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NUCLEAR FACILITY DESIGN CONSIDERATIONS - VOL 2 OF 3

Main Category:	Nuclear Engineering
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Course #:	NUC-114
Course Content:	102 pgs
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NUC-114 EXAM PREVIEW

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Exam Preview:

1. Deuterium is the lightest of the naturally occurring radioactive nuclides. Tritium is produced in the upper atmosphere as a result of cascade reactions between incoming cosmic rays and elemental nitrogen.
   a. True
   b. False
2. Using TABLE I. Typical confinement provisions for RLWFs, which of the following options below is the secondary confinement provision for a low-level liquid waste facility?
   a. Soil barrier or process building
   b. Storage vessel or basin or treatment system
   c. Dike or berm around vessel or dike or berm
   d. Storage building or process building
3. ^{238}Pu presents special design challenges because of its high specific activity.
   a. True
   b. False
4. During the short-term period following emplacement when short-lived nuclides dominate the hazards associated with a RSWFs disposal facility, the engineered system of barriers should remain effective and should contain the emplaced wastes. Typically, this time period is considered to include at least ____ years but not more than 1,000 years following permanent closure.
   a. 200
   b. 300
   c. 400
   d. 500

5. Using TABLE II. The isotopes of Hydrogen, which of the following hydrogen isotopes has a natural abundance of approximately 99.985%?
    a. Protium
    b. Deuterium
    c. Tritium
    d. Yttrium
6. Using FIGURE 6. Relative radiation resistance of base oils, and the surrounding reference material, which of the following base oils offers the highest radiation resistance from the choices below?
    a. Phosphates
    b. Silicons
    c. Mineral Oils
    d. Ethers
7. According to the reference material, by the late 1950s, it had firmly been established that exposures to tritium in the form of tritiated water vapor (i.e., T2O, or, more correctly, HTO) could be as much as 100,000 times more hazardous than comparable exposures to tritium gas.
    a. True
    b. False
8. The use of __ valves in series on containers filled with tritium, which are to be disconnected from the tritium apparatus on a routine basis, has been in common use for several years.
    a. 2
    b. 3
    c. 4
    d. 6
9. According to the reference material, the process of vitrification is the melting and fusing of materials to produce a glass. It is one of the preferred treatment options for many types of hazardous and radiological waste.
    a. True
    b. False
10. According to the reference material, melter life is dependent largely on how the melter is operated during its productive life. Operating temperatures of vitrification melters range between ___ and ___°C.
    a. 500,900
    b. 800,1000
    c. 1050,1250
    d. 1250,1500

ACRONYMS

ACGIH	American Conference of Governmental Industrial Hygienists
ACI	American Concrete Institute
ADP	automated data processing
AHJ	authority having jurisdiction
ALARA	as low as reasonably achievable
ANS	American National Standards
ANSI	American National Standards Institute
ASHRAE	American Society of Heating, Refrigerating, and Air Conditioning Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing & Materials
AWG	American wire gauge
AWS	American Welding Society
BP&V	Boiler and Pressure Vessel
CAD	computer-aided design
CADD	computer-aided design and drafting
CAM	continuous air monitors
CFR	Code of Federal Regulations
D&D	decontamination and decommissioning
DC	direct current
DCS	distributed control system
DNFSB	Defense Nuclear Facilities Safety Board
DOE	Department of Energy
DOE-EM	Department of Energy, Office of Environmental Management
DOE-RW	Department of Energy, Office of Civilian Radioactive Waste Management
dpm	disintegrations per minute
ER	environmental remediation
FIPS	Federal Information Processing Standards
HEPA	high-efficiency particulate air (filter)
HID	high-intensity discharge
HPS	high-pressure sodium
HVAC	heating, ventilation, and air conditioning
I/O	input/output
IAEA	International Atomic Energy Agency
IEEE	Institute of Electrical and Electronics Engineers
IFM	irradiated fissile material
IFMSF	irradiated fissile material storage facility
ISA	International Society for Measurement and Control (formerly Instrument Society of America)
LET	linear energy transfer
MIC	microbiological-influenced corrosion
NFC	National Fire Code
NFPA	National Fire Protection Association
NPH	natural phenomena hazards

NRC	Nuclear Regulatory Commission
PLC	programmable logic controller
PPHF	plutonium processing and handling facility
PSF	plutonium storage facility
plf	pounds per linear foot
ppm	parts per million
psf	pounds per square foot
psi	pounds per square inch
psia	pounds per square inch absolute
psig	pounds per square inch gauge
PTFE	polytetraflouoroethylene
PVC	polyvinyl chloride
R.G.	Regulatory Guide
RLWF	radioactive liquid waste facility
RSWF	radioactive solid waste facility
SNM	special nuclear material
SSC	structures, systems, and components
SST	safe, secured transport
STP	standard temperature and pressure
TSR	technical safety requirement
UCRF	uranium conversion and recovery facilities
UEU	unirradiated enriched uranium
UEUSF	unirradiated enriched uranium storage facility
UMTRA	Uranium Mill Tailings Remedial Action
UPHF	uranium processing and handling facility
UPS	uninterrupted power supply

ABBREVIATIONS

Ω	resistivity
µm	micron
^oC	degrees Centigrade
^oF	degrees Fahrenheit
^oK	Kelvin
Ar	argon
cal	calorie
Ci	curie
cm^2	square centimeter
cm^3	cubic centimeter
D	deuterium
g	gram
H	hydrogen
H_2O	water
keV	kiloelectron volt (joule)
kVA	kilovolt-ampere
kWh	kilowatt-hour
mCi	millicurie (becquerel)
m^3	cubic meter

mg	milligram
min	minute
mm	millimeter
MW(e)	megawatt (electrical)
N_2O	nitrous oxide
N_2	nitrogen
NO	nitric oxide
NO_2	nitrogen dioxide
O_2	oxygen
Pu(IV)	plutonium polymer
Pu^{238}	plutonium-238
PuF_4	plutonium tetrafluoride
sec	second
T	tritium (the hydrogen isotope of mass-3)
UF_6	uranium hexafluoride
UO_2	uranium oxide

SECTION 2

SPECIAL FACILITIES AND ACTIVITIES

INTRODUCTION AND SCOPE

This section of the Design Considerations Handbook provides design principles that the facility design team should consider for special facilities. These design principles have been developed as a result of design and operating experience with such facilities. These considerations should be consulted when designing facilities whose hazards and operations are similar to those discussed in this section.

2.1 PLUTONIUM PROCESSING AND HANDLING FACILITIES.

2.1.1 Introduction. A plutonium processing and handling facility (PPHF) is typically designed for the following functions involving plutonium:

- shipping and receiving;
- storage;
- chemical processing;
- recovery of scrap/residue;
- characterization, control, and accounting; and
- management of plutonium-contaminated wastes.

Note that ^{238}Pu presents special design challenges because of its high specific activity. Those considerations are not addressed here.

2.1.2 Design Considerations. The following sections provide specific design considerations for a PPHF. The design of PPHFs should consider the following features because of the special characteristics of plutonium and other materials with high specific activity or radiotoxicity.

Shipping and Receiving. A PPHF should be sited away from highly populated areas. It should also have reasonable access to major transportation networks, such as rail systems and interstate highways while maintaining safe distance. Because many state governments have the authority to designate traffic routes for shipment of radioactive material, close coordination with state and local agencies is recommended.

The shipping and receiving area in a PPHF should accommodate the convoy of safe-secured-transport (SST), including its escort vehicles. The area should be free of any obstacles (other buildings and structures) during loading and unloading of the plutonium payload to establish a clear line-of-sight by site security forces.

Radiation monitoring equipment should be available in the shipping and receiving area for surveying the radiation level on the surface of the SST and the containers during receipt of radioactive material from off-site. The shipping and receiving area may also be equipped with a decontamination port if a radiation survey indicates that the surface of a container is contaminated.

Storage. A PPHF should include a storage facility (such as a vault-type room) in the process area to provide storage and staging functions. The following features should be considered in the design of the storage facility:

- Operation of the storage vault should comply with the strict regulation of fire loading.
- Packaging and unpackaging of plutonium in the storage vault area should provide for minimizing the build-up of packaging material.
- Storage racks and shelves should be designed and constructed to meet seismic requirements.
- Spacing between storage units should be sufficient to satisfy criticality controls.
- Layout of storage racks should minimize radiation exposure to operating personnel and provide line-of-sight by safeguards and security.
- Storage racks and shelves should be constructed of noncombustible material and designed to hold the storage containers securely in place and keep them properly separated.
- Storage vault doors, racks, and containers should be designed to accommodate the application of tamper-indicating devices.
- Design of the storage vault should facilitate the ease of performing periodic inventory.
- Pyrophoric material should not be stored in a storage vault. Plutonium metal scraps (e.g., machine turnings, shavings, and fine chips) may be

chemically reactive and should be processed to plutonium oxide before they are stored in the storage vault. [Because plutonium hydrides, carbides, oxycarbides, and nitrides are reactive and potentially pyrophoric, especially in finely divided form (powder), they should be handled in dry, inert (i.e., oxygen-free) atmosphere and should be converted to oxides for prolonged storage.]

- Plutonium oxide is formed either by the reaction of the plutonium metal with oxygen in the air or by calcining plutonium compounds, such as the peroxide, oxalate, and nitrate. Plutonium oxide is generally a chemically inert powder and insensitive to self-radiation damage. However, plutonium oxide can absorb moisture from the air (depending on calcining condition), and incompletely calcined oxide could subsequently release gases, resulting in over-pressurization (bulging) a storage can. If the plutonium compounds are not completely oxidized, the subsequent oxidation process could cause a decrease in the sealed container pressure, thereby imploding a storage can. To prevent or minimize these storage problems, plutonium oxides should be stabilized as prescribed by standards for packaging plutonium for storage.
- The use of plastic bags in bag-in/bag-out operations could cause problems if the heat generated from radioactive decay melts the plastic bags after prolonged storage. The decomposition of the plastic bags could release gases that could also bulge the can. State-of-the-art container packaging methods that either preclude or minimize the use of plastic bags should be considered, especially for long-term storage of plutonium containers.
- Design of storage tanks for aqueous plutonium solutions should consider geometrically safe configurations with respect to nuclear criticality. Plutonium polymer [Pu(IV) solid] could be formed inadvertently under conditions of transient instability, and once formed, could be difficult to destroy. Polymerization in localized areas of low acidity could also occur if an acidic plutonium solution is diluted with water or steam. The plutonium polymers could clog transfer lines, interfere with ion-exchange separations, cause foaming, and constitute a criticality hazard. Detection

of the build-up of polymers and means to remove these solids should be provided in aqueous plutonium storage systems. Prolonged solution storage of significant quantities of amorphous plutonium should be avoided.

Chemical Processing. Plutonium processing operations should be conducted in the plutonium process area of the PPHF. The initial line of defense to protect workers in a process area is the confinement system, which includes enclosures, gloveboxes, conveyor lines, the ventilation system, and process piping. The primary confinement system should be designed to minimize the impact on workers and facilities. A secondary confinement barrier enclosing the primary confinement system provides contamination protection to plant personnel outside the area of secondary confinement. A tertiary confinement system, comprised of the building structure, encloses both the primary and secondary confinement barriers as well as the offices and other support areas, providing the final barrier between the potential contamination and the outside environment. Further design considerations for confinement systems are contained in Part I, Sections 1.1 and 1.2.

The following design features should be considered for facilities that process chemicals:

- The process area should be compartmented to isolate high risk areas, thereby minimizing productivity and financial loss if an accident occurs. Movement of personnel, material, and equipment between the process area and the uncontrolled area (such as the offices) should be through a controlled access area or an air lock.
- The process area should permit ease of egress and material/equipment movement to allow rapid evacuation in the event of an accident. Consideration should be given to providing a ready room near or within the process area where maintenance, operating, and monitoring personnel could be readily available. The room should be located in a low background radiation area.
- Indicators, auxiliary units, and supporting equipment control components that do not have to be adjacent to operating/process equipment should be

installed outside the radiation or contaminated areas. Equipment that requires periodic inspection, maintenance, and testing should be located in areas with the lowest possible radiation and contaminated levels. Equipment that is expected to be contaminated during operation should have provisions for both in-place maintenance and removal to an area of low radiation for repair. Maintenance areas for repair of contaminated equipment should provide for containment or confinement of radioactive material.

- To the maximum extent practicable, the process area should provide sufficient space and versatility to accommodate equipment for programmatic changes and process modifications. It should also be designed to facilitate surveillance. Irregular plant layout (with obstacles) should be avoided where possible.

Two different types of chemistry are generally employed for plutonium processing: the aqueous chemical process and pyrochemistry. Aqueous processes that are common to the plutonium production and chemical analysis are: dissolution, precipitation, liquid-liquid extraction, and oxidation-reduction reactions.

The purex process, which has been the typical aqueous process used in plutonium production, involves the extraction and purification of plutonium with tributyl phosphate. Other processes are the production of plutonium tetrafluoride (PuF_4) and the reduction of PuF_4 using calcium and iodine. The purex process should include design features to deal with the use of flam mable liquids, the potential for radiolysis, the iron catalysis of hydrogen peroxide decomposition, and the potential generation of a large volume of plutonium-contaminated wastes.

The design of facilities that employ an aqueous chemical process should consider the following features:

- Systems, structures, and components for aqueous processing should be resistant to highly corrosive liquid and entrained vapors. Depending on the process to be used, stainless steel components are acceptable for nitrate-based systems. Because stainless steel is incompatible with chlorides, special coatings for gloveboxes (e.g., KynarTM) should be considered, along with Teflon TM or derivative polymer piping, valves, pump bodies, and vessels in systems that employ chloride chemistry. Selection of in-line process controls should consider materials compatibility. Automated ion-exchange systems have been used at Los Alamos with great success.

- The sizing of process equipment is necessarily small to accommodate nuclear criticality requirements. In-process storage of feed solutions is efficiently handled in slab tanks or hollow cylindrical tanks. Pencil tanks have also been used; however, the array of such tanks is more complicated and subject to leaks. Selection of gasket, pump, and valve material should recognize the corrosive nature of process solutions. Design of tankage should recognize the potential for post-precipitation and formation of a layer of solid precipitate on the bottom of the tank. Care should be exercised in agitating such a layer if it forms because a nuclear criticality could occur. Hence, tankage with small horizontal surfaces, such as hollow cylindrical tanks, is desirable.
- Piping and valves should be located so that flammable, explosive, or toxic gases or liquids that are necessary to the process can be isolated to prevent injuring workers if an accidental release occurs. The flammable gases should be provided by a hard-piped system with the gas supply located outside the facility in cylinders to limit the total quantity available in the event of a fire or explosion.
- Radioactive liquid piping systems should be designed to avoid notches, crevices, and rough surfaces that might retain radioactive material. The piping system that collects contaminated liquids should be designed so that effluents from leaks in the system can be collected without releasing the liquids into the personnel access areas or to the environment.
- Stainless steel should be used in radioactive waste and process system piping and equipment so that smooth, nonporous, corrosion-resistant materials are in contact with the contaminated, corrosive, and radioactive liquids. The piping system should be of welded construction whenever

practicable. Flanges should be used only when absolutely necessary for servicing.

- Piping or other conduits to convey plutonium solutions or plutonium-contaminated waste liquid should be double-walled or contained within an enclosure provided with a leak-tight barrier. Any potential leakage from the primary pipe should be collected in a geometrically safe sump or tank. Wherever possible, the piping system should be designed to avoid traps that could hold plutonium solutions.

  The process and equipment supporting the pyrochemical processing of plutonium should be designed to accommodate the pyrophoricity of plutonium metal and other materials used in molten salt and molten metal processing, button break-out, and the sampling operations. Features of the design should consider the minimization of dusting from operations. A criticality-safe service vacuum system may be used to clean up dusts.

In addition to the features described above, facilities for the chemical processing of plutonium should address the following:

- The process glovebox system should be designed to minimize moisture pickup by process materials. An inert atmosphere should be considered in gloveboxes where plutonium is processed. Bag-in/bag-out operations should be conducted without compromising glovebox atmosphere integrity.
- The airborne radioactive effluents typically associated with PPHFs are furnace off-gas, airborne dust, off-gas from solvent processes, and corrosive vapor or mists from dissolvers. The design of airborne effluent systems should consider and minimize plutonium holdup at locations in off-gas and ventilation ductwork and include provisions to detect, monitor, and recover the build-up of such material.
- The capability to service equipment should be provided. Equipment should be designed to minimize plutonium holdup. Provisions should be made to remove process material from equipment and to measure plutonium holdup with minimum downtime.

- Because of the pyrophoric nature of plutonium metal, the plutonium process/handling glovebox system should be designed to accommodate glovebox fire safety. A leak detection system should be provided to detect the inleakage of air, which could change the glovebox atmosphere and lead to a plutonium fire. Certain glovebox construction components are combustible—rubber gloves, plastic bags, polyvinyl chloride (PVC) pipes, etc. Thus, the glovebox is vulnerable to involvement in fire, which, in turn, could cause the loss of glovebox integrity.

  Facility design should provide for the continuous monitoring of external radiation exposure levels in process areas (e.g., hot cells and canyons) during maintenance or repair operations. Neutron shields in the form of water jackets should be capable of being monitored for water loss.

- Gloveboxes should be equipped with quick couplings for dry chemical-type extinguishers.

Recovery of Scrap/Residue. Plutonium scrap and residue should be recovered, processed, and accounted for—to the extent practical—according to the special nuclear material (SNM) accounting requirements. To prevent the accumulation of plutonium-containing scrap or residue, space should be provided for expeditious treatment or processing of these materials to allow their return to the main process.

Plutonium could be recovered using various methods, depending on the chemical process employed. For the aqueous process, plutonium could be recovered by means of leaching and dissolution, followed by purification, evaporation, and concentration. For pyrochemistry, the recovery process would include salt flux remelting, hydriding, oxidation, and/or anion exchange.

The following features should be considered to address the recovery and handling of scrap and residue:

- Equipment for recovery and handling of scrap/residue should be designed to minimize dusting and physical losses or spillage. Vessels used for solution treatment, assay, or storage should be of geometrically safe design to preclude accidental nuclear criticality.

- Provisions should be made for crucibles and molds removed or no longer serviceable to be processed to recover or remove residual plutonium before they are discarded.
- Design should provide for the need to process plutonium scrap and residue before any subsequent disposition action is taken. Aqueous (chloride or nitrate system) and pyrochemical processing are required. Stabilization optimally involves separation of plutonium from matrix material in order to minimize the volume of material to be stored. Likewise, separation of plutonium from waste matrices will minimize the amount of transuranic waste to be shipped and placed in a waste repository. The final form of the concentrate should be a stable (but not necessarily highly purified) oxide or metal. The process includes acid dissolution (hydrochloric or nitric), some degree of purification (e.g., ion exchange), precipitation (typically as oxalate), calcining (to decompose the oxalate and produce plutonium oxide), and packaging. If a metal form is required as the end point, the temperature at which the oxalate is calcined should be kept as low as practical. Metal can be produced by direct oxide reduction with elemental calcium in a molten salt medium (calcium chloride). Means to regenerate the calcium chloride salt medium should be included in the process design.

Characterization, Control and Accounting **.** Chemical sampling and analyses should be provided to support process and operations in the process area of the PPHF. Techniques employed for the characterization of plutonium include: metallography, electron microscope, X-ray diffraction, chemical analysis, thermal analyses, and isotopic. The most common detection technique employed is the nondestructive assay detection system, which includes (1) radiation detection based on alpha, gamma radiation, and neutron activation, and (2) calorimetry, which measures the heat output of the radioactive materials.

Several pyrochemical processes are likely to be used in plutonium processing. Molten salt extraction is used to separate americium from plutonium in aged plutonium items. A “saltless” extraction process has been developed at Los Alamos that greatly reduces the amount of waste generated. Electrorefining is used to purify plutonium, leaving impurities behind in an anode “heel” that requires further processing, usually by aqueous means. Relatively pure plutonium oxide can be reduced to metallic form by direct oxide reduction, which requires dissolving plutonium oxide in a calcium chloride salt and introducing metallic calcium as the reducing agent. The resulting calcium oxide can be converted to calcium chloride by reacting with a chlorinating agent. Note that the media for pyrochemistry are typically chloride salts, and require special aqueous process equipment specifications to minimize corrosion. Likewise, the crucibles used to contain the melt also should be processed into a waste form or processed for extraction of plutonium. Exhaust ventilation, handling devices, and local furnace cooling should take thermally hot operations into account.

Design of materials management and storage systems should attempt to achieve inventory extension to the maximum extent possible; that is, to minimize the frequency with which inventory and reconciliation are necessary. This can be accomplished by use of a vault system with a long-term storage vault that can be locked down for a year or more, and a day vault that contains the items that will be used within the year. Sizing of process equipment should recognize the down-time required to complete inventory actions. These actions include cleaning out process equipment, wiping down gloveboxes, consolidating materials, conducting nondestructive assays, and reconciling inventory values.

Management of Plutonium-Contaminated Wastes. Plutonium-contaminated and radioactive wastes generated from the PPHF should be managed and handled safely and effectively. The process system should be designed to minimize the generation of wastes at the source. The waste management system should be designed to limit the release of radioactive materials to the environment.

