulate over time is a function of those particles which have not cleared from the lung, interpretation of urine excretion curves can extract particle size information. This requires extended monitoring without additional tritium intakes, which can adversely affect project management activities. To estimate intakes (and therefore doses) accurately and promptly from urine data, detailed knowledge of the particulate dissolution rates should be available. Tritiated material and particle size distribution involved in the intake, and the expected dissolution rate for that material and size, must be known. When tritiated materials and sizes are likely to range broadly, it will be difficult to provide material identification and size characterizations in the workplace. Moreover, although dissolution rate data are available for several materials, these data do not include all the possible combinations of materials and particle sizes that might be encountered.

E.5.2.2.2 Physical Diameter versus AMAD for Particulates

The ICRP and most technical papers that address respirability of particulate contamination use the term “activity median aerodynamic diameter” or “AMAD” when referring to particle size distributions. The AMAD can be very different from the physical diameter of a median particle, as it is a function of the particle’s density and shape. The relative deposition of particles in the respiratory tract is dependent on the AMAD. Particles with small AMADs tend to deposit deeper (higher percentage in the alveolar interstitial region) in the lung than larger AMAD particles; because of the slow clearance rates of the deep lung, smaller particles generally cause higher lung doses than larger ones. For particles with AMADs greater than about 10 μm, the fractional deposition in the deep respiratory region is considered negligible. The larger sized particles tend to be deposited in the upper respiratory region where they are rapidly removed to the gut and eliminated in feces. The lung dose per unit intake is less for larger AMAD particles than for smaller AMAD particles. Table E.5-6 shows a comparison of physical diameter versus AMAD (calculated per Cheng 1999a) for a variety of materials from low density to high density.

Table E.5-6. Physical diameter versus AMAD for monodisperse particle distribution ($\sigma_g = 1$)

Physical Diameter (μm) vs. AMAD (μm)					
Base Material*	**AMAD = 10**	**AMAD = 5**	**AMAD = 2**	**AMAD = 1**	**AMAD = 0.5**
Organic–[~$(CH_2)_n$]	12.9	6.48	2.61	1.31	0.668
Rust–[~FeO(OH)]	5.87	2.92	1.15	0.559	0.265
Ti H_2	6.17	3.07	1.21	0.591	0.282
Zr H_2	4.76	2.36	0.920	0.441	0.202
Hf H_2	3.53	1.74	0.666	0.310	0.133

* Variable amounts of elemental hydrogen are isotopically tritium.

NOTE: The organic row listed in this table is for insoluble OBT. In other words the row is for OBT that is particulate in nature, to include oil mists. See Section 5.2.5 for a discussion of soluble and insoluble OBT.

Particle size distributions which are "monodisperse" are comprised of particles which are all the same size. The parameter σ_g is the geometric standard deviation of the distribution. When $\sigma_g = 1$, no variability in the size distribution exists. When $\sigma_g > 1$, the distribution is called "polydisperse."

Subsequent text in this chapter investigates the dose from tritiated particulates as a function of different particle size distribution assumptions, from 0.5 to 10 µm AMAD. Dose variability over this range can then be compared to the dose from a 1 µm default size. Sub-micrometer particle size distributions are only produced by hot processes such as combustion (Cheng 1999b). This assertion is supported by Newton, 1987, and Dorrian, 1995. These references show that high energy processes such as oxyacetylene torch, electric arc cut rod, and plasma torch produce sub-micrometer particle size distributions, while lower energy processes such as band saw, reciprocating saw, and grinder produce larger particle size distributions. High energy cutting of tritium contaminated objects will release tritium from the objects in the form of HTO. Evaluation of particle sizes from 0.5 to 10 µm is therefore expected to be an appropriate evaluation of most respirable particulate STCs.

E.5.2.2.3 Dissolution Rate

The dissolution rate for a tritiated material is the rate at which tritium is released from the material to physiological (lung) fluid as HTO. For a given tritiated particulate intake, the dissolution rate affects the amount of tritium that remains in the lung to impart dose there. The effect of a varying dissolution rate assumption on the dose assigned from a given intake would be the same regardless of the method used to determine intake. Therefore, uncertainty in the dissolution rate also cannot be used to choose a preferred data collection method (urine bioassay vs. fecal bioassay vs. air concentration values). ICRP 71 gives DCFs for tritiated particulate intakes that vary from 0.3 to 14 times the DCF for HTO intakes, depending on the Absorption Type, which is primarily a function of the dissolution rate. Assuming the slowest dissolution rate for an intake would be a conservative assumption, in lieu of characterizations to the contrary, but would not grossly overestimate dose for a given intake. This is the basis behind applying ICRP 66 absorption type S ("slow," 99.9% dissolving with a halftime of 7,000 days) in the derivation of tritiated particulate DCF in Section E.5.2.2.

E.5.2.2.4 Default Parameters for ITPs

Dose conversion factors for ITPs are dependent on the particle size distribution and absorption characteristics of the aerosols in question. ICRP 66 provides guidelines for default assumptions to be used when detailed characterization data are unavailable. In many cases, the characterization process may require breach of contaminated systems. The assumptions listed in Table 5-7 should therefore be used to assess dose from suspected ITP intakes. These assumptions are conservative and consistent with ICRP 66 recommendations for assessing occupational exposure.

Table E.5-7: Default assumptions for insoluble tritiated particulate intakes

Mode of Intake	**Inhalation**
Particle Size	5μm AMAD
Geometric Standard Deviation (σ_g)	2.5
Shape Factor	1.5
Absorption Type	S
Weighting Factors	10 CFR Part 835
Biokinetic Model	ICRP 78

NOTE: The default particle size assumption of 5 μm AMAD is consistent with 10 CFR Part 835. However, values of the various self-absorption factors, dose conversion factors, and air concentration values are calculated for 1 μm and other values of AMAD. Such values may be used when justified.

E.5.2.2.5 Computer Codes

The Radiation Protection Division of the United Kingdom's Health Protection Agency (formerly the National Radiological Protection Board of Great Britain (NPRB)) has developed a computer program, called IMBA, (Integrated Modules for Bioassay Analysis) to facilitate the use of the recent ICRP biokinetic models such as the ICRP 66 Human Respiratory Tract Model. This model is discussed in Section 5.1. ICRP 71 (which defaults to a 1 μm AMAD polydisperse particle size distribution with a material density of 3 g/cm^3) also uses the ICRP 66 Human Respiratory Tract Model to calculate DCF. The DCFs for a 1 μm AMAD polydisperse particle size distribution and a material density of 3 g/cm^3 listed in ICRP 71 are the same as those calculated with IMBA. IMBA can also be used to refine the conclusions of this appendix, and to assess the dose from ITPs of known particle sizes and material densities or dissolution rates other than absorption type S. However, adjustment for self-absorption of beta particles and their energy still has to be performed as described previously to obtain DCF_os.

NOTE: A computer program called LuDEP (developed by the NRPB) was used at the Mound site to perform many of the calculations used in this appendix such as the variability of the DCFs for ITPs of various particle sizes. IMBA is the successor to LuDEP.

ITP dose evaluations may be calculated either by using IMBA directly, or by using DCFs (organ Committed Equivalent Dose and E_{50}) that are developed in this appendix (or that can be calculated using IMBA).

E.5.2.3 Self-Absorption Factor (SAF) for ITPs

Within the ITP, the mass of the particulate absorbs a portion of the beta particles emitted during tritium decay. The fraction of beta particles that escapes the ITP is called the “self-absorption factor for beta particles” (SAFβ). When analyzing air monitoring samples for ITPs using LSC, it is necessary to consider SAFβ, since ITPs do not dissolve appreciably and release tritium to LSC cocktail. Only beta radiation that escapes the particles in LSC cocktail is available for detection. The SAFβ is used to estimate the "actual" ITP activity that would be “observed” when samples are counted via LSC. This use of SAFβ has been confirmed experimentally (Kropf, 1998), although the supporting evidence is limited.

To assess dose to the lung from ITPs properly, the “self-absorption factor for energy” (SAFe) is an important consideration. Absorbed dose is a measure of the energy deposited (per unit mass) in tissue. Only the beta energy that escapes the particles contributes to dose. SAFe is the fraction of energy that escapes the particles. Because LuDEP does not account for SAFe, refinement of the results of LuDEP computations is necessary.

The SAFβ and SAFe for given physical particle sizes are calculated numerically here by the methods described by Kropf (1998). The determinations of SAFβ and SAFe are dependent on the electron density of a given material, which is proportional to $A/(Z\Delta)$, where A is the atomic mass of the material’s empirical formula, Z is the number of protons in the empirical formula, and Δ is the density.

The values employed to calculate the self-absorption factors (SAFs) for several representative materials are given in Table E.5-8 below. DOE has handled tritides of several metals, considered to be insoluble, which do not appear in the table (in the interest of brevity), including scandium, yttrium, and alloys of titanium and lanthanum. The derivation of SAFs for a representative but broad range of tritiated particulate materials is the prime focus of this section. The materials investigated were selected to represent a wide range of self-absorption, itself a function of the parameter $A/(Z\Delta)$. The materials in the tables of this section have values of this parameter which vary from 0.2 to 1.9. Parameter values for scandium, yttrium, and lanthanum ditritides are 0.72, 0.52, and 0.47, respectively. [Values for lanthanum alloys are slightly larger than those for lanthanum because of the inclusion of lighter elements. Values for iron/titanium alloys are close to those for titanium as given in the tables.] With respect to SAFs, the materials included in the tables therefore bracket the metals and alloys not included.

Table E.5-8: Constants for calculating SAFs for various monodisperse (σg = 1) ITPs

Base Material*	A	Z	Δ	$A/(Z\Delta)$
Organic [~$(CH_2)_n$]	14	8	0.9	1.944
Rust [~FeO(OH)]	89	43	3	0.690
Ti H_2	47.9	22	3.9	0.558
Zr H_2	91.22	40	6.49	0.351
Hf H_2	178.49	72	11.68	0.212

* Variable amounts of elemental hydrogen are isotopically tritium.

NOTE: In Table E.5-8, A and Z for organic and rust are based on the sum of the elements of the empirical formula. Δ for organic and rust are an average of a range of values from the Handbook of Chemistry and Physics (CRC, 1973). A and Z for hydrides are based on the metal only, since hydrogen is a minor contribution. Δ for hydrides are taken from Richardson, 1999 and Cheng, 1999a. Hydrogen in tritium-labeled compounds has been represented as protium only, as this has a minor effect on the ratio $A/(Z\Delta)$.

Functional representations (curve-fits) of numerically calculated SAF data (for monodisperse physical particle sizes), as a function of particle size, were further derived. These representations are a refinement of those by Kropf (1998), and more accurately fit the data at small particle sizes.

From the functional representations of SAFs for monodisperse physical particle sizes, SAFs for polydisperse AMAD particle size distributions were calculated using the methodology adapted from Kropf (1998) and Cheng (1999a). The resulting values are given in Tables E5-9 and E.5-10.

Table E.5-9: SAF_e for Various ITP Materials and Polydisperse (σ_g = 2.5) Particle Size Distributions (in AMAD)

Base Material*	A/(ZΔ)	SAF_e vs. Particle Size [AMAD(μm), σ_g=2.5]				
		10	**5**	**2**	**1**	**0.5**
Organic [~$(CH_2)_n$]	1.944	0.305	0.474	0.707	0.841	0.925
Rust [~FeO(OH)]	0.690	0.231	0.381	0.614	0.768	0.879
Ti H_2	0.558	0.215	0.362	0.594	0.753	0.871
Zr H_2	0.351	0.180	0.312	0.541	0.714	0.852
Hf H_2	0.212	0.142	0.260	0.490	0.680	0.842

* Variable amounts of elemental hydrogen are isotopically tritium.

Table E.5-10: SAFβ for Various ITP Materials and Polydisperse (σg = 2.5) Particle Size Distributions (in AMAD)

Base Material*	A/(ZΔ)	SAF_β vs. Particle Size [AMAD(μm), σ_g=2.5]				
		10	**5**	**2**	**1**	**0.5**
Organic [~$(CH_2)_n$]	1.944	0.205	0.333	0.535	0.677	0.789
Rust [~FeO(OH)]	0.690	0.140	0.244	0.439	0.602	0.745
Ti H_2	0.558	0.133	0.231	0.423	0.590	0.739
Zr H_2	0.351	0.104	0.191	0.380	0.555	0.719
Hf H_2	0.212	0.077	0.154	0.343	0.529	0.707

* Variable amounts of elemental hydrogen are isotopically tritium.

Other published computations of self-absorption factors (Richardson, 1999; Traub, 1999) largely confirm the results above. Traub (1999) states that bremsstrahlung photons (created when beta particles are self-absorbed), are not expected to be a significant contribution to dose from tritiated particulates.

E.5.2.4 Dose Conversion Factors (DCFs) and Derived Air Concentrations (DACs) for ITPs

The broad range of SAF derivations in the previous section allows correction of dose conversion factors (DCFs) for self-absorption across a broad range of materials and size distributions. The goal of these derivations and corrections in Section 5.2.2 is to determine a limiting ("worst case") self-absorption-corrected DCF among a variety of materials with wide variation in SAF. The variability of DCF as a function of particle size distribution is also assessed. This determination of a limiting DCF for ITPs allows a material-independent approach to air sampling and surface surveys following that point. A readily implemented approach, utilizing data obtained in terms of "observed" activity, is also provided.

E.5.2.4.1 Dose Conversion Factors (DCFs) for ITPs

Using LuDEP, the DCFs for committed equivalent dose to lung and E_{50} were calculated for a variety of ITP materials and particle size distributions using the default assumptions given in Table E.5-5 (except for particle size). The results are shown in Tables E.5-11 and E.5-12 below. LuDEP makes no allowance for self-absorption within particulates.

Table E.5-11: Lung Committed Equivalent Dose DCFs (Sv/Bq) for Various ITPs and Particle Sizes (AMAD, $\sigma_g = 2.5$), Absorption Type S Assumed

	Lung DCF vs. Particle Size [AMAD (μm), $\sigma_g = 2.5$]				
Base Material*	**10**	**5**	**2**	**1**	**0.5**
Organic [~$(CH_2)_n$]	4.50×10^{-10}	9.90×10^{-10}	1.65×10^{-9}	1.74×10^{-9}	1.64×10^{-9}
Rust [~FeO(OH)]	4.51×10^{-10}	1.00×10^{-9}	1.71×10^{-9}	1.98×10^{-9}	2.29×10^{-9}
Ti H_2	4.51×10^{-10}	1.00×10^{-9}	1.74×10^{-9}	2.07×10^{-9}	2.52×10^{-9}
Zr H_2	4.53×10^{-10}	1.01×10^{-9}	1.81×10^{-9}	2.30×10^{-9}	3.07×10^{-9}
Hf H_2	4.56×10^{-10}	1.03×10^{-9}	1.94×10^{-9}	2.70×10^{-9}	3.87×10^{-9}

* Variable amounts of elemental hydrogen are isotopically tritium.

NOTE: The reason the DCFs differ slightly for a given AMAD in Table E.5-11 is that the densities differ and the density affects the deposition fractions in the various lung compartments.

Table E.5-12: DCF (Sv/Bq), E_{50}, for Various ITPs and Particle Sizes (AMAD, $\sigma g = 2.5$), Absorption Type S Assumed

	E_{50} DCF vs. Particle Size [AMAD (μm), $\sigma_g = 2.5$]				
Base Material*	**10**	**5**	**2**	**1**	**0.5**
Organic [~$(CH_2)_n$]	6.10×10^{-11}	1.27×10^{-10}	2.05×10^{-10}	2.14×10^{-10}	2.00×10^{-10}
Rust [~FeO(OH)]	6.11×10^{-11}	1.28×10^{-10}	2.13×10^{-10}	2.43×10^{-10}	2.79×10^{-10}
Ti H_2	6.15×10^{-11}	1.28×10^{-10}	2.16×10^{-10}	2.54×10^{-10}	3.07×10^{-10}
Zr H_2	6.17×10^{-11}	1.3×10^{-10}	2.25×10^{-10}	2.83×10^{-10}	3.74×10^{-10}
Hf H_2	6.20×10^{-11}	1.32×10^{-10}	2.41×10^{-10}	3.30×10^{-10}	4.70×10^{-10}

* Variable amounts of elemental hydrogen are isotopically tritium.

NOTE: In Table E.5-12, the E_{50} DCF for rust, 1 µm AMAD, differs slightly from the E_{50} DCF given in ICRP 71 for absorption type S tritiated aerosols (2.6×10^{-10} Sv/Bq), even though the particle size distribution and densities are identical. The reason for this difference is that the ICRP 71 value was computed using the default “Environmental” assumptions (e.g., individual is sleeping 33.3% of the time) and the values given in the table above were computed using the default “Occupational” assumptions because this document focuses on occupational protection.

The DCFs given in Tables E.5-11 and E.5-12 were not corrected to account for SAFe. Therefore, the lung component of the DCFs is significantly exaggerated. In order to determine more realistic DCFs, it is necessary to correct the lung dose for SAFe using the values from Table E.5-9.

NOTE: This correction should only be applied to lung dose from particulates, not to HTO in the lung that results from the dissolution of the particle. However for absorption type S materials, the rate of dissolution is so slow that the contribution to the lung dose from HTO is negligible.

The correction for SAFe is made as follows:

$$DCF_{(corrected\ for\ SAF_e)} = DCF_{(uncorrected)} - 0.12 * DCF_{lung} + 0.12 * DCF_{lung} \times SAF_e \qquad \text{(Eq.E.5-9)}$$

where: $DCF_{(uncorrected)}$ is E_{50} per unit intake (Sv/Bq; values from Table 5-12)

0.12 is the ICRP 60 weighting factor for lung

DCF_{lung} is the Lung committed equivalent dose per unit intake (Sv/Bq; values from Table 5-11)

SAF_e is the corresponding value from Table 5-9

Table E.5-13 shows the E_{50} DCFs, corrected for SAF_e, for various ITPs.

Table E.5-13: DCF (Sv/Bq), E_{50}, for Various ITPs and Particle Sizes (AMAD, σ_g = 2.5), Absorption Type S Assumed, with SAFe Corrections Applied to the Lung Dose Component

	E_{50}DCF (w/ SAF_e) vs. Particle Size [AMAD (μm), σ_g = 2.5]				
Base Material*	**10**	**5**	**2**	**1**	**0.5**
Organic [~$(CH_2)_n$]	2.35×10^{-11}	6.41×10^{-11}	1.47×10^{-10}	1.81×10^{-10}	1.85×10^{-10}
Rust [~FeO(OH)]	1.95×10^{-11}	5.35×10^{-11}	1.34×10^{-10}	1.88×10^{-10}	2.45×10^{-10}
Ti H_2	1.90×10^{-11}	5.17×10^{-11}	1.32×10^{-10}	1.93×10^{-10}	2.67×10^{-10}
Zr H_2	1.71×10^{-11}	4.62×10^{-11}	1.25×10^{-10}	2.03×10^{-10}	3.19×10^{-10}
Hf H_2	1.51×10^{-11}	4.03×10^{-11}	1.22×10^{-10}	2.27×10^{-10}	3.97×10^{-10}

* Variable amounts of elemental hydrogen are isotopically tritium.

E_{50} DCFs in Table E.5-13 are now smaller than the E_{50} DCFs in Table 5-12, because of the corrections for particulate energy self-absorption. In order to use the values shown in Table E.5-13, the “actual” activity inhaled has to be known. Air monitoring can be performed via batch air filtration sampling and subsequent LSC analysis of the collected ITPs. An indicator of air concentration potential can also be provided by pre-job or in-job contamination surveys. To determine the “actual” activity on a filter, it would be necessary to dissolve the ITPs. Otherwise, self-absorption of the beta radiation from tritium leads to underestimates of the airborne activity. However, many ITPs (especially absorption type S) are extremely difficult to dissolve, requiring strong acids for extended periods of time. E_{50} DCFs for ITPs of absorption type S, when determined based on “actual” activity, are found to be highly dependent upon particle size. Table E.5-13 data indicate that variations for a given material can be as much as a factor of 30 for particle sizes from 0.5 to 10 μm AMAD.

Fortunately, it is not necessary to rely on “actual” activity or to dissolve ITPs prior to counting samples. Not only is dissolution cumbersome, but “observed” activity on filters counted by LSC without dissolution is a much better indicator of dose or dose potential than is “actual” activity. The reason for this is that (due to self-absorption) only the beta radiation that escapes ITPs is capable of causing lung dose and, if the ITPs are not dissolved prior to counting, only the beta radiation that escapes is counted.

Table E.5-14 shows E_{50} DCFs in terms of Sv per “observed” Bq when undissolved ITPs are counted via LSC. The values in the table were calculated by dividing the E_{50} DCFs in Table E.5-13 by the corresponding SAFβs from Table E.5-10. The term “DCF_o” is used to represent DCFs that are based on “observed” activity, as determined by LSC counting without dissolution.

Table E.5-14: DCF_o (Sv/observed Bq), E_{50}, for various ITPs and particle sizes (AMAD, $\sigma_g = 2.5$), Absorption Type S assumed

	E_{50} DCF_o vs. Particle Size [AMAD (μm), $\sigma_g = 2.5$]				
Base Material*	**10**	**5**	**2**	**1**	**0.5**
Organic [~$(CH_2)_n$]	1.14×10^{-10}	1.92×10^{-10}	2.75×10^{-10}	2.67×10^{-10}	2.35×10^{-10}
Rust [~FeO(OH)]	1.39×10^{-10}	2.19×10^{-10}	3.04×10^{-10}	3.12×10^{-10}	3.29×10^{-10}
Ti H_2	1.42×10^{-10}	2.24×10^{-10}	3.11×10^{-10}	3.27×10^{-10}	3.62×10^{-10}
Zr H_2	1.64×10^{-10}	2.42×10^{-10}	3.30×10^{-10}	3.66×10^{-10}	4.44×10^{-10}
Hf H_2	1.97×10^{-10}	2.61×10^{-10}	3.55×10^{-10}	4.29×10^{-10}	5.61×10^{-10}

* Variable amounts of elemental hydrogen are isotopically tritium.

Table E.5-14 data indicate that the E_{50} DCF_os, based on “observed” activity of ITPs of absorption type S, are less dependent on particle size than are the E_{50} DCFs, based on “actual” activity, from Table E.5-13. The assumption of a single particle size distribution would result in much smaller errors than noted above; for example, when using “observed” activity, E_{50} DCFs vary only by a factor of about 5 for particles ranging in sizes from 0.5 to 10 μm AMAD. Compare this with a factor of 25 when using “actual” activity.

The 10 CFR Part 835 recommended default particle size for occupational exposure is 5 μm AMAD ($\sigma_g = 2.5$). From Table E.5-14, the most conservative E_{50} DCF_o for 5 μm AMAD is about 2.6×10^{-10} Sv per observed Bq (9.7 × 102 rem/Ci_o). This is the value that should be used to compute E_{50} assessments for inhalation intakes of tritiated particulates, based on air monitoring and observed results from that monitoring.

To calculate internal dose using the various dose conversion factors for ITPs, it is necessary to estimate the intake of ITPs. Intake of ITPs (I) can be inferred from air sampling results using equations E5-10 and E5-11. (See DOE-STD 1121-98 for considerations to be applied when using air sampling results to determine internal dose.)

To calculate intake from air sampler measurements use equation 5-10.

$$I(Bq) = \frac{A_F(Bq) \cdot BR(m^3/hr)}{FR(m^3/hr) \cdot T_C(hr)} \cdot T_E(hr) \qquad \text{(Eq E.5-10)}$$

where

A_F is the activity on the filter sample (either observed or actual)
BR is the breathing rate of the worker
FR is the flow rate of the air sampler
T_E is the time period the worker was exposed to air containing radioactive material sampled
T_C is the time over which the sampler was operated.

Note: It is important to make sure that TC reflects the time the sampler was operating in an atmosphere expected to contain STPs. Including periods when the sampler is not in an STP containing atmosphere in the determination of TC will tend to reduce the estimated STP concentration in air and thus, lead to an underestimate of intake.

To calculate intake from personal air sampler measurements use equation E.5-11.

$$I(Bq) = \frac{A_F(Bq) \cdot BR(m^3/hr)}{FR(m^3/hr)} \qquad \text{(Eq E.5-11)}$$

NOTE: To ensure that intake is not underestimated, the sampler should never be turned off when a worker is in an atmosphere expected to contain STPs.

NOTE: Equations E.5-10 and E.5-11 assume 100% collection efficiency. The filters used at the Mound Site (Mound 2000) had collection efficiencies that ranged between 0.981 – 0.9999 for 0.3 to 10 um particles. ANSI N13-1 (ANSI 1999) contains a table of collection efficiencies for various types of filters.

Various types of internal doses for STCs can then be calculated from the product of equations E.5-10 or E.5-11 and the various DCFs provided in this appendix. For example, the DCFs in Table E.5-14 table can be used to determine the E_{50} from inhalation of ITPs of various sizes and composition using the observed activity on an air filter. In addition, Annex A contains DCFs for STPs of density 11.7 gm/cm3 (Hf) for absorption types S, M, and F and AMADs of 1 and 5 µm.

To calculate internal doses (using air sampling results) from inhalation of ITP species that have particle sizes, densities, and compositions that differ from those cases addressed in this appendix, it will be necessary to obtain appropriate values for the self-absorption factors for beta and energy, as they are not provided in Tables E.5-9 and E.5-10. The observed air sample activity would then have to be converted to actual activity by dividing it by the appropriate SAFβ or by otherwise determining the total activity in the STPs in the sample. Then use either equation E.5-10 or E.5-11, to calculate the intake in terms of actual activity.

Next, using a computer code such as IMBA, the E_{50} can be calculated. The component of the E_{50} resulting from exposure to the lung would have to be corrected for energy absorption using the appropriate SAFe in a manner analogous to that in equation E.5-9.

NOTE: If the base material and particle size are not known, parameters and associated SAFs for the most conservative material (i.e., HfH_2), 1 μm AMAD particle size ($\sigma_g = 2.5$), and absorption type S should be used for dose computations (See Table E.5-7 for default assumptions) [absorption type S is believed to be sufficiently conservative: the DCF_o for 5 μm AMAD polydisperse HfH_2 is 2.61×10^{-10} Sv/Bq_o (assuming absorption type S). Deviations from these parameters should only be allowed in situations where material and particle size are well known.

As noted above, the variability of E_{50} DCF_o with material or particle size is not great (range is factor of 5). Therefore, using a conservative assumption of material and particle size distribution (5 μm AMAD) is accurate within that factor for any material or particle size distribution; this assumption can be readily implemented when air monitoring is the method of intake assessment. A factor of dose overestimation (~ 2 – 10) is further applied when all captured tritiated particulates are assumed to be absorption type S and are measured by LSC.

E.5.2.4.2 DACs for ITPs

Derived Air Concentrations (DACs) can now also be computed; these are concentrations of tritiated particulate aerosols which, when inhaled, impart prescribed doses over a given time (i.e., 0.05 Sv in 2,000 hrs).

The equation below converts dose conversion factors (DCFs) to DACs.

$$DAC\left(\frac{Bq}{m^3}\right) = \frac{0.05 \quad (Sv)}{DCF \quad \left(\frac{Sv}{Bq}\right)} \cdot \frac{1}{2{,}400 \quad (m^3)} \qquad \text{(Eq. E.5-12)}$$

If the DCF_o is used in place of DCF in Eq. E.5-12, then this formula can be used to calculate DACo. Accordingly, the most conservative DACo for ITPs (based on the E_{50} DCF_o determined in Section 5.2.4.1 for ITPs [4.3×10^{-10} Sv/Bqo]) is 4.8×10^4 Bqo/m^3 (1.2 μCio/m^3, 1.3×10^{-6} μCio/cm^3).

Appendix A to the June 7, 2008 amendments to 10 CFR Part 835 contains DACs for radiological protection against insoluble STCs including ITPs. These DACs assume a particle of density 11.7 gm/cm^3, are based on observed activity and were calculated for absorption types S, M, and F. The methods contained in this appendix were used to calculate these DACs. The DCFs for these DACs are listed in Annex A to this appendix.

E.5.2.5 Biokinetic Model for Soluble OBT

For soluble forms of OBT, particularly solvents and the small molecule component of OBT oil, the assumption can be made that all activity deposited is instantaneously absorbed into the body (ICRP 78, 1997). Soluble OBT is, therefore, amenable to available urine bioassay. However, the biokinetic model for soluble OBT is different from that for HTO. Table E.5-15 shows the difference in fractional uptake and biological halftime for soluble OBTs and HTO (ICRP 78, 1997).

Table 5-15. Fractional Uptake and Biological Halftime for Soluble OBT and HTO (ICRP 78, 1997)

	Fractional uptake with 10-day biological $T_{1/2}$	**Fractional uptake with 40-day biological $T_{1/2}$**
HTO	97%	3%
OBT	50%	50%

E.5.2.5.1 Intake and Dose Assessment for Soluble OBT

Soluble OBT solvents or oil components do not cause self-absorption of tritium beta activity as do solid STCs. A DCF and DAC can therefore be expressed in terms of actual activity. The E_{50} DCF that should be used for a soluble OBT intake is 4.1×10^{-11} Sv/Bq (152 rem/Ci) (ICRP 78), unless a more appropriate DCF is derived and documented for the type of soluble OBT encountered in the workplace.

To calculate the DAC for vapors of soluble OBT, there are two pathways that must be considered (which are similar to HTO pathways), inhalation and skin absorption. If there is no evidence that a soluble OBT is absorbed through the skin, then uptake via skin absorption is zero and the DAC can be calculated using Eq E.5-12. Using the DCF for OBTs from ICRP 78 of 4.1×10^{-11} Sv/Bq, a DAC of 5.0×10^5 Bq/m^3 is calculated.

If there is evidence that an OBT is absorbed via the skin the following approach should be used to calculate the DAC. ICRP 30 states that for exposure to airborne HTO, two-thirds (⅔) of the intake is from inhalation and one-third (⅓) is from skin absorption. Therefore, the total intake of airborne HTO is 1.5 times the inhaled intake. To determine a DAC, the ICRP 30 assumptions for uptake of airborne HTO are assumed here to be valid for soluble OBT; i.e., ⅔ of the intake is due to inhalation and ⅓ is due to skin absorption. Therefore, equation E.5-12 should be divided by a factor of 1.5 to account for the OBT taken into the body from skin absorption. Accordingly, dividing the result of the calculation from the previous paragraph by 1.5, the calculated DAC for soluble OBT that accounts for skin absorption is 3.3×10^5 Bq/m^3 (9.1 μCi/m^3 or 9.1×10^{-6} μCi/cm^3).

Soluble forms of OBTs are considered to be instantaneously absorbed into the body; therefore, urinalysis can be used to assess soluble OBT intakes. This approach is adequate to demonstrate compliance with the dose limits established in Subpart C of 10 CFR Part 835. To determine the soluble OBT intake, the individual's average urine excretion volume (if not available, the ICRP 23 Reference Man value of 1.4 liter per day should be used) and the ICRP 78 expected excretion rate per unit intake of soluble OBT should be used.

Appendix A to the June 7, 2008 amendments to 10 CFR Part 835 contains the DAC for radiological protection against soluble STCs such as soluble OBT. This DAC is based on actual activity and does not consider uptake by the skin. The DCF for this DAC is listed in Annex A to this appendix.

Internal soluble OBT doses should be assessed by either:

1. Using the E_{50} DCF listed above for soluble OBT (IMBA may also be used to calculate individual organ CDE DCFs); or
2. Using IMBA to convert the intake amount to final E_{50} (IMBA may also be used to calculate individual organ CDEs).

NOTE: IMBA can accommodate ingestion intakes of soluble materials, as well as inhalations of particulates, and can be used to document the soluble OBT dose calculations. Using the ICRP 78 biokinetic model for soluble OBT above, IMBA calculates 4.2×10^{-11} Sv E_{50} per Bq intake by ingestion, which is the same as the ICRP 78 E_{50} DCF for soluble OBT intake by ingestion. This value is slightly larger than the DCF for soluble OBT (based on inhalation) quoted at the beginning of this section (4.1×10^{-11} Sv/Bq). Therefore, using IMBA for soluble OBT intakes provides a slight overestimate of the E_{50}, compared to that which would be obtained by simply using the DCF listed above.

E.6 WORKPLACE CONTROLS

In establishing workplace controls for STCs, the extent of workplace controls should be commensurate with the hazard presented by the specific STC for which protection is required. For certain STCs, such as certain types of organically bound tritium, existing technology permits institution of a radiological control system that is commensurate with systems used to protect workers from exposure to the oxide and gaseous forms of tritium. Alternatively, establishing a level of workplace control commensurate with the hazard presented by other types of STCs (such as ITPs) poses a significant challenge because of difficulties in identifying and characterizing these types of

STCs. As a result of such limitations in individual and workplace monitoring, the system of workplace controls should be conservative in order to demonstrate that workers are adequately protected.

E.6.1 Design/Engineered Controls

Current regulations and standard practices for occupational radiation protection require a primary reliance on design and engineered controls for protection, unless such controls are impractical or ineffective. The distinction between design/engineered controls and administrative controls is not always definitive. For example, an item of protective clothing that blocks the incident radiation is functioning as a form of shielding, but is most commonly considered an administrative control. Likewise, a temporary HEPA-filtered ventilation system may require adherence to administrative procedures to ensure its proper installation and operation, but is most commonly considered an engineered control. In this appendix, the term “design/engineered controls” includes such controls as shielding systems, confinement systems, and ventilation systems. Those controls that require direct intervention and proper operation by individual users, such as respiratory protection, protective clothing, and access controls, are considered to be administrative controls.

Because the physical behavior of ITPs mimics that of similar particulates, there are no design/engineered controls that are unique to ITP contamination in the workplace. Design/engineered controls for ITPs should be evaluated and applied in the same manner as for other particulate contaminants, with particular attention to such factors as likelihood of contamination spread, particle sizes, contamination levels, likely individual doses, planned and potential activities and events, and planned future uses for the affected area. Consistent with the as low as reasonably achievable (ALARA) process, consideration should be given to efforts that may be necessary to recover from the use of engineered control systems for STC contamination protection, such as special decontamination, storage and handling needs, purchase of new equipment, and special waste handling needs.

E.6.2 Administrative Controls

Following development and implementation of appropriate design features, including engineered controls, administrative controls are typically developed and applied to ensure that the design features are properly installed, maintained, and operated. In addition, administrative controls provide an additional level of safety above and beyond that provided by the design features alone, consistent with the ALARA process.

E.6.2.1 Administrative Systems

An adequate structure of administrative systems is necessary to ensure that projects are completed safely, satisfactorily, and in compliance with applicable requirements. The actual components of the administrative systems, and the content of those systems, is dependent on a number of factors, including the nature of the work to be performed, education, skills and training of the individuals performing the work, and the magnitude of any associated hazards. Management oversight is required to achieve the appropriate balance between the various components of the administrative systems. For example, certain high-hazard work may require extensive pre-job planning and rigid adherence to detailed written procedures and radiological work permits (RWPs). For lower hazard work, or more routine and repetitive work, it may be appropriate to place more reliance on employee training and experience. The appropriateness of the balance achieved should be routinely assured through a rigid system of management oversight that closely examines the work processes, and detects potential problems before they adversely affect safety and achievement of the facility's mission.

10 CFR 835.501 requires written authorizations to enter into and perform work in radiological areas. While there is a great deal of flexibility provided for establishing the exact nature of and details provided in these authorizations, they generally fall into one or more of the classifications discussed below.

E.6.2.1.1 Policies and Procedures

Facility requirements for safe work with STCs should be clearly delineated in written policies and procedures. These requirements may be established in either generally applicable procedures, or in procedures that are established and limited in scope to address STC affected areas and topics only. For example, requirements for controlling access to STC-contaminated areas may be specified in procedures that apply only to the specified areas, or the requirements may be established as a sub-section of a generally applicable procedure that addresses control of access to a wide variety of hazardous areas. Likewise, there need not be a separate ALARA policy addressing only exposure to STCs if the existing ALARA policy adequately addresses all radiation exposures, including STC exposure.

In addition to any procedures that may be implemented to ensure compliance with 10 CFR 835.501, 10 CFR 835.104 requires that written procedures be developed and implemented as necessary to ensure compliance with the applicable regulatory requirements of that rule. The procedures should be written in sufficient detail to ensure that the user is able to complete the relevant task properly, and in compliance with applicable requirements. The style of writing should be appropriate to the assigned procedure user.

Facilities should have written procedures to address the following issues associated with STCs, in addition to the procedural controls required for the remainder of the radiological control program:

- Methods for identifying and quantifying STC levels in the workplace;
- Methods for controlling the spread of airborne and surface STC contamination, including design features and engineered controls;
- Methods for controlling and handling materials (e.g., protective clothing, respirators, tools, etc.) that are exposed to STC contamination;
- Methods for tracking those areas of the facility affected by STC contamination;
- Methods for controlling individual exposure to STCs;
- Radiological and activity based criteria for which STC controls are required;
- Criteria for selecting individuals for participation in the STC-internal dose monitoring program;
- Methods for performing STC monitoring;
- Individual STC dose assessment techniques;
- Criteria for selecting individuals to participate in enhanced training required for STC control; and
- Requirements for assessing STC-control program effectiveness, identifying program shortcomings, and implementing programmatic changes as necessary to ensure continuing improvement.

Written procedures form only one component of an adequate administrative system. Active management involvement is required to determine the proper balance between the components of the administrative system. For example, details regarding techniques used to perform repetitive actions may be more appropriately addressed through radiation safety training. Certain job-specific actions may be more appropriately addressed in radiological work permits rather than written procedures.

E.6.2.1.2 Technical Work Documents

While generally applicable written procedures are frequently used to establish standardized methods where consistency of techniques or results (e.g., regulatory compliance) is important, technical work documents (TWDs) are often used to address measures associated with specific tasks. Included within the scope of technical work documents are certain specific procedures, work packages, and research plans.

TWDs should be developed and implemented for any task that requires more detailed written guidance than that provided in the generally applicable procedures.

Chapter 3 of the Radiological Control Standard (RCS) provides guidance for implementing a system of TWDs.

E.6.2.1.3 Radiological Work Permits

Radiological Work Permits (RWPs) are used to establish specific radiological control requirements for work in areas that exceed, or have the potential to exceed specified radiological hazard levels. This would usually include tasks that require specific worker actions due to the likelihood of severe radiological consequences, such as surface contamination levels or airborne radioactivity levels exceeding 100 times the applicable Appendix A or Appendix D levels, respectively. The RWP should address those issues discussed in the RCS, with emphasis on conditions and controls that are related to the presence of STCs. Chapter 3 of the RCS provides guidance for developing and implementing RWPs.

E.6.2.1.4 Administrative System Integration

The components of the administrative system are usually linked. For example, the administrative system may address the requirements for repairing a component located in a contamination area as follows:

A written procedure establishes standard requirements for entering a contamination area;

Another written procedure establishes criteria (radiological conditions, work activities, etc.) that require specific tasks to be performed in accordance with an RWP/TWD;

- The TWD establishes specific requirements (e.g., valve operation sequences, torque settings, sequence of operations, etc.) for performing the task; and

- The RWP provides more detailed radiological control requirements for repairing the component within the specified contamination area.

E.6.2.1.5 Pre- and Post-Job Briefings

For higher hazard work, many facilities include requirements for pre- and post-job briefings in the RWP. The pre-job briefing provides an opportunity to discuss the details of the job, its hazards, and required controls with affected workers. The post-job briefing provides an opportunity to review the completed work, identify both strengths and weaknesses associated with its planning and performance, and institutionalize these lessons for future reference. Briefings may also be conducted during job performance to analyze the current status and address emergent issues.

The thresholds and processes associated with briefings for STC work need not differ from those for other jobs. However, special attention should be paid to STC identification and control issues. Mechanisms should be in place to capture any lessons learned, not only for similar work, but for both similar and dissimilar work involving STC hazards. Specific guidance is provided in Chapter 3 of the RCS.

E.6.2.2 Administrative Control Levels

Facilities typically use a system of administrative control levels to limit individual doses and to direct management attention toward those individuals, work groups, and/or activities that result in the most significant doses. Management establishes the value(s) of the administrative control level(s) on a yearly basis, such that line management and workers are challenged to control, and find innovative ways to reduce, their individual and workgroup doses. Greater levels of control are achieved by establishing administrative control levels on a work group-specific basis, reflecting management attention to specific workgroup activities and hazards. As individual doses approach the established administrative control level, successively higher levels of management approval are required to raise the level, thereby bringing greater management attention to ALARA issues in the workplace. Administrative control levels should be set at a level that is challenging, but achievable in view of the planned activities and the radiological conditions in the workplace. However, if it is never necessary to consider authorizing an individual to exceed the previously established administrative control level (thus requiring active management attention), the level probably has been set too high. On the other hand, if the administrative control level is set at an unnecessarily stringent level, costly delays in planned activities may result, reflecting inattention to the cost/benefit aspect of ALARA planning.

The presence of STCs in the workplace does not profoundly affect the administrative control level program. In setting or extending administrative control levels, consideration should be given to the levels of surface and airborne STC contamination, the individual doses likely to result from that contamination, uncertainties in relating workplace conditions to actual doses, and any delays in determination of individual doses following exposure to STCs. The administrative control level should be set in a manner that challenges individuals and organizations to seek innovative means of reducing their doses, while avoiding unnecessary interference with scheduled activities.

Chapter 2 of the RCS provides specific guidance for implementing a system of administrative control levels.

E.6.2.3 Decontamination

When it is necessary to perform work in an STC-contaminated area or on STC-contaminated equipment, consideration should be given to performing a thorough decontamination of the area/equipment prior to performing the required work. This assessment should be conducted in a manner consistent with the ALARA process. A thorough decontamination effort may eliminate any significant STC contamination levels, thus eliminating the need for STC controls for the remainder of the evolution. This may in turn reduce collateral impacts, such as the need to launder STC-contaminated protective clothing, maintain STC-contaminated respirators and ventilation systems, and determine STC doses for all individuals involved in the work. These benefits should be balanced against the costs of performing the decontamination (including the collateral impacts discussed above), the likelihood of success, the likelihood and possible impact of spreading STC contamination during the decontamination operation. With either approach, consideration should be given to the costs and difficulties associated with handling any resulting waste products.

E.6.2.4 Personal Protective Equipment

Because STCs can exist in particulate form, the personal protective equipment (PPE) used for ITP control is not unique – that equipment used for control of other radioactive particulates in the workplace should be sufficient for ITP control also. Different from respiratory protection for exposure to HT and HTO, air purifying respirators may be used for ITP work. Consideration should be given to the need for special cleaning and handling needs for STC-contaminated PPE to reduce the probability of spreading STC contamination to previously unaffected areas or to minimizing the handling and cleaning of STC contaminated PPE. These needs may be met by establishing separate facilities for STC-contaminated PPE, by using disposable anti-contamination clothing, or by ensuring the handling areas are free of STC contamination before handling non-STC-contaminated PPE. In most cases, the cost of establishing special facilities for handling STC-contaminated PPE will be prohibitive. If offsite facilities are used for cleaning/processing of PPE (e.g., contracted laundry facilities for contaminated protective clothing), the contractor should be made aware of the possibility of STC contamination so that he may implement an effective STC control program at his facility.

E.6.2.5 Area Posting

10 CFR Part 835 requires that certain areas be posted to alert individuals to the presence of specific conditions, including specified levels of:

- penetrating external radiation;
- removable surface contamination; and
- airborne radioactivity

These requirements are augmented by requirements for posting areas where radioactive items or containers of radioactive material are used, handled, or stored, and requirements for posting areas to which access is controlled for radiation protection purposes.

Because tritium, including STCs, is not typically considered an external radiation hazard, posting for penetrating external radiation hazards is not expected to be an issue for areas where STCs are used, handled or stored.

The need for posting to alert individuals to the presence of removable contamination, airborne radioactivity, and radioactive items or containers of radioactive materials, consisting in whole or part of STCs, depends on the results of efforts to identify the presence, and quantify levels of, STCs in specified areas. General guidance for establishing area postings for radiological hazards is provided in Chapter 12 of DOE G 441.1-1C, *Radiation Protection Programs Guide for Use with Title 10, Code of Federal Regulations, Part 835,* Occupational Radiation Protection. This guidance is applicable to posting for the presence of STCs; however, the STC surface or airborne contamination level should be evaluated consistent with the guidance provided in Sections E.3.1 and E.3.2 of this appendix. Section E.3.2.1.2 provides guidance pertinent to areas or objects where tritium is bound tightly to surfaces and/or to the matrix below such surfaces. This guidance is applicable to efforts to evaluate the total (fixed plus removable) STC activity levels, which are used as a basis for radioactive material area posting.

Circumstances may arise in which it is desirable to establish a controlled area due solely to the presence of STCs. 10 CFR Part 835 requires that controlled areas be established whenever radiological or radioactive material areas are present, but does not establish specific criteria that define the controlled area boundaries. Controlled area postings should be established to enclose any area where it is likely that personnel activities in the area will result in the spread of:

- STC removable surface contamination beyond the current area boundaries at levels exceeding one tenth of the levels specified in Appendix D of 10 CFR Part 835; or

- STC airborne radioactivity beyond the current area boundaries at levels exceeding one tenth of the airborne concentration value specified in Sections E.5.2.4.2 and E.5.2.5.1 of this Appendix.

The controlled area boundaries should be established so that all of the affected area is encompassed, and area monitoring should be performed on a periodic basis to verify the adequacy of the boundaries. The area boundaries should be altered as necessary, based on the results of the area monitoring.

It is often desirable to restrict access to STC-contaminated areas, to specified individuals or workgroups so as to limit the scope of the STC dose assessment program. If this has been done, posting for STC-contaminated areas should provide additional information that will allow individuals to identify the area as one affected by STCs, and to take appropriate action to avoid or facilitate entry, as appropriate.

E.6.2.6 Access Control

NOTE: The access control measures discussed in this section are in addition to the radiation safety training measures discussed in Section E.6.2.9 of this appendix.

Access control measures may be necessary to control individual access to areas where STC surface contamination or airborne radioactivity is present. 10 CFR Part 835 requires entry control measures for all radiological areas. The established controls should be appropriate for the radiological conditions in the area and the planned activities, some of which may cause changes in the radiological conditions or in the conditions (e.g., relative position, exposure time, contaminant resuspension) under which individuals are exposed. DOE has provided guidance in the RCS.

For low levels of STC contamination (e.g., surface contamination levels less than the 10 CFR Part 835 Appendix D values; airborne radioactivity levels resulting in an intake less than 12 DAC-hours in a week [0.3 times the DAC values]), only rudimentary entry control measures are necessary. These measures may include basic engineered and administrative controls.

NOTE: Although repeated exposure to STCs in unposted areas with airborne radioactive materials could lead to a dose of up to 1500 mrem (E_{50}) in a year, DOE requirements in 10 CFR Part 835 specify initiation of individual and workplace monitoring when annual exposures are expected to exceed 40 DAC-hours (100 mrem). In addition, 10 CFR Part 835 requires the institution of the ALARA process when establishing workplace controls for radiation exposure during routine operations. Thus, it is unlikely planned exposures exceeding 100 mrem would occur without knowledge of either the site or the facility's radiological protection organization.

If the STC contamination levels are elevated (e.g., surface contamination levels exceed the 10 CFR Part 835 Appendix D values, but are less than 10 times those values; exposures to airborne radioactivity could exceed 12 DAC-hours, but are less than 120 DAC-hours, per week), more aggressive measures should be implemented. Such measures may include both design features and administrative controls, such as filtered ventilation systems, physical barriers to both individual access and contamination spread, area decontamination, individual sign-in (written or electronic), written procedures and/or radiological work permits, as appropriate, designated access routes, etc. These controls should be considered in addition to those discussed above for lower levels of contamination. Note that these levels correspond to those levels defining a radiological area as provided in 10 CFR 835.2(a); therefore, specific actions are required to ensure compliance.

If STC contamination levels are very high (e.g., surface contamination levels exceed ten times the 10 CFR Part 835 Appendix D values; airborne radioactivity levels exceed ten times the DAC values), then the most aggressive contamination control measures should be considered and implemented, consistent with the ALARA process.

See section E.3.2.1.2 for guidance relating to areas containing locations where tritium is bound tightly to the surface and/or to the matrix below the surface.

E.6.2.7 Radioactive Material Labeling

Labeling of radioactive items and containers of radioactive material is required under some circumstances to alert individuals to the radiological hazards associated with those items or containers. DOE has provided appropriate guidance for implementing these requirements in Chapter 12 of DOE G 441.1-1C, *Radiation Protection Programs Guide for Use with Title 10, Code of Federal Regulations, Part 835,* Occupational Radiation Protection, and in the RCS.

The thresholds under which radioactive material labeling is required are provided in 10 CFR 835.605, which requires labeling when the total quantity of radioactive material exceeds one-tenth of the 10 CFR Part 835 Appendix E values. Appendix E does not currently provide a specific value for STCs, but does provide a value for tritium. This value is applicable to radioactive items and containers of radioactive material containing STCs.

It is often desirable to restrict access to STC-contaminated materials to specified individuals or workgroups so as to limit the scope of the STC dose assessment program. If this has been done, labels for STC-contaminated materials should provide additional information that will allow individuals to identify the material as affected by STCs, and to take appropriate action to avoid exposure or facilitate proper handling.

10 CFR 835.606 establishes several conditions under which labeling of radioactive items and containers of radioactive material are not required. When using these exceptions, caution should be exercised to ensure that individuals working in the vicinity of radioactive items and containers of radioactive material are provided appropriate information in the absence of radioactive material labels. For example, inclusion of appropriate information on area postings or in employee training or pre-job briefings may prove to be an effective substitute for radioactive material labeling under the specified conditions.

Note that the 10 CFR Part 835 radioactive material labeling requirements are applicable only to “items and containers.” Piles and areas of granular solids, such as soil and sand, are generally not considered to be “items” or “containers”; therefore, labeling of such piles and areas may not be required or practical. However, consideration should be given to the need for posting such piles or areas as contaminated areas consistent with 10 CFR 835.603. See Chapter 12 of DOE G 441.1-1C for further guidance on application of the 10 CFR Part 835 posting requirements to areas of soil and similar materials.

Another situation for which labeling may not be required by 10 CFR Part 835, but for which labeling would be useful, is for materials that have been exposed to tritium long enough for the tritum to bind tightly in the matrix of the material. Labels could be used to warn individuals that operations performed on these materials that generate particles could create an airborne radiological hazard.

E.6.2.8 Contaminated Material Control

Appropriate controls on the storage, use, and movement of materials contaminated with STCs are necessary to limit the spread of STC contamination to other materials, areas, and individuals. Such a spread would result in a need to expand the scope of the STC monitoring program beyond its initial boundaries. This is undesirable due to both the possible individual and environmental exposures that may occur, and the relative expense and complexity of the STC monitoring and control program.

Subpart L of 10 CFR Part 835 establishes DOE's regulatory requirements for control of contaminated material and equipment. In general, the regulations require that material and equipment that is located in a contamination, high contamination, or airborne radioactivity area be monitored for the presence of surface contamination prior to release to the controlled area. The regulations also establish provisions allowing for release of contaminated materials under certain controlled conditions.

Material and equipment that is located in areas where STC contamination is known to exist should be monitored consistent with the requirements of 10 CFR Part 835 Subpart L prior to release to the controlled area. The monitoring should be performed consistent with Sections E.3.1 and E.3.2 of this appendix and should be capable of identifying STC contamination at or below the 10 CFR Part 835 Appendix D values.

If STC contamination is discovered on material and equipment that is to be moved, or that is located in an area that is to be released from contaminated area status, efforts should be made to both contain the contamination, and to warn individuals of its presence. All contaminated items should be securely packaged or wrapped in such a manner that the STC contamination will not be inadvertently disturbed by individual activities, environmental forces (such as air or liquid flow), or work activities (such as the application of mechanical force or heat). Packages and wrapping materials should be appropriate for the surface to be contained and the expected environment. Programs should be established to periodically inspect packages and wrapping materials to ensure they do not degrade in long-term storage or use. If degradation is noted, surface contamination monitoring should be performed to detect and determine the magnitude of the release of STC contamination. The package or wrapping material should be repaired or replaced, as appropriate.

Material and equipment that are contaminated with STCs should be clearly labeled to indicate the type and extent of the contamination, and thereby facilitate tracking of the items and ensure implementation of appropriate controls and monitoring during activities involving the contaminated items. If practical, a log of STC-contaminated material and equipment should be maintained. Such a document may prove useful when planning work activities and associated monitoring and control activities. STC-contaminated material and equipment that is no longer in use should be decontaminated and released or disposed of as soon as practicable. This will reduce the likelihood of spreading STC contamination during storage periods, and as a result, any unplanned or unauthorized access to the material.

If practicable, separate spaces and facilities should be dedicated to storage, use, and decontamination of STC-contaminated material and equipment. Likewise, consideration should be given to laundering STC-contaminated protective clothing and other launderable items separately. These measures will reduce the likelihood of spreading STC contamination to previously unaffected areas, systems, and items, necessitating expansion of the STC monitoring and control program, and will facilitate control over these items. Should it be necessary to use, handle, or store STC-contaminated items in common areas, then it will be necessary to expand the STC contamination monitoring program to encompass affected surfaces and systems, such as ventilation and drain systems and surrounding areas. Similarly, areas, items, and systems that are dedicated to STC-contamination purposes, such as decontamination areas and laundry equipment, should be routinely monitored for residual STC contamination.

See section 3.2.1.2 for guidance relating to areas containing materials or equipment where tritium is bound tightly to the surface and/or to the matrix below the surface.

E.6.2.9 Training

E.6.2.9.1 General Employee and Radiological Worker Training

Because of the specialized control and monitoring requirements that are necessary for STCs, enhanced training should be provided to individuals entering affected areas, or working in the vicinity of affected items to ensure they are equipped to recognize and respond to STC hazards.

Subpart J of 10 CFR Part 835 establishes regulatory requirements for radiation safety training for DOE activities. Chapter 14 of DOE G 441.1-1C, *Radiation Protection Programs Guide for Use with Title 10, Code of Federal Regulations, Part 835,* Occupational Radiation Protection, provides guidance for achieving compliance with these requirements. The training requirements are generally divided into two levels, depending on the types of hazards encountered. General Employee Radiological Training is provided to individuals who enter only the controlled area and receive minimal exposure to radiation. Radiological Worker Training is provided to individuals who enter radiological areas or perform radiological work that is likely to result in doses exceeding 0.1 rem in a year. These requirements are augmented by 10 CFR 835.103, which requires that all individuals who are responsible for developing and implementing measures necessary for ensuring compliance with the requirements of 10 CFR Part 835 have the appropriate education, training, and skills to discharge their responsibilities.

For those activities affected by radiological hazards arising from STCs, General Employee Radiological Training and Radiological Worker Training should include material on the following topics, to the extent appropriate for the target audience:

- The physical and radiological characteristics of STCs;
- Radiological hazards associated with exposure to STCs;
- Measures used to monitor (areas and individuals) for STC contamination;
- Limits and action levels associated with STCs;
- Radiological controls implemented to control the spread of STC contamination;
- Special considerations for work planning and execution;
- Special warning signs and labels used in areas affected by STCs, and appropriate responses to those signs and labels;
- Any special records or reports associated with STCs that may be generated and retrieved; and
- Appropriate responses to emergency situations involving STCs.

Although members of the public entering the controlled area are subject to the radiation safety training requirements of 10 CFR 835.901, these requirements may be met, in part, through provision of an escort or area entry controls to ensure the individual’s safety and compliance with the documented RPP. If members of the public are granted unescorted access to the controlled area, then radiation safety training is required. DOE does not expect members of the public to receive any significant exposure to STCs; therefore, the required training should be augmented by provisions to identify, and a warning to avoid entering, the affected areas.

E.6.2.9.2 Training of Radiological Control Staff

Individuals in the radiological control organization will require specialized training to allow them to develop and coordinate implementation of effective programs for STC control. Training should be provided to affected radiological control organization managers, supervisors, engineers, administrators, and technicians. The training should address the same subjects as those specified above, but in more detail as is appropriate to the individuals’ responsibilities.

E.6.2.9.3 Training of Facility Managers, Supervisors and Work Planners

Managers and supervisors should complete the same or higher level of radiation safety training as the individuals who report to them. Managers and supervisors should receive additional training that will prepare them to address employee concerns about radiological hazards, including occupational exposure to STCs. Managers, supervisors, and work planners should also receive training that will prepare them to complete any measures that are necessary to execute their responsibilities for coordinating, planning, and completing work in areas affected by STCs.

E.6.2.10 Work Processes

DOE’s requirements for safety management are established in the DOE P 450.4 series of directives and guides focusing on Integrated Safety Management (ISM). The ISM system addresses the following seven guiding principles:

1. Line management responsibility for safety
2. Clearly defined roles and responsibilities
3. Personnel competence commensurate with responsibilities
4. Balance of priorities
5. Identification of safety standards and requirements
6. Hazard controls tailored to the work being performed
7. Operations authorization

For any particular facility, most of these guiding principles are unaffected by the presence of STCs; that is, the implementation of the guiding principles spans a wide range of issues and is not hazard- or isotope-specific. The STC protection and control program should not be separated from other radiological and industrial safety programs; it should be fully integrated with related safety programs so that the hazards associated with STCs are considered in a manner consistent with those arising from other sources of workplace hazards. However, the provisions of this appendix should be considered in developing safety programs for STCs. For example, Guiding Principle #1 will require consideration of the training guidance provided above. Likewise, Guiding Principles # 6 and #7 will require consideration of the results of the workplace monitoring program to provide an accurate assessment of the relative hazards arising from the presence of STCs in the workplace.

Finally, it is important that implementation of work processes be carried out in a manner that involves all categories of individuals. These categories should include workers, supervisors and managers as well as individuals from the various safety disciplines who will participate in operations involving exposure to STCs.