Process liquid waste should be collected in the liquid waste treatment system and contained in geometrically safe vessels for temporary storage, sampling, and neutralization. Liquid waste should be concentrated by evaporation, and off-gas from the liquid waste evaporator should be sampled for radioactive materials and hazardous chemicals before release to the environment. The concentrator bottom should be collected and solidified in containers with content meeting the waste acceptance criteria of existing or potential waste disposal site(s). Explosive or highly flammable materials should not be stored in proximity to these wastes. U.S. NRC R.G. 3.10, *Liquid Waste Treatment System Design Guide for Plutonium Processing and Fuel Fabrication Plants,* provides useful guidance that should be considered.

2.2 PLUTONIUM STORAGE FACILITIES.

2.2.1 Introduction. PSFs typically contain strategic amounts of plutonium. The guidance contained in this section applies to facilities where strategic amounts of plutonium or significant quantities of other transuranic radionuclides, such as neptunium and californium, are stored. This section does not apply to “in process” or “in use” material, to material in assembly cells for use in weapons, or to material that is packaged in approved containers awaiting either transportation or disposition upon receipt.

Note that ^{238}Pu presents special design challenges because of its high specific activity. Those considerations are not addressed here.

2.2.2 Design Considerations. The design of PSFs should accommodate all planned plutonium handling and storage activities (e.g., analysis, shipping and receiving operations, packaging, and unpackaging). Provisions should be made to minimize the build-up of packaged materials or packaging materials. Receiving operations involving removal of radioactive material from protective shipping containers should be performed in an unpackaging room.

Facility design, to the maximum extent practical, should:

- provide sufficient versatility to accommodate equipment for programmatic changes and modifications and for multi-shift operations,
- provide sufficient spacing between compartments to facilitate relocation and maintenance of equipment in case of manual or automatic storage operations, and
- facilitate expeditious identification, inventory, placement, and retrieval of storage containers.

Facility layout should provide for efficient cleaning, maintenance, and ease of inspection and should consider the requirements for secure location of storage containers, traffic control, and segregation. Door locations should be coordinated with aisles to facilitate access to stored material, for loading and unloading of material, for use of firefighting equipment, and for compliance with NFC NFPA 101, *Life Safety Code*. Bumpers should be provided where necessary to minimize potential damage to the structure of racks from handling equipment. New storage facilities should be physically separated from process operations, storage of nonnuclear materials, flammable or explosive materials or equipment, and functions not directly required for storage operations.

Combustible packaging materials should be stored in metal containers or structures outside a PSF in a location that will not endanger the storage facility or stored material if a fire occurs in the packaging material. No hazardous gases or liquids should be used in PSFs. No natural gas or other fossil fuels should be used for heating purposes unless the heating occurs in a separate building that is clearly isolated from the primary facility.

The design should provide for sufficient spacing and arrangement of compartments and/or containers to facilitate the taking of inventories. Vault doors, racks, and containers should accommodate the application of tamper-indicating devices. Adequate space for measurement should be provided for the required inventory verification and/or confirmation. An automated vault surveillance system should be provided where excessive radiation exposure would result from entry for material control and accountability purposes. The design of the vault and/or system should facilitate inventory requirements.

Those areas of the facility where SNM is stored (e.g., plutonium product storage) should be located in the least accessible (to an intrusion force) area of the plant.

Design of storage tanks for aqueous solutions of plutonium should ensure that they are geometrically favorable with respect to nuclear criticality. When there is a tendency for solids to precipitate, vessels should be instrumented to detect the build-up of solids and designed to facilitate removal of solids.

The ventilation system should be designed to provide adequate heat rejection capacity. DOE-STD-3013-2012, *Stabilization, Packaging, and Storage of Plutonium-Bearing Materials,* provides guidance regarding containers for storage of plutonium oxide and metal containing greater than 50 percent plutonium.

Suitable physical compartmentalization should be considered to limit the quantity of stored materials in each compartment to safe levels, to provide the necessary access features and controls, and satisfy loss limitation criteria.

Cautionary systems (e.g., visual or audible alarms or other warning systems) or interlocks should be considered to prevent inadvertent entry into hazardous areas. Safety alarm systems should annunciate inside and outside the PSF to identify hazardous areas to anyone present in either area. The need for visual alarm devices within the facility, in addition to audible alarm devices, should be considered.

Storage racks should be noncombustible and designed to hold storage containers securely in place, maintain proper separation of storage containers, and maintain structural integrity under normal operational conditions, anticipated events, and accident conditions.

2.3 UNIRRADIATED ENRICHED URANIUM STORAGE FACILITIES.

2.3.1 Introduction. UEUSFs are used to store unirradiated enriched uranium in a solid, liquid, or gaseous form. UEUSF activities may include shipping, receiving, handling, packaging, and unpackaging.

2.3.2 Design Considerations. The design should accomplish the following:

- Accommodate planned UEU handling and storage activities (e.g., analysis, shipping and receiving operations, packaging, and unpackaging). The build-up of packaged materials or packaging materials should be minimized. Receiving operations involving removal of radioactive material from protective shipping containers should be performed in the unpackaging room(s).
- Incorporate into the design ALARA concepts to minimize overall effects on workers, the public, and the environment.
- Provide sufficient versatility to accommodate equipment for programmatic changes, programmatic modifications, and multishift operations.

- Provide sufficient spacing between compartments to facilitate relocation and maintenance of equipment and ease of manual or automatic storage operations.
- Facilitate expeditious identification, inventory, placement, and retrieval of storage containers.
- Provide for sufficient spacing and arrangement of compartments and/or containers to facilitate the taking of inventories. Vault doors, racks, and containers should be designed to accommodate the application of tamper-indicating devices. Adequate space for measurement capability should be provided for the required inventory verification and/or confirmation. The design of the vault system should facilitate inventory requirements. Those areas of the facility where SNM is stored should be located in the least accessible area of the plant.

To expedite recovery from accidents and provide facility versatility, modular construction concepts should be used, where feasible.

No hazardous gases or liquids should be used in UEUSFs. No natural gas for heating purposes should be used unless the heating occurs in a separate building that is clearly isolated from the primary facility.

New storage facilities should be physically separated from process operations, storage of nonnuclear materials or equipment, and functions not directly required for storage operations.

Combustible packaging materials should be stored in metal containers or structures outside a UEUSF in a location that should not endanger the storage facility or stored material should a fire occur in the packaging material. The need to provide automatic fire suppression systems for these areas should be considered.

Facility layout should provide for efficient cleaning, maintenance, and ease of inspection. Layout of floor and access areas should consider the requirements for secure location of storage containers, traffic control, and segregation. Suitable physical compartmentalization should be considered to limit the quantity of stored materials in each compartment to safe levels, to provide the necessary access features and controls, and to satisfy loss limitation criteria. Bumpers should be provided where necessary to minimize potential damage to the structure or racks from handling equipment.

Design of storage tanks for aqueous solutions of enriched uranium should ensure that they are geometrically favorable with respect to nuclear criticality safety. When there is a tendency for solids to precipitate, vessels should be instrumented to detect the build-up of solids and designed to facilitate removal of solids.

Cautionary systems (e.g., visible or audible alarms or other warning systems) or interlocks should be considered to prevent inadvertent entry into hazardous areas. Safety alarm systems should annunciate inside and outside the UEUSF to identify hazardous areas to anyone present in either area. The need for visual alarm devices within the facility, in addition to audible alarm devices, should be considered.

Storage racks should be noncombustible and designed to hold storage containers securely in place, ensure proper separation of storage containers, and maintain structural integrity under normal operations, anticipated operational occurrences, and accident conditions.

Door locations should be coordinated with aisles to facilitate access to stored material, for loading and unloading of material, for use of firefighting equipment, and for compliance with NFC NFPA 101.

Airborne radioactive wastes associated with UEUSFs that should be considered during the design include but are not limited to the airborne releases associated with the venting of storage containers. Cladding or canning failure during dry storage is also a source of such wastes.

2.4 URANIUM PROCESSING AND HANDLING FACILITIES.

2.4.1 Introduction. A UPHF is a facility that receives feed material from sources such as a conversion facility, a reprocessing facility, or fuel/target storage material. A UPHF processes, handles, and produces products such as UO_2, UF_6, uranium metal, reactor fuel assemblies, target assemblies, and nuclear weapons components.

This section is not process-specific. It is applicable to facilities that handle and process uranium; however, it is principally directed at facilities that process and handle uranium enriched in ^{235}U.

2.4.2 Design Considerations. The design of processing facilities should consider inclusion of the design features described below. Design requirements vary significantly depending on the characteristics of the uranium, the type of processing and handling activities, and the characteristics of the site.

- Materials of different uranium assays should be handled in physically different trains of equipment even though duplication of equipment results. If this is not possible, the equipment should be sized for criticality control of the most restrictive condition.
- A definite isotopic specification for reactor returns should be established before facility design is started for refabrication of enriched uranium that has been irradiated and reprocessed.

Metallurgical processes and ceramic materials processing are the two principal types of processes for fabrication of uranium products. The hazards associated with each of these processes should be considered during the design of the fire protection, ventilation, and confinement systems. In addition, the chemical toxicity of uranium should be considered during the design of the facility. The design should provide specific control and isolation of flammable, toxic, and explosive gases, chemicals, and materials admitted to the areas of the facility.

- The design should provide space for shielding, both permanent and temporary, of personnel and/or remote operations of equipment and processes.
- The primary confinement system should be constructed of fire-resistant materials, and the process equipment and process being confined should be designed to prevent or minimize the probability of potential flammable or explosive conditions. Confinement enclosures for flammable metals should be designed with self-contained fire protection and extinguishing equipment; in some cases, inert atmospheres may be desirable within the enclosures.

- To the extent practical, discrete processing steps should be performed in individual process confinements to reduce the amount of hazardous material that can be released by a single or local failure of the confinement system. Process and auxiliary system differential pressure should be maintained to inhibit back-flow of hazardous materials into auxiliary systems.
- Process operations that involve oxide powder or that can generate powder or dust should be provided with special confinement to prevent the spread of contamination. Facility design should preclude the handling of uranium oxides in large open rooms.
- Airborne radioactive wastes typically associated with UPHFs that should be considered during the design include but are not limited to airborne particulate material generated by fabrication processes (e.g., airborne grinding dust). Nuclear criticality safety should be considered in the design of the airborne effluent system.
- When inert confinement system atmospheres are used, moisture removal systems should be considered to maintain long-term stability of packaged material. Small-volume process enclosures should be designed to prevent the enclosed atmosphere from being pressurized by rapid insertion of gloves into the enclosure.

2.5 IRRADIATED FISSILE MATERIAL STORAGE FACILITIES.

2.5.1 Introduction. IFMSFs are self-contained installations for storage of highly radioactive fissile material (e.g., spent fuel and target elements) that has been exposed to a neutron fluence, usually in a nuclear reactor. The irradiated material should be properly clad or canned when received so that leakage from the assemblies is minimized and remains within specified limits. The IFMSF stores the material in a manner that ensures the integrity of the cladding or canning. The stored material is shipped to facilities such as a hot laboratory or high-level solid radioactive waste facility.

This section applies to a water-pool type of storage facility. Dry-type and spent fuel storage facilities that are part of a reactor facility are not covered by this section.

2.5.2 **Design Considerations.** The design of IFMSFs should consider inclusion of the design features described below. Design requirements vary significantly depending on the characteristics of the material, the type of storage, and the characteristics of the site.

- The cooling water system for a water-pool type IFMSF should perform its required functions during normal and anticipated operating conditions and should be capable of limiting the maximum pool temperature. If pool boiling is used as an emergency cooling mechanism, the ventilation system design should consider the quantity of vapor being generated. Concrete and structural design should consider elevated temperatures. Drainage of condensate should be considered in the structure and equipment.
- If the emergency makeup system is not permanently installed, the time required to implement its operation should be conservatively less than the time required to lower the pool water level to the minimum allowable depth or raise the pool temperature to boiling.
- A pool water cleanup system should be provided to maintain water clarity, provide long-term cladding integrity, maintain structural integrity of the storage racks and other submerged structures, and minimize exposure rates and airborne contamination levels on the operating floor to ALARA levels. The piping configuration for the pool cooling water and cleanup system should be designed to eliminate the possibility of siphoning the pool water to a level below the minimum depth required for shielding and/or cooling. Cooling and cleanup systems should consider material deposition and plate-out in piping and equipment.
- The design should also consider the inclusion of filters capable of being either remotely back-flushed or designed so that cartridges can be removed directly into a shielded container. Instrumentation for periodic functional testing of the pool-water cleanup system and heat-exchanger

performance should be considered. The design should use containerized or modularized filters to reduce exposure during maintenance. Filter change-out prior to build-up of radiation levels should be considered.

- The normal water level of the storage pool should be at or near the final design grade level. The water level necessary for in-storage radiation shielding should be at or below grade.
- For water-pool type facilities, the design professional should consider providing the pool liner with a leakage collection system that will allow leakage detection and limit absorption of contaminated pool water by concrete structures.
- A system should be incorporated to detect leakage from stored IFM in the event of a cladding or canning failure that could allow the escape of fission products and other radioactive material greater than specified limits. This system should include the following:
    - Sampling of coolant allows identification of an individual leaking assembly.
    - System components, piping, and instrumentation are appropriately shielded to maintain operator exposures within guidelines and use ALARA design principles to minimize overall exposures.
    - The storage facility provides for the temporary storage of a leaky assembly. These provisions should limit the spread of contamination by a leaky assembly and provide adequate cooling and shielding of the assembly.
- The IFMSF should provide for the interim canning of leaking assemblies until disposal.
- Special design features should be considered for safe loading, removal, and handling of IFM. These systems and equipment should protect against the dropping of shipping casks, IFM assemblies, and other items onto the stored IFM. In water-pool type facilities, damage to the pool during loading and unloading operations should not allow the pool level to drop below the minimum allowable depth. Consideration should be given

to features that will prevent breaching the pool integrity if a shipping cask is dropped.

- Exhaust systems for pool areas should be HEPA-filtered. Other types of air-cleaning devices (adsorbers) should be considered.
- Airborne radioactive wastes typically associated with IFMSFs that should be considered include but are not limited to airborne releases associated with the venting of transport casks and storage vessels. Cladding or canning failure during long-term wet or dry storage is also a source of airborne radioactive wastes.
- Ventilation system design should consider the evaporation, mixing, and condensation of potentially tritiated sources in collection systems above and around pool areas.

2.6 REPROCESSING FACILITIES.

2.6.1 Introduction. A reprocessing facility is typically designed to recover uranium, plutonium, and other selected actinides and selected fission products from irradiated fissile fuel material and target material. The reprocessing facility is typically designed to separate these materials from each other and from any remaining actinides and fission products.

2.6.2 Design Considerations. The design of reprocessing facilities should consider inclusion of the features described below. Design requirements vary significantly depending on the material (fuel) characteristics, the reprocessing technique, and the characteristics of the site.

- Process system and auxiliary system differential pressure should be maintained to inhibit back-flow of contamination into auxiliary systems. The process equipment for transferring toxic and corrosive fluids should use vacuum and gravity where possible. Pumps and jets should have pressure capacity no greater than 10 percent above needed transfer capacity.
- The integrity of process equipment off-gas treatment systems should be ensured for normal operations, anticipated operational occurrences, and accidents.

- The use of directed airflow and back-flow prevention features to feed areas (i.e., shear and dissolver areas) should be considered.
- Mechanical chopper and dissolver off-gas and other process vents should be treated by an off-gas treatment system for removal of nuclides. As a minimum, the treatment system should be designed for particulate removal and should control the release of airborne radionuclides. In addition, the design should incorporate ALARA concepts to minimize impacts on operators and the public/environment.
- Radioiodine adsorber units in the exhaust ventilation/off-gas system should be considered to reduce the radioiodine concentration in the effluent. Additionally, these releases should be ALARA. (See ASME AG-1 for adsorber selection considerations.)

To reduce the amount of hazardous material that can be released if the process equipment fails, the following design provisions should be considered:

- grouping or compartmentalizing process equipment to form units that can isolate the process inventory into modular units;
- the capability to detect leakage from process equipment; and
- selection of the method (e.g., manual, remote-manual, or automatic) of performing corrective actions (e.g., process shutdown) according to the potential hazards associated with a particular release.

Design features that should be considered for maintenance of the confinement systems include the following:

- the use of electrical equipment that precludes or minimizes the introduction of an ignition source in flammable or potentially flammable locations;
- support and protection systems (such as fire protection systems) that do not promote the failure of the principal confinement systems; and
- provisions for sprinklers, water fog, or other suitable systems within the secondary confinement to provide for rapid heat removal and minimum

pressurization of the process cell or canyon and to minimize the loading of ventilation system filters with combustion products.

Process equipment should be designed to operate under process conditions that prevent or minimize the potential for explosive chemical reactions (e.g., solvent vapor explosions, nitrate-solvent reactions). Process system design should provide for all fission product oxidation states expected during processing (e.g., suppression of the volatilization of ruthenium or the prevention of iodate formation).

Systems should be provided to reduce the likelihood and consequences of pressurizing a primary confinement component as a result of an accident.

Airborne radioactive effluents typically associated with reprocessing facilities that should be considered during the design include but are not limited to dissolver off-gas, process vessel vents, and high-level liquid radioactive waste collection and storage tank vents. Effluent system designs should preclude the holdup or collection of fissile material and other material capable of sustaining a chain reaction in portions of the system that are not geometrically favorable. Nuclear criticality safety should be considered in the design of airborne radioactive effluent systems. U.S. NRC R.G. 3.20, *Process Off-Gas Systems for Fuel Reprocessing Plants,* and R.G. 3.32, *General Design Guide for Ventilation Systems for Fuel Reprocessing Plants,* provide useful design guidance that should be considered.

2.7 URANIUM CONVERSION AND RECOVERY FACILITIES.

2.7.1 Introduction. UCRFs receive feed materials (such as UF_6, uranyl nitrate, or UO_3), process these materials chemically, and produce uranium metal, UO_2, and UF_6. Uranium recovery facilities receive and handle scrap feed materials that are of different types, shapes, sizes, uranium contents, and enrichments. The kind of scrap and therefore the process to facilitate recovery of uranium may vary daily. This section is not process-specific, but is principally directed at facilities that produce feed materials for UPHFs and those facilities that recover uranium from scrap provided by UPHFs.

2.7.2 Design Considerations. The design of UCRFs should consider the features described below. Design requirements vary significantly depending on the material characteristics, the type of recovery and conversion processes used, and the characteristics of the site.

- The design should provide special control and isolation of flammable, toxic, and explosive gases, chemicals, and materials admitted to the areas of the facility.
- To the extent practical, the primary confinement system should be constructed of fire-resistant materials, and the process equipment and process being confined should be designed to prevent or reduce the potential for flammable or explosive conditions. Confinement enclosures for flammable metals should be designed with self-contained fire protection and extinguishing equipment; in some cases, inert atmospheres may be desirable within the enclosures.
- Work that could subject personnel to possible inhalation exposures should be performed in process confinement enclosures. Gloveboxes should be the preferred enclosure, but are not always practical. Alternative systems may have to be considered.
- To the extent practical, discrete processing steps should be performed in individual process confinements to reduce the amount of hazardous material that can be released by a single or local failure of the confinement system. Process and auxiliary system differential pressure should be maintained to inhibit back-flow of hazardous materials into auxiliary systems.
- Equipment design should include appropriate interlocks to prevent spills and cross-contamination.
- The design of process systems should minimize the production of scrap and waste.
- Geometric restrictions for nuclear criticality safety should apply to various units of equipment for the different processes used. In addition, other considerations, such as sufficient agitation in a process vessel to prevent the settling of uranium material, should be considered for nuclear criticality safety.

- Leakage of enriched uranium material from processing equipment should be prevented. Design considerations should include, but not be limited to, the use of corrosion-resistant construction materials and features less vulnerable to leakage (e.g., of flanged and/or welded construction).
- Use of thermal insulation on the equipment that processes uranium solutions of high enrichment should be minimized because it absorbs the solution in the event a leak occurs. The uranium-impregnated insulation would be subject to scrap recovery operations. Because the insulation is considered a "full reflector," the equipment together with the insulation may not be geometrically favorable for highly enriched uranium solutions.
- Storage tanks for aqueous solution of enriched uranium should be designed to ensure favorable geometry with respect to nuclear criticality safety. Where there is a tendency for solids to precipitate, vessels should be instrumented to detect settling of solids and be designed to facilitate periodic removal of solids.
- Airborne radioactive wastes typically associated with UCRFs that should be considered during the design include but are not limited to airborne particulate material generated during processing (e.g., airborne grinding dust) and vapors and gases used or generated during the processing. Nuclear criticality safety should be considered in the design of the airborne effluent system.

Uranium Conversion Facilities. Piping systems, surge vessels, and control instruments with associated piping that carry UF_6 gas should be equipped with heat tracing or heated enclosures wherever necessary to prevent solidification of UF_6. Steam may be used as the primary heating agent where low-enrichment material (less than or equal to 2 percent ^{235}U) is involved. At higher enrichments, a dry radiant heat source should be the preferred means of supplying the heating requirements.