ANNEX A – VALUES OF SIGNIFICANT DOSIMETRIC PROPERTIES OF STCS

Derived Air Concentration (DAC) Values

DAC for HTO$_2$ (Includes a 50% allowance for skin absorption)	2 x 10^{-5} µCi/ml 7 x10^5 Bq/m^3
DAC for HT$_2$	2 x 10^{-1} µCi/ml 9 x 10^9 Bq/m^3
DAC for ITPs (If host material not known)	4.8 x 10^4 Bq$_o$/m^3
AMAD = 1 µm	1.2 x 10^{-6} µCi$_o$/ml
DAC$_o$ for ITPs (If host material not known)	8.0 x 10^4 Bq$_o$/m^3
AMAD = 5 µm	2.0 x 10^{-6} µCi$_o$/ml
DAC for soluble OBTs (Divide by 1.5 if skin absorption expected but magnitude not known)	5.0 x 10^5 Bq/m^3 1.4 x 10^{-5} µCi/ml

Dose Conversion Factors (DCFs) for Inhalation of HTO and STCs

HTO (Does not include an allowance for skin absorption. Multiply estimated intake by 1.5 to account for skin absorption when calculating derived exposure controls)[1]	1.8 x 10^{-11} Sv/Bq 6.7 x 10^{-2} mrem/ µCi
ITPs AMAD = 1 µm	
ITP Type F Density = 11.7 gm/cm^3	1.3 x 10^{-11} Sv/Bq$_o$ 4.8 x 10^{-2} mrem/ µCi$_o$
ITP Type M Density = 11.7 gm/cm^3	7.6 x 10^{-11} Sv/Bq$_o$ 2.8 x 10^{-1} mrem/ µCi$_o$
ITP absorption type S Density = 11.7 gm/cm^3 (use this DCF$_o$ if host material not known)	4.3 x 10^{-10} Sv/Bq$_o$ 6.7 x 10^{-2} mrem/ µCi$_o$
DCF$_o$ (Ti, Zr, Hf, rust, Organic [oil droplets]) absorption type S	See Table 5-14
ITPs AMAD = 5 µm	
ITP Type F Density = 11.7 gm/cm^3	5.5 x 10^{-11} Sv/Bq$_o$ 2.0 x 10^{-1} mrem/ µCi$_o$
ITP Type M Density = 11.7 gm/cm^3	8.6 x 10^{-11} Sv/Bq$_o$ 3.1 x 10^{-1} mrem/ µCi$_o$
ITP absorption type S Density = 11.7 gm/cm^3 (use this DCF$_o$ if host material not known)	2.6 x 10^{-10} Sv/Bq$_o$ 9.6 x 10^{-1} mrem/ µCi$_o$
DCF$_o$ (Ti, Zr, Hf, rust, Organic [oil droplets]) absorption type S	See Table 5-14
Soluble OBT Compounds (Adjustment for skin absorption, if applicable, should be accounted for when calculating DACs [see section E.5.2.5.1])	
OBT[2]	4.1 x 10^{-11} Sv/Bq 1.5 x 10^{-1} mrem/ µCi

Tritiated Methane[3] ($CH_{4-x}T_x$)	1.8×10^{-13} Sv/Bq 6.7×10^{-2} mrem/ µCi

[1] ICRP 1979
[2] ICRP 1995
[3] ICRP 1998b

ANNEX B – INSOLUBLE METAL TRITIDE BENCHMARK

This is a benchmark calculation of dose and excretion following an intake of a type M Insoluble Metal Tritide (IMT) using the ICRP 66 respiratory tract model and ICRP 78 systemic model. The calculations were implemented in MathCad 2000[98] and Excel 97[99]. (Note that when performing internal dose estimated for the purpose of demonstrating compliance with 10 CFR Part 835, the tissues weighting factors specified in 10 CFR Part 835 must be used.) A basic knowledge of Mathcad syntax will facilitate the study of this benchmark, but anyone having experience with a higher level language (e.g., FORTRAN) should be able to follow the calculations.

First, load in definitions and subroutines from the file FUNCTIONS.MCD. The content of this file is given in the end of this appendix.

The physical decay constant for tritium in units of 1/day is given by

$$\lambda = \frac{\ln(2)}{(12.3 \cdot 365)}$$

The deposition of the SMT aerosol in the compartments of the respiratory tract is a function of the AMAD of the aerosol and is calculated with the subroutine “Depo” (which is defined in the next appendix). For occupational exposure, an AMAD of 5 :m is assumed.

$$\mathrm{AMAD} = 5$$

$$\mathrm{D} = \mathrm{Depo}(\mathrm{AMAD})$$

The initial content q0 of the compartments are defined below. The initial content of any compartment not explicitly defined has a value of zero because the initial content of the last compartment (i.e., urine) is defined as zero[100]. The intake of the SMT I_{SMT} is defined as unity and the

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[98] Mathsoft, Inc. 101 Main Street, Cambridge MA.

[99] Microsoft

[100] This is a characteristic of Mathcad.

intake of HTO I_{HTO} as zero. Intakes of HTO are modeled as direct instantaneous depositions into the blood compartment.

$$I_{smt} \equiv 1$$

$$I_{hto} \equiv 0$$

$$i = AI1 .. ET1$$

$$q0_i = I_{smt} \cdot D_i$$

$$q0_{urine} = 0$$

$$q0_{blood} = I_{hto}$$

The lung dissolution rate constants and f_1 are defined below. All rate constants are in units of 1/days. Defaults are for type M SMT are: f_r=0.1, s_r=100, s_s=0.005, f_1=0.1. For type S: f_r=0.001, s_r=100, s_s=0.0001, f_1=0.01.

$$f_r = 0.1$$

$$s_r = 100$$

$$s_s = 0.005$$

$$s_p = s_s + f_r \cdot (s_r - s_s)$$

$$s_{pt} = (1 - f_r) \cdot (s_r - s_s)$$

$$s_t = s_s$$

$$f_1 = 1 \cdot 10^{-1}$$

Setting the total removal rate constant for urine to zero defines a zero matrix that has dimensions of urine x urine (39 x 39 for this model). The individual transfer rate constants for the respiratory tract are then defined per ICRP 66.

$k_{\mathrm{urine,\,urine}} = 0$

$k_{\mathrm{AI1,\,bb1}} = 0.02$

$k_{\mathrm{bb2,\,BB1}} = 0.03$

$k_{\mathrm{ETseq,\,LNet}} = 0.001$

$k_{\mathrm{AI1,\,blood}} = s_{\mathrm{p}}$

$k_{\mathrm{bb2,\,blood}} = s_{\mathrm{p}}$

$k_{\mathrm{ETseq,\,blood}} = s_{\mathrm{p}}$

$k_{\mathrm{AI1,\,TAI1}} = s_{\mathrm{pt}}$

$k_{\mathrm{bb2,\,Tbb2}} = s_{\mathrm{pt}}$

$k_{\mathrm{ETseq,\,TETseq}} = s_{\mathrm{pt}}$

$k_{\mathrm{AI2,\,bb1}} = 0.001$

$k_{\mathrm{BB1,\,ET2}} = 10$

$k_{\mathrm{BBseq,\,LNth}} = 0.01$

$k_{\mathrm{AI2,\,blood}} = s_{\mathrm{p}}$

$k_{\mathrm{BB1,\,blood}} = s_{\mathrm{p}}$

$k_{\mathrm{BBseq,\,blood}} = s_{\mathrm{p}}$

$k_{\mathrm{AI2,\,TAI2}} = s_{\mathrm{pt}}$

$k_{\mathrm{BB1,\,TBB1}} = s_{\mathrm{pt}}$

$k_{\mathrm{BBseq,\,TBBseq}} = s_{\mathrm{pt}}$

$k_{\mathrm{AI3,\,bb1}} = 0.0001$

$k_{\mathrm{BB2,\,ET2}} = 0.03$

$k_{\mathrm{bbseq,\,LNth}} = 0.01$

$k_{\mathrm{AI3,\,LNth}} = 0.00002$

$k_{\mathrm{BB2,\,blood}} = s_{\mathrm{p}}$

$k_{\mathrm{bbseq,\,blood}} = s_{\mathrm{p}}$

$k_{\mathrm{AI3,\,blood}} = s_{\mathrm{p}}$

$k_{\mathrm{BB2,\,TBB2}} = s_{\mathrm{pt}}$

$k_{\mathrm{bbseq,\,Tbbseq}} = s_{\mathrm{pt}}$

$k_{\mathrm{AI3,\,TAI3}} = s_{\mathrm{pt}}$

$k_{\mathrm{ET2,\,S}} = 100$

$k_{\mathrm{LNth,\,TLNth}} = s_{\mathrm{pt}}$

$k_{\mathrm{bb1,\,BB1}} = 2$

$k_{\mathrm{ET2,\,blood}} = s_{\mathrm{p}}$

$k_{\mathrm{LNet,\,blood}} = s_{\mathrm{p}}$

$k_{\mathrm{bb1,\,blood}} = s_{\mathrm{p}}$

$k_{\mathrm{ET2,\,TET2}} = s_{\mathrm{pt}}$

$k_{\mathrm{LNet,\,TLNet}} = s_{\mathrm{pt}}$

$k_{\mathrm{bb1,\,Tbb1}} = s_{\mathrm{pt}}$

$k_{\mathrm{ET1,\,ENV}} = 1$

$k_{\mathrm{LNth,\,blood}} = s_{\mathrm{p}}$

The transfer rate constants for the transformed respiratory tract compartments are defined:

$k_{\mathrm{TAI1,Tbb1}} = k_{\mathrm{AI1,bb1}}$

$k_{\mathrm{TBB2,TET2}} = k_{\mathrm{BB2,ET2}}$

$k_{\mathrm{TAI1,blood}} = s_t$

$k_{\mathrm{TBB2,blood}} = s_t$

$k_{\mathrm{TAI2,Tbb1}} = k_{\mathrm{AI2,bb1}}$

$k_{\mathrm{TET2,S}} = k_{\mathrm{ET2,S}}$

$k_{\mathrm{TAI2,blood}} = s_t$

$k_{\mathrm{TET2,blood}} = s_t$

$k_{\mathrm{TAI3,Tbb1}} = k_{\mathrm{AI3,bb1}}$

$k_{\mathrm{TETseq,TLNet}} = k_{\mathrm{ETseq,LNet}}$

$k_{\mathrm{TAI3,TLNth}} = k_{\mathrm{AI3,LNth}}$

$k_{\mathrm{TETseq,blood}} = s_t$

$k_{\mathrm{TAI3,blood}} = s_t$

$k_{\mathrm{TBBseq,TLNth}} = k_{\mathrm{BBseq,LNth}}$

$k_{\mathrm{Tbb1,TBB1}} = k_{\mathrm{bb1,BB1}}$

$k_{\mathrm{TBBseq,blood}} = s_t$

$k_{\mathrm{Tbb1,blood}} = s_t$

$k_{\mathrm{Tbbseq,TLNth}} = k_{\mathrm{bbseq,LNth}}$

$k_{\mathrm{Tbb2,TBB1}} = k_{\mathrm{bb2,BB1}}$

$k_{\mathrm{Tbbseq,blood}} = s_t$

$k_{\mathrm{Tbb2,blood}} = s_t$

$k_{\mathrm{TLNet,blood}} = s_t$

$k_{\mathrm{TBB1,TET2}} = k_{\mathrm{BB1,ET2}}$

$k_{\mathrm{TLNth,blood}} = s_t$

$k_{\mathrm{TBB1,blood}} = s_t$

The transfer rate constants for the systemic compartments, urine, feces, and other excreta are defined:

$$k_{\mathrm{hto,bladder}} = \frac{1.4}{3} \cdot \frac{\ln(2)}{10}$$

$$k_{\mathrm{hto,ULI}} = \frac{0.1}{3} \cdot \frac{\ln(2)}{10}$$

$$k_{\mathrm{obt,bladder}} = \frac{\ln(2)}{40}$$

$$k_{\mathrm{blood,obt}} = 0.03 \cdot \frac{\ln(2)}{0.25}$$

$$k_{\mathrm{blood,hto}} = 0.97 \cdot \frac{\ln(2)}{0.25}$$

$$k_{\mathrm{hto,excreta}} = \frac{1.5}{3} \cdot \frac{\ln(2)}{10}$$

$$k_{\mathrm{bladder,urine}} = 12$$

Finally, the transfer rate constants for the GI tract are defined:

$$k_{\mathrm{S,SI}} = 24$$

$$k_{\mathrm{SI,ULI}} = 6$$

$$k_{\mathrm{SI,blood}} = \frac{k_{\mathrm{SI,ULI}} \cdot f_1}{1 - f_1}$$

$$k_{\mathrm{ULI,LLI}} = \frac{24}{13}$$

$$k_{\mathrm{LLI,feces}} = 1$$

ANNEX C – MATHCAD DEFINITIONS AND SUBROUTINES

As an alternative to calculating each total removal rate constant individually, one may use the function *total*, which sums the rate constants in each row and assigns the sum to the diagonal element of the rate matrix. The function is defined in the next appendix.

$$k = \mathrm{total}(k, \lambda)$$

The eigenvalues and coefficients for each compartment are calculated. Note that the content of each compartment at time t is defined in the function q(t,comp) by a sum of 39 exponential terms.

$$e = \mathrm{eigenvals}\left(k^{T}\right)$$

$$C = \mathrm{coeff}(k, q0)$$

$$q(t, \mathrm{comp}) = \sum_{i=1}^{\mathrm{cols}(k)} C_{\mathrm{comp}, i} \cdot \exp\left[\left(e_i\right) \cdot t\right]$$

The content of all compartments at time t=0 calculated with the function q_{all} should equal the sum of what was deposited in the body.

$$q_{all}(t) = \sum_{j=1}^{\mathrm{cols}(k)} q(t, j)$$

In other words, if everything goes OK, what goes into the system

$$\sum_{i=AI1}^{\mathrm{urine}} q0_i = 0.81957$$

should equal the material that is in the system at t=0 days for a radioactive material(or any other time for insoluble material).

$$q_{all}(0) = 0.81957$$

The function Decays is used to calculate the number of decays that occur in each compartment of the biokinetic model.

```
Decays(x) = | $s \leftarrow 0$
            | for $j \in AI1 .. urine$
            |     $s \leftarrow s - \frac{C_{x,j}}{e_j} \cdot \left( \exp(e_j \cdot 0) - \exp(e_j \cdot 50 \cdot 365) \right)$ if $\left| C_{x,j} \right| > 0$
            | $Decays \leftarrow s \cdot 24 \cdot 3600$
```

Note that a dosimetric source organ may consist of more than one biokinetic compartment. For example, the number of decays on the surfaces of ET2 is the sum of the decays in the ET2 and TET2 biokinetic compartments.

$$U_{sET1sur} = Decays(ET1)$$

$$U_{sET2sur} = Decays(ET2) + Decays(TET2)$$

$$U_{sET2seq} = Decays(ETseq) + Decays(TETseq)$$

$$U_{sBBsol} = Decays(BB2) + Decays(TBB2)$$

$$U_{sBBgel} = Decays(BB1) + Decays(TBB1)$$

$$U_{sBBseq} = Decays(BBseq) + Decays(TBBseq)$$

$$U_{sbbsol} = Decays(bb2) + Decays(Tbb2)$$

$$U_{sbbgel} = Decays(bb1) + Decays(Tbb1)$$

$$U_{sbbseq} = Decays(bbseq) + Decays(Tbbseq)$$

$$a = Decays(TAI2) + (Decays(AI3) + Decays(TAI3))$$

$$b = Decays(AI1) + Decays(TAI1) + Decays(AI2)$$

$$U_{sAI} = a + b$$

$$U_{sLNet} = Decays(LNet) + Decays(TLNet)$$

$$U_{sLNth} = Decays(LNth) + Decays(TLNth)$$

$$U_{sS} = Decays(S)$$

$$U_{sSI} = Decays(SI)$$

$$U_{sULI} = Decays(ULI)$$

$$U_{sLLI} = Decays(LLI)$$

$$U_{sUBcont} = Decays(bladder)$$

$$U_{sTissue} = Decays(obt) + Decays(hto) + Decays(blood)$$

In summary, the decays in the source organs are given in array U:

U =

	1
1	29241.9
2	362.4
3	2522.4
4	489.5
5	311.3
6	13098
7	637.9
8	952.5
9	9697.9
10	396.2
11	531041
12	2311.1
13	355.5
14	1510
15	5436
16	17820.4
17	32894.2
18	142132

The dose to each target organ is the sum of the decays in each organ and the appropriate SEE:

$$\text{target} = \text{tAdrenals} \,..\, \text{tUterus}$$

$$H_{\text{target}} = \sum_{\text{source} = \text{sET1sur}}^{\text{sTissue}} U_{\text{source}} \cdot SEE_{\text{target, source}}$$

The array H is loaded into an Excel 97 spreadsheet to complete the dose calculation. The organ doses are combined as shown in the spreadsheet to give ICRP 60 50-year committed effective dose in units of Sv/Bq.

ICRP 60 Committed Effective Dose in Sv

	Ht	A	A x Ht	Wt	Mass (g)	Ht x Mass	Ht x Wt
Adrenals	1.88E-12				14	2.63E-11	
UB_Wall	1.88E-12			0.050			9.40E-14
Bone_Sur	1.88E-12			0.010			1.88E-14
Brain	1.88E-12				1400	2.63E-09	
Breasts	1.88E-12			0.050			9.40E-14
St_Wall	4.63E-12			0.120			5.56E-13
SI_Wall	8.07E-12				640	5.16E-09	
ULI_Wall	3.88E-11	0.570	2.21E-11				
LLI_Wall	1.13E-10	0.430	4.85E-11				
Kidneys	1.88E-12				310	5.83E-10	
Liver	1.88E-12			0.050			9.40E-14
ET1-bas	1.88E-12	0.001	1.88E-15				
ET2-bas	1.91E-12	0.998	1.90E-12				
LN-ET	3.16E-11	0.001	3.16E-14				
BBi-bas	1.88E-12	0.167	3.13E-13				
BBi-sec	1.88E-12	0.167	3.13E-13				
bbe-sec	1.98E-12	0.333	6.58E-13				
AI	4.42E-10	0.333	1.47E-10				
LN-Th	7.20E-11	0.001	7.20E-14				
Muscle	1.88E-12				28000	5.27E-08	
Ovaries	1.88E-12						
Pancreas	1.88E-12				100	1.88E-10	
R_Marrow	1.88E-12			0.120			2.26E-13
Skin	1.88E-12			0.010			1.88E-14
Spleen	1.88E-12				180	3.38E-10	
Testes	1.88E-12			0.200			3.76E-13
Thymus	1.88E-12				20	3.76E-11	
Thyroid	1.88E-12			0.050			9.40E-14
Uterus	1.88E-12				80	1.50E-10	
Composite Organs							
lungs	1.49E-10			0.120			1.78E-11
ET	1.94E-12				15	2.91E-11	
colon	7.06E-11			0.120			8.47E-12
oesophagus	1.88E-12			0.050			9.40E-14
remainder	2.01E-12			0.050			1.00E-13
				1.000	30759	6.18E-08	
effective							**2.81E-11**

The 24-hour incremental urinary and fecal excretion functions are defined as shown below. Note that the increment excreted in urine and feces is calculated with “insoluble” tritium. The incremental excretion is then adjusted for radioactive decay. Because we are starting with radioactive tritium, the incremental excretion is adjusted to insoluble tritium first.

$$\Delta e_u(t) = \exp(-\lambda \cdot t) \cdot \left[q(t, \text{urine}) \cdot \exp(\lambda \cdot t) - q(t-1, \text{urine}) \cdot \exp[\lambda \cdot (t-1)] \right]$$

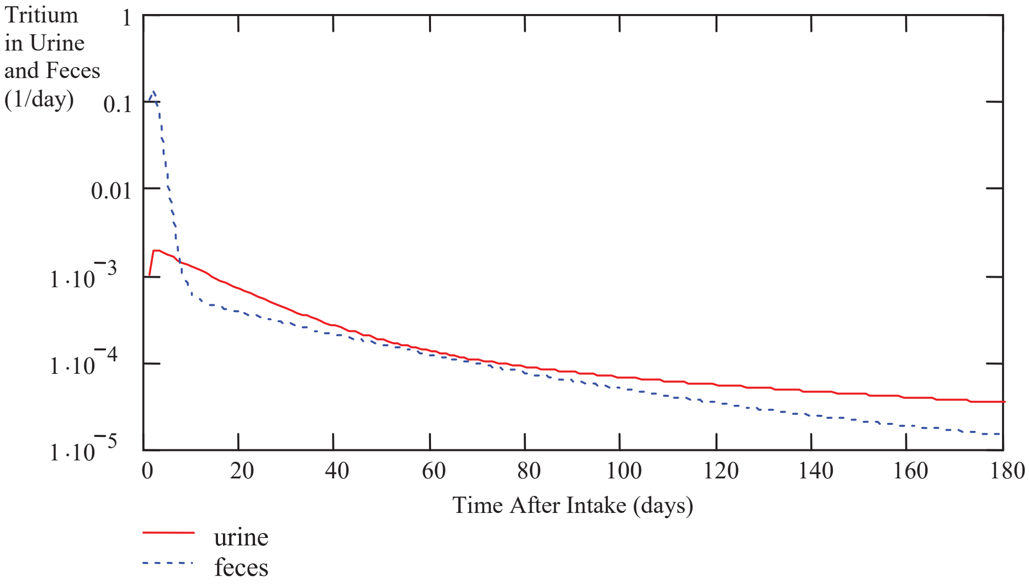
$$\Delta e_f(t) = \exp(-\lambda \cdot t) \cdot \left[q(t, \text{feces}) \cdot \exp(\lambda \cdot t) - q(t-1, \text{feces}) \cdot \exp[\lambda \cdot (t-1)] \right]$$

The following plot shows the 24-hour urinary and fecal excretion for days 1 through 180 after an acute inhalation intake of 5 µm AMAD type M IMT.

$$t = 1 .. 180$$

$$e_{u_t} = \Delta e_u(t)$$

$$e_{f_t} = \Delta e_f(t)$$



In Mathcad, the first element is designated as being in the zeroth row and zeroth column, i.e., having the (0,0) coordinate. When defining a matrix for biokinetic models it is useful for the matrix to begin with the (1,1) coordinate. This is accomplished by setting the built-in variable ORIGIN to 1.

$$\text{ORIGIN} \equiv 1$$

The compartments in the biokinetic model are assigned names to clarify their use in the definition of the rate matrix. For example, rather than define the transfer rate constant $k_{1,4}$, we define it as $k_{\text{AI1},\text{bb1}}$. The respiratory tract compartments are:

AI1 = 1
AI2 = 2
AI3 = 3
bb1 = 4
bb2 = 5
bbseq = 6
BB1 = 7
BB2 = 8
BBseq = 9
ET2 = 10
ETseq = 11
ET1 = 12
LNet = 13
LNth = 14
ENV = 15

The transformed respiratory tract compartments are:

TAI1 = 16
TAI2 = 17
TAI3 = 18
Tbb1 = 19
Tbb2 = 20
Tbbseq = 21
TBB1 = 22
TBB2 = 23
TBBseq = 24
TLNth = 25
TET2 = 26
TETseq = 27
TLNet = 28

Finally, the GI tract, systemic compartments, and excreta compartments are:

S = 29
SI = 30
ULI = 31
LLI = 32
feces = 33
blood = 34
bladder = 35
excreta = 36
obt = 37
hto = 38
urine = 39

The sources (beginning with "s") in the dosimetric model are assigned names to clarify their use in the definition of the SEE and dose matrices:

sET1sur = 1
sET2sur = 2
sET2seq = 3
sLNet = 4
sBBgel = 5
sBBsol = 6
sBBseq = 7
sbbgel = 8
sbbsol = 9
sbbseq = 10
sAI = 11
sLNth = 12
sUBcont = 13
sS = 14
sSI = 15
sULI = 16
sLLI = 17
sTissue = 18

Likewise, the targets (beginning with "t") in the dosimetric model are assigned numbers to clarify their use in the definition of the SEE and dose matrices:

tAdrenals = 1
tUB = 2
tBS = 3
tBrain = 4
tBreasts = 5
tS = 6
tSI = 7
tULI = 8
tLLI = 9
tKidneys = 10
tLiver = 11
tET1bas = 12
tET2bas = 13
tLNet = 14
tBBbas = 15
tBBsec = 16
tbbsec = 17
tAI = 18

tLNth = 19
tMuscle = 20
tOvaries = 21
tPancreas = 22
tRM = 23
tSkin = 24
tSpleen = 25
tTestes = 26
tThymus = 27
tThyroid = 28
tUterus = 29

Once the transfer rate constants are defined, the total removal rate constants may be calculated by summing across the rows of the rate matrix:

```
$total(k, \lambda) =$
    $K \leftarrow k$
    for $comp \in 1..cols(k)$
        $K_{comp,comp} \leftarrow 0$
        for $j \in 1..cols(k)$
            $K_{comp,comp} \leftarrow K_{comp,comp} + k_{comp,j}$ if $comp \neq j$
        $K_{comp,comp} \leftarrow -(K_{comp,comp} + \lambda)$
    $K$
```

The coefficients and rate constants for the retention functions are calculated by the eigensystem method. This approach to solving systems of ODEs is described by G. G. Killough and K. F. Eckerman in "A Conversational Eigenanalysis Program for Solving Differential Equations," in Computer Applications in Health Physics, Proceedings of the Seventeenth Midyear Topical Symposium of the Health Physics Society, 1984 (Killough 1984). Excellent examples may be found in "The Linear Algebra Problem Solver" by the Research and Education Association, 1980, gives excellent examples and discussion of the method (see problems 18-6 through 18-11). Given the initial content of each compartment at t=0 and the rate matrix, the following function calculates the coefficients for the retention functions:

```
coeff(k, q0) = | q0 ← submatrix(q0, 1, rows(k), 1, 1)
               | V ← eigenvecs(k^T)
               | M ← lsolve(V, q0)
               | for j ∈ 1.. cols(k)
               |    for i ∈ 1.. cols(k)
               |       C_{i,j} ← V_{i,j}·M_j
               | C
```

Deposition fractions for aerosol AMADs in the range of 0.001 to 10 microns are calculated by interpolating from the respiratory tract deposition table given in the ICRP CD-ROM[101]. The ICRP 66 respiratory tract model for occupational exposure (light work) was used to calculate the values given in this table:

D =	0.001	0.003	0.01	0.03	0.1	0.3
AI1	1.359E-04	6.714E-03	8.229E-02	1.525E-01	8.688E-02	4.458E-02
AI2	2.718E-04	1.343E-02	1.646E-01	3.051E-01	1.738E-01	8.916E-02
AI3	4.530E-05	2.238E-03	2.743E-02	5.085E-02	2.896E-02	1.486E-02
bbf	2.262E-02	8.770E-02	1.291E-01	7.188E-02	3.328E-02	1.523E-02
bbs	2.294E-02	8.895E-02	1.309E-01	7.290E-02	3.376E-02	1.544E-02
bbseq	3.212E-04	1.245E-03	1.833E-03	1.021E-03	4.726E-04	2.162E-04
BBf	3.117E-02	4.420E-02	2.685E-02	1.035E-02	4.746E-03	3.260E-03
BBs	3.161E-02	4.483E-02	2.723E-02	1.049E-02	4.813E-03	3.293E-03
BBseq	4.425E-04	6.276E-04	3.812E-04	1.469E-04	6.738E-05	4.619E-05
ET2	4.390E-01	3.494E-01	1.735E-01	6.994E-02	3.218E-02	5.820E-02
ETseq	2.196E-04	1.748E-04	8.680E-05	3.499E-05	1.610E-05	2.912E-05
ET1	4.433E-01	3.327E-01	1.539E-01	6.248E-02	3.071E-02	5.217E-02

The following function performs a linear interpolation of the values in the table to calculate the deposition fractions in each compartment of the respiratory tract.

```
Depo(AMAD) = | d ← submatrix(D, 1, 13, 2, 11)
             | for i ∈ 2.. 13
             |    Depo_{i-1} ← linterp[(d^T)^⟨1⟩, (d^T)^⟨i⟩, AMAD]
             | Depo
```

---

[101] The ICRP Database of Dose Coefficients for Workers and Members of the General Public, Version 1.0 (Pergamon Press: New York) 1998 (ICRP 1998a).

All SEEs are from SEECAL[102] except for the SEE for the bladder wall <--- bladder contents, which is set to zero in accordance with ICRP 67[103]. The SEE have the units of SV/decay. The SEE for the body tissues as source and all body tissues and organs as targets is defined by

$$\text{target} = \text{tAdrenals} .. \text{tUterus}$$

$$\text{SEE}_{\text{target},\,\text{sTissue}} = 1.323 \cdot 10^{-17}$$

Specific target-source combinations for the respiratory and GI tract are:

$$\text{SEE}_{\text{tET2bas},\,\text{sET2seq}} = 1.088 \cdot 10^{-17}$$

$$\text{SEE}_{\text{tLNet},\,\text{sLNet}} = 6.069 \cdot 10^{-14}$$

$$\text{SEE}_{\text{tbbsec},\,\text{sbbsol}} = 9.853 \cdot 10^{-18}$$

$$\text{SEE}_{\text{tAI},\,\text{sAI}} = 8.276 \cdot 10^{-16}$$

$$\text{SEE}_{\text{tAI},\,\text{sLNth}} = 4.138 \cdot 10^{-16}$$

$$\text{SEE}_{\text{tLNth},\,\text{sLNth}} = 3.035 \cdot 10^{-14}$$

$$\text{SEE}_{\text{tS},\,\text{sS}} = 1.821 \cdot 10^{-15}$$

$$\text{SEE}_{\text{tSI},\,\text{sSI}} = 1.138 \cdot 10^{-15}$$

$$\text{SEE}_{\text{tULI},\,\text{sULI}} = 2.069 \cdot 10^{-15}$$

$$\text{SEE}_{\text{tLLI},\,\text{sLLI}} = 3.372 \cdot 10^{-15}$$

---

[102] M. Cristy and K. F. Eckerman, *SEECAL: Program to Calculate Age-Dependent Specific Effective Energies*, ORNL/TM-12351 (Cristy 1993).

[103] "Urinary excretion will not result in a significant additional dose to the bladder wall which is assumed to receive the same dose as other tissues," Table C-1.1, ICRP 67 (ICRP 1994b).

ANNEX D - PARAMETERS FOR DISSOLUTION OF INSOLUBLE TRITIATED PARTICULATE MATERIALS

(From Mound Technical Basis Document for Stable Tritiated and Particulate and Organically Bound Tritium (Mound 2000))

The following table compiles literature data for dissolution of tritium into physiological fluid from various forms of tritium-containing particulate materials. Parameters are given for retention of tritium in the source material as described by (typically) the double exponential:

$$S(t) = F_r \exp(-R_r t) + F_S \exp(-R_S t) \qquad \text{(eq. D-1)}$$

Where: S(t) = the tritium fraction retained in source material at time t

F_r = the fraction of rapidly dissolving material ("fast component")

F_S = the fraction of slowly dissolving material ("slow component")

R_r = the dissolution rate of rapidly dissolving material

R_S = the dissolution rate of slowly dissolving material

Particle size characteristics and literature references are also provided.