Uranium Recovery Facilities. The design of a uranium recovery facility should be approached on a case-by-case basis, considering possible forms of scrap and different assays of material that could be received for processing and possible methods that could be used for enriched uranium recovery. The following design features should be considered:

- Materials of different uranium assays should be handled in physically different trains of equipment even though duplication of equipment results. If this is not possible, the equipment should be sized for criticality control of the greatest uranium enrichment.
- For enriched uranium that has been irradiated and reprocessed, a definitive isotopic specification for the uranium should be adopted before facility design is begun.
- In addition to provisions for handling uranium and other radioactive materials such as trace quantities of fission products and transuranics, the design should provide for the safe handling of other hazardous materials (e.g., acids, bases, organic solvents, fluorine, hydrogen, hydrogen fluoride, and magnesium) used or generated during recovery operations.

2.8 RADIOACTIVE LIQUID WASTE FACILITIES.

2.8.1 **Introduction.** Radioactive liquid waste facilities (RLWFs) store, treat, and dispose of radioactive liquid wastes generated by facilities and activities. This waste includes low-level, high-level, and transuranic-contaminated (to include enriched uranium and ^{233}U) waste. An RLWF may be a separate facility or an adjunct to another facility. RLWFs may include waste treatment activities that separate solid and liquid waste constituents with provisions for disposing of noncontaminated waste.

2.8.2 **Design Considerations.** The design of RLWFs should consider the features described below. Design requirements vary significantly depending on waste characteristics, waste management techniques, and site characteristics.

- The use of multiple barriers should be emphasized when necessary to restrict the movement of radioactive liquid waste that has the potential for human contact or for reducing groundwater quality below requirements.
- Measurement and analysis capability should be provided to determine the volume and radioactivity of wastes fed to collection tank(s). Provisions

should be made for analyzing liquids prior to transfer. Each transfer line should be identified individually. Instrumentation and control systems should be used to provide monitoring and control capabilities associated with confinement, nuclear criticality safety, and/or radiation protection.

- Individual lines should be used for each waste stream fed to central collection tanks, where necessary, to prevent chemical reactions or introduction of contaminants such as complexing agents that could interfere with waste decontamination. The use of traps in radioactive liquid waste lines should be avoided, and piping should be designed to minimize entrapment and build-up of solids in the system. Bypasses that would allow waste streams to be routed around collection tanks should be avoided. The radioactive liquid waste treatment system should contain no bypasses or drains through which waste may inadvertently be released directly to the environment.
- Basic liquid waste treatment concepts include volume reduction, immobilization of radioactive material, change of composition, and removal of radioactive material from waste. The waste treatment concept(s) for a particular application should be selected on a case-by-case basis. To the extent practical, features should be included to allow volume reduction and/or waste solidification (immobilization) to forms required for long-term isolation.
- Provisions should be made to adjust liquid waste characteristics prior to treatment to minimize adverse chemical reactions in the treatment system.
- Recirculating closed-loop cooling systems should be used for facilities and equipment associated with the storage or treatment of high-heat, high-level radioactive liquid waste.
- Provisions should be made for the continuous monitoring and recording of radioactivity, flow volume, pH, and other parameters required for material control and proper waste treatment operations while each volume of industrial waste is being received by an on-site treatment plant. This

monitoring allows optimum control of waste treatment operations and helps prevent unintended off-site releases.

- Liquid process wastes containing radioactive or other hazardous material should be collected and monitored near the source of generation before batch transfer through appropriate pipelines or tank transfer to a liquid waste treatment plant or area. Radiation, liquid level, or conductivity detectors should be provided in collection systems. Monitoring not only provides information useful for planning efficient waste treatment operations, but also can serve as an indicator of unintended fluctuations in process operations.
- The airborne radioactive waste sources typically associated with RLWFs and RSWFs that should be considered during the design include but are not limited to radioactive liquid waste process vessel vents, high-level liquid radioactive waste collection and storage tank vents, airborne effluents from process system vents, and fission product gases. Effluent system designs should preclude the holdup or collection of fissile material or other material capable of sustaining a chain reaction in portions of the system that are not geometrically favorable. Nuclear criticality safety should be considered in the design of airborne effluent systems.
- Provisions should be made to handle combustible gasses generated during waste handling and/or storage.
- Consideration should be given to condensation and deposition of aerosols formed in vent lines.

Liquid Waste Confinement Systems. The following provisions are typical for an RLWF confinement system (see Table I). The actual confinement system requirements for a specific RLWF should be determined on a case-by-case basis.

- The degree of confinement required in a RLWF is both storage-specific and process-specific, but in either case should suit the most restrictive case anticipated.

- The primary confinement system consisting of the process equipment and/or primary storage tanks should operate under process conditions that prevent or minimize the potential of explosive chemical reactions.
- Spills, overflow, or leakage from storage vessels or other primary confinement structures should be collected and retained within a suitable secondary confinement structure (e.g., secondary vessel, dike or berm, elevated threshold within a storage or process building, etc.). The secondary confinement structure should be able to retain the maximum radioactive liquid waste inventory that may be released by a spill, overflow, or leak from the primary confinement structure. For outdoor applications, the capacity should also include maximum predicted precipitation. The structure should also be designed to preclude overtopping due to wave action from the primary vessel failure and, in outdoor applications, to wind-driven wave action. The capability should exist to transfer collected liquid from the secondary confinement structure to a suitable storage location.

TABLE I. Typical confinement provisions for RLWFs.

Material being confined	Primary	Secondary	Tertiary
High-level liquid waste	Primary storage vessel[1] or treatment system equipment[3]	Secondary storage vessel[1] or process cell	Soil barrier[2] or process building[7]
Low-level liquid waste	Storage vessel' or basin[6] or treatment system[5]	Dike or berm[8] around vessel or dike or berm	None
Transuranic waste	Storage vessel' or treatment system[5]	Storage building[7] or process building[7]	None

[1] Double-wall underground storage tanks and transfer piping are typically used to establish primary and secondary confinement barriers. Primary storage tanks have condensers and/or filters in their vent stream. The space between tanks is also ventilated and the exhaust is filtered.

[2] Soil barrier is the engineered backfill material and natural setting surrounding the waste storage tanks. A monitoring capability should be available to detect leakage from the storage tanks into the soil.

[3] Typical treatment equipment includes waste calciner, evaporator, or waste fractionization equipment. Treatment also occurs within the storage vessel (e.g., precipitation).

[4] Single-wall storage tank.

[5] Typical treatment concepts include volume reduction, immobilization of radioactive material, change of composition, and removal of radioactive material from waste.

[6] Interim storage in retention or settling basins.

[7] With elevated threshold or other means of confinement.

[8] When dikes or berms are considered, use of an impervious membrane should be considered to minimize the cost of cleanup should a spill occur.

High-Level Liquid Waste Confinement. Design of a high-level liquid waste confinement system should consider the following:

- Tank and piping systems used for high-level liquid waste collection, treatment, and storage should be of welded construction to the extent practical. Construction materials should be selected to minimize all forms of corrosion. Consideration should be given to stress relieving, welding parameter controls, etc., depending on the materials used. Fatigue failure should be a design consideration where temperature cycling is required (i.e., evaporator systems, etc.).

- Potential non-uniform distribution of decay heat caused by solids in the waste should be considered in the design of storage tanks and any associated cooling system. Agitation of tank contents should be provided, when necessary, to control waste temperature.
- Double-walled piping, multi-pipe encasements, and double-walled tanks should be considered to establish the primary and secondary confinement boundaries in underground portions of high-level liquid waste systems. Provisions should be made to detect leakage from the primary confinement to the interspace.
- Installation of spare pipelines between transfer points should be considered.

Process and waste storage vessels should be vented through appropriate treatment systems that control the release of radioactive material in gaseous effluents, ensuring these releases are ALARA. Design of these systems should consider the following:

- Off-gas should be suitably pretreated upstream of off-gas treatment equipment to remove or reduce the concentration of chemicals that may adversely affect system operation.
- The venting system should prevent overpressure or vacuum conditions from occurring within vessels.
- The venting system should prevent the build-up of hydrogen from radiolysis.
- Tank overflows should be directed to collection systems.

Integrity of the primary confinement boundary should be determined by some or all of the following measures:

- vessel inventory monitoring (e.g., liquid level sensors);
- on-line leakage monitoring for the interspace of double-walled vessels (e.g., airborne activity monitors, sump level sensors, conductivity cells);
- leakage monitoring outside confinement vessels (e.g., surveillance wells to detect leakage into ground water);

- capability for periodic visual surveillance, including remote visual surveillance with closed-circuit television;
- periodic evaluation of test coupons of primary tank construction materials installed before the tank was placed in service; and
- other surveillance or testing measures, as appropriate.

Low-Level Liquid Waste Confinement. The following should apply to the low-level liquid waste confinement system:

- An impervious dike or berm around the process system should provide secondary confinement for low-level liquid wastes.
- Process and waste storage vessel vents should be provided.
- Retention basins should be lined, fenced, and posted with appropriate radiation warning signs. A system for monitoring radionuclide migration from the basin should be available.
- An impervious berm or dike should be capable of retaining the maximum radioactive liquid waste inventory that may be released by a leak or failure of a primary confinement vessel. A capability should exist to transfer waste that has leaked into the secondary confinement.
- A means of removing rain or snow from the secondary confinement area should be provided unless rain or snow is precluded from entry to the confinement area. Monitoring or testing of the removed rain or snow should be considered.

Transuranic-Contaminated Liquid Waste Confinement. The following features should be considered in the design of a transuranic-contaminated (to include enriched uranium and ^{233}U) liquid waste confinement system:

- A storage or process building should provide secondary confinement for transuranic-contaminated liquid wastes.
- Tank and piping systems used for transuranic-contaminated waste collection, treatment, and storage should be of welded construction to the extent practical. Construction materials should be selected to minimize all forms of corrosion. Consideration should be given to stress relieving,

welding parameter controls, etc., depending on the materials used. Fatigue failure should be a design consideration where temperature cycling is required (i.e., evaporator systems, etc.).

- Process and waste storage vessel vents should be considered.

Nuclear criticality safety should be considered in the design of primary and secondary confinement structures and components.

2.9 RADIOACTIVE SOLID WASTE FACILITIES.

2.9.1 Introduction. Radioactive solid waste facilities (RSWFs) are used to store, treat, and dispose of the range of solid waste generated by DOE facilities and activities. This waste contains high-level, low-level, and transuranic-contaminated solid waste including radioactive-mixed waste. An RSWF may be a separate facility or an adjunct to another facility.

2.9.2 Design Considerations. The design of RSWFs should consider the design features described below. Design requirements vary significantly depending on waste characteristics, waste management techniques, and site characteristics.

- Cooling water systems or cooling air systems should be provided, where required, for facilities and equipment associated with the interim storage or treatment of high-level radioactive solid waste, and to maintain the long-term integrity of the primary confinement boundary. To the extent practical, passive cooling means should be used for air cooling systems.
- Instrumentation and control systems should be required at an RSWF to provide monitoring and control capabilities associated with confinement, nuclear criticality safety, and radiation protection.

High-Level Waste Disposal Facility Confinement. During the short-term period following emplacement when short-lived nuclides dominate the hazards associated with a disposal facility, the engineered system of barriers should remain effective and should contain the emplaced wastes. Typically, this time period is considered to include at least 300 years but not more than 1,000 years following permanent closure.

Technical criteria associated with the engineered system of barriers should address the following:

- establishment of a high-integrity confinement system during emplacement (to limit the rate of release of radionuclides from the system),
- in situ stresses affecting the engineered system of barriers,
- corrosion affecting the engineered system of barriers,
- radiological effects on barrier integrity, and
- contact with groundwater.

During the long-term period, reliance should not be placed on the engineered system of barriers to contain emplaced waste. Confinement during the long-term period should be accomplished by the geologic setting. Technical criteria associated with the geologic setting should address the following:

- leaching characteristics of waste and waste binders;
- site and soil characteristics, including fractures, porosity, hydraulic conductivity, sorption, hydraulic gradient, and thermal gradient;
- long-term geologic stability;
- groundwater travel time;
- absence of resources that would be an incentive for human intrusion; and
- stability of rock mass.

The facility should allow retrieval of wastes during the 50-year period following emplacement and before permanent closure of the facility.

Low-Level Waste Disposal Facility Confinement. Low-level solid waste that is disposed to the ground should be confined by a site-specific system of barriers that may include—but not necessarily be limited to—waste form, waste packaging, and the geologic setting.

When site permeability characteristics do not provide the required confinement capabilities, the confinement system should be augmented by the following:

- constructing low permeability walls around the low-level waste,

- lining the walls and bottom of the excavated area with low permeability material, and
- other suitable methods for reducing permeability.

Means should be provided to minimize contact of emplaced low-level waste with water. Active water-control measures should not be required following permanent closure. Typical requirements for water control are as follows:

- Placing a layer of highly permeable material (e.g., sand, gravel) beneath the low-level waste to channel any percolating water to a sump.
- Mounding the soil surface to facilitate surface water runoff.
- Use of a suitable low-permeability cover material (e.g., clay) over the disposal area to prevent contact of the waste by infiltrating rainwater. This cover material should be protected by a layer of overburden (e.g., sand, gravel, top soil).
- A site diversion system for surface water runoff during operation of the facility. (This system should not be required following site permanent closure.)
- Temporary protective covers (e.g., a tarpaulin) before the completion of the natural in-place soil barrier over the low-level waste.
- Revegetation of the overburden layer.
- Other suitable and reliable means for minimizing water contact with low-level waste.

Solid Waste Confinement Systems. The following provisions are typical for an RSWF confinement system. The actual confinement system requirements for a specific RSWF should be determined on a case-by-case basis.

- In general, the primary confinement should be the radioactive solid waste process system equipment and associated off-gas or vent systems during the treatment stage of processing. In special cases, such as RSWF where the processes or storage include corrosive or noxious materials, the radioactive solid waste process or storage system should be totally enclosed and provided with its own ventilation system and off-gas cleanup system. In such cases, the radioactive solid waste process or storage system should be treated as the primary confinement system. Depending on the waste being processed and stored, the primary confinement and secondary confinement should consist of a site-specific engineered system of barriers (e.g., drums, liners, concrete casks).

- Secondary confinement for radioactive solid waste during treatment should consist of a process cell or building and its ventilation system; secondary confinement for radioactive solid waste during interim storage should be provided by a storage building or structure.
- Tertiary confinements are not needed in most cases for radioactive solid waste during the treatment or interim storage phase of the radioactive solid waste management process. Tertiary confinement for radioactive solid waste is typically considered to be the geologic structure of the site.
- In addition to these principal confinement systems, features such as change rooms and special access ways should be used to minimize the spread of radioactive contamination within the facility.

Primary Confinement System. The primary confinement system consists of process system equipment and its associated ventilation and off-gas system, storage containers, or other waste and site-specific engineered barriers.

Secondary Confinement System. The secondary confinement system consists of the process cell barriers and the ventilation systems associated with the cells or building, or a storage building or structure. In some cases, a drum, cask, or other waste and site-specific engineered barrier should provide secondary confinement.

- Penetrations of the secondary confinement should have positive seals to prevent migration of contamination out of the secondary confinement area.
- Process cells should be supplied with ventilation air from the building ventilation system, and should be provided with exhaust ventilation to control ventilation flow in the event of a credible breach in the secondary confinement barrier. Pressure in the compartments should be negative with respect to the building ventilation system. Special features (e.g., air

locks or enclosed vestibules) should be considered for access through secondary and tertiary confinement barriers.

Tertiary Confinement System. The natural geologic setting comprises the tertiary confinement system. The tertiary confinement system should meet the following performance objectives:

- following permanent closure, ongoing site maintenance should not be needed, and
- in the absence of unplanned natural processes or human contact with a low-level waste disposal facility, calculated contaminant levels in groundwater at the site boundary should not exceed the maximum contaminant levels established in Federal statutes.

2.10 TRITIUM FACILITIES. The DOE Tritium Focus Group has issued DOE HDBK-1129-2008, *Tritium Handling and Safe Storage.* This Handbook provides reference and background information that should be considered during the design of tritium handling and storage facilities.

2.10.1 Introduction. The design and operational philosophy of the older tritium facilities focused on worker protection. The tritium handling equipment was located in airflow hoods, and any releases from the equipment went into the ventilation system, up the stack, and directly into the environment. As long as local airflow requirements were maintained at the proper levels, exposures to workers were low to nonexistent. The high stacks maintained exposures to workers and personnel working or living downwind of such releases well below acceptable levels.

By the mid-to-late 1960s, more modern operational philosophies began to emerge. The design philosophy changed and placed the equipment that handled substantial quantities of tritium into gloveboxes; the gloveboxes in turn were equipped with their own, individualized cleanup systems. Although the initial intent of this type of change was to reduce tritium emissions to the environment to near zero by eliminating large releases, the tritium emissions were only reduced by some 10 to 25 percent. Thus, the primary lesson learned from this type of operational change strongly suggested that most tritium emissions to the environment did not come from large releases, but from the background releases from the facilities themselves.

By the early 1970s, entirely new facility designs began to emerge. Having adapted to the lessons learned from the earlier operational changes, these newer facility designs also placed the bulk of the equipment that handled substantial quantities of tritium into gloveboxes. Rather than having individualized glovebox cleanup systems, the cleanup systems used in the newer facility designs heavily emphasized the use of a centralized cleanup system for glovebox operations. Evacuable covers were sometimes placed over glovebox glove ports to minimize the potential for permeation from the gloveboxes to the working rooms. Additional cleanup systems were installed to handle the emissions from the glove port covers and other vacuum systems that were not compatible with the centralized glovebox cleanup system. Real-time monitoring capabilities were added to the gloveboxes and cleanup systems to track their respective reliabilities. Rapid response, real-time monitoring capabilities were added to the room air monitoring capabilities to protect the worker further. Rapid response, real-time monitoring capabilities were also added to the stack exhaust monitoring capabilities to more reliably monitor releases to the environment. And, in some cases, specialized cleanup systems were added at the room air ventilation level to allow for the cleanup of large releases into a working room, before the tritium released into the ventilation system was finally released to the stack.

With more than 25 years of operational experience with various types of newer facility designs, it is clear that the bulk of the emissions from all tritium facilities over the last 25 years have come from the background emissions of the facilities themselves.

To better understand how these emissions come about, an appreciation for the behavior of tritium in the facilities is necessary. Once understood, prospective designers and engineers will begin to understand what they can and cannot control through design innovations and techniques.

2.10.2 **Sources of Tritium.** Tritium is the lightest of the naturally occurring radioactive nuclides. Tritium is produced in the upper atmosphere as a result of cascade reactions between incoming cosmic rays and elemental nitrogen.

In its simplest form, this type of reaction can be written as

$$ {}^{14}_{7}\mathrm{N} + {}^{1}_{0}\mathrm{n} \rightarrow {}^{12}_{6}\mathrm{C} + {}^{3}_{1}\mathrm{H} \qquad (1)$$

Tritium is also produced in the sun as a sub-set of the proton-proton chain of fusion reactions. Although a steady stream of tritium near the surface of the sun is ejected out into space (along with many other types of particles) on the solar wind, much larger streams are ejected out into space during solar flares and prominences. Being much more energetic than its solar wind counterparts, tritium produced in this manner is injected directly into the earth's upper atmosphere as the earth moves along in its orbit. Regardless of the method of introduction, however, estimates suggest that the natural production rate for tritium is about 4 x 10^6 Ci/yr, which, in turn, results in a steady-state, natural production inventory of about 7 x 10^7 Ci.

Tritium is also introduced into the environment through a number of man-made sources. The largest of these, atmospheric nuclear testing, added approximately 8 x 10^9 Ci to the environment between 1945 and 1975.

Because the half-life of tritium is relatively short, much of the tritium produced in this manner has long since decayed. However, tritium introduced into the environment as a result of atmospheric testing increased the natural background levels by more than two orders of magnitude, and, in spite of its relatively short half-life, the natural background levels of tritium in the environment will not return to normal until sometime between the years 2020 and 2030.

Tritium levels in the environment cannot truly return to background levels, however, because of a number of additional man-made sources. Tritium is also produced as a ternary fission product, within the fuel rods of nuclear reactors, at a rate of 1-2 x 10^4 Ci/1,000 MW(e). (Although much of the tritium produced in this manner remains trapped within the matrix of the fuel rods, estimates suggest that recovery of this tritium could reach levels of 1 x 10^6 Ci/yr.) Light-water and heavy-water moderated reactors produce another 500-1,000 to 1 x 10^6 Ci/yr, respectively, for each 1,000 MW of electrical power. Commercial producers of radioluminescent and neutron generator devices also release about 1 x 10^6 Ci/yr. Thus, tritium facilities operate within a background of tritium from a variety of sources.

2.10.3 The Relative Abundance of Tritium . The isotopes of elemental hydrogen have long been recognized as being special—so special, in fact, that each has been given its own chemical name and symbol. Protium, for example, is the name given to the hydrogen isotope of mass-1, and the chemical symbol for protium is H. Deuterium is the name given to the hydrogen isotope of mass-2; the chemical symbol for deuterium is D. Tritium is the name given to the hydrogen isotope of mass-3. Its chemical symbol is T.