Source Material	**F_r**	**R_r, d^{-1}**	**F_s**	**R_s, d^{-1}**	**Size Parameters (μm)**	**Reference**
TiT$_x$	0.55	4.6	0.35 (a)	0.093 (a)	CMD=0.95 σ_g=1.93 AMD=3.5 AMAD=5.7	Cheng (1999a)
TiT$_x$	0.24	0.71	0.76	0.021	Same as above	Cheng (1997)
TiT$_x$	-	-	1.0	0.0019	CMD=103 σ_g=1.58 AMD=193 AMAD=312	Cheng (1997)
ZrT$_x$	0.48	0.0165	0.52	0.0018	CMD=0.32 σ_g=2.24 AMD=2.25 AMAD=4.78	Cheng(1999b) Kropf (1998)
HfT$_x$ (b)	0.0006	0.0095	0.9994	2.3E-6	CMD=1.04 σ_g=2.41 AMD=10.6 AMAD=30	Cheng (1999c)
HfT$_x$ (b)	0.00015 (c)	0.02 (c)	0.9997	1E-7	Same as above	Cheng (1999c)
HfT$_x$	0.008	0.2	0.992	~5E-6	"respirable"	Ref 4 in McConville (1995)
Glass	0.23	0.079	0.16	0.0032	CMD=3.8 σ_g=2.8 AMD~20 AMAD~19	Cool (1983) [Sample KMS2] (d)
Glass	0.27	0.23	0.027	0.03	Same as above	Cool (1983) [Sample KMS1] (e)
Luminous Paint	0.12	~10	0.88	0.019	----	Rudran (1988)

(a) An additional slow component (F~0.1 and R<0.01) is apparent from Fig 2 in reference.

(b) Represent each of two separate data sets within this study.

(c) An additional fast component of F=0.00015 and R=100 occurred for this HfT$_X$ sample.

(d) Parameters are recalculated from Fig 1 of reference. An additional slow component of F=0.61 and unspecified R is apparent. Particle sizes are from Cool (1984).

(e) Parameters are recalculated from Table 2 of reference, assuming 30% dissolution at day 100. An additional slow component of F=0.70 and unspecified R is apparent. Particle sizes are from Cool (1984).

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APPENDIX F1: Program Management Guide

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INTRODUCTION

Purpose and Scope

This beginning section discusses in general recommendations for the implementation of radiation safety training. Course specific guidance follows

This Appendix describes a recommended implementation process for conducting the radiation safety training required by Title 10 Code of Federal Regulations *Occupational Radiation Protection*, (10 CFR Part 835) Subpart J and as outlined in the DOE standard DOE-STD-1098-2008, *Radiological Control* (RCS). The Appendix is to assist those individuals, both within the Department of Energy (DOE) and Managing and Operating (M&O) contractors, identified as having responsibility for implementing the training required by 10 CFR Part 835 and recommended by the RCS.

Management Guide Content

The management guide is divided into the following sections:

- Introduction
- Instructional Materials Development
- Training Program Standards and Policies
- Course Specific Information

Training Program Goal

The goal of the training program is to provide a baseline knowledge for those individuals completing the training. Use of the DOE developed material provides personnel with the information necessary to perform their assigned duties at a predetermined level of expertise. Implementing the training program helps ensure consistent and appropriate training of personnel.

Organizational Relationships and Reporting Structure

The DOE Office of Worker Safety and Health Policy (AU-11) is responsible for approving and maintaining the DOE developed training materials associated with the training program.

The establishment of a comprehensive and effective contractor site radiological control training program is the responsibility of line management and their subordinates. The training function can be performed by a separate training organization, but the responsibility for quality and effectiveness rests with the line management.

Copies of this Appendix may be obtained from the DOE Radiation Safety Training Home Page Internet site.

Instructional Materials Development

Target Audience

Course instructional materials were developed for specific employees who are responsible for knowing or using the knowledge or skills for each course. With this in mind, the participant should never ask the question, "Why do I need to learn this?" However, this question is often asked when the participant cannot apply the content of the program. It is the responsibility of management to select and send workers to training who need the content of the program. When workers can benefit from the course, they can be motivated to learn the content and apply it on their jobs. Care should be taken to read the course descriptions along with the information about who should attend. Participants and DOE facilities alike will not benefit from workers attending training programs unsuitable for their needs.

Prerequisites

A background and foundation of knowledge facilitates the trainee in learning new knowledge or skills. It is much easier to learn new material if it can be connected or associated to what was previously learned or experienced. Curriculum developers who have been involved in preparing instructional materials for the core training know this and have established what is referred to as "prerequisites" for each course.

Certain competencies or experiences of participants were also identified as necessary prior to participants attending a course. Without these competencies or experiences, the participants would be at a great disadvantage and could be easily discouraged and possibly fail the course. It is not fair to the other participants, the unprepared participant, and the instructor to have this misunderstanding.

Training Material

Training materials for the program consist of lesson plans, study guides and handouts. The training content should be presented in its entirety. Overhead transparencies are provided in support of the training content and may be substituted or supplemented with updated or site-specific information.

Supplemental material and training aids may be developed to address site-specific radiological concerns and to suit individual training styles. References are cited in each lesson plan and may be used as a resource in preparing site-specific information and training aids.

Each site is responsible for establishing a method to differentiate the site-specific information from the DOE developed lesson plan material. When additional or site-specific information is added to the text of the core lesson plan material, a method should be used to differentiate site information from DOE developed material.

Exemptions

Qualified personnel can be exempted from training if they have satisfactorily completed training programs, (i.e., facility, college or university, military, or vendor programs) comparable in instructional objectives, content, and performance criteria. Documentation of the applicable and exempted portions of training should be maintained.

Training Program Standards and Policies

Qualification of Instructors

The technical instructor plays a key role in the safe and efficient operation of DOE facilities. Workers should be well qualified and have a thorough understanding of the facility's operation, such as processing, handling, and storage of materials, and maintenance of equipment. Workers should know how to correctly perform their duties and why they are doing them. They should know how their actions influence other worker's responsibilities. Because workers' actions are so critical to their own safety and the safety of others, their trainers should be of the highest caliber. The technical instructor should understand thoroughly all aspects of the subjects being taught and the relationship of the subject content to the total facility. Additionally, the instructor should have the skills and knowledge to employ the instructional methods and techniques that will enhance learning and successful job performance. While the required technical and instructional qualifications are listed separately, it is the combination of these two factors that produces a qualified technical instructor.

The qualifications are based on the best industry practices that employ performance-based techniques to ensure that workers receive the highest quality training possible. This is possible only when technical instructors possess the technical competence and instructional skills to perform assigned instructional duties in a manner that promotes safe and reliable DOE facility operations.

Technical Qualifications

Instructors should possess technical competence (theoretical and practical knowledge along with work experience) in the subject areas in which they conduct training. The foundation for determining the instructor's technical qualifications is based on two factors:

- the trainees being instructed, and
- the subject being presented

The following is an example of a target audience, subject to be taught, and instructor technical qualifications.

TARGET AUDIENCE	SUBJECT BEING TAUGHT	INSTRUCTOR TECHNICAL QUALIFICATIONS
Tritium Facilities Personnel, Visitors, DOE Employees	Tritium Hazards and Safety Training	Demonstrated knowledge and skills in radiation protection, above the level to be achieved by the trainees, as evidenced by previous training/education and through job performance, AND Completion of all qualification requirements for the senior-level radiation protection technician position at the trainees' facility or a similar facility.

Methods for verifying the appropriate level of technical competence may include the review of prior training and education, observation, and evaluation of recent related job performance, and oral or written examination. Other factors that may be appropriate for consideration include DOE, NRC, or other government qualification, certification by the American Board of Health Physics and/or registration by the National Registry of Radiation Protection Technologists, vendor or facility certification, and most importantly, job experience. To maintain technical competence, a technical instructor should continue to perform satisfactorily on the job and participate in continuing technical training.

Instructional Capability and Qualifications

Qualifications of instructional capability should be based on demonstrated performance of the instructional tasks for the specific course requirements and the instructor's position. Successful completion of instructor training and education programs, as well as an evaluation of on-the-job performance, is necessary for verification of instructional capability. Instructional capability qualification should be granted as the successful completion of an approved professional development program for training instructors. The program should contain theory and practice of instructional skills and techniques, adult learning, planning, conducting, and evaluating classroom, simulator, laboratory, and on-the-job training activities.

Illustrated talks, demonstrations, discussions, role playing, case studies, coaching, and individual projects and presentations should be used as the principal instructional methods for presenting the instructional training program. Each instructional method should incorporate the applicable performance-based principles and practices. Every effort should be made to apply the content to actual on-the-job experience or to simulate the content in the classroom/laboratory. The appropriate methodology required to present the instructional content will indicate a required level of instructional qualification and skill.

Current instructors' training, education, and job performance should be reviewed to determine their training needs for particular courses. Based on this review, management may provide exemptions based on demonstrated proficiency in performing technical instructor's tasks.

Through training or experience, technical instructors should be able to*:

- Review instructional materials and modify them to fully meet the needs of the training group.
- Arrange the training facility (classroom/laboratory or other instructional setting) to meet the requirements for the training sessions.
- Effectively communicate, verbally and non-verbally, lessons to enhance learning.
- Invoke student interaction through questions and student activities.
- Use appropriate instructional materials and visual aids to meet the lesson objectives.
- Administer performance and written tests.
- Ensure that evaluation materials and class rosters are maintained and forwarded to the appropriate administrative personnel.
- Evaluate training program effectiveness.
- Modify training materials based on evaluation of training program.

*Stein, F. *Instructor Competencies: The Standards.* International Board of Standards for Training, Performance and Instruction; 1992.

Selection of Instructors

Selection of instructors should be based on the technical and instructional qualifications specified in the Course Specific Information section of this guide. In addition to technical and instructional qualifications, oral and written communication skills, and interpersonal skills, should be included in the process of selecting and approving instructors.

Since selection of instructors is an important task, those who share in the responsibility for ensuring program effectiveness should:

- interview possible instructors to ensure they understand the importance of the roles and responsibilities of technical instructors and are willing to accept and fulfill their responsibilities in a professional manner

- maintain records of previous training, education, and work experience

Procedures for program evaluation will include documentation of providing qualified instructors for generic and site-specific training programs.

Test Administration

A test bank of questions for this training, with site specific information, should be developed and the content validated. As the test banks are used, statistical validation of the test bank should be performed in order to fully refine the questions and make the tests as effective as possible. The questions contained in the test bank are linked directly to the objectives for each course. In this way, trainee weaknesses can be readily identified and remedial procedures can be put into place. The test outcomes can also be used to document competence and the acquisition of knowledge.

The test banks should also be used by the instructors to identify possible weaknesses in the instruction. If numerous trainees fail to correctly answer a valid set of questions for an objective, the instruction for that objective needs to be reviewed for deficiencies.

Written examinations should generally be used to demonstrate satisfactory completion of theoretical classroom instruction. The following are some minimal recommendations for the test banks and tests:

- Tests are randomly generated from the test bank.
- Test items represent all objectives in the course.
- All test bank items are content-validated by a subject matter expert.
- Test banks are secured and are not released either before or after the test is administered.
- Trainees receive feedback on their test performance.
- Test banks undergo statistical analysis.

- For the first administrations of tests, a minimum passing score of 80% should be required for a passing score. As statistical analysis of test results is performed, a more accurate percentage for a passing score should be identified.

Test administration is critical in accurately assessing the trainee's acquisition of knowledge being tested. Generally, the following rules should be followed.

- Tests should be announced at the beginning of the training sessions.
- Instructors should continuously monitor trainees during completion of tests.
- All tests and answers should be collected at the conclusion of each test.
- No notes can be made by trainees concerning the test items.
- No talking (aside from questions) should be allowed.
- Answers to questions during a test should be provided but answers to test items should not be provided or alluded to.
- Where possible, multiple versions of each test should be produced from the test bank for each test administration.
- After test completion, trainees may turn in their materials and leave the room while other trainees complete their tests.
- Trainee scores on the tests should be held as confidential.

Program Records and Administration

Training records and documentation shall meet the requirements of 10 CFR 835.704.

Training Program Development/Change Requests

All requests for program changes and revisions should be sent to the DOE Technical Standards Program using the form "Document Improvement Proposal" provided at the conclusion of the material, as provided on the DOE Technical Standards website.

Audits (internal and external)

Internal verification of training effectiveness should be accomplished through senior instructor or supervisor observation of practical applications and discussions of course material. All results should be documented and maintained by the organization responsible for Radiological Control training.

The training program materials and processes is evaluated on a periodic basis by DOE-HQ. The evaluation includes a comparison of program elements with applicable industry standards and requirements.

Evaluating Training Program Effectiveness

Verification of the effectiveness of Radiological Training for Tritium Facilities should be accomplished by surveying a limited subset of former students in the workplace. This evaluation should include observation of practical applications and discussion of the course material. DOE has issued guidelines for evaluating the effectiveness of radiological training through the DOE Operations Offices and DOE Field Offices. These guidelines are available as an attachment to the Program Management Guide of DOE-HDBK-1122-99, Radiological Control Technician Training.

For additional guidance, refer to DOE STD 1070-94, A Guide for Evaluation of Nuclear Facility Training Programs. The guidelines contained in these documents are relevant for the establishment and implementation of post-training evaluation programs.

Course-Specific Information

Purpose

This section of the program management guide is to assist those individuals assigned responsibility for implementing the Radiological Training for Tritium Facilities. Standardized implementation of this training helps ensure consistent and appropriate training for all personnel.

Course Goal

Upon completion of this training, the student will have a basic understanding of the characteristics of tritium and understand the precautions and safeguards needed for working in a tritium facility.

Target Audience

Individuals who have assigned duties in tritium facilities.

Course Description

This course illustrates and reinforces the skills and knowledge needed to provide personnel with an understanding of the characteristics of tritium and the precautions needed for working with or around it in a DOE facility. This course is designed to meet Article 663 of the RCS for individuals who have assigned duties in a tritium facility.

Prerequisites

Training which is considered commensurate with site-specific hazards. Radiological Worker II or the equivalent is recommended prior to receiving or concurrently with this tritium facilities safety training.

Some of those basic concepts contained in Radiological Worker II are used extensively throughout this training.

Length

2 - 4 hours (depending on site-specific information)

Test Bank

Test banks, as applicable, should be developed by the sites, incorporating site-specific information.

Retraining

Retraining is not required for this course.

Instructor Qualifications

Instructors of this course have a major role in making it successful and meeting the specified objectives. Instructors should have related experience and be technically competent. In this course it is imperative that the instructor have the background and experience of working in a tritium facility. Instructors should be able to relate their own work experience to the workers in a tritium facility. Instructors should be able to answer specific questions and use a variety of instructional material to meet the objectives.

<u>Education:</u>

Minimum of B.S. degree in Health Physics or related discipline is <u>preferred</u>.

<u>Experience:</u>

At least five years of applied radiological protection experience in an operating radiological facility is preferred. The areas of experience should include:

- Tritium hazards
- Radiological controls associated with tritium
- Conducting surveys and monitoring for tritium

Comprehensive knowledge of Federal regulations and guidance and best nuclear industry practices pertaining to radiological protection.

Materials Checklist

The following checklist should be used to ensure all training materials are available. The Program Management Guide, Instructor's Guide, Student's Guide, and Lessons Learned section are provided in Word 2003 format. The Overhead Transparencies are provided in Power Point 2003 for windows format.

- Program Management Guide
- Instructor's Guide
- Student's Guide
- Overhead Transparencies

The following checklist should be used before training is provided to ensure that equipment is available and working.

- Overhead projector
- Screen
- Flip chart
- Markers

APPENDIX F2: Instructor's Guide

Core Course Material

Course Goal:	Upon completion of this training the student will have a basic understanding of the characteristics of tritium and understand the precautions and safeguards needed for working in a tritium facility as outlined in the course objectives.
Target Audience:	Individuals who have assigned duties in Tritium Facilities.
Description:	This course illustrates and reinforces the skills and knowledge needed to provide personnel with an understanding of the characteristics of tritium and the precautions needed for working with or around it in a DOE facility. This course is designed to be consistent with Article 663 of the DOE RSC for individuals who have assigned duties in a tritium facility.
Prerequisites:	Training which is considered commensurate with site-specific hazards. Radiological Worker II or the equivalent is recommended prior to receiving or concurrently with this tritium facilities safety training. Some of the basic concepts contained in Radiological Worker II are used extensively throughout this training.
Length:	2 - 4 hours (depending on site-specific information)

Course Objectives:

EO1 IDENTIFY the following properties of tritium:
- physical/chemical
- radioactive.

EO2 IDENTIFY sources of tritium:
- natural
- byproduct
- weapons.

EO3 IDENTIFY uses of tritium:
- weapons applications
- research
- fusion energy production

- industrial/commercial.

EO4 IDENTIFY modes of tritium exposure:
- inhalation
- ingestion
- absorption

EO5 IDENTIFY the biological effects of tritium:
- biological half-life
- dose.

EO6 IDENTIFY the radiological control methods for tritium:
- engineered
- administrative
- surface contamination limits
- personal protective equipment
- application of ALARA principles.

EO7 IDENTIFY methods for monitoring for tritium:
- air sampling for worker safety
- contamination surveys
- bioassay.

EO8 IDENTIFY tritium waste minimization and handling techniques. (Site-Specific)

EO9 IDENTIFY alarms and proper response to abnormal conditions in the tritium facility. (Site-Specific)

Training Aids: Overhead transparencies (may be supplemented or substituted with updated or site-specific information)

Equipment Needs:
- Overhead projector
- Screen
- Flip chart
- Markers

Student Materials: Students Guide

LESSON SUMMARY

Introduction	
Welcome students to the course.	Show Title OT-1
Introduce self and instructor team.	
Define logistics.	
- safety briefing - exits - restrooms - hours - breaks - sign-in sheets - test - accountability - end of course evaluation	
Remind the participants that they need to have completed Radiological Worker Training prior to this course. They should be familiar with terms like rem, contamination, etc.	
Course Goal	
At the end of this course, the participant should be able to demonstrate a basic understanding of the characteristics of tritium and precautions for working in a tritium facility as outlined in the course objectives. We will attempt to answer the following questions:	Show OT 2
- Tritium, what is it? - How does it behave? - What are the benefits and hazards? - How do we control it?	
State Course Objectives	Show OT 3, OT 4, and OT 5

Course Content

Briefly review the content of the course, noting that there is a logical sequence ("flow"), and that as you present the material you will relate the material covered to the circumstances that they can expect to find in the facility workplace and procedures. (You will be inserting site-specific tritium information.)

Show OT 6

1. Properties of Tritium
2. Sources and Uses of Tritium
3. Modes of Exposure and Biological Behavior of Tritium
4. Radiological Controls for Tritium
5. Monitoring for Tritium
6. Tritium Waste Minimization and Handling
7. Response to Abnormal Conditions in the Tritium Facility
8. Lessons Learned
9. Summary and Review

Notes

I. PROPERTIES OF TRITIUM - *What is tritium?*

Show OT 7

Tritium is a radioactive isotope of hydrogen (H). Hydrogen is the lightest and most abundant element in the universe. Hydrogen has only one proton in its nucleus. Tritium has, in addition to the single proton, two neutrons in its nucleus. This makes it three times heavier than the most common form of hydrogen.

Show OT 8

A. Isotopes of Hydrogen

Show OT 9

1. Protium (H) (99.985% natural abundance)
2. Deuterium = hydrogen + 1 neutron (D) (0.015% natural abundance)
3. Tritium = hydrogen + 2 neutrons (T)

B. Symbol for Tritium

Show OT 10

Tritium is designated as:

$$\mathrm{T}, \mathrm{H\text{-}3}, \text{or } {}^{3}_{1}\mathrm{H}$$

C. Chemical Properties of Hydrogen/Tritium

Tritium "behaves" just like hydrogen chemically because it has one proton and one electron like ordinary hydrogen.

Reaction rates may vary from hydrogen.

	Notes
1. Substitution	Show OT 11
Tritium atoms can easily substitute for hydrogen atoms. Examples:	
a. elemental hydrogen (tritium gas, HT, DT, or T_2)	
b. tritiated water (tritium oxide: HTO, DTO, or T_2O)	Show OT 12
c. organically bound tritium: These compounds may result in more dose per intake than tritiated water.	Show OT 13 - Tritium can replace a loosely bounded hydrogen atom of an organic molecule.
d. metals and tritium (metal tritides): These compounds may result in more dose per intake than tritiated water. Accordingly, additional controls may be needed, such as special air monitoring or enhanced personal protective equipment.	Show OT 14 - Hydride storage of elemental tritium is common. Uranium Hydride is most frequently used for this purpose. Tritium is released by heating the metals to 400 degrees C for 1 atm. dissociation pressure. Refer to RCTP 2001-02 and Health Physics Journal September 2001 for more information.
2. Solubility	Instructor should emphasize that the biological exchange of tritium with hydrogen in the body is of concern.
Exchanges with hydrogen in organic and other materials (oils, plastics, etc.)	
3. Flammability	Show OT 15 - The range of flammable concentrations of tritium in air is between 4% and 75% by volume in air.
Tritium gas is flammable and can explode under certain conditions.	

Notes

D. Physical Properties

1. Diffusion

Tritium gas is lighter than air and diffuses rapidly in air.

Show OT 16

Show OT 17

2. Permeability

Show OT 18

Tritium gas permeates through most materials, that is, it travels through them by way of spaces or interstices in them. The rate depends upon the material and its thickness. Tritium's radioactive, chemically reducing and diffusive properties result in degradation of many useful polymeric materials, pump oils, plastics, and o-rings. This can cause loss of mechanical functions in certain situations within a short period of time.

$$^{3}_{1}\text{H decays to } ^{3}_{2}\text{He} + \beta^{-} + \overline{\upsilon}$$

E. Radioactive Decay of Tritium

$$^{3}_{2}He = \text{ Helium}$$

$$\overline{\upsilon} = \text{anti-neutrino}$$

$$\beta^{-} = \text{ beta minus}$$

Show OT 19

Tritium "decays" by emitting a beta particle and becoming an atom of helium.

1. Beta Energy

   A very low energy beta is emitted.

Max energy = 18.6 keV
Ave. energy = 5.7 keV

2. Beta Range

   Tritium is not an "external" radiation hazard.

Instructor should emphasize the radiological controls are to keep tritium outside the body. Explain what is meant by "external" radiation hazard.

   a. Travels less than 1/4 inch in air

   b. Cannot penetrate through the dead layer of the skin

   c. Cannot penetrate clothing or gloves

3. Half-life

   Radioactive half-life = 12.3 years.

Show OT 20

F. X-Ray Production

Tritium betas can produce low energy x rays. Because the beta particle is of such low energy the x rays it may produce are not very penetrating and are not normally considered a hazard.

G. Fusion

Fusion is the process of two nuclei joining together. The energy source in most stars is nuclear fusion of hydrogen isotopes.

SOURCES AND USES OF TRITIUM

How is tritium produced and what is it used for?

A. Sources of Tritium

There are three primary sources of tritium in our environment. Tritium is present in our environment from both man-made and natural sources as discussed below. Natural tritium is indistinguishable from man-made tritium.

1. Natural Sources

Tritium occurs naturally. It is formed by the reactions between cosmic rays and the nitrogen in the upper atmosphere. Nitrogen makes up 80% of the earth's atmosphere.

Cosmic rays generate approximately 4 million curies of tritium per year. With tritium being continually produced and at the same time decaying, the natural tritium in our environment is about 70 million curies.

Notes

$${}^{2}_{1}\mathrm{H} + {}^{2}_{1}\mathrm{H} \rightarrow {}^{3}_{2}\mathrm{H}e + {}^{1}_{0}\mathrm{n}$$

$$1\ \mathrm{mev\ per\ fusion}$$

Show OT 21

$${}^{3}_{1}\mathrm{H} + {}^{2}_{1}\mathrm{H} \rightarrow {}^{4}_{2}\mathrm{H}e + {}^{1}_{0}\mathrm{n}$$

$$17\ \mathrm{mev\ per\ fusion}$$

Show OT 22

Show OT 23

Modes of production:

1. The nitrogen nucleus captures a neutron and decays to tritium and carbon.

$${}^{14}_{7}\mathrm{N} + {}^{1}_{0}\mathrm{n}\ \mathrm{yields}$$

$${}^{3}_{1}\mathrm{H} + {}^{12}_{6}\mathrm{C}$$

$${}^{2}_{1}\mathrm{H} + {}^{2}_{1}\mathrm{H}\ \mathrm{yields}$$

$${}^{3}_{1}\mathrm{H} + {}^{1}_{1}\mathrm{H}$$

2. Two hydrogen-2 atoms (deuterium) combine to form tritium and normal hydrogen.

Notes

2. Byproduct of Reactor Sources

   a. Lithium-7 captures a neutron and decays with tritium as a product.

   b. Boron captures a neutron and decays with tritium as a product.

   c. Activation of deuterium in water.

   d. Ternary fission - A fission resulting in three fission products, one of which is tritium. This process has a 0.1% yield.

   The production of tritium from power reactors around the world is less than one-half that naturally formed (approximately 1 to 2 million curies a year).

$$^{7}_{3}\mathrm{Li} + {}^{1}_{0}\mathrm{n} \text{ yields}$$

$$^{3}_{1}\mathrm{H} + {}^{1}_{0}\mathrm{n} + \alpha$$

$$^{10}_{5}\mathrm{B} + {}^{1}_{0}\mathrm{n} \text{ yields } {}^{3}_{1}\mathrm{H} + 2\alpha$$

Show OT 24

$$^{2}_{1}\mathrm{H} + {}^{1}_{0}\mathrm{n} \text{ yields } {}^{3}_{1}\mathrm{H}$$

3. Weapons Testing as a Source

   The amount of tritium in the world from weapons testing has been steadily declining since the 1970's when atmospheric testing was curtailed.

   Atmospheric testing from 1945-1975 produced about 8 billion curies. This has decayed to about 400 million curies.

Show OT 25

Notes

4. DOE Production of Tritium

DOE has produced tritium at the Savannah River Site with the use of a reactor. Tritium is commercially available from Canada and the European Community for non-weapons use.

$${}^{6}_{3}\mathrm{Li} + {}^{1}_{0}\mathrm{n} \text{ yields } {}^{3}_{1}\mathrm{H} + \alpha$$

B. Uses of Tritium

Show OT 26

1. Consumer Products

Instructor - emphasize that tritium has use in everyday activities.

   a. <u>Gaseous tritium light sources:</u>
      - exit signs (1 Ci to tens of Ci's)
      - aviation landing aids (30-165 Ci per light)

      (1 curie of tritium has a mass of approximately 0.1 mg)

Approximately 100,000 Ci per runway

   b. <u>Luminizing industry</u>: Self-luminous compounds for dials (several mCi's) and controls as well as other general industry uses.

2. Research - Tritium Labeling

Show OT 27

Tracers for medical and laboratory research.

3. Department of Energy

Show OT 28

   a. <u>Weapons development and applications</u>

   b. <u>Fusion energy</u>: As a fuel source

Notes

III. MODES OF EXPOSURE AND BIOLOGICAL BEHAVIOR OF TRITIUM - *How can I be exposed to tritium and what would it do to me?*

Show OT 29

A. External Dose

Show OT 30

- Tritium is not an external hazard.
- Tritium is not a dose concern if it is located outside the body.
- Tritium betas will not penetrate a dosimeter.
- We are interested only in tritium inside the body.

The tritium beta when it is stopped (particularly in high atomic numbered materials, e.g., lead) will produce a low energy x-ray. This x-ray cannot penetrate into the body, because of the low energy of the x-ray.

There may be rare circumstances where external dose should be considered (i.e., using large quantities of tritium).

B. Internal Dose

Tritium is an "internal" radiation hazard.

C. Modes of Exposure

Show OT 31

Tritium can deliver a radiation dose if it gets inside our bodies. Modes of entry include:

- inhalation
- absorption
- ingestion

1. Inhalation

Show OT 32

a. Tritium gas: is only slightly incorporated into the body when inhaled. Most tritium gas inhaled is subsequently exhaled. There are other chemical and physical processes to convert tritium gas to tritiated water.

An extremely low percent (in the order of 0.005%) of tritium gas inhaled is converted to tritiated water prior to being exhaled.

b. Tritiated water vapor: Nearly 100% of tritiated water vapor inhaled is incorporated into body fluids/tissues.

c. Hazard: Exposure to tritiated water is approximately 10,000 times more hazardous than exposure to elemental tritium gas.

Tritiated water vapor is more hazardous due to its ability to be almost 100% assimilated, increasing the dose received by the individual.

d. Special Tritium Compounds: consist of organically bound tritium and tritium particulate aerosols.

Exposure to organically bound tritium can be up to approximately 13 times more hazardous than exposure to HTO.

Exposure to tritium particulate aerosols can deliver up to approximately 20 times more dose to the whole body than exposure to HTO. The dose to the lungs from tritium particulate aerosols could be 2 orders of magnitude higher than from HTO.

2. Ingestion

Ingestion may occur by eating, drinking, chewing tobacco, and applying makeup where tritium contamination is present. Always wash hands thoroughly when leaving areas where there is a potential for contamination, and never eat, drink etc. where tritium contamination may be present.

Notes

3. Absorption

Show OT 33

Absorption is also a hazard because an individual can receive, in certain situations, 1/3 of their uptake from absorption through the skin if not properly using personal protective equipment.

The assumption is that an additional 50% of the inhalation intake is absorbed through the skin.

a. Tritium gas: There is negligible skin absorption for tritium gas.

b. Tritiated water: Tritiated water can be absorbed through the skin. It has been observed that moisture on hands enhances absorption.

This is an important fact for maintenance workers to know.

c. Solvents: Some solvents (organically bound tritium) can also go through the skin.

Insert site specific information concerning gloves.

Tritium will penetrate gloves; therefore gloves should be changed at a prescribed routine basis.

D. Biological Behavior of Uptakes of Tritium

Tritiated water in the body acts just like water.

1. Distribution

Show OT 34

Tritiated water is rapidly and uniformly distributed throughout the entire body. The Committed Effective Dose from an uptake of one curie of tritiated water is 63 rem. Soluble

Typical production reactor coolant has approximately 10 Ci/L of tritium. So 100 ml of such coolant has about 1 Ci of tritium. If approximately 8 ml was ingested a committed effective dose of 5 rem (the annual limit) would result.

organically bound tritium behaves in the body in a similar manner.

Stable tritium particulate aerosols and insoluble organically bound tritium behave in the body in a similar manner as the particulate to which the tritium is bound.

2. Biological Half-life

Show OT 35

Half of the HTO is eliminated from the body in about 10 days. Through normal biological processes, it would take 70 days or more to eliminate 99% of the assimilated tritiated water.