Protium is by far the most abundant of the hydrogen isotopes. Deuterium follows next with a relative abundance of about 1 atom of deuterium for every 6,600 atoms of protium; that is, the D to H ratio (D:H) is about 1:6,600. Tritium is the least common hydrogen isotope. The relative abundance of naturally occurring tritium (i.e., tritium produced in the upper atmosphere and tritium injected directly by the sun) has been estimated to be on the order of 1 tritium atom for every 10^{18} protium atoms. The introduction of man-made tritium into the environment, particularly as a result of atmospheric testing, has raised this level approximately one order of magnitude so that the ambient T to H ratio is now approximately $1:10^{17}$.

The names, commonly used symbols, atomic masses, and relative natural abundances of the hydrogen isotopes are summarized in Table II.

TABLE II. The isotopes of Hydrogen.

Name	Chemical Symbol	Atomic Mass	Natural Abundance (Percent)	Natural Abundance (x:H Ratio)
Protium	H	1.007 825 03	99.985 %	1:1
Deuterium	D	2.014 101 78	0.015 %	1:6,600
Tritium	T	3.016 049 26*	very low	1:1017

* Calculated

2.10.4 The Radioactive Decay of Tritium.

Generic. As the lightest of the pure beta emitters, tritium decays with the emission of a low-energy beta particle and an anti-neutrino; i.e.,

$$ {}^{3}_{1}\mathrm{H} \rightarrow {}^{3}_{2}\mathrm{He} + \beta^{-} + \upsilon_{e} \qquad (2)$$

Tritium decays with a half-life of 12.3232 ± 0.0043 mean solar years or, using 365.2425 mean solar (days) per mean solar year, 4,500.96 ± 1.57 days. The specific activity of tritium is approximately 9,619 Ci/g, and/or 1.040×10^{-4} g/Ci. In addition, the activity density (i.e., the specific activity per unit volume) for tritium gas (T2) is 2.589 Ci/cm^3, under standard temperature and pressure (STP) conditions (i.e., 1 atmosphere of pressure at 0EC), and/or 2.372 Ci/cm^3 at 25EC. Under STP conditions, it can also be shown that these values translate to 58,023 Ci/g-mole and 29,012 Ci/g-atom, respectively.

Beta Emissions. Beta particles interact with matter by colliding with bound electrons in the surrounding medium. In each collision, the beta particle loses energy as electrons are stripped from molecular fragments (ionization) or promoted to an excited state (bremsstrahlung production). Because the rate of energy loss per unit path length (linear energy transfer, or LET) increases as the velocity of the beta particle slows, a distinct maximum range can be associated with beta particles of known initial energy.

The beta decay energy spectrum for tritium is shown in Figure 1. The maximum energy of the tritium beta is 18.591 ± 0.059 keV. The average energy is 5.685± 0.008 keV. The maximum range of the tritium beta (i.e., the mass attenuation coefficient) is 0.58 mg/cm^2.

The adsorption of energy from beta particles that emanate from a point source of tritium has been shown to occur nearly exponentially with distance. This is a result of the shape of the beta spectrum as it is subdivided into ranges that correspond with subgroups of initial kinetic energies. As a consequence, the fraction of energy absorbed, F, can be expressed as

$$F=1-e^{-(\mu/\rho)(\rho)(x)} \qquad (3)$$

where μ/ is the mass attenuation coefficient of the surrounding material, is the density of the surrounding material, and x is the thickness of the surrounding material.

For incremental energy absorption calculations, Equation (3) can be restated as

$$F = 1 - e^{-\mu x} \qquad (3a)$$

where μ (i.e., the linear attenuation coefficient) is the product of the mass attenuation coefficient (μ/ρ) and the density (ρ), and x is the incremental thickness of choice. In gases at 25°C, at atmospheric pressure, for example, the linear attenuation coefficients for the gases hydrogen (H_2), nitrogen (N_2), and argon (Ar), are 1.81 cm^{-1}, 11.0 cm^{-1}, and 12.9 cm^{-1}, respectively. A 5-mm thickness of air will absorb 99.6 percent of tritium betas. A comparable thickness of hydrogen (or tritium) gas will absorb only 60 percent of the tritium betas.

Absorption coefficients for other media can be estimated by applying correction factors to the relative stopping power (the scattering probability) of the material of interest. For the most part, these will be directly proportional to ratios of electron densities. Examples of tritium beta ranges are shown in Table III. The values shown for tritium gas and for air are stated as STP values.

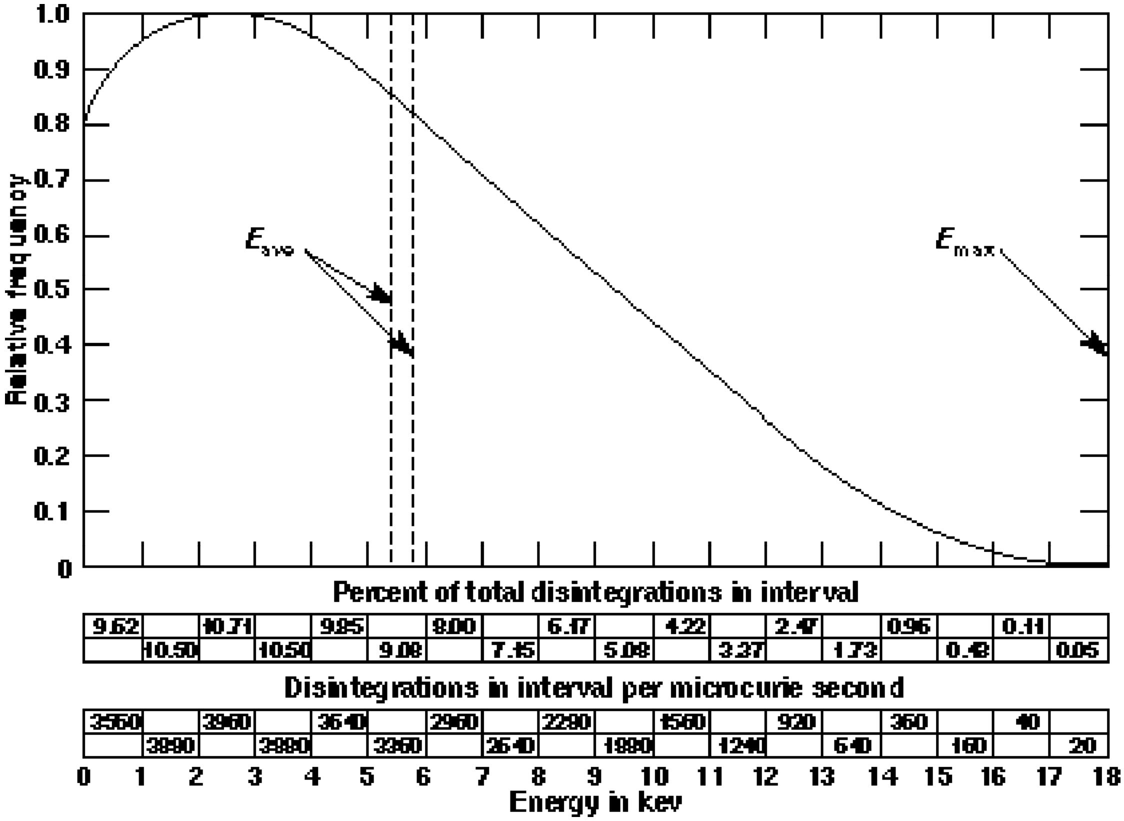

FIGURE 1. Tritium beta decay energy spectrum.

TABLE III. Approximate ranges of tritium betas.

Material	Beta Energy	Range
Tritium Gas	Average	0.26 cm
Tritium Gas	Maximum	3.2 cm
Air	Average	0.04 cm
Water (Liquid)	Average	0.42 µm
Water (Liquid)	Maximum	5.2 µm
Stainless Steel	Average	0.06 µm

Photon Emissions. No nuclear electromagnetic emissions (gamma emissions) are involved in the decay scheme for tritium although it is worth noting that tritium does produce bremsstrahlung (braking radiation) as its beta particles are decelerated through interactions with nearby matter.

2.10.5 The Chemical Properties of Tritium.

Generic. Although the chemical properties of tritium have been described in great detail, three distinctive types of chemical reactions and one underlying principle in particular are worth noting here. The reaction types are solubility reactions, exchange reactions, and radiolysis reactions. The underlying principle is Le Chatelier's Principle. An overview of these types of reactions and Le Chatelier's Principle is presented below.

Solubility Reactions. 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. On the atomic or molecular scale, hydrogen-like atoms, diatomic hydrogen-like species, or larger, hydrogen-like-bearing molecules tend to dissolve interstitially (i.e., they diffuse into the crystalline structure, locating themselves inside the normal lattice work of the internal structure). Schematically, such reactions can easily be described in terms of the generic reactions:

$$H_2 + \text{Material} \rightarrow 2H + \text{Material}, \qquad (4a)$$

$$^{x}H_{y} + \text{Material} \rightarrow {}^{x}H_{y} + \text{Material, and} \qquad (4b)$$

$$^{3}_{1}\mathrm{H} + \text{Material} \rightarrow {}^{3}_{1}\mathrm{H} + \text{Material}. \qquad (4c)$$

Theoretically, however, the underlying mechanics are much more complex. For example, of the generic reactions shown above, none are shown as being reversible. From a chemical perspective, none of these reactions is technically correct because, in most dissolution reactions, the solute that goes in can be expected to be the same solute that comes out. From an operational standpoint, however, experience has shown that, regardless of the tritiated compound that enters into the reaction, an HTO (i.e., a tritiated water vapor) component can be expected to come out. Presumably, this is due to catalytic effects and/or exchange effects that derive from the outward migration of the tritiated species through molecular layers of water vapor that are bound to the surface of the material.

Exchange Reactions. Driven primarily by isotope effects, exchange reactions involving tritium can be expected to occur at a relatively rapid pace. Moreover, the speed at which reactions of this type can occur can be further enhanced by the addition of energy from radioactive decay. For tritium, therefore, reactions similar to the following can be expected, and they can be expected to reach equilibrium in time frames that range from seconds to hours:

$CH_4 + 2T_2 \rightarrow CT_4 + 2H_2$, and (5)

$2H_2O + T_2 \rightarrow 2HTO + H_2$. (6)

Equation (5) describes the preferential form of tritium as it exists in nature in the earth's upper atmosphere. Equation (6) describes the preferential form of tritium as it exists in nature in the earth's lower atmosphere (i.e., in a terrestrial environment).

Equation (6) is particularly important because it describes the formation of tritiated water vapor (i.e., HTO) without the involvement of free oxygen (i.e., with no O2). A comparable reaction that would involve free oxygen would take the form of a classic inorganic chemical reaction, such as

$H_2 + T_2 + O_2 \rightarrow 2HTO$. (7)

But, because a classic inorganic chemical reaction like that depicted in Equation (7) can be expected to reach equilibrium in a time frame that ranges from many hours to several days under the conditions normally found in nature, classic inorganic chemical reactions of this type are not necessary for this discussion.

Radiolysis Reactions. It was noted previously in Section 2.10.4 that the range of the tritium beta is very short. As a consequence, it follows that virtually all of the energy involved in tritium beta decay will be deposited in the immediate vicinity of the atoms undergoing decay. When the medium surrounding the decaying atoms is tritium gas, tritiated water, or tritiated water vapor in equilibrium with its isotopic counterparts, reactions such as those presented in Equations (8) and/or (9) bellow can be expected to dominate. When the medium surrounding the decaying atoms is not a medium that would normally be expected to contain tritium, however, an entire spectrum of radiolysis reactions can be expected to occur.

For typical, day-to-day operations, the most common type of radiolysis reactions in the tritium community can be expected to occur at the interface between the air above a tritium contaminated surface and tritium contaminated surface itself. For these types of reactions, some of the energy involved in the tritium decay process can be expected to convert the nitrogen and oxygen components in the air immediately above the surface (i.e., the individual N_2 and O_2 components in the air) into the basic generic oxides of nitrogen, such as nitric oxide, nitrous oxide and nitrogen peroxide (i.e., NO, N_20, and NO_2, respectively). As the energy deposition process continues, it can also be expected that these simpler oxides will be converted into more complex oxides, such as nitrites and nitrates (i.e., NO_2s and NO_3s, respectively). Because all nitrite and nitrate compounds are readily soluble in water (and/or water vapor), it can further be expected that a relatively large percentage of the available nitrites and nitrates in the overpressure gases will be absorbed into, and/or dissolved into, the mono-molecular layers of water vapor that are actually part of the surface. (See "Modeling the Behavior of Tritium" in Section 2.10.6, below.) With the available nitrites and nitrates now an integral part of the mono-molecular layers of water vapor, it can finally be expected that the most common type of radiolysis-driven reactions should result in the gradual, low-level build-up of tritiated nitrous and nitric acids on the surfaces of most tritium contaminated materials. Over the long-term, it should further be expected that these tritiated nitrous and nitric acids will be broken down into tritiated ammonia compound dissolved into the surface bound layers of water vapor.

For the most part, this particular type of reaction sequence does not normally present itself as a problem in day-to-day tritium operations because (1) the overall production efficiency for these types of reactions is relatively low, and (2) the materials used for the construction of most tritium-handling systems are not susceptible to attack by nitrous and/or nitric acids or by tritiated ammonia compounds. By contrast, however, it should be noted that other types of radiolysis-driven reactions can expected to occur with tritium in the presence of compounds containing chlorides and/or fluorides, and these can easily lead to chloride/fluoride induced stress corrosion cracking. (See, for example, the discussion on "Organics" in Section 2.10.6, below.)

One additional point that is worth noting about radiolysis-driven reactions is that their long-term potential for causing damage should not be underestimated. Although the overall production efficiency for these types of reactions might be expected to be relatively low, the generation of products from these types of reactions can, on the other hand, be expected to occur continuously over relatively long periods of time (i.e., 10-20 years, or more). As a consequence, the long-term effects from these types of reactions can be difficult to predict, especially because very little is known about the long-term, synergistic effects of low-level, tritium micro-chemistry. (See Section 2.10.6.)

Le Chatelier's Principle. 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 like those depicted in Equations (5) and (6), Le Chatelier's Principle states that, when the background tritium levels are increased in nature (by atmospheric testing, for example), the reactions will be shifted to the right in order to adjust to the new equilibrium conditions by readjusting to the naturally occurring isotopic ratios. Thus, we get reactions of the type

$$CH_4 + 2T_2 \rightarrow CT_4 + 2H_2\text{, and} \qquad (5a)$$

$$2H_2O + T_2 \rightarrow 2HTO + H_2. \qquad (6a)$$

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

$$CH_4 + 2T_2 \rightarrow CT_4 + 2H_2\text{, and} \qquad (5)$$

$$2H_2O + T_2 \rightarrow 2HTO + H_2. \qquad (6)$$

By itself, Le Chatelier's Principle is a very powerful tool. When applied singularly, or to a sequential set of reactions like those depicted in Equations (5), (5a), and (5) again, and/or (6), (6a), and (6) again, Le Chatelier's Principle shows that exchange reactions of the types depicted above tend to behave as springs, constantly flexing back-and-forth, constantly 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 (H_2, D_2, T_2, and all combinations thereof), can be expected to dissolve to some extent in virtually all materials, Le Chatelier's Principle can be expected to work equally as well on solubility reactions, like those shown above in the generic Equations (4a), (4b), and (4c). These will be covered in more detail under the heading of Bulk Contamination Modeling (see below).

2.10.6 **Modeling the Behavior of Tritium.** Any model of the behavior of tritium starts with the assumption that all three hydrogen isotopes coexist in nature, in equilibrium with each other, in the nominal isotopic ratios described above in Table III. To this is added the consequences predicted by Le Chatelier's Principle. From both, we get the fundamental relationship,

$$2H_2 + 2D_2 + 2T_2 \rightarrow H_2 + HD + D_2 + HT + DT + T_2. \quad (8)$$

In a terrestrial environment, virtually all of the tritium that exists in nature exists as water or water vapor. Correcting this situation for the natural conversion to water and/or water vapor, Equation (8) becomes

$$2H_2O + 2D_2O + 2T_2O \rightarrow H_2O + HDO + D_2O + HTO + DTO + T_2O. \quad (9)$$

It can also be assumed that the surfaces of all terrestrially bound objects are coated with a series of mono-molecular layers of water vapor. In the final step, it can be assumed 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. (See Figure 2.)

Surface Contamination Modeling. When an overpressure of tritium is added to the system (i.e., the surface, in this case), a perturbation is introduced to the system, and Le Chatelier's Principle indicates that the tritium levels in the mono-molecular layers of water will be shifted to the right; that is,

$$2H_2O + 2D_2O + 2T_2O \rightarrow H_2O + HDO + D_2O + HTO + DTO + T_2O.(9a)$$

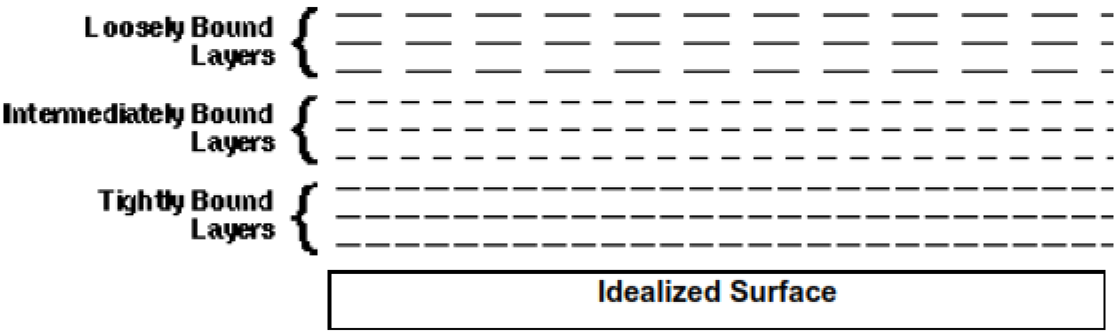

FIGURE 2. Idealized surface showing idealized mono-molecular layers of water vapor.

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; that is,

$$2H_2O + 2D_2O + 2T_2O \rightarrow H_2O + HDO + D_2O + HTO + DTO + T_2O. \qquad (9)$$

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."

Plate-Out Expectations. 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 be more difficult than those described above.

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, itself; see "Bulk Contamination Modeling," below. Under such circumstances, the surface contamination will range from relatively high to very high levels (i.e., from a few tens of thousand to several hundred thousand dpm/100 cm^2), and that mechanical efforts expended to decontaminate such surfaces could be very difficult.

Outgassing Expectations. 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 (5) and (6), (5a) and (6a), (8), (9), and (9a) 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 should not be underestimated. The duration of the trailing portion of the release should not be underestimated either. 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.

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. Under such circumstances, the situation reverts back to the circumstances described above under the heading of “Plate-Out Expectations.” With no additional influx of tritium, tritium incorporated into all of the mono-molecular layers of water vapor will eventually return to background levels, without human intervention, regardless of the method or level of contamination.

Bulk Contamination Modeling. 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,

$$2H_2O + 2D_2O + 2T_2O \rightarrow H_2O + HDO + D_2O + HTO + DTO + T_2O. \qquad (9a)$$

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 actual 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.

Materials Compatibility Issues. 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. For simple solubility reactions, such as

$$H_2 + \text{Material} \rightarrow 2H + \text{Material}, \qquad (4a)$$

$$^{x}H_{y} + \text{Material} \rightarrow {}^{x}H_{y} + \text{Material, and} \qquad (4b)$$

$$^{3}_{1}H + \text{Material} \rightarrow {}^{3}_{1}H + \text{Material}. \qquad (4c)$$

basic compatibility issues should be considered. As a general rule, the solubility of tritium in pure metals and/or ceramics should have 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 should be 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.]

Pressure Considerations. 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.

Temperature Considerations. 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

$$\frac{D \cdot t}{L^2} = 0.45 \qquad (10)$$

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

$$D = 4.7 \times 10^{-3} e^{(-12,900/RT)} \qquad (10a)$$

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), Equation (10) indicates that it will take about 875 years to reach steady-state permeation, at a temperature of 25^oC. At 100^oC, the time frame will be reduced to about 11 years, and at 500^oC, it only takes about 12 hours.

Organics. With the introduction of organic materials into any tritium handling system, the matrix of solubility considerations becomes complicated to its maximum extent because the simple solubility reactions, such as those shown above as Equations (4a), (4b), and (4c), are no longer working by themselves. With the availability of free tritium dissolved into the internal volume of the organic material, the molecular surroundings of the organic material see a local perturbation in their own internal systems, and Le Chatelier’s Principle indicates that the system will adjust to the perturbation with the establishment of a new equilibrium. Under such circumstances, exchange reactions can be expected to dominate over simple solubility reactions, and the available tritium can be expected to replace the available protium in any—and all—available sites. Once the tritium has been incorporated into the structure of the organic material, the structure begins to break down from the inside out, primarily as a result of the tritium decay energy.

The specific activity of tritium gas at atmospheric pressure and 25^oC is 2.372 Ci/cm^3 . The expected range of the average energy tritium beta particle in unit density material is only 0.42 µm. This means that all energy from the decay of the dissolved tritium is deposited directly into the surrounding material. At 2.732 Ci/cm^3, this becomes equivalent to 2.88 × 104 rads/hr.

The general rule for elastomers used for sealing is that total radiation levels of 107 rads represent the warning point that elastomers may be losing their ability to maintain a seal. At 10^8 rads, virtually all elastomers used for sealing lose their ability to maintain a seal. Typical failures occur as a result of compression set (i.e., the elastomer becomes brittle and loses its ability to spring back). At 10^6 rads, on the other hand, the total damage is relatively minor, and most elastomers maintain their ability to maintain a seal. At 10^7 rads, the ability of an elastomer to maintain a seal becomes totally dependent on the chemical compounding of the elastomer in question. It only takes about 2 weeks for an elastomer to receive 10^7 rads at a dose rate of 2.88×10^4 rads/hr. Elastomers cannot be used for sealing where they might be exposed to high concentrations of tritium.

Similar analogies can be drawn for all organic materials. The preferred rule of thumb is the use of all organic materials should be discouraged wherever they might be exposed to tritium. Since this is neither possible nor practical, the relative radiation resistance for several elastomers, thermoplastic resins, thermosetting resins, and base oils is shown graphically in Figures 3 through 6, respectively.

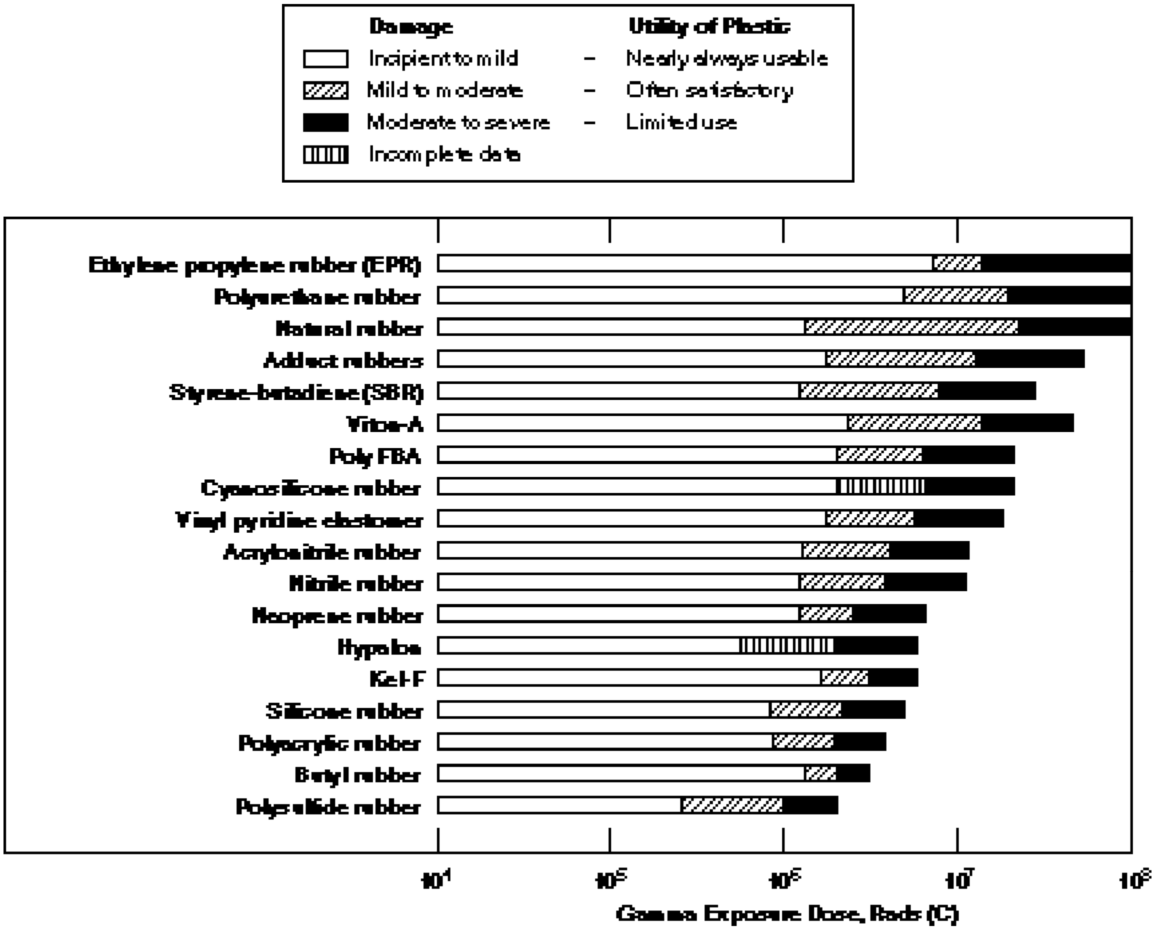

FIGURE 3. Relative radiation resistance of elastomers.

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[1] 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

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[1] The proper chemical name for neoprene is "chlorobutadiene."

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^{o}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.

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^{o}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^{TM} shavings indicated no failures after 3,200 hours, it was concluded that fluorides were being leached out of the Teflon^{TM} and deposited directly into the bodies of the stainless steel test vessels. An interesting sideline to this test is that, after the tests, the Teflon^{TM} shavings showed no obvious signs of radiation damage (i.e., no apparent discoloration or other change from the original condition.

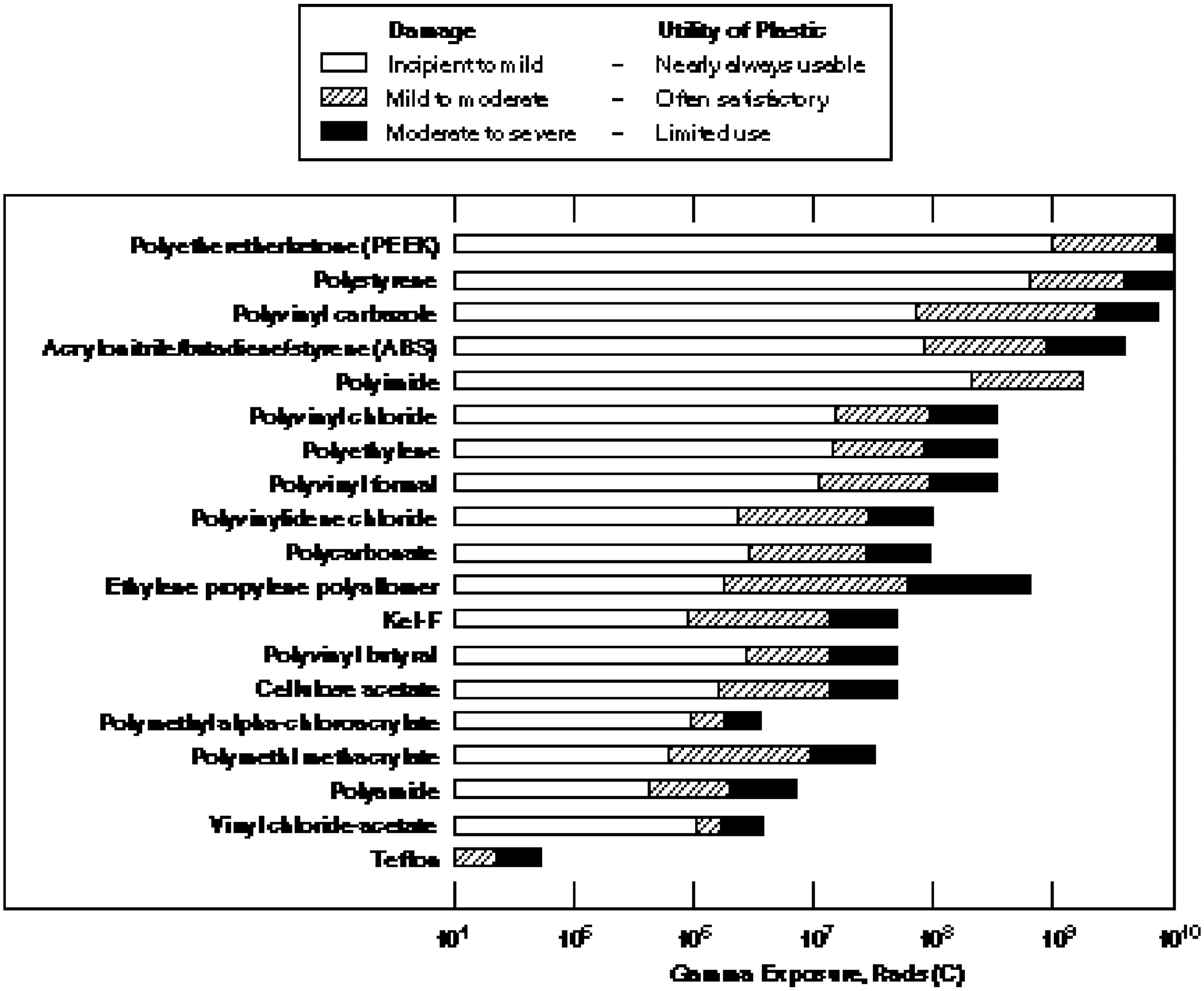

FIGURE 4. Relative radiation resistance of thermoplastic resins.

Outgassing from Bulk Materials. Discussions on the outgassing from bulk materials can be subdivided into two parts: outgassing from surfaces that have been wetted with tritium and outgassing from surfaces that have not been wetted with tritium. For surfaces that have been wetted with tritium, the behavior of the outgassing should be virtually identical to that described above.

For surfaces that have not been wetted with tritium, it should be assumed that the source of the outgassing is from tritium that has been dissolved in the body of the parent material.

As the saturation level in the body of the bulk material is reached, the dissolved tritium begins to emerge from the unexposed side of the material surface, where it then begins to move through the mono-molecular layers of water vapor on that side. In the initial stages, the pattern of the tritium moving into these mono-molecular layers tends to resemble the reverse of that described in the surface contamination model described above (i.e., the tritium is incorporated first into the very tightly bound, near-surface layers, then into the intermediate layers, and finally into the loosely bound, outer layers). As the tritium saturation levels in the body of the bulk material gradually reach steady-state, the tritium levels

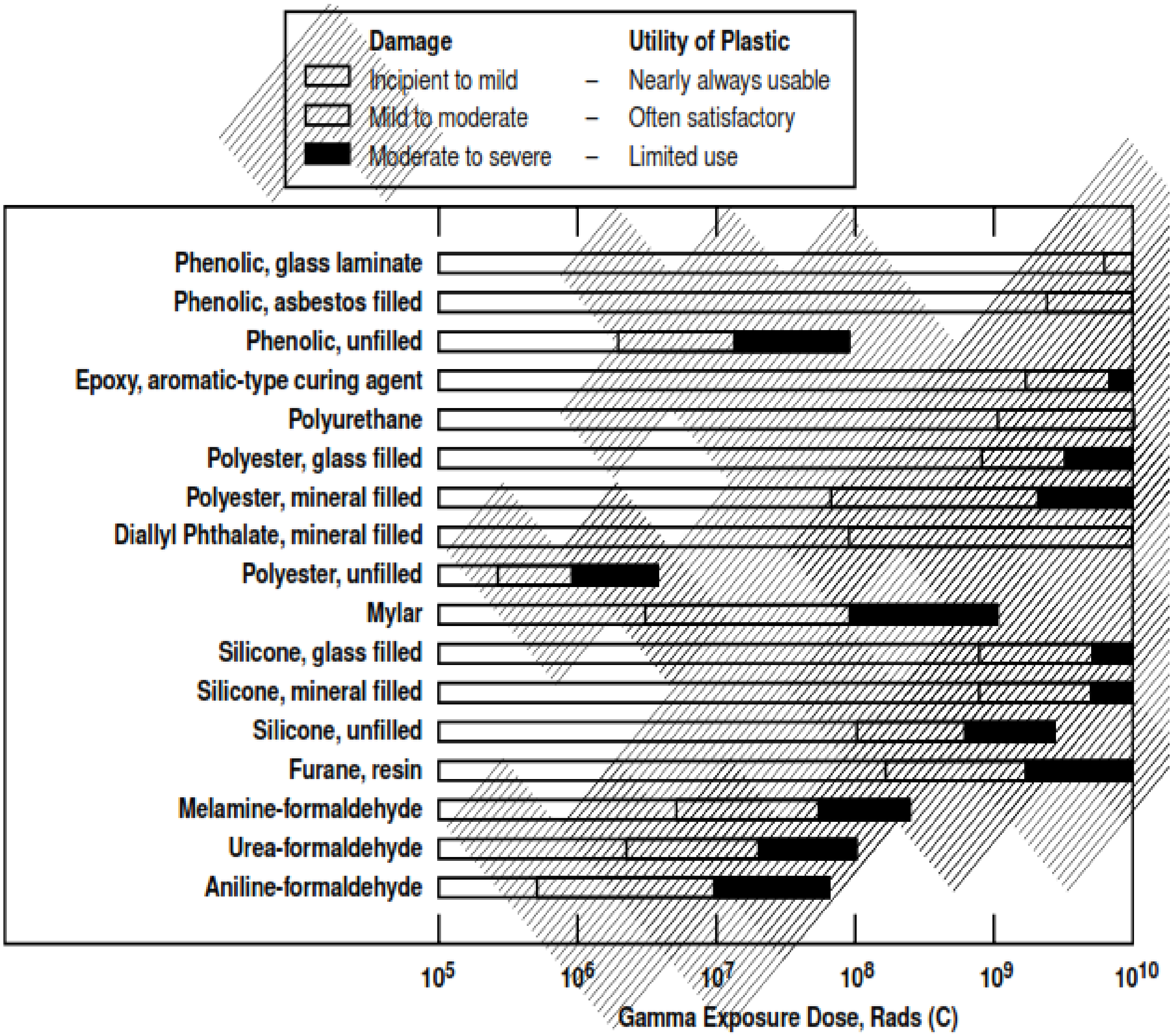

FIGURE 5. Relative radiation resistance of thermosetting resins.

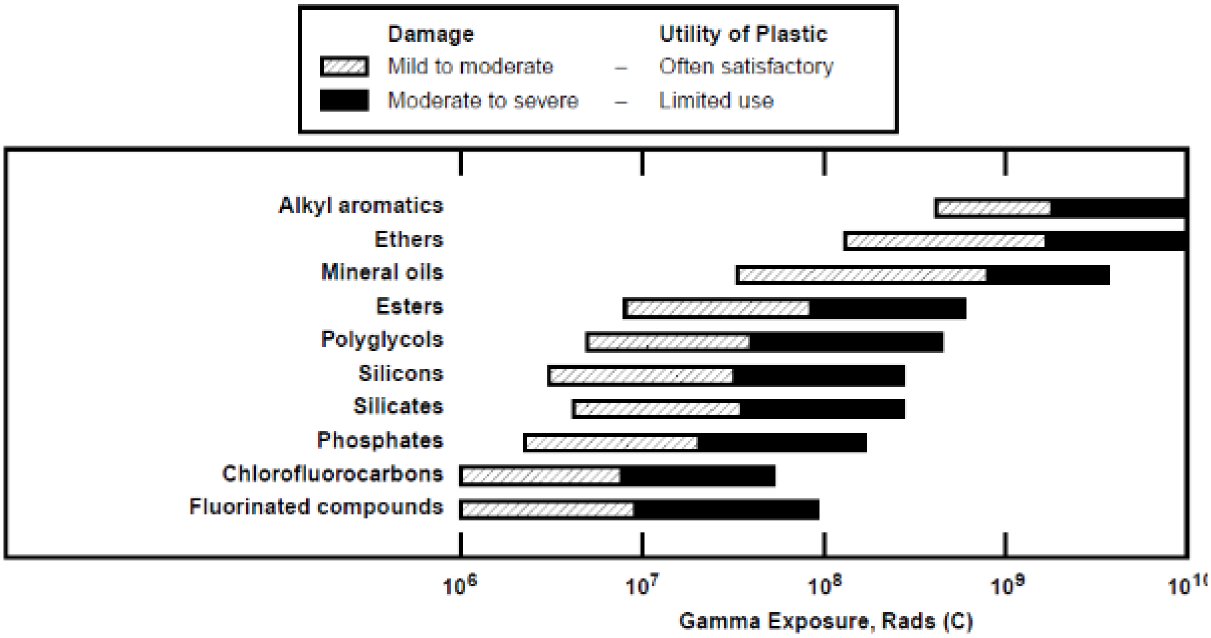

FIGURE 6. Relative radiation resistance of base oils.

moving into the mono-molecular layers gradually build over time, and the pattern slowly changes from one of a reverse surface contamination model to one of a reverse outgassing model (i.e., the level of outgassing from any given surface can be expected to increase until it too reaches a steady-state, equilibrium level with its own local environment).

2.10.7 **<u>The Development of Tritium Technology.</u>** In the preceding sections, we have taken a look at the basic physical and chemical properties of tritium that designers have to contend with but have little or no control over. In reality, these are the things that designers need to accommodate. The sections that follow describe the design and operational philosophies that have evolved over the years.

During the first two decades of the development of tritium technology, operational techniques were designed primarily to protect the worker from exposures to tritium. By the late 1950s, it had firmly been established that exposures to tritium in the form of tritiated water vapor (i.e., T_2O, or, more correctly, HTO) could be as much as 25,000 times more hazardous than

comparable exposures to tritium gas (HT, or, more correctly, T_2). Although real-time monitoring systems had been developed to detect the presence of tritium in the working environment, the monitoring systems of the time could not differentiate between the more hazardous chemical species (HTO) and the less hazardous chemical species (HT). As a consequence, the basic philosophy for worker protection revolved around the use of high-volume-airflow, single pass ventilation systems.

The high-volume-airflow, single-pass ventilation systems of the time were intended to provide several air changes per hour, throughout an entire building, 24 hours a day, 365 days a year. (See Figure 7.) From the perspective of the building, outside air was brought into the building by the supply fans; incoming air was then conditioned for comfort, passed through the working spaces, and pushed out into the environment by the exhaust fans through an elevated stack. From the perspective of the worker, individual airflow requirements were segmented—major tritium handling systems were enclosed in high velocity air hoods; work

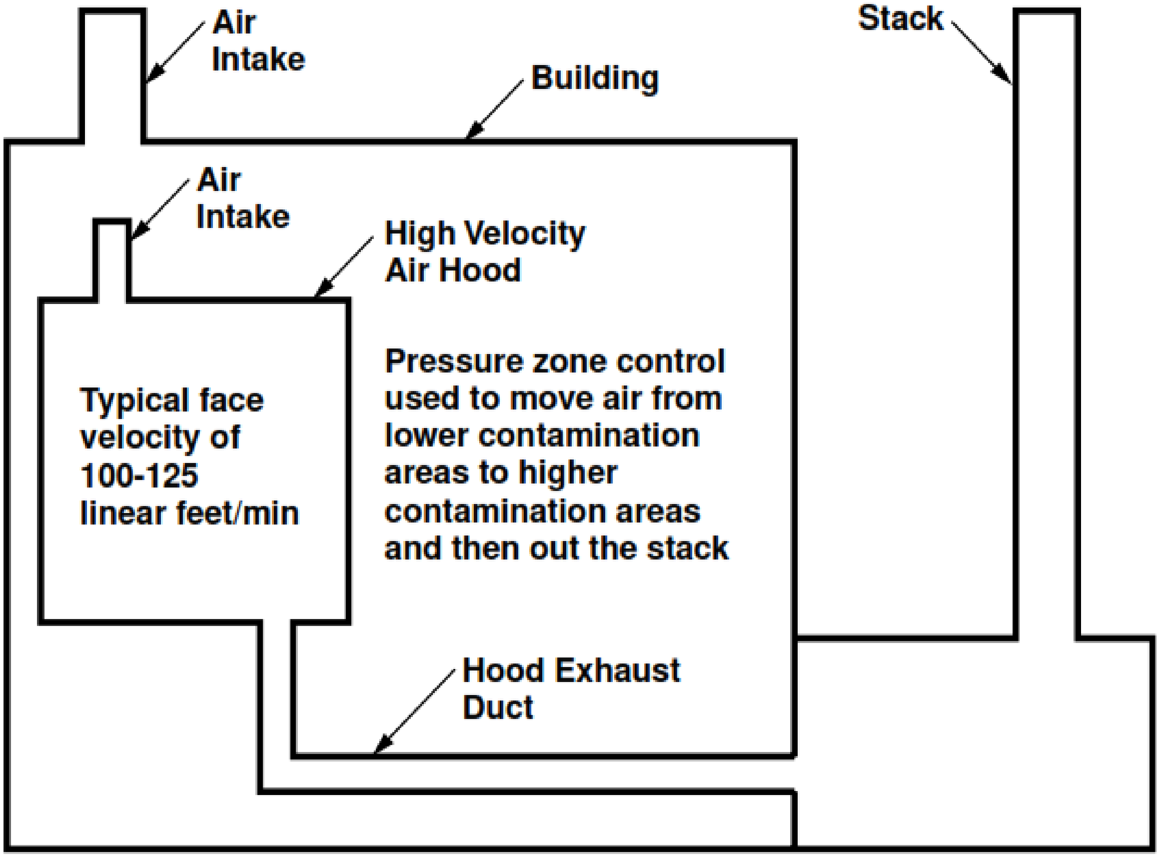

FIGURE 7. Typical tritium facility single-pass ventilation system.

was performed through gloves in the ventilated hood enclosure doors, or through accessible hood openings. The high air velocity hoods were maintained at a pressure that was negative with respect to the surrounding room spaces; most importantly, any tritium releases that occurred due to normal operations, component failure, and/or operator error would occur inside the hoods. The high-velocity air flowing through the building, and then through the hoods, would sweep any released tritium away from the worker.