Stable tritium particulate aerosols and insoluble organically bound tritium behave in the body in a similar manner as the particulate to which the tritium is bound and would have a longer half-life.

E. Medical Treatment

Show OT 36

Insert site-specific policy.

The biological half-life can be shortened by increasing the water elimination in the body. Therefore, individuals who have uptakes of tritium are encouraged to drink water. Drinking copious amounts of water should not be done without a physician's guidance. Certain medical conditions may be affected by liquid intake.

IV. RADIOLOGICAL CONTROLS FOR TRITIUM - *How can I protect myself from exposures to tritium?*

Show OT 37

Tritium can be present in a variety of chemical forms. Ongoing research indicates that the form of tritium which gives the highest dose (per unit intake) is tritium particulate aerosols. However, these compounds are not found in the quantities and various locations as tritiated water. If we can rule out special tritium compounds as being of concern, we assume it may be tritiated water.

Show OT 38

The preferred hierarchy of control is as follows:

- engineered
- administrative
- personal protective equipment

A. Engineered Controls

Show OT 39

1. Containment and Confinement

Containment or confinement is a series of physical barriers, minimizes exposure of workers.

Confinements such as gloveboxes are almost always used when handling large quantities of tritium. However, hoods are acceptable for handling small quantities, such as in a laboratory.

Instructor should insert site specific information concerning what is considered large or small quantities for handling tritium.

2. Airflow

Show OT 40

Maintaining negative ventilation is essential for the safe operation of a tritium facility. Airflow should be from areas of LEAST to MOST contamination.

May need to use site specific terminology.

3. Local Exhaust Ventilation

The primary advantage of local exhaust ventilation techniques is the removal of airborne tritium, regardless of its release rate or chemical or physical form. In addition, these techniques use relatively low volume rates compared to normal ventilation requirements.

4. Dilution Ventilation

Dilution ventilation is the once-through flow technique of exchanging outside air for inside air for comfort and the reduction of airborne sources.

Tritium is considered valuable and needs to be recovered. (i.e. not to send up the stack)

5. Storage

Tritium can be stored in storage beds. Metal tritide and uranium hydride are the most common for these storage systems. Tritium is generally released by heating the metal tritide.

B. **Administrative Controls**

Show OT 41

There are many administrative controls to reduce doses. The following are just a few that should apply to all sites:

Ask students to name other ways to prevent or reduce doses.

Insert site specific posting requirements.

- limitation of access time
- procedures/RWPs
- postings

For tritium and tritium compounds, 10 CFR Part 835 Appendix D requires posting contamination areas based on removable contamination values of 10,000 dpm/100cm^2.

C. Personal Protective Equipment

Show OT 42

1. Air Supplied Suits

Because of the absorption through the skin associated with the use of respirators and other breathing apparatus, air-supplied plastic suits that completely enclose the body are widely used by facilities that handle large quantities of tritium.

2. Protective Clothing

Discuss site specific requirements

Protective clothing (PC), or anti-contamination clothing (anti-Cs), is used to minimize the spread of contamination from contaminated to clean areas.

In many operations, the hands and forearms of workers are vulnerable to contact with tritium surface contamination. The proper selection of gloves and glove materials is important. In many instances a plastic/water proof suit is required.

Insert site specific information concerning selection of PCS.
Even with a plastic or water-proof PC suit, a stay time may be assigned due to the ability of tritium to permeate through plastic.

V. MONITORING FOR TRITIUM - *How do I know if tritium is present?*

Show OT 43

Notes

A. Personnel Monitoring

Show OT 44

1. External Dose

Insert site specific requirements

Dosimeters are not typically used to monitor for radiological doses resulting from tritium. The weak energy beta radiation will not penetrate the dosimeter.

(insert any site specific monitoring procedures)

2. Internal Dose

Show OT 45

The best method used to determine if an individual has an uptake of HTO or soluble organically bound tritium is by bioassay (urinalysis). Routine urine samples, collected at some predetermined frequency and counted for the tritium content, provide a very sensitive measurement of tritium in the body. This is especially true if the time of uptake is known.

Air sampling results may be used to assess dose from other types of special tritium compounds.

3. Routine Versus Non-Routine Bioassay Monitoring

1. Prospective: Prospective bioassay is routine bioassay.

   a. Routine: Routine urinalysis is conducted on a preset periodic basis.

   b. Non-routine or Special: Non-routine bioassay is done whenever a potential exposure to tritium is suspected.

(insert site specific bioassay procedures)

B. Workplace Monitoring

Show OT 46

Air monitoring and surface contamination surveys are used to verify that loose contamination is not present. They provide an early indication of potential problems.

1. Airborne Tritium Monitoring

Show OT 47

Airborne tritium monitoring is used for:

Insert and discuss site specific information concerning equipment.

a. Prompt detection of airborne contamination for worker protection;

b. Determination of the status of processes; and

c. Identification of any leaks in primary or secondary containments or confinements.

2. Contamination Surveys

Despite contamination control measures, tritium is easily spread.

Show OT 48

All workplaces shall be surveyed for contamination levels on a regularly scheduled basis. The frequency of such surveys will depend on the potential for dispersement of the tritium-contaminated material in the area and the quantity of tritium in the area. During routine surveys, all work enclosures, work surfaces, floors, equipment, etc., within the workplace should be surveyed.

At the surface contamination levels in 10 CFR Part 835 Appendix D, tritium is difficult to measure directly because the low-energy beta is readily absorbed in air and the window of the detector. Normal frisking methods (use of pancake probe) will not detect tritiated water.

Show OT 49

Surfaces are normally wiped (smeared) with a small paper, either dry or moistened, which will pick up loose tritiated material. The wipe is then normally counted by liquid scintillation techniques.

VI. TRITIUM WASTE MINIMIZATION AND HANDLING- *Why is it important and how do you minimize tritium waste?*

Show OT 50

This material maybe covered in Radiological Worker I or II training. Emphasis should be on site specific information. It is not necessary to repeat information.

(insert site specific controls)

Methods below are given as a reference only.

A. Minimizing Tritium-Contaminated Waste

1. Avoid generating wastes by not bringing unnecessary material into the controlled area.

2. Whenever possible, avoid use of porous materials or those known to be highly permeable to tritium.

3. Designate an area to store contaminated tools for reuse or consider all tools in the area to be contaminated.

4. Plan work so that, whenever possible, construction and clean maintenance can be done in a clean area.

5. When transporting items contaminated with tritium, adequate contamination control such as wrapping in plastic or placing in sealed containers should be considered.

Insert site-specific information concerning procedures for transporting tritium contaminated items.

B. Minimizing Mixed Waste

(insert site specific controls)

Methods below are given as a reference only.

1. Use non-hazardous cleaning materials for decontamination whenever possible.

2. Segregate radioactive-only from hazardous-only at the source.

3. Explore the use of other materials which are non-hazardous for use in radiological areas to prevent the generation of mixed waste.

VII. RESPONSE TO ABNORMAL CONDITIONS IN THE TRITIUM FACILITY- *What should I do in an emergency involving tritium?*

To properly deal with unexpected adverse situations occurring in a tritium facility, a well-thought-out response program and personnel trained to execute the response should be in place.

Show OT 51
This material may be covered in Radiological Worker I or II training. Emphasis should be on site specific information. It is not necessary to repeat information.

Notes

A. Abnormal Conditions

Abnormal conditions in a tritium facility could include:

- fire/explosion
- natural disaster
- tritium releases
- other hazards

Personnel found contaminated should follow site specific decontamination procedures which would typically include showering with cold water and the use of mild detergents.

B. Facility Alarms

(insert site specific alarms and emergency response procedures)

C. Facility Emergency Responses

Insert site specific emergency plan and hazards assessment documents.

(insert site specific alarms and emergency response procedures)

LESSONS LEARNED

Show OT 52

(insert site specific lessons learned)

SUMMARY AND REVIEW

Show OT 53

OVERHEADS

(in Power Point folder)

Overhead 1

Radiological Training for Tritium Facilities

Overhead Transparencies

DEPARTMENT OF ENERGY • UNITED STATES OF AMERICA

Overhead 2

- ♦ Tritium - what is it?
- ♦ How does it behave?
- ♦ What are the benefits and hazards?
- ♦ How do we control it?

Overhead 3

Radiological Training for Tritium Facilities

Objectives

1. **IDENTIFY the following properties of tritium**
   - **physical/chemical**
   - **radioactivity**

2. **IDENTIFY sources of tritium**
   - **natural**
   - **by-product**
   - **weapons**

3. **IDENTIFY uses of tritium**
   - **weapons applications**
   - **research**
   - **fusion energy production**
   - **industrial/commercial**

Overhead 4

Objectives

(Continued)

4. **IDENTIFY modes of tritium exposure**
   - **inhalation**
   - **ingestion**
   - **absorption**

5. **IDENTIFY biological effects of tritium**
   - **dose**
   - **biological half -life**

6. **IDENTIFY radiological control methods for tritium**
   - **engineered**
   - **surface contamination limits**
   - **personal protective equipment**
   - **application of ALARA principles**
   - **administrative**

Overhead 5

Objectives

(Continued)

7. IDENTIFY methods for monitoring tritium
    - air sampling for worker safety
    - contamination surveys
    - bioassay
8. IDENTIFY tritium waste minimization and handling procedures
9. IDENTIFY alarms and proper response to abnormal conditions in the tritium facility

Overhead 6

Outline

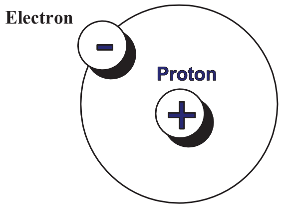
1. Properties of tritium
2. Sources and uses of tritium
3. Modes of exposure and biological effects of tritium
4. Radiological controls for tritium
5. Monitoring for tritium
6. Tritium waste minimization and handling
7. Response to abnormal condition in the tritium facility
8. Lessons learned
9. Summary and Review

Overhead 7

What is Tritium?
Properties of Tritium
H-3

Overhead 8

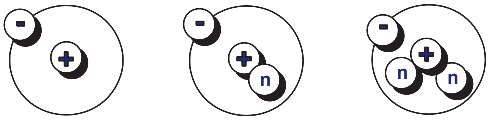
All hydrogen atoms have one proton (H)



Overhead 9

The number of neutrons may vary

ISOTOPES of hydrogen



Protium H — No neutrons — **H**

Deuterium D — 1 neutron — **D**

Tritium T — 2 neutrons — **T**

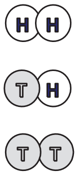
Overhead 10

Tritium is designated as:

T or H-3 or $^{3}_{1}H$

Tritium can easily substitute for protium (H)

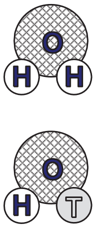
Overhead 11



Normal Hydrogen Gas	H_2
Tritiated Gas	HT
Tritium Gas	TT or T_2

Tritium behaves just like hydrogen

Overhead 12



Normal Water	H_2O
Tritiated Water	HTO

Overhead 13

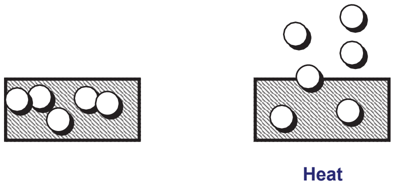
Tritium can form organic compounds

$C_6H_{12}+T_2$ "Benzene" → $C_6H_{11}T+HT$ or

CH_4+T_2 "Methane" → CH_3T+HT

Overhead 14

Tritium can be captured in metallic compounds called "Hydrides" and then released by heating



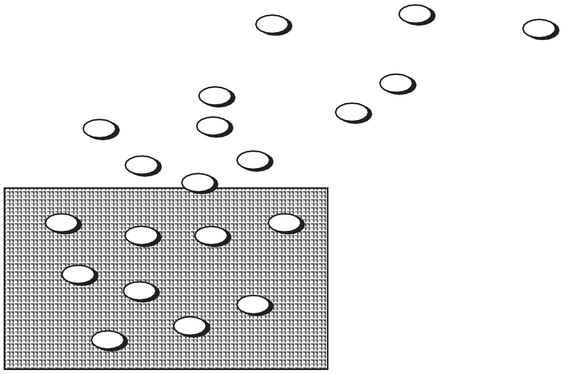
Overhead 15

Hydrogen/tritium gas may be flammable

Overhead 16

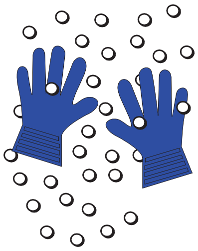
Hydrogen/tritium gas is lighter than air

Hydrogen/tritium gas rapidly disperses in air



Overhead 17

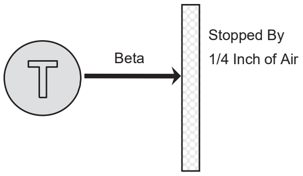
Hydrogen/tritium gas and tritiated water penetrates through most materials, including gloves



Overhead 18

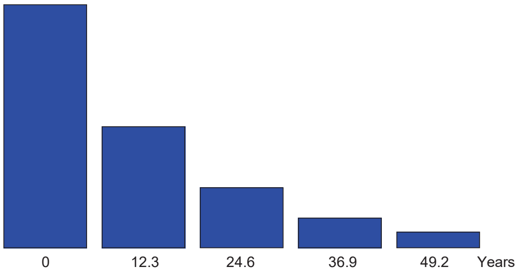
Tritium decays by emitting a weak beta particle

Overhead 19



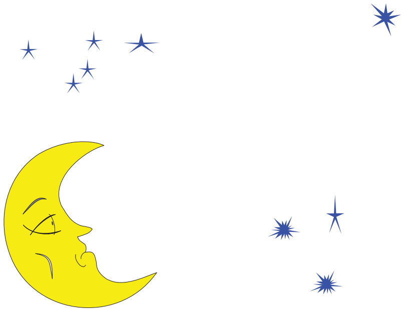
The radioactive half-life of tritium is 12.3 years

Overhead 20



Overhead 21

Fusion of hydrogen and other light element isotopes fuels the stars

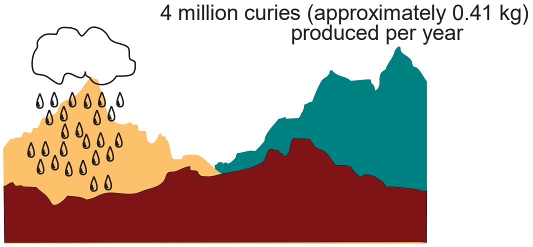


Overhead 22

Sources of Tritium

Overhead 23

Tritium is produced naturally by cosmic rays and is carried to the earth's surface by rain



World inventory of natural tritium from cosmic ray interactions is approximately 70 million curies.

Overhead 24

Tritium is formed in reactors by ternary fission and activation of light elements

Commercial production - **1 million curies per year**

Nuclear reactors - **1-2 million curies per year**

Overhead 25

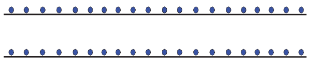
Environmental tritium from atmospheric testing will be at natural levels by about 2030

The world inventory of tritium from atmospheric testing is approximately 200 million curies .

Overhead 26

Tritium is used commercially in a variety of products

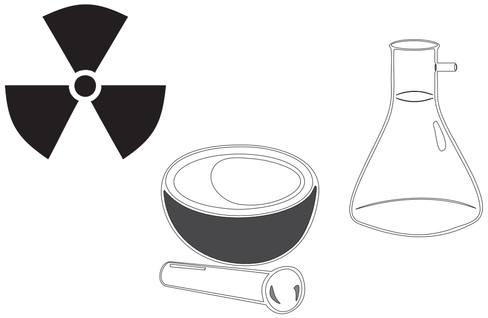
Illuminating Signs



Runway Lights

Overhead 27

Tritium is used extensively in biology and medical research



Overhead 28

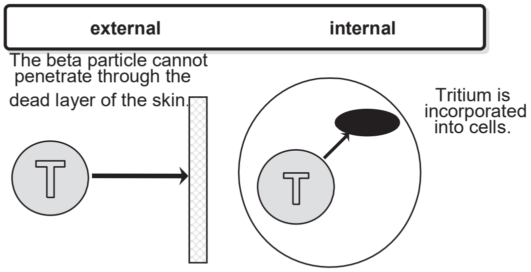
Tritium is the fuel in fusion reactions

Overhead 29

Modes of Exposure
and
Biological Behavior of
Tritium

Overhead 30

Tritium is NOT an external radiation hazard-It is an internal hazard when taken into the body



external

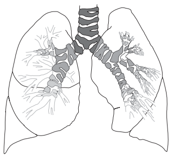
internal

The beta particle cannot penetrate through the dead layer of the skin.

Tritium is incorporated into cells.

Overhead 31

The main pathways for uptake of tritium are:

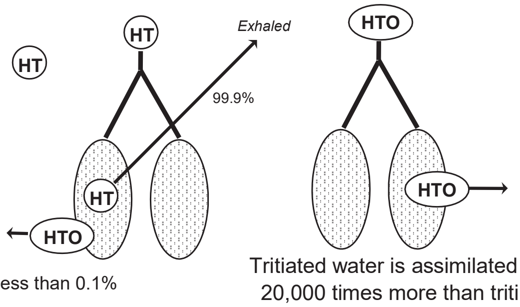


Inhalation

Absorption

Overhead 32

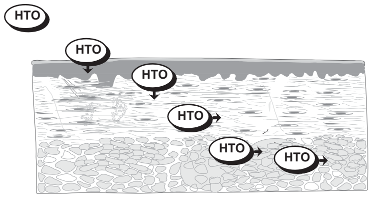
Tritiated water vapor is more hazardous than tritium gas, as it is readily incorporated into the body



Tritiated water is assimilated in the body
20,000 times more than tritium gas

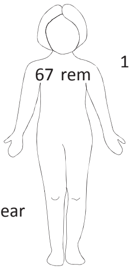
Overhead 33

Tritiated water can also be absorbed through skin into body water.



Overhead34

Tritium taken into the body is distributed throughout the body.

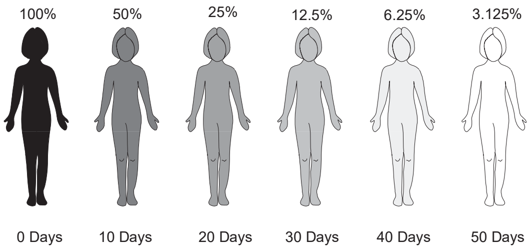


1 Curie of tritiated water

10 CFR Part 835
establishes 5 rem/year
as the occupational
dose limit.

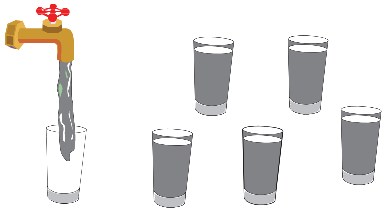
Overhead 35

The biological half-life of tritium is about 10 days.



Overhead36

The elimination of tritium from the body can be increased by increasing water intake



Overhead 37

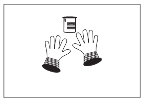
Radiological Controls

Engineered
Administrative
Personal Protective Equipment

Overhead 38

Engineered controls are preferred over administrative ones.

Engineered

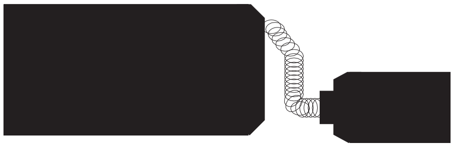


Glovebox

Administrative

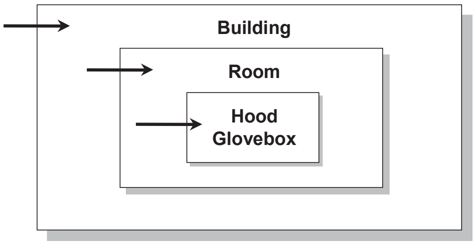
Overhead 39

Larger quantities of tritium are handled in engineered containment systems.



Overhead 40

Airflow should be from areas of LEAST to MOST contamination in tritium facilities



Overhead 41

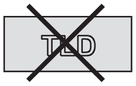
Access controls may also be administrative.

Overhead 43

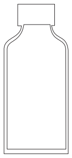
Monitoring for Tritium

Overhead 44

Your dosimeter is NOT used to monitor tritium dose which results from assimilation of tritiated water in the body.



TLDs are used for external exposure. Tritium betas cannot penetrate the TLD case.



Bioassay is used to determine tritium dose.

Overhead 45

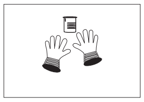
Urinalysis is used to monitor for possible uptakes of tritium.

Both: Routine and
Non-Routine

Overhead 46

Workplace monitoring for tritium

Monitor



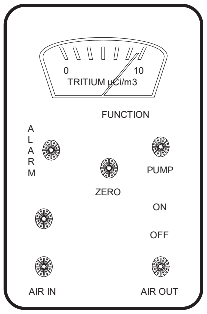
Glovebox

Swipes

Personal Air Sampler

Overhead 47

Area fixed air monitors and tritium sniffers are used to detect tritium leaks/releases.



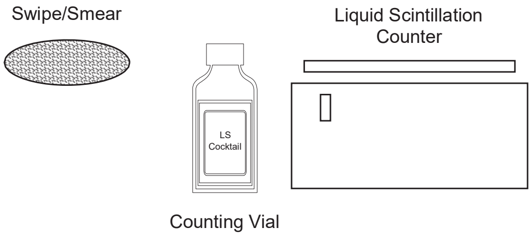
Overhead 48

Tritium spreads easily, therefore contamination control requires constant vigilance.

Tritium is like sand at the beach...
It gets into everything!

Overhead 49

Tritium contamination is detected and measured by taking swipes/smears.



Overhead 50

Tritium Waste Minimization and Handling

- Avoid generating waste by not bringing unnecessary material into the controlled area.
- Whenever possible, avoid use of porous materials or those known to be highly permeable to tritium.
- Designate an area to store contaminated tools for reuse.
- Plan work so that, whenever possible, construction and clean maintenance can be done in a clean area.

Overhead 51

Response to Abnormal Conditions

Overhead 52

Lessons Learned

Overhead 53

Summary and Review

APPENDIX F3: Student's Guide

Course Objectives:

EO1 IDENTIFY the following properties of tritium:
- physical/chemical
- radioactive.

EO2 IDENTIFY sources of tritium:
- natural
- byproduct
- weapons.

EO3 IDENTIFY uses of tritium:
- weapons applications
- research
- fusion energy production
- industrial/commercial.

EO4 IDENTIFY modes of tritium exposure:
- inhalation
- ingestion
- absorption

EO5 IDENTIFY the biological effects of tritium:
- biological half-life
- dose.

EO6 IDENTIFY the radiological control methods for tritium:
- engineered
- administrative
- surface contamination limits
- personal protective equipment
- application of ALARA principles.

EO7 IDENTIFY methods for monitoring for tritium:
- air sampling for worker safety
- contamination surveys
- bioassay.

EO8 IDENTIFY tritium waste minimization and handling techniques. (Site-Specific)

EO9 IDENTIFY alarms and proper response to abnormal conditions in the tritium facility. (Site-Specific)

Notes

I. PROPERTIES OF TRITIUM - What is tritium?

Tritium is a radioactive isotope of hydrogen (H). Hydrogen is the lightest and most abundant element in the universe. Hydrogen has only one proton in its nucleus. Tritium has, in addition to the single proton, two neutrons in its nucleus. This makes it three times heavier than the most common form of hydrogen.

A. Isotopes of Hydrogen

1. Protium (H) (99.985% natural abundance)
2. Deuterium = hydrogen + 1 neutron (D) (0.015% natural abundance)
3. Tritium = hydrogen + 2 neutrons (T)

B. Symbol for Tritium

Tritium is designated as:

$$\mathrm{T}, \mathrm{H\text{-}3}, \text{or } {}^{3}_{1}\mathrm{H}$$

C. Chemical Properties of Hydrogen/ Tritium

Tritium "behaves" just like hydrogen chemically because it has one proton and one electron like ordinary hydrogen.

Notes

1. Substitution

   Tritium atoms can easily substitute for hydrogen atoms. Examples:

   a. elemental hydrogen (tritium gas, HT, DT, or T_2)

   b. tritiated water (tritium oxide: HTO, DTO, or T_2O)

   c. organically bound tritium: These compounds may result in more dose per intake than tritiated water.

   d. metals and tritium (metal tritides): These compounds may result in more dose per intake than tritiated water. Accordingly, additional controls may be needed, such as special air monitoring or enhanced personal protective equipment.

2. Solubility

   Exchanges with hydrogen in organic and other materials (oils, plastics, etc.)

3. Flammability

   Tritium gas is flammable and can explode under certain conditions.

Notes

D. Physical Properties

1. Diffusion

   Tritium gas is lighter than air and diffuses rapidly in air.

2. Permeability

   Tritium gas permeates through most materials, that is, it travels through them by way of spaces or interstices in them. The rate depends upon the material and its thickness. Tritium's radioactive, chemically reducing and diffusive properties result in degradation of many useful polymeric materials, pump oils, plastics, and o-rings. This can cause loss of mechanical functions in certain situations within a short period of time.

$$ {}^{3}_{1}\mathrm{H}\ \text{decays to}\ {}^{3}_{2}\mathrm{He} + \beta^{-} + \bar{\nu} $$

E. Radioactive Decay of Tritium

${}^{3}_{2}He$ = Helium

$\bar{\nu}$ = antineutrino

β^{-} = beta minus

Tritium "decays" by emitting a beta particle and becoming an atom of helium.

Notes

1. Beta Energy

A very low energy beta is emitted.

2. Beta Range

Tritium is not an "external" radiation hazard.

   a. Travels less than 1/4 inch in air
   b. Cannot penetrate through the dead layer of the skin
   c. Cannot penetrate clothing or gloves

3. Half-life

Radioactive half-life = 12.3 years.

F. X-Ray Production

Tritium betas can produce low-energy x-rays. Because the beta particle is of such low energy, the x-rays it may produce are not very penetrating and are not normally considered a hazard.

Notes

G. Fusion

Fusion is the process of two nuclei joining together. The energy source in most stars is nuclear fusion of hydrogen isotopes.

SOURCES AND USES OF TRITIUM

How is tritium produced, and what is it used for?

A. Sources of Tritium

There are three primary sources of tritium in our environment. Tritium is present in our environment from both man-made and natural sources as discussed below. Natural tritium is indistinguishable from man-made tritium.

1. Natural Sources

Tritium occurs naturally. It is formed by the reactions between cosmic rays and the nitrogen in the upper atmosphere. Nitrogen makes up 80% of the earths atmosphere.

Cosmic rays generate approximately 4 million curies of tritium per year. With tritium being continually produced and at the same time decaying, the natural tritium in our environment is about 70 million curies.

Notes

2. Byproduct of Reactor Sources

   a. Lithium-7 captures a neutron and decays with tritium as a product.

   b. Boron captures a neutron and decays with tritium as a product.

   c. Activation of deuterium in water.

   d. Ternary fission - A fission resulting in three fission products, one of which is tritium. This process has a 0.1% yield.

   The production of tritium from power reactors around the world is less than one-half that naturally formed (approximately 1 to 2 million curies a year).

3. Weapons Testing as a Source

   The amount of tritium in the world from weapons testing has been steadily declining since the 1970's when atmospheric testing was curtailed.

   Atmospheric testing from 1945-1975 produced about 8 billion curies. This has decayed to about 400 million curies.

Notes

4. DOE Production of Tritium

DOE has produced tritium at the Savannah River Site with the use of a reactor. Tritium is commercially available from Canada and the European Community for non-weapons use.

B. Uses of Tritium

1. Consumer Products

   a. Gaseous tritium light sources
      - exit signs (1 Ci to tens of Ci's)
      - aviation landing aids (30-165 Ci per light)

      (1 curie of tritium has a mass of approximately 0.1 mg)

   b. Luminizing industry: Self-luminous compounds for dials (several mCi's) and controls as well as other general industry uses.

2. Research - Tritium Labeling

   Tracers for medical and laboratory research.

3. Department of Energy

   a. Weapons development and applications

   b. Fusion energy: As a fuel source

Notes

III. MODES OF EXPOSURE AND BIOLOGICAL BEHAVIOR OF TRITIUM - How can I be exposed to tritium and what would it do to me?

A. External Dose

- Tritium is not an external hazard.
- Tritium is not a dose concern if it is located outside the body.
- Tritium betas will not penetrate a dosimeter.
- We are interested only in tritium inside the body.

B. Internal Dose

Tritium is an "internal" radiation hazard.

C. Modes of Exposure

Tritium can deliver a radiation dose if it gets inside our bodies. Modes of entry include:

- inhalation
- absorption
- ingestion

Notes

1. Inhalation

   a. Tritium gas: is only slightly incorporated into the body when inhaled. Most tritium gas inhaled is subsequently exhaled. There are other chemical and physical processes to convert tritium gas to tritiated water.

   b. Tritiated water vapor: Nearly 100% of tritiated water vapor inhaled is incorporated into body fluids/tissues.

   c. Hazard: Exposure to tritiated water is approximately 10,000 times more hazardous than exposure to elemental tritium gas.

   d. Special Tritium Compounds: consist of organically bound tritium and tritium particulate aerosols.

   Exposure to organically bound tritium can be up to approximately 13 times more hazardous than exposure to HTO.

   Exposure to tritium particulate aerosols can deliver up to approximately 20 times more dose to the whole body than exposure to HTO. The dose to the lungs from tritium particulate aerosols could be 2 orders of magnitude higher than from HTO.

2. Ingestion

   Ingestion may occur by eating, drinking, chewing tobacco, or applying makeup where tritium

Notes

contamination is present. Always wash hands thoroughly when leaving areas where there is a potential for contamination, and never eat, drink, etc. where tritium contamination may be present.

Notes

3. Absorption

Absorption is also a hazard because an individual can receive, in certain situations, 1/3 of their uptake from absorption through the skin if not properly using personal protective equipment.

a. Tritium gas: There is negligible skin absorption for tritium gas.

b. Tritiated water: Tritiated water can be absorbed through the skin. It has been observed that moisture on hands enhances absorption.

c. Solvents: Some solvents (organically bound tritium) can also go through the skin.