Although this combination of single-pass ventilation systems and high air velocity hoods was used quite extensively—and quite successfully—for basic worker protection, the protection of personnel living or working downwind was dependent on the massive dilution factors that could be gained by the use of such systems.

By the end of the first decade, it gradually became obvious that the protection of personnel living or working downwind could no longer remain dependent on massive dilution factors. In their initial attempts at controlling tritium releases, a newer generation of tritium design personnel developed a plan to control releases by tightening design controls and material selection requirements. After only a few years, however, it also became obvious that the techniques developed, although helpful, would not be completely successful. While many valuable lessons were learned, tritium releases continued to occur despite more stringent design, material, and performance requirements.

By the late 1960s and early 1970s, it finally became obvious that a more realistic tritium operating philosophy would have to be developed, one that would not only protect the worker, but would also substantially increase the basic protection of the public and the environment.

2.10.8 **Confinement Systems vs. Containment Systems.** Within DOE, the term “confinement” has been used to describe the generic use of barriers, systems, and structures that have been specifically designed to limit the dispersion of radioactive materials within a given facility. Typical examples of confinement barriers and systems include the use of fume hoods, ventilated gloveboxes, air locks, etc.

Because most DOE facilities have, over the years, been associated with the handling of solid or particulate materials, HEPA filters have long been used on ventilation exhaust systems to minimize or eliminate the release of radioactive materials to the environment.

For tritium, however, the concept of “confinement” has little meaning because (1) tritium is a gas, (2) the use of HEPA filters on exhaust systems can do nothing to prevent the release of tritium to the environment, and (3) the classic concept of confinement, from the perspective of the facility, reduces back to the use of the high volume airflow, single pass ventilation systems described above. Thus, standard terminology has evolved that clearly differentiates between the use of “confinement” systems and the use of “containment” systems.

Primary Containment Systems. The innermost barrier that separates tritium from its immediate outside environment, primary containment systems for tritium handling typically consist of piping, valves, containment vessels, pumps, transducers, etc. Because these components can be expected to have their interior surfaces fully wetted with high concentrations of tritium gas under vacuum conditions and/or at high pressures, materials considerations become important, as do the methods of construction.

Because the primary containment system in question may or may not be situated in a secondary containment environment, the rule-of-thumb for modern primary tritium containment systems is to assume that there is no secondary containment system, and that the primary containment system is the only barrier between the tritium and the outside environment. The materials and construction issues associated with primary containment systems for tritium handling should follow the guidance set forth in the ASME *Boiler and Pressure Vessel Code* or a similar code. In most cases, the materials of choice will be a 300-series stainless steel, preferably a low carbon variety (e.g., 304L or 316L). Welding is the preferred method of construction wherever possible. Prior to being placed into service, individual portions of the system should be certified to a maximum leakage rate of 10^{-6} to 10^{-7} cm^3/sec (helium), at a minimum of 125 percent of the maximum expected operating pressure.

Secondary Containment Systems. The use of secondary containment systems in the DOE tritium complex tends to vary a great deal, depending upon when the systems were put into service and their projected use. Depending on their use, secondary containment systems can be expected to vary in complexity, from the relatively simple “jacket concept” used to house primary containment system plumbing and containment vessels, to much more complex glovebox concepts, where the gloveboxes in turn are connected to specialized cleanup systems. Depending on the amount of tritium at risk, construction requirements for secondary containment systems can also be expected to vary widely, from high-quality systems, to intermediate-quality systems, to low-quality systems. An overview of each type of system is described below, along with its intended function.

When gloveboxes are used for secondary confinement, the following design features should be considered:

- Air should not be used for the atmosphere of a recirculating tritium glovebox because of the potential for the formation of explosive mixtures of hydrogen gas. Argon or nitrogen is recommended.
- The glovebox atmosphere should be maintained at a pressure lower than that of the surroundings and diffusion-resistant material should be used to the maximum extent possible to limit tritium leakage.

High-Quality Secondary Containment Systems. A typical example of a high-quality, secondary containment system can be represented by the “jacket concept,” in which the plumbing associated with a primary containment system is completely enclosed inside an independent secondary container. High-quality, secondary containment jackets are most often used to enclose primary containment system plumbing when the primary system plumbing is used to connect systems between one glovebox and another inside a room, between one system and another inside a building, and between one system and another when the systems reside in different buildings. High-quality, secondary jackets are also used quite extensively to enclose primary-containment-system, high-pressure vessels, where the failure of the primary system vessel might be expected to release large quantities of tritium. (See, for example, Figure 8.) High-quality, secondary jackets are also used to enclose individual primary containment system components that are temperature cycled on a routine basis. In this case, however, the purpose of the high-quality secondary containment jacket is to capture any tritium that has permeated through the walls of the primary system component as a direct result of the temperature cycling.

In each of the examples cited above, substantial amounts of tritium can be at risk at any given time. Under such circumstances, the rule-of-thumb for the design of high-quality, secondary containment systems is to assume that there is no primary containment system, and that the secondary containment system is the only barrier between the tritium and the outside environment. By default, therefore, the secondary containment system design requirements become identical to those used for primary containment systems; accordingly, the materials and construction issues associated with these types of secondary systems should also follow the guidance set forth in Section III of the *ASME Boiler and Pressure Vessel Code*.

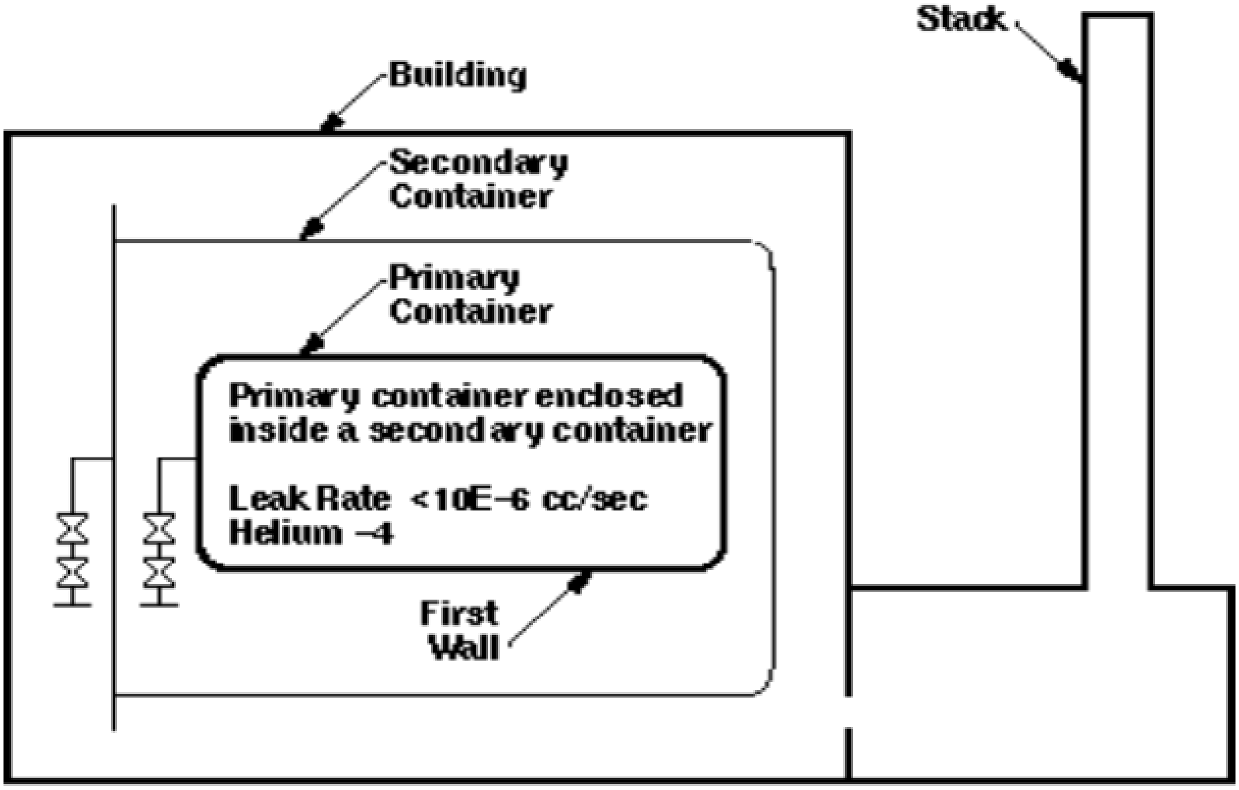

FIGURE 8. High-quality secondary containment.

From a materials and construction perspective, there should be virtually no difference between the requirements for the primary containment and secondary containment designs. In most cases, the materials of choice will be a 300-series stainless steel, preferably a low-carbon variety (e.g., 304L or 316L). Welding will be the preferred method of construction wherever possible. Prior to being placed into service, individual portions of the system will be certified to a maximum leakage rate of 10^{-6} to 10^{-7} cm^3/sec (helium), at a minimum of 125 percent of the maximum expected pressure.

From an operational perspective, however, the design philosophy should be that the void volume between the primary containment vessel outer wall and the secondary containment jacket inner wall in such systems is generally evacuated during service. Thus, if tritium is released into these secondary containers, there are no dilution gases present, and any leakage from the secondary container is in the same form as that contained in the primary container. Following a release into a high-quality system, the tritium can be recovered in almost the original purity without dilution by other gases by pumping it into another primary container. Several days can elapse during the recovery process without a significant release of tritium to the environment.

Intermediate-quality Secondary Containment Systems **.** It is neither practical nor possible to enclose all primary tritium systems inside high-quality, non-diluting, evacuated, secondary containers. Therefore, intermediate-quality, secondary containment systems, such as gloveboxes, are used to help minimize any tritium releases into the facility and/or out to the environment. Because gloveboxes can be constructed to accommodate all kinds of equipment, most primary containment systems are housed inside gloveboxes in modern tritium facilities. (See Figure 9.) When properly designed, gloveboxes allow access to the primary system equipment for ease of operation and maintenance. Gloveboxes are referred to as intermediate-quality systems because the design requirements do not have to meet codes and standards as stringent as the *ASME Boiler and Pressure Vessel Code*. Gloveboxes should, however, be designed in accordance with the recommendations of the American Glovebox Society.

The purpose of intermediate-quality secondary containment systems, such as gloveboxes, is to prevent the immediate release of tritium into the room or out to the environment in the event of breach of the primary containment system. Following a release into an intermediate-quality secondary containment system, the tritium is generally recovered in the tritium removal sections of the cleanup system (see below) and several hours can elapse during the recovery process without a significant release of tritium to the room or the environment.

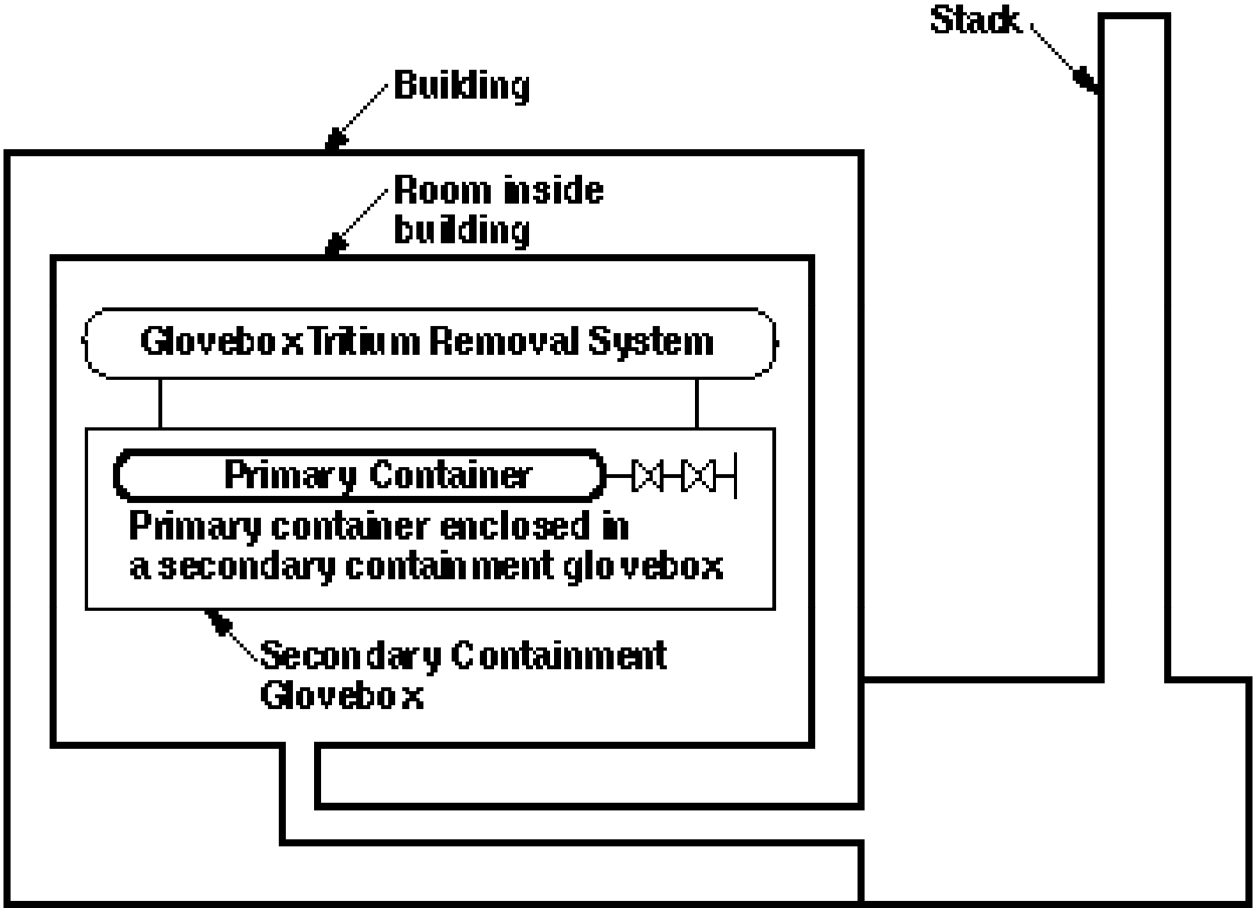

FIGURE 9. Intermediate-quality secondary containment.

In a typical working environment, tritium gloveboxes are operated at a pressure that is slightly negative with respect to the pressure in the room (i.e., on the order of a few tenths of an inch of water column). This allows for greater comfort for the worker while he or she is working through the gloveports. This is supposed to prevent the leakage of tritium from the box into the room. 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.

Because the true leakage rates of most tritium gloveboxes can generally be certified to be no more than 10^{-2} to 10^{-3} cm^3/sec, the ingress of air into the box environment is a problem to be addressed. For this reason, 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. Because one of the undesired impurities will always include tritium (as T_2, HT, and/or HTO), the cleanup systems should remove free tritium from the glovebox gases. In most cases, the cleanup systems will be designed to remove free tritium down to the part-per-million to part-per-billion level. Because this is still equivalent to tritium concentrations that range from 2.6 Ci/m^3 down to 2.6 mCi/m^3, 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 previously under the heading of “Outgassing Expectations.” The chemical form of the tritium release will be as HTO.

Low-Quality Secondary Containment Systems. In some situations, tritium facilities are equipped with low-quality containment systems, such as full-scale buildings, or one or more rooms inside a building. (See Figure 10.) As a general rule, containment systems will only be installed as part of a conversion project, a retrofit project, or an add-on project, where the building or room in question was not initially intended for tritium handling.

Because the containment boundaries for systems are represented by the floors, walls, ceilings, doors, and windows of the rooms or the building in question, tritium leakage rates from containment systems cannot be stated accurately. Like the intermediate-quality secondary containment systems described above, the purpose of low-quality secondary containment systems is to prevent the immediate release of tritium to the environment in the event of a breach of the primary containment system.

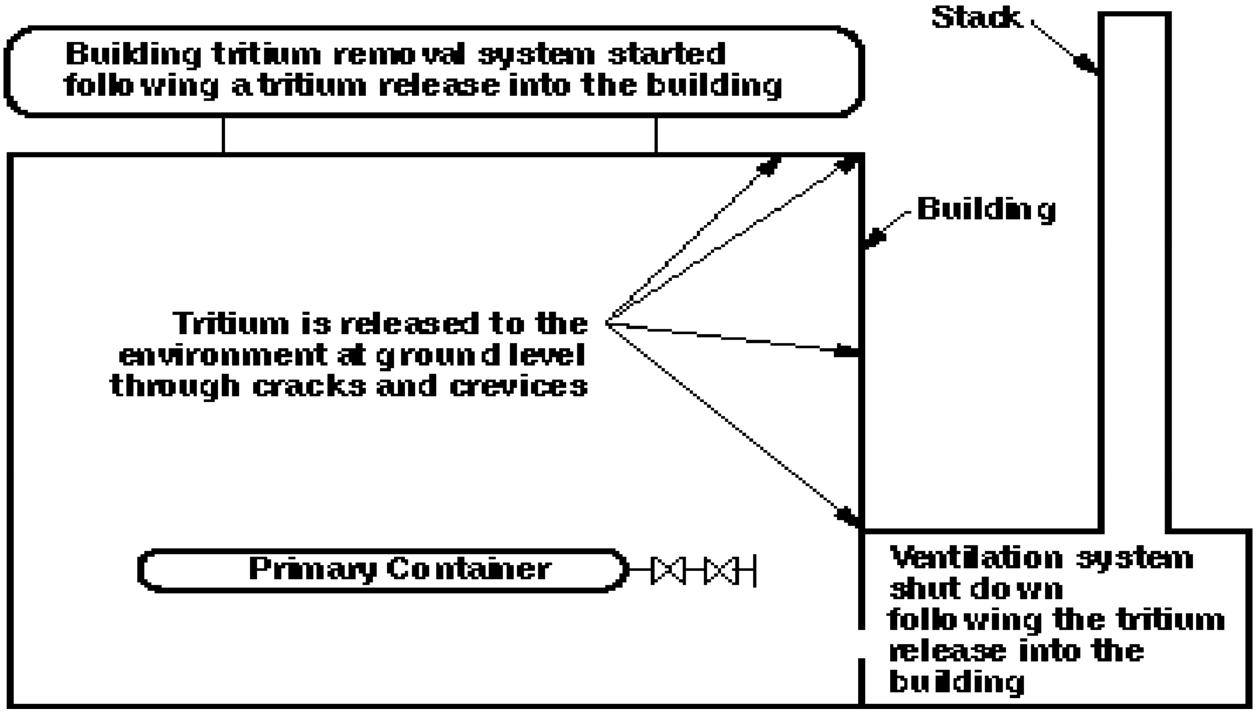

FIGURE 10. Low-quality secondary containment.

Independent of the scale of the operation, the tritium removal systems used for these types of containment systems should be capable of a very high rate of throughput, and they should be examined with great care prior to selecting the containment systems for use. For example, for a small building with a 100 m^3 volume, on the order of 20 X 18 X 10 feet, a 1-g tritium release into the building will result in a volumetric air concentration of about 100 Ci/m^3. If the building has a true leakage rate on the order of 100 cm^3/sec, the tritium leakage rate out of the building will be about 0.6 Ci/min·g.

With the building ventilation system shut down, no benefits will be gained from massive dilution factors, and the tritium will be released to the environment at ground level through the walls, ceilings, doors, windows, etc., of the building. Expanding on the details, the released tritium will convert quite quickly to HTO (i.e., the more hazardous form of tritium); the tritium will thoroughly contaminate the building, adjacent rooms in the building, and the areas immediately surrounding the building. In addition, the tritium released to the environment will be poorly mixed with the environment, and, as a consequence, exposures to personnel living or working downwind could be expected to be relatively high.

In spite of their apparent drawbacks, it should be noted that low-quality containment systems *do* have a place in the broader spectrum of tritium containment strategies, particularly when used as an integral part of conversion projects, retrofit projects, or add-on projects, and particularly when other primary/secondary containment system combinations intended for use have a very low probability of release.

Tertiary Containment Systems. In most tritium facilities, the term "tertiary containment" is something of a misnomer. It usually refers to an ultraconservative approach where a high-quality primary system is housed in an intermediate-quality secondary containment system, such as a glovebox, which is equipped with a cleanup system. Both of these are then housed in a low-quality containment system, such as a room, which is independently connected to a cleanup system. In most cases, the cleanup system associated with both the glovebox and the room are the same cleanup system.