Tritium will penetrate gloves; therefore gloves should be changed at a prescribed routine basis.

D. Biological Behavior of Uptakes of Tritium

Tritiated water in the body acts just like water.

1. Distribution

Tritiated water is rapidly and uniformly distributed throughout the entire body. The Committed Effective Dose from an uptake of one curie of tritiated water is 63 rem. Soluble organically bound tritium behaves in the body in a similar manner.

Stable tritium particulate aerosols and insoluble organically bound tritium behave in the body in a

Notes

similar manner as the particulate to which the tritium is bound.

Notes

2. Biological Half-life

Half of the HTO is eliminated from the body in about 10 days. Through normal biological processes, it would take 70 days or more to eliminate 99% of the assimilated tritiated water.

Stable tritium particulate aerosols and insoluble organically bound tritium behave in the body in a similar manner as the particulate to which the tritium is bound and would have a longer half-life.

E. Medical Treatment

The biological half-life can be shortened by increasing the water elimination in the body. Therefore, individuals who have uptakes of tritium are encouraged to drink water. Drinking copious amounts of water should not be done without a physician's guidance. Certain medical conditions may be affected by liquid intake.

IV. RADIOLOGICAL CONTROLS FOR TRITIUM - How can I protect myself from exposures to tritium?

Tritium can be present in a variety of chemical forms. Ongoing research indicates that the form of tritium which gives the highest dose (per unit intake) is tritium particulate aerosols. However, these compounds are not found in the quantities and various locations as tritiated water. If we can rule out special tritium compounds as being of concern, we assume it may be tritiated water.

The preferred hierarchy of control is as follows:

- engineered
- administrative
- personal protective equipment

Notes

Notes

A. Engineered Controls

1. Containment and Confinement

Containment or confinement is a series of physical barriers, minimizes exposure of workers.

Confinements such as gloveboxes are almost always used when handling large quantities of tritium. However, hoods are acceptable for handling small quantities, such as in a laboratory.

2. Airflow

Maintaining negative ventilation is essential for the safe operation of a tritium facility. Airflow should be from areas of LEAST to MOST contamination.

3. Local Exhaust Ventilation

The primary advantage of local exhaust ventilation techniques is the removal of airborne tritium, regardless of its release rate or chemical or physical form. In addition, these techniques use relatively low volume rates compared to normal ventilation requirements.

Notes

4. Dilution Ventilation

Dilution ventilation is the once-through flow technique of exchanging outside air for inside air for comfort and the reduction of airborne sources.

5. Storage

Tritium can be stored in storage beds. Metal tritide and uranium hydride are the most common for these storage systems. Tritium is generally released by heating the metal tritide.

B. Administrative Controls

There are many administrative controls to reduce doses. The following are just a few that should apply to all sites:

- limitation of access time
- procedures/RWPs
- postings

For tritium and tritium compounds, 10 CFR Part 835 Appendix D requires posting contamination areas based on removable contamination values of 10,000 dpm/100cm^2.

Notes

C. Personal Protective Equipment

1. Air-Supplied Suits

Because of the absorption through the skin associated with the use of respirators and other breathing apparatus, air-supplied plastic suits that completely enclose the body are widely used by facilities that handle large quantities of tritium.

2. Protective Clothing

Protective clothing (PC), or anti-contamination clothing (anti-Cs), is used to minimize the spread of contamination from contaminated to clean areas.

In many operations, the hands and forearms of workers are vulnerable to contact with tritium surface contamination. The proper selection of gloves and glove materials is important. In many instances a plastic/water proof suit is required.

Notes

V. Monitoring for Tritium: How do I know if tritium is present?

A. Personnel Monitoring

External Dose

Dosimeters are not typically used to monitor for radiological doses resulting from tritium. The weak energy beta radiation will not penetrate the dosimeter.

insert any site specific monitoring procedures)

2. Internal Dose

The best method used to determine if an individual has an uptake of HTO or soluble organically bound tritium is by bioassay (urinalysis). Routine urine samples, collected at some predetermined frequency and counted for the tritium content, provide a very sensitive measurement of tritium in the body. This is especially true if the time of uptake is known.

Air sampling results may be used to assess dose from other types of special tritium compounds.

3. Routine Versus Non-Routine Bioassay Monitoring

Notes

a. Routine: Routine urinalysis is conducted on a preset periodic basis.

b. Non-routine or Special: Non-routine bioassay is done whenever a potential exposure to tritium is suspected.

(insert site-specific bioassay procedures)

B. Workplace Monitoring

Air monitoring and surface contamination surveys are used to verify that loose contamination is not present. They provide an early indication of potential problems.

1. Airborne Tritium Monitoring

Airborne tritium monitoring is used for:

a. Prompt detection of airborne contamination for worker protection;

b. Determination of the status of processes; and

c. Identification of any leaks in primary or secondary containments or confinements.

2. Contamination Surveys

Notes

Despite contamination control measures, tritium is easily spread.

All workplaces shall be surveyed for contamination levels on a regularly scheduled basis. The frequency of such surveys will depend on the potential for dispersement of the tritium-contaminated material in the area and the quantity of tritium in the area. During routine surveys, all work enclosures, work surfaces, floors, equipment, etc., within the workplace should be surveyed.

Notes

At the surface contamination levels in 10 CFR Part 835 Appendix D, tritium is difficult to measure directly because the low-energy beta is readily absorbed in air and the window of the detector. Normal frisking methods (use of pancake probe) will not detect tritiated water.

Surfaces are normally wiped (smeared) with a small paper, either dry or moistened, which will pick up loose tritiated material. The wipe is then normally counted by liquid scintillation techniques.

VI. TRITIUM WASTE MINIMIZATION AND HANDLING- Why is it important and how do you minimize tritium waste?

(insert site specific controls)

Methods below are given as a reference only.

A. Minimizing Tritium-Contaminated Waste

1. Avoid generating wastes by not bringing unnecessary material into the controlled area.
2. Whenever possible, avoid use of porous materials or those known to be highly permeable to tritium.
3. Designate an area to store contaminated tools for reuse or consider all tools in the area to be contaminated.

Notes

4. Plan work so that, whenever possible, construction and clean maintenance can be done in a clean area.

5. When transporting items contaminated with tritium, adequate contamination control such as wrapping in plastic or placing in sealed containers should be considered.

B. Minimizing Mixed Waste

(insert site-specific controls)

Methods below are given as a reference only.

1. Use non-hazardous cleaning materials for decontamination whenever possible.

2. Segregate radioactive-only from hazardous-only at the source.

3. Explore the use of other materials which are non-hazardous for use in radiological areas to prevent the generation of mixed waste.

VII. RESPONSE TO ABNORMAL CONDITIONS IN THE TRITIUM FACILITY- What should I do in an emergency involving tritium?

To properly deal with unexpected adverse situations occurring in a tritium facility, a well-thought-out response program and personnel trained to execute the response should be in place.

Notes

A. Abnormal Conditions

Abnormal conditions in a tritium facility could include:

- fire/explosion
- natural disaster
- tritium releases
- other hazards

Personnel found contaminated should follow site-specific decontamination procedures, which would typically include showering with cold water and the use of mild detergents.

B. Facility Alarms

(insert site-specific alarms and emergency response procedures)

C. Facility Emergency Responses

(insert site-specific alarms and emergency response procedures)

VIII. LESSONS LEARNED

(insert site-specific lessons learned)

IX. SUMMARY AND REVIEW

APPENDIX G: PRIMER ON TRITIUM

OVERVIEW

This Appendix (Primer) contains an introduction and sections on the following topics:

- Radiological Fundamentals
- Physical and Chemical Properties of Tritium
- Biological Properties of Tritium
- Tritium Monitoring
- Radiological Control and Protection Practices
- Emergency Response

This document contains selected information from the *Health Physics Manual of Good Practices for Tritium Facilities*, which was prepared by EG&G Mound Applied Technologies at the Mound tritium facility. The authors acknowledge their expertise and experience.

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INTRODUCTION

This Primer is designed for use by operations and maintenance personnel to improve their knowledge of tritium safe handling practices. It is applicable to many job classifications and can be used as a reference for classroom work or for self-study. It is presented in general terms so that it can be used throughout the DOE Complex.

The information in this Primer should enable the reader to do the following:

- Describe methods of measuring airborne tritium concentration.
- List the types of protective clothing that are effective against tritium uptake from surface and airborne contamination.
- Name two methods of reducing the body dose after a tritium uptake.
- Describe the most common method for determining the amount of tritium uptake in the body.
- Describe the steps to take following an accidental release of airborne tritium.
- Describe the damage to metals that results from absorption of tritium.
- Explain how washing hands or showering in cold water helps reduce tritium uptake.
- Describe how tritium exchanges with normal hydrogen in water and hydrocarbons.

The organization of the Primer is as indicated in the Overview. The following section contains background information on "Radiological Fundamentals." Those familiar with these topics may elect to skip this section and begin reading at the section entitled "Physical and Chemical Properties of Tritium." Additional information about tritium is available from the sources listed in the "Bibliography" section.

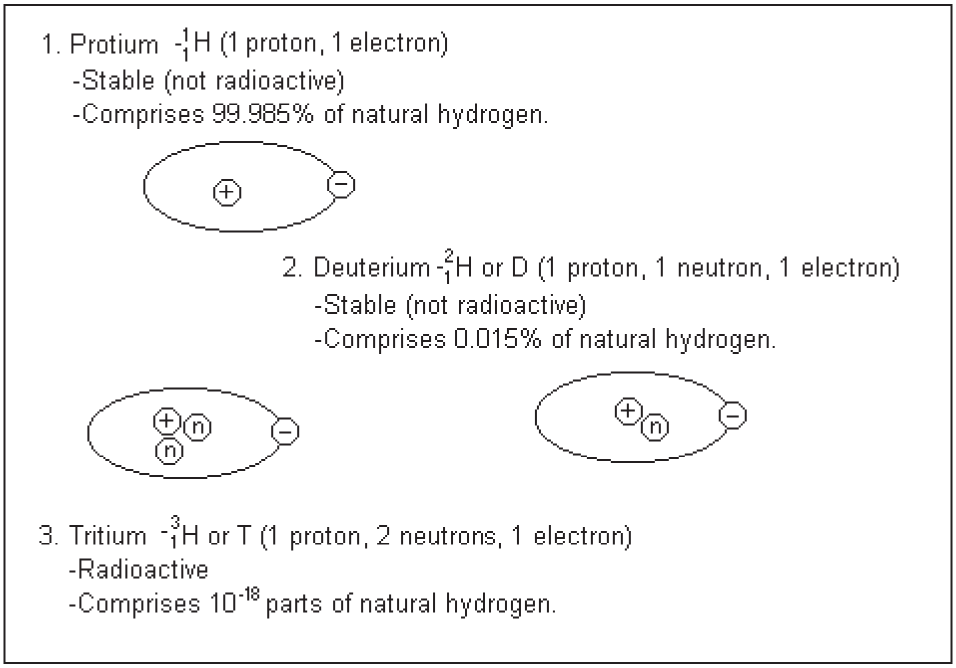
RADIOLOGICAL FUNDAMENTALS

This section provides a review of radiological fundamentals. The reader is assumed to be familiar with this information from radiological worker training. The section discusses hydrogen and its isotopes and describes basic radiological concepts.

Hydrogen and Its Isotopes

Atomic nuclei of a particular element (such as hydrogen or oxygen) have the same number of protons (positively charged), but may have a different number of neutrons (no net charge). Those that have a different number of neutrons are isotopes of that element. Most elements exist in nature in several isotopic forms. For example, hydrogen has one proton. The isotopes of hydrogen either have no neutrons (normal hydrogen, called protium), one neutron (deuterium), or two neutrons (tritium) (Figure G-1). Although isotopes of an element have almost the same chemical properties, the nuclear properties can be quite different.

Figure G-1: Hydrogen Isotopes



1. Protium -${}^{1}_{1}H$ (1 proton, 1 electron)
   -Stable (not radioactive)
   -Comprises 99.985% of natural hydrogen.

2. Deuterium -${}^{2}_{1}H$ or D (1 proton, 1 neutron, 1 electron)
   -Stable (not radioactive)
   -Comprises 0.015% of natural hydrogen.

3. Tritium -${}^{3}_{1}H$ or T (1 proton, 2 neutrons, 1 electron)
   -Radioactive
   -Comprises 10^{-18} parts of natural hydrogen.

Nuclear notation uses the chemical symbol (H for hydrogen) and an arrangement of subscripts and superscripts. The total number of protons and neutrons is shown as a superscript: ${}^{1}H$ for protium, ${}^{2}H$ for deuterium, and ${}^{3}H$ for tritium. The number of protons (which identifies the element) is shown as a subscript. However, the common practice of using H, D, and T for these isotopes, respectively, will be followed in this document, except where nuclear reactions are illustrated.

The atomic masses, symbols, and natural abundances of the three isotopes of hydrogen are given in Table G-1.

Table G-1: Hydrogen Isotopes

Symbol				
Physical	Common	Name	(%) Natural Abundance	Mass (mass units)
${}^{1}_{1}H$	H	Protium	99.985	1.007825
${}^{2}_{1}H$	D	Deuterium	0.015	2.01400
${}^{3}_{1}H$	T	Tritium	1×10^{-18}	3.01605

Sources of Tritium

Tritium occurs naturally in the environment. Reactions between cosmic radiation and gases in the upper atmosphere produce most of the world's natural tritium. For example,

$${}^{14}_{7}N + {}^{1}_{0}n \quad {}^{3}_{1}H + {}^{12}_{6}C \qquad \rightarrow$$

Tritium converts into water and reaches the earth's surface as rain. An estimated production rate of 4 × 10^6 Ci/yr results in a world steady-state natural inventory of ~70 × 10^6 Ci[104].

In addition, commercial producers of radioluminescent and neutron generator devices release about 1 × 10^6 Ci/yr. Atmospheric nuclear test explosions from 1945 to 1975 added about 8 × 10^9 Ci of tritium to the environment, much of which has since decayed. However, about 5 × 10^8 Ci remain in the environment, mostly diluted in the oceans. Underground nuclear tests appear to add little tritium to the atmosphere. The nuclear power and defense industries now release 1-2 × 10^6 Ci/yr, a small fraction of which comes from light-water reactors. Tritium is also a by-product of light-water and heavy-water nuclear reactor operation. In their coolants, these reactors produce about 500 to 1,000 and 2 × 10^6 Ci/yr, respectively, for every 1,000 MW(e) of power. Tritium is a fission product within nuclear fuel, generated at a rate of 1-2 × 10^4 Ci per year/1000 MW(e). U.S. DOE reactors have produced tritium by the neutron bombardment of lithium (Li).

Stable and Unstable Nuclides

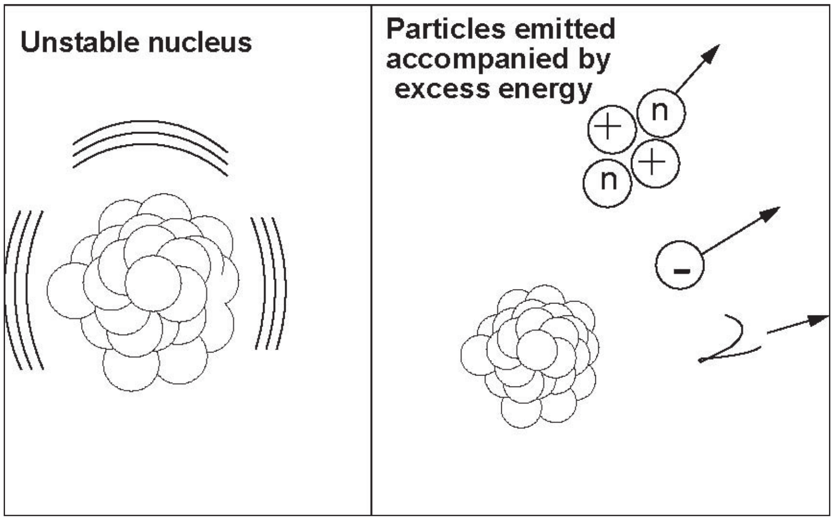
In the lighter elements, the ratio of neutrons to protons in their stable nuclei is usually 1 (one neutron for every proton). For the heavier elements, this ratio gradually increases to about 1.5 (1.5 neutrons for every proton). Although one cannot always predict from its ratio whether an isotope is stable or unstable, the relationship between the number of protons and neutrons is extremely important.

When an isotope is unstable, its nucleus will emit rays or particles or it may split into two different nuclei. Some combinations of neutrons and protons lead to stable nuclei. If there are too many or too few neutrons, the resulting nucleus is not stable. This unstable nucleus tries to become more stable by releasing excess energy. Atoms with unstable nuclei are radioactive. The process of nuclei releasing this energy is referred to as radioactive decay or disintegration (Figure G-2). If a nucleus is still unstable after radioactive decay, further decay will occur.

---

[104] The curie (Ci) is a unit of activity defined as 3.7×10^{10} disintegrations per second (dps). A more basic unit is 1 dps, which is the definition of the becquerel (Bq). Throughout this Primer, the curie will be used instead of the becquerel.

Figure G-2: Radioactive Decay

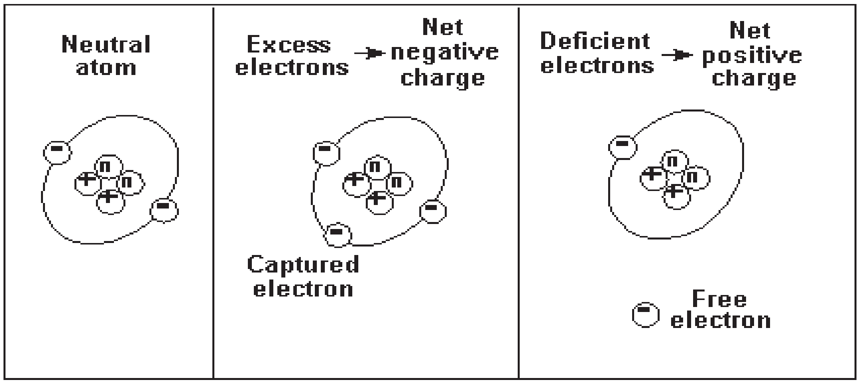


Ions and Ionization

Atoms can combine chemically to form molecules. Atoms and molecules are surrounded by orbiting electrons (negatively charged). If the number of electrons equals the total number of protons (positively charged) in the nucleus, the atom or molecule is neutral (uncharged).

Electrically charged atoms or molecules are called *ions*. Ions are either positively or negatively charged, depending on the number of orbiting electrons relative to the number of protons in the nucleus. As shown in Figure G-3, ions with more electrons than protons are negatively charged, while ions with more protons than electrons are positively charged. The process of breaking a neutral atom or molecule into electrically charged parts is called *ionization*. This process requires energy. Ionization removes electrons from the atom, or molecule, leaving an ion with a positive charge. The negatively charged electron (which can attach itself to a neutral atom or molecule) and the positively charged ion are called an ion pair. Radiation that causes ionization is called *ionizing radiation*.

Figure G-3: Neutral and Ionized Atoms



Types of Radiation

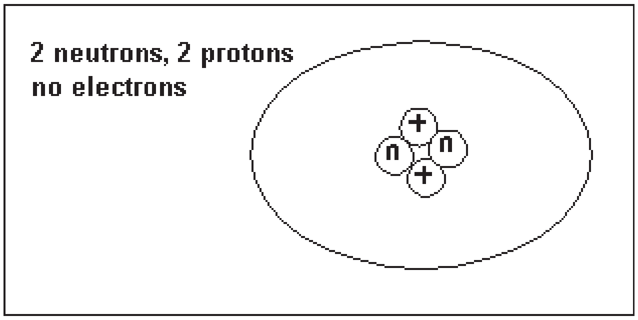
There are four basic types of ionizing radiation emitted from nuclei: alpha particles, beta particles, gamma rays, and neutrons.

- **alpha** particle (α)—consists of two protons and two neutrons and is the same as the nucleus of a helium atom ($^{4}_{2}He$) (Figure 4). Generally, only the heavy nuclides can emit alpha particles.

A typical example of an α -emitting nuclide is uranium-238:

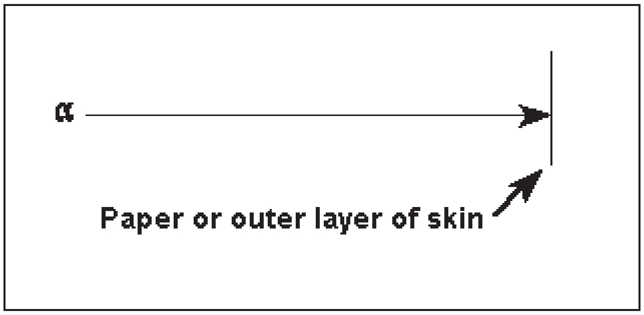
$$^{238}_{92}U \rightarrow {}^{234}_{90}Th + \alpha + \text{energy.}$$

Figure G-4: Alpha Particle



The mass of an alpha particle is about four times the mass of a single neutron or proton, and has a positive charge of +2 (it has no electrons). This positive charge causes the alpha particle (2 neutrons, 2 protons 0 electrons) to ionize nearby atoms as it passes through body tissue. The strong positive charge and its relatively slow speed (resulting from its large mass) causes the alpha particle to interact strongly with orbiting electrons of atoms and molecules and to lose large amounts of energy in a short distance. This limits the penetrating ability of the alpha particle, making it easy to stop. A few centimeters of air, a sheet of paper, or the outer layer of skin stops alpha particles (Figure G-5).

Figure G-5: Alpha Shielding



Alpha particles are not an external radiation hazard because they are easily stopped by protective clothing or the outer layer of skin. However, if an alpha emitter is inhaled or ingested, it becomes an internal radiation hazard. Because the source is in close contact with body tissue, the alpha particle will dissipate its energy in a short distance of the tissue.

- **beta particle** (β) is equivalent to an electron except for its source. Beta-emitting nuclides have too many neutrons. A neutron emits a β particle, and the neutron is then converted to a proton. Tritium decay provides a good example of this process:

$$ {}_{1}^{3}H \rightarrow \beta + {}_{2}^{3}He + \text{energy.} $$

A β particle is identical to an electron, and its mass and charge are the same as those of an electron. As in the case of alpha particles, beta particles ionize atoms by removing electrons from their orbits. This reaction occurs from charged particle interactions or "collisions" with orbiting electrons.

Beta particles penetrate further than alpha particles of the same energy. A high-energy beta particle can penetrate a few centimeters of organic tissue. The higher the energy is, the greater the penetrating ability. However, low-energy beta particles of tritium can be shielded by skin, paper, or only about 6 mm of air.

- **gamma ray** (γ) is emitted when the nucleus of a nuclide releases stored energy without releasing a particle. Many gamma-emitters are found among the products of nuclear fission.

During pure γ emission the nucleus does not emit particles or change its nuclear structure or chemical characteristics. For instance,

$$ {}_{Z}^{A}X \rightarrow \gamma + {}_{Z}^{A}X . $$

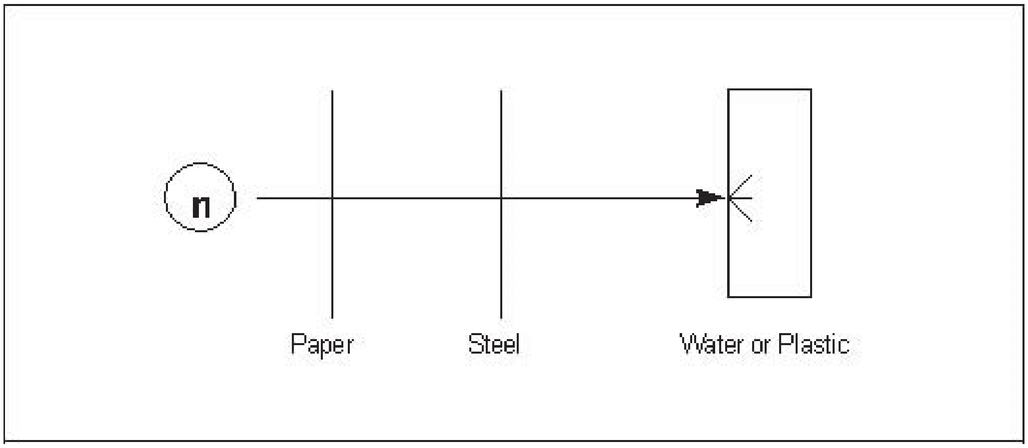
Gamma radiation is in the form of electromagnetic waves (or photons). Gamma rays are similar to x-rays, but they differ in their origin and energy. Gamma rays originate within the nucleus, and x-rays originate outside the nucleus.

Gamma rays have a very high penetrating power because they have no charge or mass. Depending on their energy, a stream of gamma rays may penetrate with gradually diminishing intensity through several inches of concrete or similar material. They can be shielded effectively by very dense materials, such as lead and uranium. Gamma rays are a whole-body hazard. That is, because of their penetrating ability, the damage caused by gamma rays is not restricted to any particular body organ.

- **neutron** (n)—may be emitted spontaneously by heavy nuclei during fission or may be emitted during radioactive decay. They are uncharged particles that have mass and a high penetrating ability.

A neutron has about 2,000 times the mass of an electron, but only one-fourth the mass of an alpha particle. Neutrons are difficult to stop because they lack a charge. Neutrons mainly interact with matter by striking hydrogen nuclei or interacting with the nucleus of atoms. These collisions generally cause charged particles or other radiation to be emitted. These particles may then ionize other atoms. Collisions between neutrons and hydrogen nuclei (protons) are effective in stopping or slowing down high-energy neutrons. Neutrons are best shielded by materials with a high hydrogen content, such as water or plastic (see Figure G-6).

Figure G-17 Neutron Shielding



Radioactivity

As radioactive isotopes decay, the number of radioactive nuclei decreases. The time required for half of the nuclei in a sample of a specific radioactive isotope to undergo decay is called its (physical) half-life (Figure G-7). Each radioactive isotope has its own characteristic half-life. Radioactive isotopes decay to less than 1% of their original quantity after about seven half-lives.

Half-lives vary widely with different radionuclides, as shown by the following examples:

^{16}N -7.35 seconds

^{3}H -12.43 years

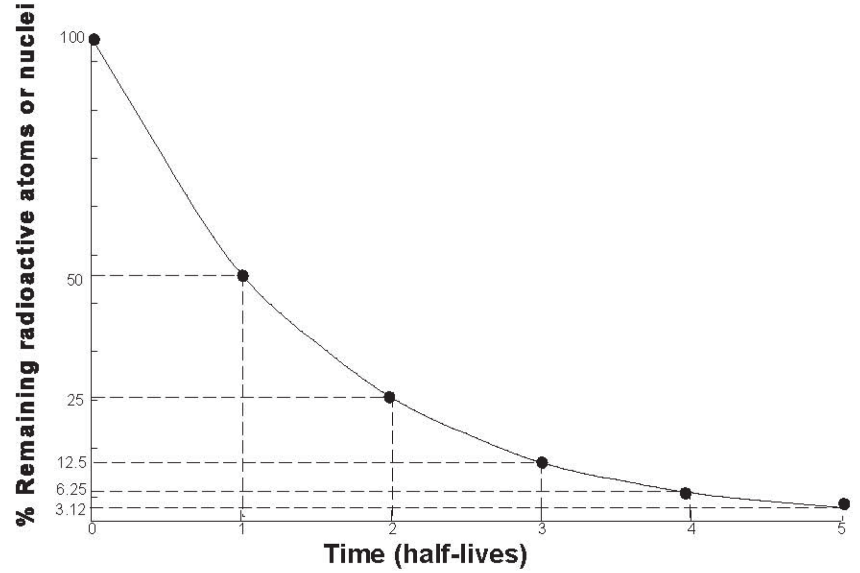
^{238}U -4.5 × 10^9 years (4.5 billion years)

The activity of a radioactive isotope sample is defined as the number of nuclei that decay per unit of time.

It has been shown that for a pure radioactive isotope the number of nuclei decaying per unit time (rate of decay) is proportional to the number of nuclei available to decay. If the substance is not being replenished, its activity will decrease accordingly. Therefore, in terms of half-life, the remaining activity after a period of time can be expressed as follows:

$$A_t = A_0 \times (1/2)^n \text{ or } A_0 / A_t = 2^n$$

Figure G-7: Half Life



example where:

A_0 = the initial activity (Ci)
A_t = activity after elapsed time, t (Ci)
n = number of half-lives during elapsed time, equal to $t/T_{1/2}$,
t = elapsed time (time units)
$T_{1/2}$ = half-life (time units).
For example, assume we have stored 10,000 Ci of tritium.
How many curies will be left after 50 years of storage?
A_0 = 10,000 Ci; n = 50/12.43 = 4.02
$A_t = A_0 (1/2)^n$
$A_t = 10{,}000 (1/2)^{4.02}$
= 10,000 × 0.062
= 620 Ci (approximately)

PHYSICAL AND CHEMICAL PROPERTIES OF TRITIUM

This section reviews the nuclear properties of tritium and discusses some of the physical and chemical properties that are important in understanding tritium handling, containment, and contamination control.

Nuclear and Radioactive Properties

Being an isotope of hydrogen, tritium has many of the properties of ordinary hydrogen (such as chemical reactions, permeability, and absorption). Differences may occur because the decaying tritium atoms can speed up (catalyze) reactions of undecayed tritium, or because atoms that have undergone decay have changed into helium atoms (^{3}He). Additionally, small differences in chemical reaction rates may result from the relative masses of the isotopes.

Some of the useful properties of tritium are listed in Table G-2. Note that the properties listed are those of molecules of TT (or T2) i.e., two tritium atoms bound together. The specific activity and power density of HT (a hydrogen and tritium atom bound together) and DT (a deuterium and tritium atom bound together) are approximately one-half those for T2. The activity density of HT and DT is exactly one-half that of T2.