In theory, the purpose of tertiary containment systems is to prevent the immediate release of tritium to the environment in the event of a catastrophic (and simultaneous) failure of both the primary and secondary containment systems. In practice, however, these systems have never been used and, in fact, they probably should not be used in an automatic mode in the event of a catastrophic (and simultaneous) failure of both the primary and secondary containment systems. Under such circumstances, the primary emphasis should be on the health and safety of the worker, and potential rescue efforts should not be hindered by shutting down the ventilation system to those parts of the building where it is needed the most. After appropriate first-aid and rescue efforts have been completed, consideration might then be given to the use of these types of systems in a manual mode. But, as was noted in the previous section, the use of such systems can be expected to have its own undesirable effects, and the actual use of such systems could make a bad situation even worse.

2.10.9 Tritium Removal Systems. In most tritium facilities, secondary containment systems are connected to, or equipped with, tritium removal systems. The primary purpose of these removal systems is to recover any tritium that has been released from its associated primary containment system and to prevent the release of that tritium to the environment. Like the secondary containment systems they are connected to, the operational aspects of individual tritium removal systems vary greatly among facilities, and among different types of containment systems in any single tritium facility.

The simplest type of tritium removal system can be found in those applications where high-quality primary containment systems are enclosed in high-quality secondary containment systems. In most applications, the high-quality secondary containment systems are operated under vacuum conditions when the primary and secondary containment systems are actually in service. The tritium removal systems, in this case, can be as simple as a vacuum transfer system designed to move any tritium that escapes from the primary system to an appropriate collection point for later processing. Under such circumstances, the actual transfer of the tritium can be allowed to proceed quite slowly because, at an allowable leakage rate of 10^{-6} to 10^{-7} cm^3/sec from the secondary, the total leakage rate from the secondary system should be less than 0.25 Ci/day.

More complex approaches should be taken when the primary systems are enclosed in intermediate-quality, secondary containment systems (e.g., a glovebox) because, in these types of applications, the ingress of tritium into the glovebox atmosphere is not the only factor that should be considered. Additional consideration should also be given to the maintenance of the glovebox operational atmosphere, the total volume of the containment system, the total volume of the cleanup system, the ability of the cleanup system to remove tritium from the glovebox atmosphere as a function of flow rate, the overall leakage rates into and out of the glovebox, the overall leakage rates into and out of the cleanup system, and operating temperature requirements for each of the operational components.

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.

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

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

(5) The water is removed from the gas stream by passing the gas stream through molecular sieve traps. A typical example of this type of cleanup system is shown schematically in Figure 11.

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 to10^8:1 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 part-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,

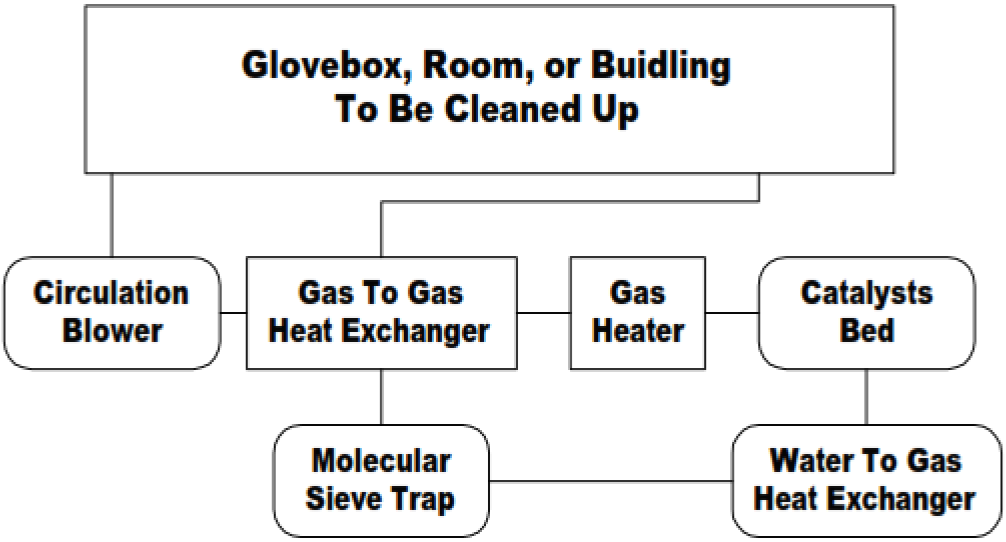

FIGURE 11. Typical tritium removal system flow schematic.

Although the operation of a typical catalyst/molecular sieve tritium removal system, like that described above, can be very effective, the operation of such systems can also have drawbacks. In the standard mode of operation, for example, little or no gas flows through either the glovebox or the cleanup system for relatively long periods of time because both sides of this type of containment scheme tend to be activated on an "on demand" basis only. A substantial drop in the catalyst bed temperature should be expected when the system is turned on (i.e., the flow on the inside of the catalyst bed goes from a static situation to several hundred cubic feet per minute in a matter of a few seconds). Under such circumstances, the temperature in the bed can easily drop below that which is required to break down tritiated methanes, and the cleanup system will have to be run long enough to allow the system temperatures to re-equilibrate. When cleanup systems have to run for relatively long periods of time, leakage rates into and out of the glovebox in question, and leakage rates into and out of the cleanup system tend to become additional factors to consider in the overall design of the facility. When the parameters are appropriately balanced, calculations indicate that the flow rates for these types of cleanup systems (i.e., high-quality primary containment system environments in combination with intermediate-quality secondary system environments) should be high enough to remove the bulk of the tritium within a few hours to a few tens of hours.

Calculations further indicate that the overall release rate to the environment will be 1 Ci for every 3 to 30 hours of operation.

When similar calculations are performed on the use of a high-quality, primary containment system in combination with a low-quality secondary system, the results indicate that the flow rates should be high enough to remove the bulk of the tritium within a few minutes to a few hours.

Double-Valve Design For Transportable Containers. The use of two valves in series on containers filled with tritium, which are to be disconnected from the tritium apparatus on a routine basis, has been in common use for several years. The idea behind this is straightforward. Valve seats fail and 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 a single valve is used and the valve seat develops a leak during storage, the container should be connected to a manifold, the container port uncapped, and the tritium 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, especially if elastomeric seats have been used (not recommended). Double valves should be used where the container valve seat is exposed to tritium for long periods of time. Experienced tritium handlers assume that valve seats can, do, and will fail, and that they have failed until proved otherwise. Figure 12 explains how to check double valving for leakage.

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 that 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 to remove the component and these 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 the new flange seals prior to placing the new component in service.

Figure 13 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 that have been placed to allow 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.

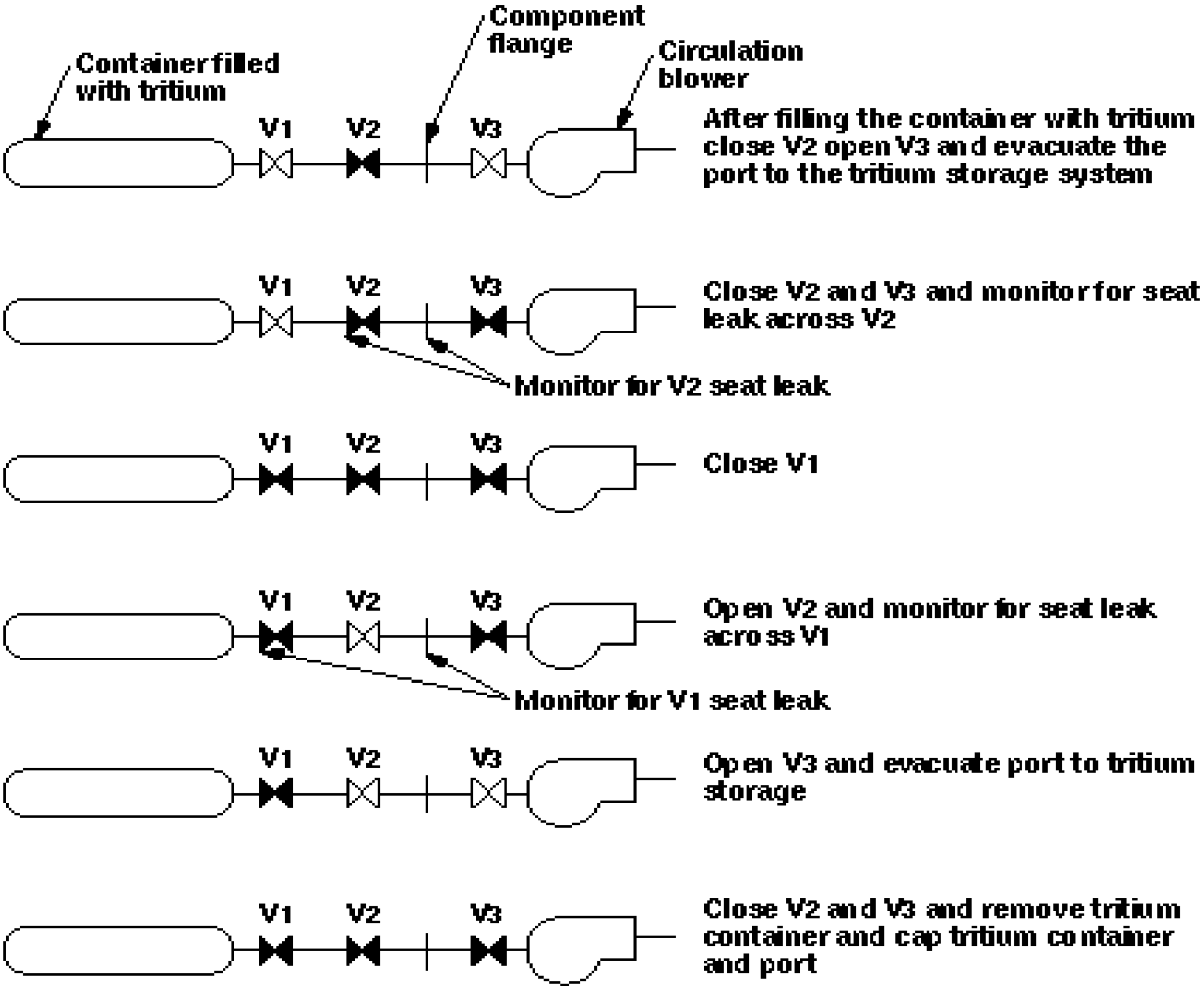

FIGURE 12. Use of double-valved containers.

In operation, the two sets of valves are closed to isolate the component from the rest of the tritium manifold. A vacuum pump is then connected to the two purge ports and the purge port valves are opened and the gases trapped between the isolation valves is evacuated to remove the tritium-contaminated gases. In most applications, air is then allowed to enter the purged volume and the evacuation operation is then repeated.

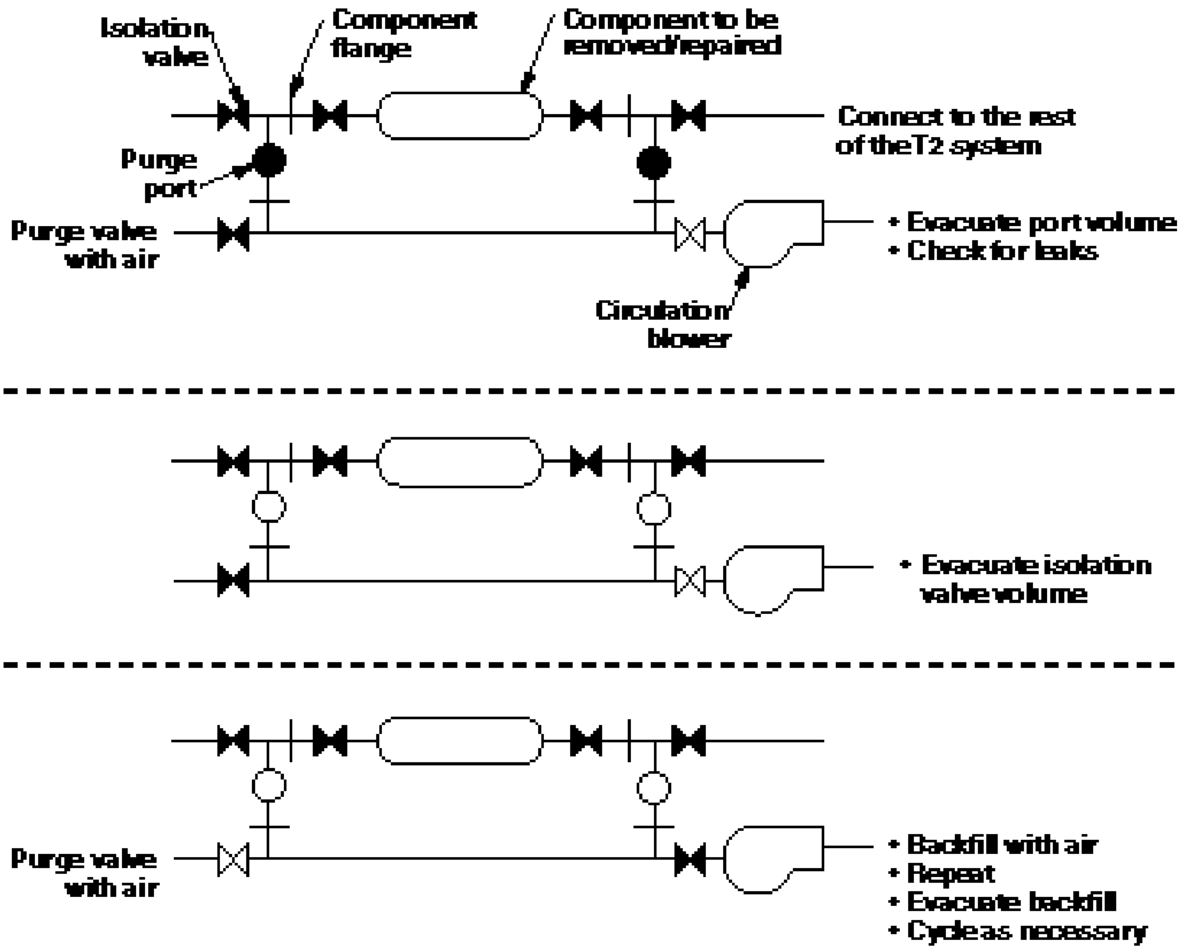

FIGURE 13. Purge ports and isolation valves.

Other gases like argon and nitrogen may be used, but in most cases air is 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 exchanges with the hydrogen and H_20 contained in the ambient air and is pumped out during the next purge cycle.

This evacuate/backfill cycle is repeated from 3 to 6 times to remove as much of the tritium in the gases and from the surface of the volume as possible before disconnecting the component. Three ambient air purge backfill cycles are typical and in practice more than six purge backfill cycles have not proven to be beneficial.

Designers should evaluate all maintenance operations in tritium facilities that offer a potential for significant exposure; such facilities should, where practical, be designed for remote repair or service. In those instances in which remote or enclosed maintenance cannot be achieved, a compressed-air breathing air system should be provided to support the use of supplied-air suits by maintenance personnel. However, every effort should be made to allow routine maintenance activities to be conducted without the need for supplied breathing air. Systems should be designed, to the extent practical, to minimize the conversion of elemental tritium to releasable tritium oxide, which poses a greater radiological hazard than elemental tritium. Although shielding may not be required to maintain occupational radiation exposures ALARA, shielding may be required for other radionuclides that present a direct radiation hazard. Shielding may be required in facilities that handle irradiated tritium production assemblies. Area radiation monitoring should be provided as appropriate.

2.11 FUSION TEST FACILITIES.

2.11.1 Introduction. Fusion facilities include magnetic confinement and inertial confinement fusion devices. Fusion devices range from experimental machines intended to operate below the break-even point to experimental or demonstration facilities intended to operate at or beyond the break-even point.

2.11.2 Design Considerations. The design of fusion test facilities should consider inclusion of the design features described below. Design requirements vary significantly depending on the characteristics of the test facility, the type of fuel, and the site characteristics.

Cooling systems should be provided, as required, for removal of heat from the fusion machine first wall (vacuum vessel), blanket, or other ancillary equipment.

To reduce the amount of tritium released by a single equipment failure, the design should include the following provisions, as practical:

- the capability of completely isolating areas that house equipment containing tritium from areas normally occupied by personnel,

- the capability of isolating sources of tritium, and
- the location of tritium monitors to allow the prompt detection of conditions requiring corrective or protective actions.

When the severity of accidents requires that a containment structure be used, the design pressure and temperature for fusion machine secondary confinement or containment should be determined considering the effects of energy transport and chemical reactions that may occur following the failure of a fluid system inside the containment. The containment should allow periodic leak-rate testing. To minimize the release of hazardous materials to the atmosphere, means should be provided to isolate the primary containment following accidents that release hazardous material to the containment atmosphere.

The design of fusion facilities' secondary confinement or containment where large quantities of tritium are used should consider inclusion of an emergency tritium cleanup system to mitigate the consequences of an accident involving a failure of a tritium system pressure boundary.

A secondary confinement system should be used for tritium auxiliary and other systems containing hazardous materials that are located outside the fusion machine secondary confinement building or containment structure.

The secondary confinement system should be capable of collecting and processing tritium leakage that may occur from tritium auxiliary systems.

Systems for fusion test facilities should, to the extent practical, minimize the conversion of elemental tritium to releasable tritium oxide, which poses a greater radiological hazard. Furthermore, to reduce radiation exposure due to inhalation and adsorption of tritium through the skin, the designer should consider using an independent compressed-air breathing air system to support the air-supplied suits used by personnel while maintaining tritium systems and equipment. Facility design should incorporate shielding where the potential for substantial neutron or gamma radiation may exist. Area tritium monitors should be considered.

2.12 DESIGN OF FACILITIES TO FACILITATE ULTIMATE DECONTAMINATION AND DECOMMISSIONING.

2.12.1 **Introduction.** Facility design should include features that will facilitate decontamination for future decommissioning, increase the potential for other uses, or both. Design of the areas in a facility that may become contaminated with radioactive or other hazardous materials under normal or abnormal operating conditions should incorporate measures to simplify future decontamination. For example, soil sampling could be considered prior to construction so that a baseline is created, against which sampling results during decommissioning can be compared.

Designs consistent with the program requirements of DOE O 435.1 Chg 1 Reaffirmed 2007, *Radioactive Waste Management*, should be developed during the planning and design phases, based upon either a proposed decommissioning method or a proposed method for conversion to other possible uses. Certain design features can readily be implemented to reduce cross-contamination of the facility and enhance the ease with which the facility can be decontaminated. For example, air filters should be strategically placed in the ventilation systems to control the spread of contamination into either the exhaust or supply ducts, depending on conditions. Walls, ceilings, and floors should be finished with easily cleaned materials; in some areas (e.g., hot cells and process equipment rooms), metal liners may be preferred. Construction joints and crevices should be caulked or sealed to prevent the accumulation of contaminated materials at inaccessible locations. The use of hazardous material should be minimized to preclude the generation of mixed waste.

The following sections address design features that should be considered for radioactive and hazardous materials processing and handling facilities.

2.12.2 **Equipment Selection and Location.** Equipment and configurations should be selected that preclude, to the extent practicable, the accumulation of radioactive or other hazardous materials at curves, turns, and joints in the piping, especially in hard-to-reach or inaccessible locations. The design should avoid the use of built-in crud traps, such as flanged couplings, dead legs, etc.

Construction materials and surface finishes should be considered to minimize porosity, crevices, rough machine marks, etc., to limit the possibility of tightly adherent contamination and to facilitate ease of decontamination. An example of this is electro-polished stainless steel liners.

Localized liquid transfer systems should be considered that avoid long runs of contaminated piping to the extent practical, with special attention to the design features necessary to maintain the integrity of joints in buried pipelines. Local processing (solidification) of liquid wastes should be considered when designing piping and liquid disposal systems.

The design should provide for full draining of contaminated piping systems by including the installation of low-point drains, pump drains, tank vent systems, and drain systems, and the elimination of dead legs between valves in system designs. A valuable tool in this design effort would be a 3-D computer-assisted design and drafting system to confirm that all low points can be drained and that dead legs are not designed into the system.

The design should allow access for easier dismantlement, cut-up (segmentation), removal, and packaging of contaminated equipment from the facility. Modular process equipment packages lend themselves well to this approach. The design should minimize pipe spring due to residual stress.

Lifting lugs should be considered as a part of equipment designs to better facilitate equipment removal both during maintenance and decommissioning. The ease with which a remotely operated device could be used in the rigging effort should also be considered.

2.12.3 Building Layout (to Facilitate Decontamination and Decommissioning).

Facility designs should limit the spread of contamination and simplify periodic decontamination and ultimate facility decommissioning and disposal or reuse, especially where radioactive or other hazardous materials will be used or will result from facility operations.

Areas for work with like radioactive or other hazardous materials should be located together to simplify solutions to problems of air supply and exhaust, waste disposal, decontamination, and cross-contamination. However, design criteria should specify minimum potential for cross-contamination between radiological and hazardous materials. In addition, areas where radioactive materials are used should be designed for ease of decontamination during building use and for decommissioning at the end of the building's life cycle.

Use of modular, separable confinements for radioactive and/or hazardous materials should be considered to preclude contamination of structural components or non-process support equipment and to allow for easy removal.