Table G-2: Important Nuclear Properties of Tritium

Half-Life	12.43 yrs
Specific Activity	9,545 Ci/g
Power Density	0.328 W/g
Activity Density	
(T_2 gas, 1 atm, 0 degrees C)	2.589 Ci/cm^3
(T_2 gas, 1 atm, 25 degrees C)	2.372 Ci/cm^3

Penetration Depths of Beta Particles

The penetration and absorption of beta particles in a material are important factors for detecting tritium and understanding the mechanisms by which tritium can degrade materials. A beta particle interacts with matter by colliding with electrons in the surrounding material. In each collision, the beta particle may lose several electron volts (keV)105 of energy, and the electron is stripped from its atom (ionization) or promoted to an excited state. The beta particle has a finite penetration depth that depends on its energy. Recall that tritium undergoes beta decay according to the following equation:

$$^{3}_{1}H \rightarrow \beta + {}^{3}_{2}He + \text{energy}$$

The helium daughter ($^{3}_{2}He$) is stable, but lighter than common helium ($^{4}_{2}He$). The decay energy is constant (18.6 keV), but is shared between the beta particle and an antineutrino (a tiny particle). The result is that not all beta particles have the same energy. The average energy is 5.7 keV. Consequently,

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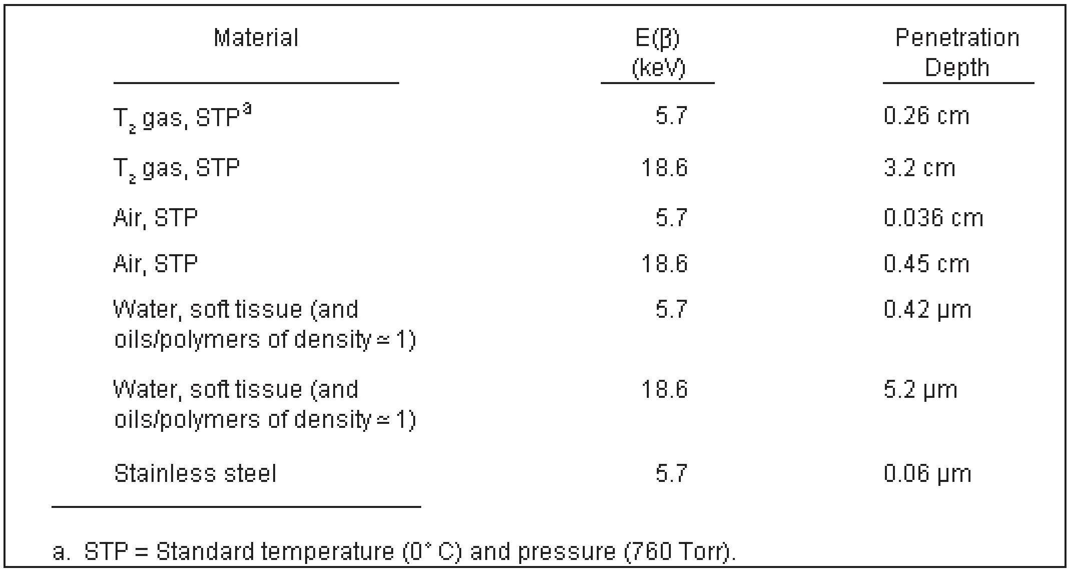
[105] An electron volt is a small unit of energy used in descriptions of nuclear and chemical reactions. It equals the energy gained by an electron when it moves across a potential of 1 volt.

not all tritium betas have the same penetration depth in a given material. Where beta ranges are given, it is customary to list both the highest energy and the average, most representative energy, as listed in Table G-3.

Table G-3: Penetration Depths of Tritium Betas



Material	E(β) (keV)	Penetration Depth
T_2 gas, STP[a]	5.7	0.26 cm
T_2 gas, STP	18.6	3.2 cm
Air, STP	5.7	0.036 cm
Air, STP	18.6	0.45 cm
Water, soft tissue (and oils/polymers of density ≃ 1)	5.7	0.42 µm
Water, soft tissue (and oils/polymers of density ≃ 1)	18.6	5.2 µm
Stainless steel	5.7	0.06 µm

a. STP = Standard temperature (0° C) and pressure (760 Torr).

With one unimportant exception, tritium is the weakest beta emitter known. The range of the most energetic tritium beta particles is only about 5 mm in air or 0.005 mm in water or soft tissue. This range makes it a nonhazard outside the body, but presents a detection problem. Where other radioisotopes can be detected by virtue of their penetrating radiation, tritium has to be introduced directly inside the detector or counter to be measured.

Chemical Properties

Laboratories that have large quantities of tritium usually handle it in the form of HT. However, at any time the tritium may be stored on metal getter beds (such as titanium, zirconium, or uranium). These beds form weak chemical compounds with hydrogen. Some of the beds are stable in air; and others are not and can only be used in certain atmospheres. The tritium is released (or delivered) by heating the beds to the required temperature.

Laboratories may also handle tritiated gases (such as ammonia and methane) and other compounds. By far the most common of these is HTO, which is formed from HT whenever it is exposed to oxygen or water vapor. The conversion reactions are oxidation and exchange:

oxidation

$2HT + O_2 \rightarrow 2HTO$

$2T_2 + O_2 \rightarrow 2TO_2$

exchange

$HT + H_2O \rightarrow H_2 + HTO$

$T_2 + H_2O \rightarrow HT + HTO$

These reaction rates are increased by radiation (from nearby tritium at high concentrations), heat, or the presence of metal catalysts (especially palladium or platinum). All chemical reactions involving hydrogen can also be performed with tritium, sometimes at a higher rate if the tritium concentration is high enough to catalyze the reaction. One of the most important reactions occurs when a tritium atom exchanges with a loosely bonded hydrogen atom of an organic molecule. However, where HT is dissolved in water (H2O), the exchange process is fairly slow because the hydrogen in is tightly bonded and the reaction is not catalyzed.

Contamination

Tritium as HT or HTO will readily adsorb onto the surface of most metals (such as stainless steel, copper, or aluminum), plastics, and rubbers. The tritium will remain fairly close to the surface unless the metal is heated to a high temperature. At room temperature, permeation into these metals is usually extremely slow.

In the case of metal contamination, the tritium remains on or very close to the surface. The contamination can be removed with water or water vapor if the surface is contaminated with HTO or with hydrogen (H_2 or D_2) if the contamination is HT. Heating also speeds up the decontamination process. The initial application of heat to surfaces can also be used to prevent or lessen the contamination by HT or HTO. Metal surfaces exposed to high pressures of HT or HTO for extended periods, especially at high temperatures, may allow enough penetration to cause structural damage to the metal. This is especially true if the decaying tritium causes a buildup of helium within the structure of the metal.

If adsorbed onto hydrogenous material, the tritium will easily permeate into the material. The HTO will move much more rapidly into the bulk material than will HT. The permeation rate varies with the type of material and is accelerated by increasing the temperature. As a result of this movement, plastics and rubbers exposed to tritium (especially as HTO) are readily contaminated deep into the bulk material and are impossible to decontaminate completely. After a period of time, the tritium exchanges with bulk hydrogen and presents little biological risk.

Highly contaminated metal or plastic surfaces may release some of the loosely-bound tritium immediately after exposure to the contaminating tritiated atmosphere or liquid. This is referred to as outgassing. The personnel risk from outgassing tritium is generally much less than that from making unprotected skin contact with the outgassing surface.

BIOLOGICAL PROPERTIES OF TRITIUM

At most tritium facilities, the most commonly encountered forms of tritium are tritium gas (HT) and tritium oxide (HTO). Other forms of tritium may be present, such as metal tritides, tritiated pump oil, and tritiated gases such as methane and ammonia. As noted earlier, deuterated and tritiated compounds generally have the same chemical properties as their protium counterparts, although some minor isotopic differences in reaction rates exist. These various tritiated compounds have a wide range of metabolic properties in humans under similar exposure conditions. For example, inhaled tritium gas is only slightly incorporated into the body during exposure, and the remainder is rapidly removed (by exhalation) following the exposure. On the other hand, tritiated water vapor is readily taken up and retained in the body water. In this Primer, we will address only those compounds likely to be found at DOE laboratories: gaseous tritium, tritiated water, other tritiated species, metallic getters, and other tritiated liquids and gases.

Metabolism of Gaseous Tritium

During a brief exposure to tritium gas, the gas is inhaled and a small amount is dissolved in the bloodstream. The dissolved gas circulates in the bloodstream before being exhaled along with the gaseous waste products (carbon dioxide) and normal water vapor. If the exposure persists, the gas will reach other body fluids. A small percentage of the gaseous tritium is converted to the oxide (HTO), most likely by oxidation in the gastrointestinal tract. Early experiments involving human exposure to a concentration of 9 µCi/mL resulted in an increase in the HTO concentration in urine of 7.7 × 10-3 µCi/mL per hour of exposure. Although independent of the breathing rate, this conversion can be expressed as the ratio of the HTO buildup to the tritium inhaled as HT at a nominal breathing rate (20 L/min). In this context, the conversion is 0.003% of the total gaseous tritium inhaled. More recent experiments with six volunteers resulted in a conversion of 0.005%. For gaseous tritium exposures, there are two doses: (a) a lung dose from the tritium in the air inside the lung and (b) a whole body dose from the tritium gas that has been converted to HTO. The tritiated water converted from the gas in the body behaves as an exposure to tritiated water.

Intake of gaseous tritium through the skin has been found to be negligible compared with that from inhalation. Small amounts of tritium can enter the skin through unprotected contact with contaminated metal surfaces, which results in organically bound tritium in skin and in urine. Ordinarily this is not a serious problem because surfaces highly contaminated with tritium gas are inaccessible to skin contact. Also, most tritium exposed to air will be converted to the oxide form (water vapor) before the internal surfaces of equipment are handled during maintenance or repair operations.

Metabolism of Tritiated Water

The biological incorporation (uptake) of airborne HTO can be extremely efficient: up to 99% of inhaled HTO is taken into the body by the circulating blood. Ingested liquid HTO is also almost completely absorbed by the gastrointestinal tract and quickly appears in the blood stream. Within minutes, it can be found in varying concentrations in the organs, fluids, and tissues of the body. Skin absorption of airborne HTO is also important, especially during hot weather, because of the normal movement of water through the skin. For skin temperatures between 30 and 40 degrees C, the absorption of HTO is about 50% of that for HTO by inhalation (assuming an average breathing rate associated with light work, 20 L/min). No matter how it is absorbed, the HTO will be uniformly distributed in all biological fluids within one to two hours. This tritium has a retention that is characteristic of water. In addition, a small fraction of the tritium is incorporated into easily exchanged hydrogen sites in organic molecules. Hence, retention of tritiated water can be described as the sum of several terms: one characteristic of body water, and one or more longer-term components that represent tritium incorporated into organic hydrogen sites.

Metabolism of Other Tritiated Species

Most tritium handled in laboratories is in the form of tritiated gas or tritium oxide. However, tritium handling operations may form other compounds, such as tritiated hydrocarbons and metal tritides. Tritium may also contaminate surfaces and liquids such as pump oil. These materials may present special safe handling problems.

Metallic Getters

Although many metals are commonly used for gettering (chemically combining with) tritium, little information on their metabolic properties is available. Some of these compounds (such as uranium tritide and lithium tritide) are unstable in air. For these, exposure to air produces different results. Uranium tritide, being pyrophoric, releases large quantities of tritiated water; lithium tritide, a hydroxyl scavenger, releases mostly tritium gas.

Tritides of metals (such as titanium, niobium, and zirconium) are stable in air. For particles of these tritides, the primary organ of concern is the lungs. Some of the tritium may leach out in the lung fluids and then be incorporated into the body water. These particles may also produce organically bound tritium from contact with lung tissue, which would further complicate the metabolic process. However, in laboratories where such tritiated metals are handled, the possibility for exposure to airborne particulates of these metals is extremely remote except in accident situations.

Tritiated Liquids

Next to HTO, the most common tritiated liquid is tritiated vacuum pump oil. Experience at DOE facilities has shown that the specific activities of pump oils can easily range from a few mCi/L to a few tens of Ci/L. The wide range in specific activities may result from variations in the tritium concentration and total throughput of tritium. Depending on the history of these pumps, the tritium may be found as HT, HTO, or tritiated hydrocarbons.

Next to pump oils, the next most common group is tritiated solvents. All solvents can be absorbed through the skin and are relatively volatile and toxic. The overall toxicity of tritiated solvents is usually dominated by the chemical nature of the solvent.

Other Tritiated Gases

If tritium is released in a nitrogen- or air-filled glovebox, other tritiated gases may be formed, such as ammonia and methane. The conversion of tritium to tritiated ammonia is small unless the tritium concentration is very high. The toxicity of these gases is not believed to be greater than that of tritium oxide.

Biological Half-Life of HTO

Studies of biological elimination rates of body water in humans date back to 1934, when the body water turnover rate was measured using HDO. Since that time, several additional studies have been conducted with HDO and HTO. A simple average of the data suggests a value of 9.5 days for the measured biological half-life of water in the body with a deviation of ±50%. Calculations based on total fluid intake indicate a similar value. This is reasonable because the turnover rate of HTO should be identical to that of body water. In other words, the biological half-life of tritium is a function of the average daily throughput of water.

The biological half-life of HTO has been studied when outdoor temperatures varied at the time of tritium uptake. The data suggest that biological half-lives are shorter in warmer months. For example, the 7.5-day half-life measured in southern Nigeria is not surprising because the mean outdoor temperature there averages 27 degrees C. In contrast, an average 9.5-day half-life was measured in North America, where the mean outdoor temperature averages 17 degrees C. Such findings are consistent with metabolic pathways involving sensible and insensible perspiration. As such, the skin absorption and perspiration pathways can become an important part of body water exchange routes. It is important to note that personnel who are perspiring will have a greater absorption of tritium from contact with tritiated surfaces. For planning purposes, it is customary to use an average half-life of 10 days. However, it is not used to calculate doses from actual exposures.

Prolonged exposures can be expected to affect the biological half-life. Tritium's interaction with organic hydrogen can result in additional half-life components ranging from 21 to 30 days and 250 to 550 days. The shorter duration indicates that organic molecules in the body retain tritium relatively briefly. The longer duration indicates long-term retention by other compounds in the body that do not readily exchange hydrogen or that metabolize more slowly. However, the overall contribution from organically bound tritium is relatively small, that is, less than about 5% for acute exposures and about 10% for chronic exposures. Methods used to compute the annual limits on intake of air and water specify only the body water component and include the assumption of a 10-day biological half-life, as mentioned above.

Bioassay and Internal Dosimetry

Exposure to tritium oxide (HTO) is by far the most important type of tritium exposure. The HTO enters the body by inhalation or skin absorption. When immersed in tritiated water vapor, the body takes in approximately twice as much tritium through the lungs as through the skin. Once in the body, it is circulated by the blood stream and finds its way into fluids both inside and outside the cells.

According to International Commission on Radiological Protection (ICRP), the derived air concentration (DAC)[106] for tritium gas (HT) and HTO are 200,000 $\mu Ci/m^3$ and 20 $\mu Ci/m^3$, respectively. The ratio of these DACs (10,000) is based on the fraction of the gas converted to HTO in the lines. During exposure to HT, a small fraction of the tritium exchanges in the lung and is transferred by the blood to the gastrointestinal tract where it is oxidized by enzymes. This process results in a buildup of HTO until the HT is removed by exhalation at the end of the exposure. The resultant dose from exposure to this HTO is roughly comparable to the lung dose from exposure to HT. Thus, the total effective dose from an HT exposure is about 10,000 times less than the total effective dose from an equal exposure to airborne HTO. For both HTO and HT exposures, a bioassay program that samples body water for HTO is essential for personnel monitoring at tritium facilities.

Sampling Schedule and Technique

After HTO enters the body, it is quickly distributed throughout the blood system and, within 1 to 2 hours, throughout all water in the body. Once equilibrium is established, the tritium concentration is found to be the same in samples of blood, sputum, and urine. For bioassay purposes, urine is normally used for determining tritium concentrations in body water.

Workers who may be or who have been exposed to tritium are normally required to submit urine samples for bioassay periodically. The sampling period may be daily, biweekly, or longer, depending on the potential for significant exposure.

Special urine samples are normally required after an incident or a work assignment with a high potential for exposure. After a possible exposure, the worker should empty the bladder 1 to 2 hours later. A sample taken after the bladder is emptied should be reasonably representative of the body water concentration. A sample collected before equilibrium is established will not be representative because of dilution in the bladder, or because of initial high concentration in the blood. However, any early sample may still be useful as a sign of the potential seriousness of the exposure.

A pure HT exposure is considered as a combination of a lung exposure from the HT and a whole body exposure from HTO. The HTO comes from the conversion of HT dissolved in the blood. The whole body dose can be determined as outlined above by analysis for HTO in the urine. Because the effective dose equivalents from the lung and whole body exposures are about equal, the total effective dose can be obtained conservatively by multiplying the HTO whole body dose by 2. However, in general, this is too conservative because a release of pure tritium gas with less than 0.01% HTO is highly unlikely. With only a slight fraction (~0.1%) of HTO in the air, the total effective dose is essentially the HTO whole body dose determined by bioassay.

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[106] The DAC is defined as that concentration of an airborne material, which, if a worker were exposed to it for one working year (2,000 hours), would result in a dose of 5 rem to the whole body or 50 rem to any organ or tissue.

As noted above, tritium-labeled molecules in the skin result from contact with metal surfaces contaminated with HT. This form is associated with a longer half-life. Lung exposure to airborne metal tritides may also cause unusual patterns of tritium concentrations in body water because of the slow release of tritium to the blood stream. If such exposures are possible at the facility, it is good practice to follow the elimination data carefully and to look for organically bound tritium in the urine.

The results of the bioassay measurements and their contribution to the worker's dose and general health should be shared with the worker in a timely fashion.

Dose Reduction

The committed dose following an HTO exposure is directly proportional to the biological half-life, which in turn is inversely proportional to the turnover rate of body water. This rate varies from individual to individual. Such things as temperature, humidity, work, and drinking habits may cause rate variations. Although the average biological half-life is 10 days, it can be decreased by simply increasing fluid intake, especially diuretic liquids such as coffee, tea, beer, and wine. Even though the half-life may be easily reduced to 4 to 5 days in this way, a physician should be consulted before persons are placed on a regimen that might affect their health. Chemical diuretics require medical supervision because the resultant loss of potassium and other electrolytes can be very serious if they are not replaced. Such drastic measures can result in a decrease in half-life to 1 to 2 days. Even more drastic is the use of peritoneal dialysis or a kidney dialysis machine, which may reduce the half-life to 13 and 4 hours, respectively. Such extreme techniques should be used only in life-threatening situations involving potential committed dose equivalents that would exceed about 100 rem without any treatment. Based on a 10-day half-life, the committed dose for an intake of 1 mCi of HTO is approximately 67 mrem.

Individuals whose urine concentrations exceed established limits should stop work that involves possible exposure to radiation, whether from tritium or other sources. Work restrictions are suggested or imposed to make certain that the annual dose limits for workers are not exceeded. The operating group may impose stricter limits on their staff than those imposed by the health physics group. Depending on the number of workers available and the importance of the work to be done, doses can be managed to safe levels (from 5 to 100 µCi/L in urine).

Results of bioassay sampling should be given to workers who have submitted samples as soon as they are available. The results may be posted, or the workers may be notified personally. Moreover, the results should be kept in the workers' radiation exposure records or medical files. Like any other radiation exposure, any dose in excess of the limits specified by applicable regulations must be reported to DOE.

TRITIUM MONITORING

The tritium monitoring system at a tritium handling facility is critically important to its safe operation. Operators and others at the facility need to be informed of the status of the processes, the development of any leaks in the primary or secondary containments, or of any releases to the room or environment so that protective measures and corrective action may be taken quickly. The location and degree of surface contamination are equally important to prevent accidental uptakes of tritium by personnel.

In this section, the various techniques used to monitor for tritium in gases (including air), in liquids, and on surfaces will be discussed.

Air Monitoring

Fixed ionization chamber instruments are the most widely used instruments for measuring gaseous forms of tritium in laboratory and process monitoring applications. Portable ionization chamber instruments are also used to control contamination and to supplement fixed instrument measurements. Such simple devices require only an electrically polarized ionization chamber, suitable electronics, and a method for moving the gas sample through the chamber—usually a pump. Chamber volumes typically range from a tenth to a few tens of liters, depending on the required sensitivity. The output is usually given in units of concentration (typically $\mu Ci/m^3$) or, if a commercial electrometer or picoammeter is used, in current units that should be converted to those of tritium concentration. The following rule of thumb can be used to convert current to concentration: 10^{15} × current (amps)/chamber volume (liters) = concentration ($\mu Ci/m^3$). For real-time tritium monitoring, the practical lower limits of sensitivity range from 0.1 to 10 $\mu Ci/m^3$. External background radiation or the presence of radon can lower the sensitivity of the instrument.

For measurements of low concentrations, sensitive electrometers are needed. For higher concentrations (>1 mCi/m3 for example), the requirements on the electronics can be relaxed, and smaller ion chambers may be used. Smaller chambers also need less applied voltage. Because of a greater ratio of surface area to volume, residual contamination in the chamber is more likely and is called "memory." This residual contamination elevates the background chamber current. Response times for higher level measurements can be made correspondingly shorter. However, because small chambers and chambers operated at low pressures may have significant wall effects, the above rule-of-thumb may not apply. Such instruments would have to be calibrated to determine their response.

Although most ionization chambers are of the flow-through type that requires a pump to provide the flow, a number of facilities use "open window" or "perforated wall" chambers. These chambers, which employ a dust cover to protect the chamber from particulates, allow the air or gas to penetrate through the wall to the inside chamber without the need for a pump. These instruments are used as single point monitors to monitor rooms, hoods, gloveboxes, and ducts.

Differential Air Monitoring

Because HTO is more toxic than HT ~10,000, it may be desirable to know the relative amounts of each species following a significant release into a room or to the environment. In the case of stack monitoring, discrete samples of the stack effluent should be taken using bubblers or desiccants with a catalyst for oxidizing the HT. Another technique for differential monitoring uses a desiccant cartridge in the sampling line of an ionization chamber monitor. The result is a measurement of the HT concentration. Without the cartridge, the total tritium concentration is measured. Subtraction of HT from the total produces the HTO concentration. The technique may be used with two instruments or one instrument in which the desiccant cartridge is automatically switched in and out of the sampling line.

Another technique uses a semipermeable membrane tube bundle in the sampling line to remove the HTO (preferentially over the HT), which is directed to an HTO monitor. After removing the remaining HTO with another membrane dryer, the sampled air is directed to the HT monitor. Although this technique is slower than the one requiring a desiccant cartridge, it does not require a periodic cartridge replacement. Furthermore, it can be adapted to measure tritium in both species in the presence of noble gases or other radioactive gases by adding a catalyst after the HTO dryers, followed by additional membrane dryers for the HTO. However, because of its slow response, it is more suitable for effluent or stack monitoring than for room monitoring. Because significant releases into a room are quite rare, it is easier to treat any such release as one of HTO than use complicated techniques for continuous differential monitoring.

Discrete Air Sampling

Discrete sampling differs from real-time monitoring in that the sampled gas (usually air) and is analyzed for tritium content (usually by liquid scintillation counting). The usual technique is to flow the sampled air through either a solid desiccant (molecular sieve, silica gel, or Drierite) or water or glycol bubblers. For low-flow rates (about 0.1 to 1 L/min), bubblers may be used. Bubblers are more convenient for sampling, but are less sensitive than the solid desiccant cartridges if the water in the desiccant is recovered by heating. Glycol or water may be used, but glycol is preferred for long-term sampling. In any case, the collected water is then analyzed for HTO. For differential monitoring of HTO and HT, a heated catalyst (usually a palladium sponge) is used between the HTO desiccant cartridge or bubblers and the HT cartridge or bubblers. This is currently the preferred method for monitoring stacks for reporting purposes. In a different arrangement, palladium is coated on the molecular sieve in the HT cartridge to oxidize the HT into HTO, which is then absorbed by the molecular sieve. However, this technique is used primarily for environmental monitoring.

Another technique for sampling HTO in room air is to use a "cold finger" to freeze HTO out of the air. An alcohol and dry ice mixture in a stainless steel beaker works well. To determine the concentration, the relative humidity must be known. Another sampling technique is to squeeze a soft plastic bottle several times to introduce the air (containing the HTO) into the bottle. A measured quantity of water is then introduced, and the bottle is capped and shaken. In a minute or less, essentially all the HTO is taken up by the water, which is then analyzed.

Other techniques involve placing a number of vials or other small specially designed containers of water, cocktail, or other liquid in selected locations in the area being monitored. After a period of time (usually a number of days), the liquid in the containers is analyzed. The result is qualitative (for open containers) to semiquantitative (for specially designed containers).

Process Monitoring

Ionization chambers are typically used for monitoring stacks, rooms, hoods, gloveboxes, and processes. The outputs can be used to sound alarms, activate ventilation valves, activate detritiation systems, and perform other functions. In general, it can be expected that stack, room, and hood monitors will require little nonelectronic maintenance (i.e., chamber replacement because of contamination). Under normal circumstances, the chambers are constantly flushed with clean air and are not exposed to high tritium concentrations. However, glovebox monitors can be expected to eventually become contaminated, especially if exposed to high concentrations of HTO. Process HT monitor backgrounds can also be expected to present problems if a wide range of concentrations (4 to 5 orders of magnitude) are to be measured.

Mass spectrometers, gas chromatographs, and calorimeters are the main instruments used for process monitoring. Because of their relative insensitivities, these instruments cannot detect tritium much below a few parts per million (Ci/m3). For this reason, the analytical results and the related health physics concerns must be interpreted carefully. It is not uncommon to find that samples showing no trace of tritium when analyzed on a mass spectrometer may actually have a concentration of several curies of tritium per cubic meter. In spite of their contamination problems, ionization chamber instruments are useful for measuring these lower concentrations and for providing instant indications of changing concentrations that are not possible with the more sophisticated instruments.

Surface Monitoring

Any material exposed to tritium or a tritiated compound has the potential of being contaminated. Although it is difficult to quantify tritium contamination levels, several methods are available to evaluate the extent of contamination, including smear surveys and off-gassing measurements. Good housekeeping and work practices are essential in maintaining contamination at acceptable levels.

For health or safety implications, an indication of loose, removable tritium contamination is more valuable than a measurement of the total surface contamination. Loose tritium can be transferred to the body by skin contact or inhalation if it becomes airborne. As a result, loose contamination is routinely monitored by smears, which are wiped over a surface and then analyzed by liquid scintillation or proportional counting.

The smears are typically small round filter papers used dry or wet (with water, glycol, or glycerol). Wet smears are more efficient in removing tritium, and the results are more reproducible, although the papers are usually more fragile when wet. However, results are only semiquantitative, and reproducibility within a factor of 2 agreement (for wet or dry smears) is considered satisfactory. Ordinarily, an area of 100 cm2 of the surface is wiped with the smear paper and quickly placed in a vial with about 10 mL of liquid scintillation cocktail, or 1 or 2 mL of water with the cocktail added later. The paper must be placed in liquid immediately after wiping because losses from evaporation can be considerable, especially if the paper is dry. The efficiency of the liquid scintillation cocktail is only slightly affected by the size of the swipe. Foam smears are also available commercially. These smears dissolve in most cocktails and do not interfere significantly with the normal counting efficiency.

Smears may be counted by gas-flow proportional counting. However, because of the inherent counting delays, tritium losses before counting can be significant. Moreover, counting efficiencies may be difficult to determine and may vary greatly from one sample to the next. Another drawback is potential contamination of the counting chamber when counting very "hot" smears. For all of these reasons, a liquid scintillation spectrometer is the preferred system.

An effective tritium health physics program should specify the frequency of routine smear surveys. Each facility should develop a routine surveillance program that may include daily smear surveys in laboratories, process areas, step-off pads, change rooms, and lunchrooms. In many locations within a facility, weekly or monthly routine smear surveys may be sufficient. The frequency should be dictated by operational experience and the potential for contamination. In addition to the routine survey program, special surveys should be made following spills or on potentially contaminated material being transferred to a less controlled area to prevent the spread of contamination from controlled areas.

The surface contamination levels acceptable for the release of materials from radiological areas may be found in the DOE Radiological Control Standard and DOE Order 5400.5.

Tritium Probes

In general, the total tritium contamination on a surface can be measured only by destructive techniques. When tritium penetrates a surface even slightly, it becomes undetectable because of the weak energy of its beta particles. With open-window probes operated in the Geiger Mueller (GM) or proportional regions, it is possible to measure many of the betas emitted from the surface. Quantifying that measurement in terms of the total tritium present is difficult because the history of every exposure is different. Consequently, the relative amounts of measurable and unmeasurable tritium are different.

Such monitoring probes are used to survey areas quickly before more careful monitoring by smears, or to monitor the smears themselves while in the field. The probe must be protected carefully from contamination. When monitoring a slightly contaminated surface after monitoring a highly contaminated one, contamination of the probe can be an immediate problem. Placing a disposable mask over the front face of the probe can reduce, but never eliminate this contamination, particularly if the tritium is rapidly outgassing from the surface. Sensitivity of the instrument depends on many factors, but should be about 103 to 104 dpm/cm^2.

For highly contaminated surfaces (>1 mCi/100 cm^2), a thin sodium iodide crystal or a thin-window GM tube can be used to measure the characteristic and continuous x-rays (Bremsstrahlung) emitted from the surface as a result of the interaction of the beta particle with the surface material.

Off-Gassing Measurements

Off-gassing can be measured using one of two methods. The simplest method is to "sniff" the surface for airborne tritium using a portable or fixed tritium monitor. The most reliable method, however, uses a closed-loop system of known volume and a flow-through ionization chamber monitor. By placing the sample inside the volume and measuring the change in concentration over time, tritium off-gassing rates can be determined accurately on virtually any material. The initial off-gassing rate is the required value because the equilibrium concentration may be reached quickly in a closed volume, especially if the volume is small because of recontamination by the airborne tritium.

The uptake of tritium from off-gassing materials is difficult to predict. Off-gassing tritium that is readily measured indicates contaminated equipment that should not be released for uncontrolled use.

Liquid Monitoring

Liquid is almost universally monitored by liquid scintillation counting. The liquid has to be compatible with the cocktail. Certain chemicals can degrade the cocktail. Others may retain much of the tritium; still others result in a high degree of quenching. In addition, samples that contain peroxide or that are alkaline may result in chemiluminescence that can interfere with measuring. Such samples should first be neutralized before counting. Chemiluminescence and phosphorescence both decay with time. Phosphorescence, activated by sunlight or fluorescent lighting, decays in the dark in a few minutes (fast component) to several days (slow component). Chemiluminescence, the result of chemical interaction of sample components, may take days to decay at room temperatures, but takes only hours to decay at the cold temperatures of a refrigerated liquid scintillation spectrometer. Distillations may be necessary for some samples.

For rather "hot" samples, as may be the case for vacuum pump oils, Bremsstrahlung counting may be useful. This technique may also be useful for active monitoring of "hot" liquids. Liquids may be monitored actively with scintillation flow cells, which are often made of plastic scintillator material or of glass tubing filled with anthracene crystals. However, both types are prone to memory effects that result from tritium contamination. In addition, flow cells are also prone to contamination by algae or other foreign material that can quickly degrade their counting efficiency.

Airborne Tritium

Tritium released to room air moves readily with normal air current. The room or building ventilation system should be designed to prevent the air from being carried to uncontaminated areas, such as offices or other laboratories where tritium is not allowed. For that reason, differential pressure zoning is commonly used, and released tritium is directed outside through the building stack. In some newer facilities where the large quantities of tritium are being handled, room air cleanup systems are available for emergency use. Following a significant release, the room ventilation system is effectively shut down, the room is isolated, and cleanup of room air is begun.

Secondary Containment

The most important control for preventing a release of tritium to the room atmosphere is the use of containment around the source of tritium. This containment usually takes the form of a glovebox, which is then a secondary containment if the tritium is already contained within the process plumbing, which is the primary containment. Even if the tritium is on the outside surface of a piece of equipment and located inside the glovebox, through popular usage, the box is still referred to as the secondary containment.

Gloveboxes used for tritium work typically are made of stainless steel or aluminum and use gloves made of butyl, neoprene, or Hypalon. Windows are made of glass or Lexan. In order to reduce the amount of tritium released to the atmosphere, gloveboxes where significant quantities of tritium are handled incorporate detritiation systems that process the glovebox atmosphere and remove the tritium. These detritiation systems, including the room cleanup systems mentioned above, convert released HT to HTO and collect the HTO on a molecular sieve for later recovery or burial. Newer systems use metal getters that recover HT without resorting to oxidation. These getters, which can only be used in certain glovebox atmospheres, can be heated to release and recover the HT easily.

The atmosphere in the glovebox may be air, nitrogen, argon, or helium, depending on the type of activity in the box. Even in boxes with inert gas atmospheres, small amounts of moisture and oxygen exist. Any release of tritium gas in the box will eventually be converted to the oxide. As a result, the oxide will slowly diffuse through the gloves and contaminate their outside surfaces. For that reason, personnel using gloveboxes that have had tritium releases are required to wear one or more additional pairs of disposable gloves when working in the glovebox.

Glovebox monitors are used to alert personnel of a release in the box and may be used to activate a cleanup system or to increase the rate of the cleanup process. With releases of tritium in the box, the monitor chamber will eventually develop a memory from contamination, mainly by HTO. Heated monitor chambers are useful in minimizing contamination by HTO.

The relative pressure of the glovebox atmosphere is normally kept negative in order to prevent the gloves from hanging outside the box where passersby may brush against them and to prevent tritium from escaping into the room should a leak develop in the glovebox. However, outward permeation of HTO through the gloves and inward permeation of room moisture are not affected by the pressure inside the glovebox.

Temporary Enclosures

At times, maintenance or repair work is done on equipment that cannot be moved into a glovebox or fume hood and that has a high potential to release tritium. For these activities a temporary box ("tent"), may be constructed over the equipment, and an existing cleanup system installed to process the air. Alternatively, if the tritium at risk is not significant, the enclosing atmosphere may be purged to the stack. If the enclosure is small, gloves and glove ports may be fitted to the side of the enclosure. For larger enclosures entry may be required. In such cases, personnel should work in air-supplied suits inside the enclosure.

Protection by Local Ventilation

In spite of the greater protection afforded by gloveboxes, fume hoods are commonly used at tritium facilities for handling or storing material with low quantities of tritium or with low-level contamination. Limits are generally imposed on the quantities used or stored in these hoods.

Fume hoods are also used to protect personnel at the outside door of glove-box pass boxes where materials are passed into and out of the boxes. Ideally, any tritium released in a hood from outgassing or a leaky container, for instance, is routed to the hood's exhaust duct. However, turbulence may occur at the hood entrance, resulting in backwash and possible contamination of personnel if the face velocity is not adequate for the design of the hood, the activities in the hood, or the local conditions (such as traffic in front of the hood). No hood should be used that has not been thoroughly surveyed and judged acceptable for tritium use.

For small operations local ventilation is commonly provided at the work site through a flexible ventilation duct ("elephant trunk") directed to the room exhaust system. The exhaust of these ducts is generally directed to the building ventilation exhaust system, which of itself may be adequate to supply the needed air flow for the duct without help from an additional in-line blower.

Flexible ducts can provide adequate ventilation during maintenance in a glovebox with a panel removed. In this application, a flexible duct can be connected to a gloveport before the panel is removed, and then the work can proceed safely.

Supplied-Air Respirators

In general, only supplied-air respirators are effective in preventing inhalation of airborne tritium. Two types of air-supplied respirators are available: self-contained breathing apparatus (SCBA) and full-face supplied air masks.

An SCBA, consisting of a full-face mask fed by a bottle of compressed air carried on the worker's back, provides excellent protection against HTO inhalation. Because the mask provides no protection against absorption by most of the skin, the SCBA is normally reserved for emergency use only. The protection factor of 3 or more afforded by the SCBA may be adequate for some applications. An SCBA can be used as an added precaution during certain maintenance or operations that experience has shown should not result in the release of significant amounts of HTO. Nevertheless, the potential for exposure is real, and the SCBA gives the worker time to leave the area if necessary before a skin exposure occurs.

Full-face supplied-air masks are also available. Because the air is normally supplied by a fixed-breathing-air system, they are not practical for many emergency situations and, consequently, are not as popular as SCBAs.

Supplied-Air Suits

Because of the inherent disadvantages associated with respirators and other breathing apparatus, supplied-air plastic suits that completely enclose the body are often used by facilities that handle large quantities of tritium. Although they afford reasonably complete body protection, they are slow to don and cumbersome to wear. For these reasons, they are not favored for rescue work where time and mobility are important considerations. For certain maintenance operations outside of gloveboxes with a high degree of risk, supplied-air suits may be quite useful.

For tritium work, supplied-air suits are constructed of materials that have acceptable permeation protection against HTO and provide good tear and abrasion resistance. Because of the closed environment, and the additional background noise caused by the flow of air into the suits, communication between personnel may require special equipment or methods.

Protection from Surface Contamination

Experience at tritium laboratories has shown that many tritium exposures to personnel occur as a result of contact with highly contaminated surfaces. Sudden and significant releases of airborne tritium occur mostly as the less toxic form HT and are quickly detected by portable or strategically placed, fixed tritium monitors. The result is that the exposure and uptake of airborne tritium are minimized. (Heavy-water reactors, of course, present a more significant risk of exposure to tritiated water vapor than to tritium gas.) The presence and degree of contamination may be unknown until measurements are made. Consequently, the importance of routine and special monitoring surveys for surfaces that personnel might contact cannot be overestimated.

Protective clothing worn by workers is one of the most important aspects of an effective health physics program. Because tritium can be absorbed easily through the skin or by inhalation, personnel protective equipment has to protect against both exposure routes. The following paragraphs describe protective measures and equipment.

Protective Clothing

Lab Coats and Coveralls

Lab coats and coveralls (fabric barriers) are worn in most tritium facilities. Lab coats are routinely worn to protect personal clothing. Coveralls are sometimes worn for added protection instead of a lab coat when the work is unusually dusty, dirty, or greasy. The protection afforded by lab coats and coveralls is minimal (except for short exposures) when tritium is airborne, but they are more effective in preventing skin contact with contaminated surfaces.

Disposable waterproof and water-resistant lab coats and coveralls have been tested at various laboratories. They are not popular for everyday use because of the cost and excessive discomfort inflicted on the worker. Most facilities prefer using ordinary open-weave fabrics for lab coats and coveralls and using an approved laundry for contaminated clothing. Some facilities have chosen to use disposable paper lab coats and coveralls, exchanging the costs associated with a laundry for the costs associated with replacement and waste disposal.

Shoe Covers

Although shoe covers provide protection against the spread of contamination and exposure, the routine use of shoe covers in a tritium facility is usually weighed against actual need. Shoe covers can offer both a degree of personnel protection and control over the spread of contamination on floors. However, in modern facilities where tritium is largely controlled by the use of secondary containment, shoe covers may not be required. Such facilities can easily maintain a clean laboratory environment by the use of regular smear surveys and good housekeeping. Using liquid-proof shoe covers until spills are cleaned up should be considered following spills of tritium-contaminated liquids and solids to prevent the spread of local contamination.

Gloves

In most operations, the hands and forearms of workers are vulnerable to contact with tritium surface contamination. The proper use and selection of gloves are essential. Many factors should be considered in selecting the proper type of glove. These include chemical compatibility, permeation resistance, abrasion resistance, solvent resistance, glove thickness, glove toughness, glove color, shelf life, and unit cost. Gloves are commercially available in butyl rubber, neoprene, polyvinyl chloride (PVC) plastics, latex, etc.

The most common gloves found in tritium laboratories are the light-weight, disposable short glove (usually PVC or latex) used for handling lightly contaminated equipment. Depending on the level of contamination, such gloves may be changed frequently (every 10–20 minutes), a second pair may be worn, or heavier gloves may be used instead. When using gloves for this purpose, the work should be planned so that contaminated gloves doe not spread contamination to surfaces that are being kept free of contamination.

When working in a glovebox using the box gloves, disposable gloves are worn to prevent uptake of HTO contaminating the outside of the box gloves. Again, depending on the level of contamination, more than one additional pair may be required, one of which may be a longer, surgeon's length, glove.

In spite of all the precautions normally taken, workers may occasionally be contaminated with tritium. The skin should be decontaminated as soon as possible after any potential skin exposure to minimize absorption into the body. Effective personal decontamination methods include rinsing the affected part of the body with cool water and soap. If the entire body is affected, the worker should shower with soap and water that is as cool as can be tolerated. Cool water keeps the pores of the skin closed and reduces the transfer of HTO across the skin. The importance of washing the affected skin as soon as possible after contamination cannot be overemphasized. Figure G-8[107] illustrates the effect of speed on reducing the uptake and the resultant dose. Even if gloves are worn when handling contaminated equipment or when working in contaminated glovebox gloves, it is good practice to wash the hands after removing the gloves.

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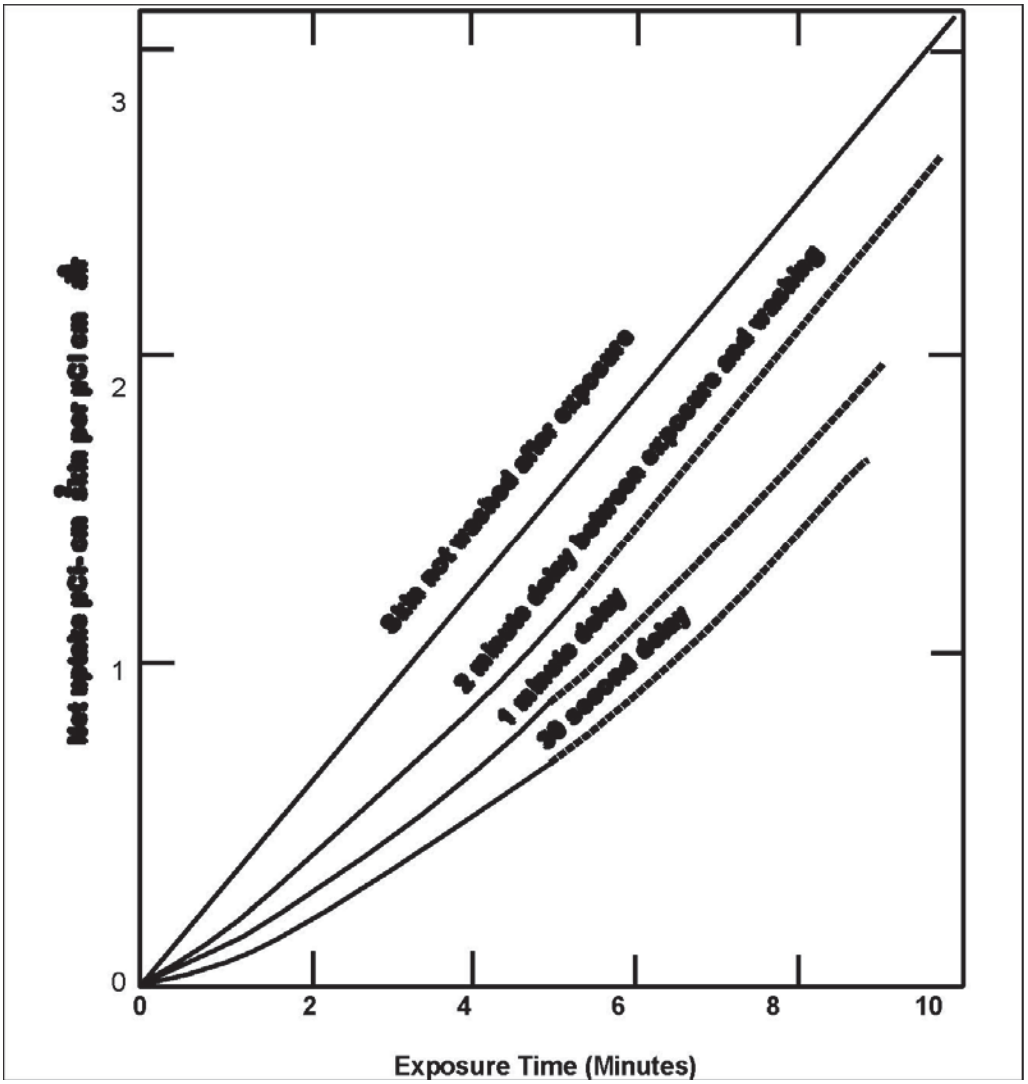
[107] W. R. Bush, *Assessing and Controlling the Hazard from Tritiated Water*, AECL-4150, Atomic Energy of Canada LTD., Chalk River, Ontario, 1972

Figure G-18: Reducing HTO Uptake



This figure shows reduction in HTO uptake by washing after exposure to HTO vapor.

EMERGENCY RESPONSE

It is important to examine the history of accidents that have occurred in tritium facilities and to consider foreseeable unplanned events in order to minimize or mitigate their effects or to prevent their taking place at all. When an accident does occur, requirements for reporting accidents should be followed.

Facilities that handle significant quantities of radioactive material have to have a site-specific emergency plan. All radiological workers at the site should be familiar with certain aspects of this plan. In addition, job assignments involving radiological hazards are typically covered by procedures and work permits that include steps for emergency situations that may arise during the course of the work. Radiological workers should be familiar with these procedures or be accompanied by a radiological control technician (RCT) to provide guidance in case of an emergency.

Emergency Steps to Take

The initial steps to be undertaken following a serious accident should include the following:

- Warning others in the vicinity
- Evacuating the laboratory if an airborne release has occurred
- Requesting any necessary assistance
- Giving urgent first aid in the event of serious injuries (This should take priority over problems that arise from contamination)
- Starting personnel decontamination procedures
- Submitting urine samples following the schedule outlined for nonroutine samples.

Decontamination of Personnel

Personnel should be decontaminated by the following procedures:

- Remove clothing thought to be contaminated
- Wash hands with soap and cool water
- Wash other parts of the body (such as face, hair, and arms) that may have been exposed to tritium, or immediately shower with cool water and soap
- If mouth-to-mouth resuscitation is to be given to a contaminated victim, the victim's mouth should first be wiped with a damp cloth.

Decontamination of Surfaces

Following a tritium spill involving a liquid with high specific activity, the area may have to be isolated and other protective measures taken before cleaning up the liquid. Monitoring for possible airborne tritium must be started to determine the need for respiratory protection or skin protection. After the spill has been cleaned up, residual contamination will remain. Depending on the level of contamination, any further steps needed to prevent the spread of contamination and reduce the level to an acceptable value should be determined.

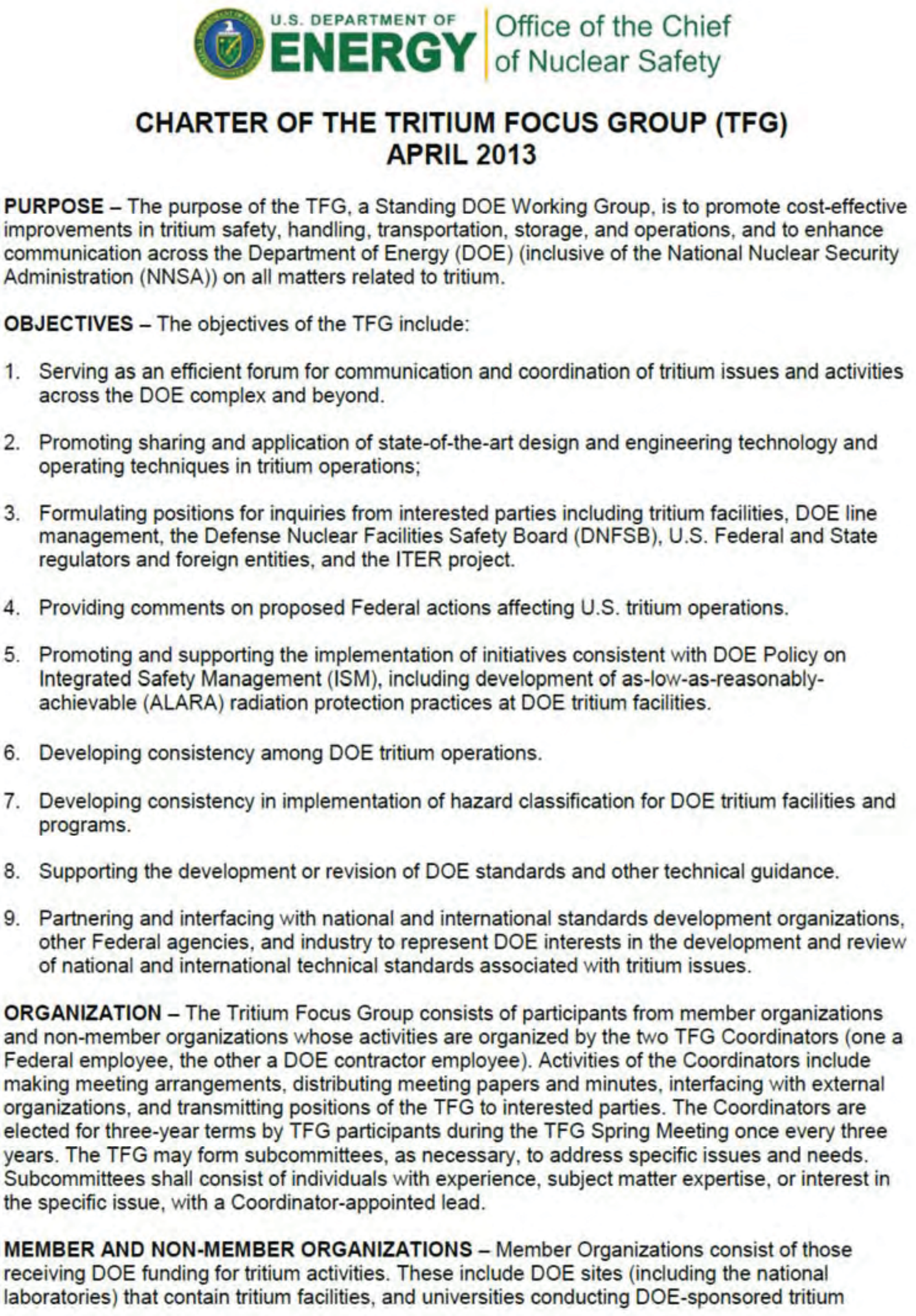
Following a release of tritium gas, surfaces would not be expected to be heavily contaminated. If tritiated water vapor is released, the contamination may be greater, depending on the amount and activity of the released vapor. In any case, smear and air surveys will be used to determine the course of action needed to control and reduce the contamination safely.

Operational Emergencies

This Primer is provided as an information resource only, and is not intended to replace any radiation worker or hazardous materials training. The Primer presents the theoretical concepts and good practices that form the basis of safe tritium handling.

Facilities that handle significant quantities of radioactive material must have a site-specific emergency plan and implementing procedures. The plan and procedures are required by DOE O 151.1C and implementing guidance on the topics of recognition, categorization/classification; protective actions; notification; consequence assessment; etc., can be found in the DOE G 151.1-series, particularly DOE G 151.1-4. All radiological workers at the site should be familiar with certain aspects of this plan, the Order, and the Guides.

APPENDIX H: CHARTER OF THE TRITIUM FOCUS GROUP



U.S. DEPARTMENT OF ENERGY | Office of the Chief of Nuclear Safety

CHARTER OF THE TRITIUM FOCUS GROUP (TFG)
APRIL 2013

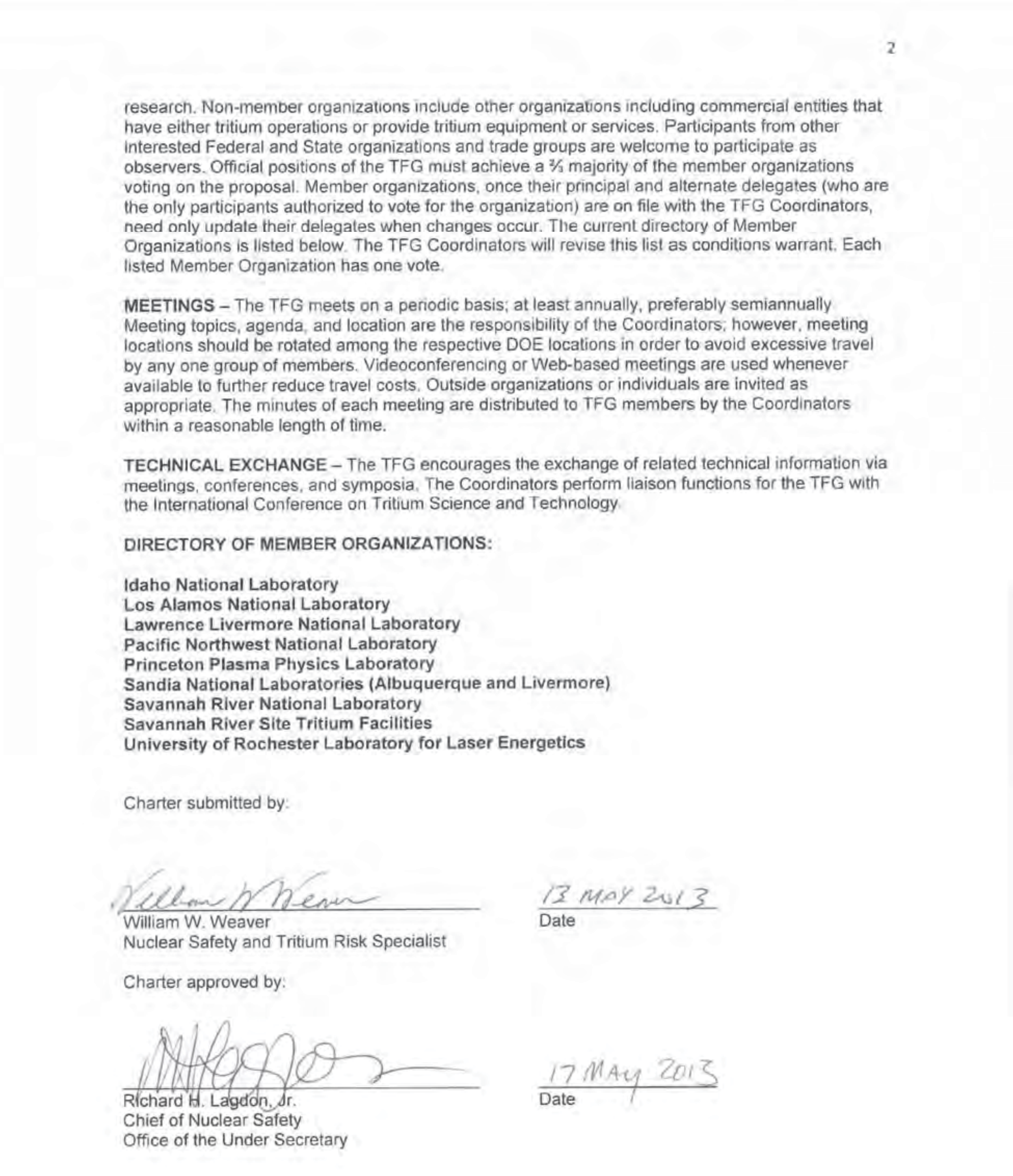
PURPOSE – The purpose of the TFG, a Standing DOE Working Group, is to promote cost-effective improvements in tritium safety, handling, transportation, storage, and operations, and to enhance communication across the Department of Energy (DOE) (inclusive of the National Nuclear Security Administration (NNSA)) on all matters related to tritium.

OBJECTIVES – The objectives of the TFG include:

1. Serving as an efficient forum for communication and coordination of tritium issues and activities across the DOE complex and beyond.
2. Promoting sharing and application of state-of-the-art design and engineering technology and operating techniques in tritium operations;
3. Formulating positions for inquiries from interested parties including tritium facilities, DOE line management, the Defense Nuclear Facilities Safety Board (DNFSB), U.S. Federal and State regulators and foreign entities, and the ITER project.
4. Providing comments on proposed Federal actions affecting U.S. tritium operations.
5. Promoting and supporting the implementation of initiatives consistent with DOE Policy on Integrated Safety Management (ISM), including development of as-low-as-reasonably-achievable (ALARA) radiation protection practices at DOE tritium facilities.
6. Developing consistency among DOE tritium operations.
7. Developing consistency in implementation of hazard classification for DOE tritium facilities and programs.
8. Supporting the development or revision of DOE standards and other technical guidance.
9. Partnering and interfacing with national and international standards development organizations, other Federal agencies, and industry to represent DOE interests in the development and review of national and international technical standards associated with tritium issues.

ORGANIZATION – The Tritium Focus Group consists of participants from member organizations and non-member organizations whose activities are organized by the two TFG Coordinators (one a Federal employee, the other a DOE contractor employee). Activities of the Coordinators include making meeting arrangements, distributing meeting papers and minutes, interfacing with external organizations, and transmitting positions of the TFG to interested parties. The Coordinators are elected for three-year terms by TFG participants during the TFG Spring Meeting once every three years. The TFG may form subcommittees, as necessary, to address specific issues and needs. Subcommittees shall consist of individuals with experience, subject matter expertise, or interest in the specific issue, with a Coordinator-appointed lead.

MEMBER AND NON-MEMBER ORGANIZATIONS – Member Organizations consist of those receiving DOE funding for tritium activities. These include DOE sites (including the national laboratories) that contain tritium facilities, and universities conducting DOE-sponsored tritium



research. Non-member organizations include other organizations including commercial entities that have either tritium operations or provide tritium equipment or services. Participants from other interested Federal and State organizations and trade groups are welcome to participate as observers. Official positions of the TFG must achieve a ⅔ majority of the member organizations voting on the proposal. Member organizations, once their principal and alternate delegates (who are the only participants authorized to vote for the organization) are on file with the TFG Coordinators, need only update their delegates when changes occur. The current directory of Member Organizations is listed below. The TFG Coordinators will revise this list as conditions warrant. Each listed Member Organization has one vote.

MEETINGS – The TFG meets on a periodic basis; at least annually, preferably semiannually. Meeting topics, agenda, and location are the responsibility of the Coordinators; however, meeting locations should be rotated among the respective DOE locations in order to avoid excessive travel by any one group of members. Videoconferencing or Web-based meetings are used whenever available to further reduce travel costs. Outside organizations or individuals are invited as appropriate. The minutes of each meeting are distributed to TFG members by the Coordinators within a reasonable length of time.

TECHNICAL EXCHANGE – The TFG encourages the exchange of related technical information via meetings, conferences, and symposia. The Coordinators perform liaison functions for the TFG with the International Conference on Tritium Science and Technology.

DIRECTORY OF MEMBER ORGANIZATIONS:

Idaho National Laboratory
Los Alamos National Laboratory
Lawrence Livermore National Laboratory
Pacific Northwest National Laboratory
Princeton Plasma Physics Laboratory
Sandia National Laboratories (Albuquerque and Livermore)
Savannah River National Laboratory
Savannah River Site Tritium Facilities
University of Rochester Laboratory for Laser Energetics

Charter submitted by:

William W. Weaver
Nuclear Safety and Tritium Risk Specialist

13 MAY 2013
Date

Charter approved by:

Richard H. Lagdon, Jr.
Chief of Nuclear Safety
Office of the Under Secretary

17 MAY 2013
Date

CONCLUDING MATERIAL

Review Activity:

HQ Offices	Field Offices	Site Offices
DR	AL	Berkeley
EE	CH	Kansas City
EM	ID	Livermore
HS	OR	Los Alamos
NA	ORP	Nevada
NE	RL	NNSA Y-12
RW	SR	Pantex
SC		

Laboratories	Facilities
Ames NL	Hanford
ANL	Kansas City
BNL	ORISE
Fermi NAL	Pantex
INL	NTS
LANL	Savannah River
LBNL	Y-12
LLNL	
PNNL	
PPPL	
SNL	
SLAC	
SRNL	
TJNAF	

Preparing Activity:
DOE-CNS_EM-40

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SAFT-1129-2014