The size and arrangement of corridors should accommodate movement of equipment for initial installation, facility operations, future replacement or removal, and ultimate decontamination and decommissioning (D&D) of the facility, including required equipment accessibility during decontamination.

Access to facilitate decontamination should be provided in areas most likely to become contaminated, such as crawl spaces, piping tunnels, and hatches into duct work.

Air exhaust filters, either roughing or HEPA, should be located as near to individual enclosures or equipment as reasonably possible to minimize long runs of internally contaminated ductwork.

Modularized radiation shielding, in lieu of or in addition to monolithic shielding walls, should be used for ease of maintenance and decommissioning.

2.12.4 **Coatings to Facilitate D&D.** To facilitate D&D, the following should be considered:

- American Society for Testing and Materials (ASTM) D4258-05 (2012), *Standard Practice for Surface Cleaning Concrete for Coating,* provides guidance for facilities that require coatings to enhance decontamination of surfaces or because of environmental conditions.
- Selection of floor and walkway coverings should consider ease of decontamination. To the extent practical, floor-to-wall interface joints should be covered for ease of decontamination.
- Bare floors, walls, and ceilings should be protected, particularly for structurally important parts of the building. Protection should be in the form of strippable or durable coatings for which effective cleaning methods have been developed.

- Surfaces in operating or process areas should have no rough or absorbent surfaces, seams, or cracks.
- For water-pool type facilities, the pool liner should be provided with a leakage detection and collection system to limit absorption of contaminated pool water by concrete structures.

2.13 D&D AND ENVIRONMENTAL REMEDIATION PROJECTS

2.13.1 Introduction. In many instances, an existing DOE facility that needs decontamination and decommissioning does not have a D&D plan and procedures in place that were implemented in the original facility design. In addition, DOE has a wide variety of facilities that will undergo D&D. These facilities are in various present-day conditions of safe layaway or operation.

Six essential functional areas should be considered during D&D plan development in order to develop a total system to solve the complex problems related to the cleanup activities associated with D&D:

- characterization,
- decontamination,
- dismantlement,
- material disposition,
- robotics, and
- regulatory compliance.

Characterization consists of four significant operational phases:

- facility characterization prior to cleanup,
- characterization during cleanup,
- characterization of waste materials, and
- site characterization after cleanup.

Each of these phases has its own unique problems and requirements and should be evaluated by the D&D plan designer to produce an efficient, economical system.

During the decontamination phase, the designer should consider various requirements and objectives in formulating the D&D plan for DOE facilities:

- permit unrestricted reuse of facility materials by recycling,
- permit reuse of the item or component in the DOE facility,
- reduce worker radiation exposure,
- avoid potential criticality accidents,
- enhance disposal, and
- separate hazardous from radioactive items and materials during decontamination.

In addition, the designer should consider several desirable objectives, which the decontamination technology and methodology should meet:

- reduction of cost,
- reduction of generated waste volumes,
- increase in productivity,
- achievement of high decontamination factors,
- capability of remote operator (if required), and
- capability for mobile operation.

During the dismantlement phase, various methodologies and site-specific requirements should be considered. The dismantlement is influenced by the type of contamination present, the level of contamination (whether remote operation is required), the facility size, and by the design features (glovebox, canyon, reactor, etc.) included in the facility.

If the designer determines that advanced, cutting-edge technology is required, operational controls should minimize spread of contaminants, generation of secondary wastes, and generation of mixed waste, and it should have ALARA features regarding worker exposures.

The area of material disposition will also require significant input by the D&D plan designer. Material disposition includes activities to recycle valuable materials for reuse and disposal of material that cannot be reused cost effectively and in a manner that protects human health and the environment. Various treatment methodologies are needed to support material disposition and should be selected by the designer to meet material disposition goals.

These methodologies include the following:

- chemical treatments,

- stabilization,
- packaging,
- refining,
- netting, and
- machining.

In addition, the D&D plan designer should investigate regulatory compliance issues. Some of these issues have various programmatic impacts, such as time constraints and procedural issues.

Environmental remediation (ER) projects are those that generally involve cleanup of soils or waste sites that are not within structures. As such, the work consists mostly of bulk moving and treating of materials. An example of this type of work is the Uranium Mill Tailings Remedial Action (UMTRA) Program. D&D projects are usually thought of in the context of work within structures. In either case, these projects often involve working with hazards that may not be completely characterized. Therefore, planning the project to minimize the work hazards is an essential part of project design. Further guidance is provided in DOE G 430.1-4, *Decommissioning Implementation Guide*.

2.13.2 **Decommissioning and Decontamination.** For cleanup activities, design criteria focus on mitigating hazard consequences that cannot be reduced further or eliminated before starting the cleanup. The basic safety objectives of protecting workers, onsite employees, and the public apply and form the basis for designing control features to mitigate the potential hazard consequences.

Hazard mitigation actions, such as applying a fixative to loose surface contaminants, will alter the form and decrease the dispersible fraction of hazardous dust. Sequencing the operation to specific rooms or building areas (segmenting) will limit the operational inventory. Limiting the number of simultaneous segmented cleanup operations to decrease complexity and avoid compounding hazards will also decrease the potential hazard consequences. Removing significant sources of inventory early in the operation will limit the time during which the hazard needs to be considered and will limit the releases from subsequent common cause or propagation events.

Design decisions for cleanup activities should consider the duration of the specific activity, the complexity of the activity, the hazard consequences, and the hardware salvaged for use at the next activity. Costly designs and equipment may need to be provided, even for short duration activities, for safe operation. To make these items cost effective, designs should include provisions to make the hardware portable (or at least easily salvageable). Designs should also consider modular items that can be installed with different modules to fit a particular application. The following sections address general design considerations.

D&D Hazards Reduction. Because the initial stages of decommissioning are generally hands-on operations in many facilities, personnel exposure issues should form a large part of the project planning efforts. The best hazard reduction is to remove the hazards as early as possible in the project for any type of hazard, thereby reducing the opportunity for exposure. The sections below provide specific examples of project planning, sequencing, lessons learned, or technology options that should be considered for hazards reduction in a D&D project.

Chemical Exposure. Bulk chemical removal as soon as possible in the project is usually an excellent choice from both a safety and regulatory viewpoint. Residual chemicals in piping systems and dead legs should always be drained and contained with industrial exposure; maximum expected volume issues should drive the draining and containment methods. Even “empty” systems can contain significant volumes of liquids within level, sagging, or residue dams that could be invisible from any external inspection of the systems. Application of ALARA principles to chemical hazards should be applied with the same rigor as to radiological exposure.

Operating from a position of knowledge is always preferred, but every project should be planned with some realization that exact chemical mixtures, volumes, and locations of liquids will not be known when the piping system is actually breached. The watchword for any chemical system clean-out or removal is to expect surprises.

These surprises can manifest themselves in–

- liquids found in a “drained” system,

- explosive gases found in drained systems,
- various concentrations or mixtures of the expected chemical actually found,
- crystals of evaporated chemicals rather than the liquids or vice versa, and
- salt cake or sludge-oozing liquids even after significant drying or solidification steps are undertaken.

The level of planning and control of the operation should take hazard level and exposure scenarios into account prior to the start of work. Any planning should always provide built-in contingency for the surprise that will occur during the decommissioning project.

Radioactive Material Exposure. Control of radiological exposure is well-established within any nuclear operations organization and those same principles of ALARA should be applied to the decommissioning work. The difference is that more unknowns should be expected, more scheduling options may be available to remove a source term, and the opportunities to spread contamination is greater during decontamination and piping system removal than during facility operations. In general, control techniques such as those listed below should be considered for decommissioning work:

- Removing a high-dose contributor earlier rather than shielding is preferred.
- Inhalation issues should be controlled as close to the source as possible.
- Engineering controls take precedence over protective clothing, and both methods are better than administrative controls.
- Exposure controls are needed until the waste leaves the site.
- Use of the same control mechanism for chemical and radiological exposure should be investigated and used whenever possible.

Industrial Hazards Exposure. The changing nature of any decommissioning project site demands not only good design and planning to eliminate as many hazards as possible, but constant awareness of the changing conditions and hazards present. Proper planning should sequence the work so that hands-on work is completed in an area, or adjacent areas, before heavy equipment demolition occurs. Other considerations include—

- extremely loud operations within an area that affect workers who are otherwise not affected;
- work that may affect airflow or air quality of other facility workers;
- overly prescribed protective clothing or administrative limits (a respirator to mitigate a potential, but highly unlikely, minor exposure may introduce a very significant risk by restricting vision or aggravating heat stress issues);
- life safety escape routes, which should be reevaluated as the facility changes (what was an emergency escape path last week may be a 30-foot hole or a collapsed structure today);
- fall protection.

Many of the old DOE structures that are being decommissioned have little or no configuration management of the original electrical systems during the surveillance and maintenance period. Electrical distribution systems could have been cannibalized in the years since operations, further aggravating the situation. This being true, one of the larger industrial hazards in a decommissioning project is encountering live electrical equipment when removing equipment, cutting conduit, or excavating around a facility even after the application of lock and tags to known sources. Removing ALL electrical power prior to the start of decommissioning within a structure should be a very high concern. Removal in this case is defined as physical breaks in the circuit by wire, transformer, or breaker removal, not by open breakers since old, poorly maintained breakers could stick or malfunction. Even the task of removing ALL electrical power can be daunting. Over the years as missions change, extra or different voltage power may have been added and not documented. The review and field investigation that removes ALL power needs to ensure the safe removal of electrical equipment and wiring systems.

2.13.3 Hazards Mitigation.

Criticality Prevention. One technique that can minimize the impact of the large inventory uncertainty would be to integrate characterization data and work plans to keep the “affected” inventory less than some predetermined value by scheduling sections of the building, systems, or components to be worked at the varying times and ensuring that waste packaging and segregation and even waste storage schemes include criticality reviews. Portable neutron or alpha instrumentation and in-process active/passive neutron interrogation technologies should be added as needed to give real-time updated information of the inventory as the project or operation proceeds. A significant technical issue that should be addressed for every waste package or configuration is the way the package density or detector/target geometry affects any neutron measurement.

Electrical Power. The use of a construction-type power system to support the decommissioning work should always be considered for anything but the simplest project. The elimination of ALL power from a structure, except that which is specifically brought in on a temporary basis, greatly reduces the chances of inadvertent encounters with live electrical equipment. These temporary systems should be sized to meet the need, colored to stand out, and built to withstand the harsh environment (rain, snow, wind, and falling objects) encountered on a demolition site.

Power or energy for cleanup operations should fit the particular activity. Cleanup work is generally the primary energy source generating an airborne condition. Where loss of power could result in unacceptable releases, consideration should be given to backup power or design of the confinement system to include a passive confinement mode sufficient to reduce releases to acceptable levels when combined with work stoppage. Warning or alarm systems should be included to ensure work stoppage upon loss of active ventilation.

Confinement. The selection of the confinement system to be used during D&D operations will be dependent on the condition of the existing building and ventilation system. It is possible that the existing structure and ventilation system is adequate without change or upgrade. Where existing systems are judged to be inadequate, the designer should consider enhancing the existing system with portable ventilation and tents within the building. The tents and ventilation would provide the active operation zoning function while the existing structure and ventilation would continue to provide the shutdown condition confinement for the remainder of the building. Care in the accident analysis and design is needed to ensure that credible accidents will not propagate beyond the work zone.

Radiation Shielding. Shielding design in cleanup activities relies on specific source reduction rather than shielding walls used in process operations. Design related to portable shielding is limited to the material (high z or low z), thickness, and features to make the shielding material easy to install, relocate, decontaminate, and remove so that it can be reused (waste minimization). The available space and shape of the object generally determines whether to shield the object or shield the worker (portable shield wall). Shielding design for buried items may have to be modular and layered because it is not always possible to determine radiation levels of buried items until they are actually uncovered.

Structural Integrity. Structures provide confinement, shielding, a stable anchor for fixed equipment, segregation or isolation, and a controlled environment (dust, humidity, and temperature control) for equipment important to safe operation. Portable equipment is generally self-standing and does not require an anchor for short-duration use. This leaves the functions of segregation and climate control. Both of these functions can be provided by a tent or, under more severe cases, sheet metal structures. The choice of structural material depends on factors such as duration of the activity, consequences of structural failure, geographical/meteorological conditions (snow or wind loading), and whether or not an existing structure is involved (D&D or soil remediation). Consideration should be given to limiting the mass of material for those structures that will be disposed of at the completion of the activity or to facilitate the ease of decontamination for those structural components that will be recovered for reuse at another activity.

2.13.4 Summary. A general D&D plan should be developed that outlines the design and operation of the activity. The plan should discuss how this combination (design and operation) will protect the worker, the public, and the environment from undue risk. General guidance is provided that suggests design features by modular, portable, and reusable characteristics to the extent safety and operational requirements will allow.

Design features provide an added resource that allows both safety and operational needs to be met. Design features mitigate the potential consequences of hazards that cannot be eliminated or reduced by other means, thus allowing operations to proceed safely. Cleanup activities should be conducted at the site of contamination using devices with design features that are flexible enough to be used at multiple sites.

The following outline demonstrates the major areas to be considered by the designer in the development of a D&D plan for a DOE facility:

- site inspection;
- site characterization;
- environmental requirements and constraints;
- predemolition planning documents;
- recommended decontamination actions/procedures;
- demolition plan:
    - demolition process,
    - demolition and disposal work breakdown structure (WBS),
    - demolition methods and equipment,
    - demolition procedure;
- disposal plan:
    - waste categories,
    - disposal methodologies,
    - post-demolition facility and land use;
- decommissioning.

2.13.5 **Environmental Remediation.** Specific performance objectives for ER activities should address timing, cost, exposure control, confinement, worker protection, and ALARA issues. Safety and operations then review the specific work to identify methods that eliminate or reduce the existing hazards by changing the hazards form, selecting a work sequence, or altering/eliminating energy sources while ensuring the performance objectives are met. Each hazard reduction method or combination of methods should be balanced against the performance objectives so that the end result is a safe, efficient, and effective operation. When existing hazards have been eliminated or reduced to the extent possible, the safety and operations engineer(s) work with the designer to develop design barriers to mitigate the impact of the remaining hazard to acceptable levels while achieving the performance objectives.

One of the major hazards of soil remediation activities is fugitive dust generation (airborne contaminants) created by the cleanup activity, wind action, or a combination of both. Large area sites can involve thousands of curies of radioactivity (total quantity) when summed over the total soil volume to be remediated (even though the concentration is microcuries or even nanocuries per gram). The planning of contaminated soil site remediation activities reduces the hazard by establishing a limit of inventory available for release. Dividing the contaminated site into segments reduces the area, thereby reducing the quantity of "contaminated dust" released per unit time. Wetting the contaminated soil during cleanup activities effectively alters the contaminant particle size, thus reducing fugitive dust generation. Conducting cleanup activities during periods of low wind also reduces dust generation and transport energy. The appropriate combination of methods to reduce the hazards should be considered before considering design barriers that would mitigate hazard consequences.

Another type of ER activity is the exhumation of previously buried or otherwise disposed-of waste material. Limiting the amount of buried material removed at any one time limits the hazards associated with the activity. This limits the contaminant inventory available for accidental release.

Confinement of hazardous particles can be achieved with systems varying from general ground cover (soil cleanup) to a complex structure with multistage ventilation (stationary cleanup "treatment" center). The confinement system needed for a specific cleanup activity should consider the work to be performed, the hazard, the complexity of the activity (multiple activities simultaneously conducted at a single location), energies available, and, if the energies are present continuously or only during actual work, the cleanup rate (hazard decrease rate), and several other factors.

General soil cleanup releases are usually chronic rather than acute. For this type of activity, wind and cleanup equipment movement provide the predominant energy sources leading to the airborne dispersion of contaminants. Barriers designed to inhibit the generation of dust (wetting or fixatives) during operations and to separate the energy from the contaminants (ground cover) during non-work periods are among the simplest forms of confinement.

Retrieval or recovery of buried waste has the potential for acute releases. Accumulation of retrieved items and in some cases individual items can contain sufficient materials to represent an accident hazard. Confinement for this type of activity could involve a vented overpack, portable hot cell, or a tent of plastic or metal construction with an associated ventilation system. Vented overpacks and hot cells provide confinement for the unusual or anticipated small percentage of packages that are uncharacteristic of the normal retrieved items. Tent confinement provides control where packages may have lost their integrity and pockets of contamination are possible or expected. Tents may also be used as weather covers for normal operation and to provide confinement only for accident conditions. The specific type of confinement used depends on the method of recovery, the type of hazard, and the confidence in the knowledge of inventory and package condition.

In cases where ER projects involve temporary structures for process confinement, especially when potentially high-activity buried waste is involved, some design considerations are common to both environmental restoration and D&D projects. For example, considerations related to electrical power, radiation shielding, structural integrity, and work process controls, are discussed in the preceding section on D&D projects.

2.14 VITRIFICATION.

2.14.1 Introduction. The process of vitrification is the melting and fusing of materials to produce a glass. It is one of the preferred treatment options for many types of hazardous and radiological waste. The general design considerations in developing a vitrification plant are discussed below.

2.14.2 General Vitrification Processes and Steps to Consider. Waste composition may make material more or less amenable to the use of vitrification as a treatment technique. It affects pretreatment required and influences the transport of material within a vitrification plant. Characteristics to consider include the following:

Physical Characteristics. Physical characteristics of the waste material, including whether the material is in solution, a slurry, or a solid, influences transport within the plant and the material's ability to form a homogenous mixture with glass formers. Imposed constraints on transport, mixing, melting the material, and the need for pre-treatment are all considerations subject to physical characteristics of the waste.

Chemical Characteristics. Certain chemicals (notably sulfur) are incompatible with vitrification and if present influences the amount of final waste produced. Other chemicals may be volatile and very difficult to retain in the glass product. Factors like pH may complicate the process of mixing a final feed material, due to rheological considerations or corrosion.

Radiological Characteristics. The radiological characteristics dictate shielding requirements, remote operations, off-gas processing, and possibly, a requirement for criticality controls.

2.14.3 Waste Extraction from Tanks. Waste extraction from tanks is the first step in the actual vitrification process. (Some in situ vitrification techniques eliminate this and many other steps discussed below, but this technology is not generally used and may often be judged unsuitable for a specific problem.) The removal of waste from tanks is the first of the process steps in moving the material toward vitrification. Some of the considerations include the following:

Pretreatment in the tank may allow adjustment of an undesirable physical or chemical characteristic to make the waste material more compatible with and amenable to future processing. Pretreatment can include pH adjustment, size reduction, mixing, and the decant and processing of dilute waste solutions on the surface of more concentrated waste deposits.

Based on Composition, experience at many radioactive waste storage facilities suggests the composition of tank wastes may make extraction techniques and extraction technology key concerns. Precipitates, agglomerates, and foreign material such as gravel have been key considerations in influencing the design of removal systems; subsequently, these considerations influence the design of facilities to process the waste. The facility design should consider the potential needs of further processing extracted material and the potential for the waste to change significantly over the life of the facility as different methods are employed for the extraction of bulk waste versus residual waste.

Based on Tank Configuration, access may be difficult and may influence removal techniques, requiring sluicers, mixing systems, etc.

Pumping techniques should consider the slurry nature of the material, both from a pickup perspective (getting the material into the pump) and the transport through piping to a processing facility.

Mixing/Homogeneity has influence in the process and process plant equipment design. In cases where feed material is very consistent, less concentration on sampling techniques and feed composition adjustment is required. If the material is nearly all in solution, less attention to slurry handling is necessary.

2.14.4 Feed Delivery. In addition to the technical considerations of pumping the material, there are operational considerations in regard to coordination of the transfer of feed material. The melter feed process necessitates a “batch feed” type system, which allows the melter to operate in a continuous feed manner with parallel processing of a new feed batch. Multiple tanks are required for a batch feed system with a minimum requirement for two tanks. One tank is required to contain the waste material for at least one feed batch. The second tank is required to mix the glass formers, other chemicals required for the waste form, and the waste material being processed. Usually the feed tank will be significantly larger than the tank in the vitrification plant receiving the waste. Adequate operational communication and interlocks to prevent tank overfill are important considerations.

2.14.5 Feed Sampling. One of the first steps in the actual vitrification process is to maintain melter feed material within an appropriate process range. This may be for glass quality reasons but is also very important to prevent unusual reactions in the melter.

The sampling process should adequately address the number and volume of samples to be collected. Often several samples may be needed to support a given laboratory analysis and the statistical evaluation of the results.

The sampling process often has many manual steps and is, therefore, error prone, particularly when slurry feeds are involved. Design should minimize the potential for error, providing reliable indications to operators and minimizing the number of manual steps. To ensure that the sample system does not introduce additional process errors, consideration should also be given to ensuring the pumping system for taking samples has the same operational characteristics as the pumping system feeding the melter.

Slurry Handling is of concern throughout the feed preparation of most facilities. Slurries present particular issues in line sizing, flow rates, valve operations, flushing allowances, and designs that accommodate plug clearing operations. These should be considered when developing slurry sampling provisions.

Consistency of Waste Feed should be considered in developing sampling plans and systems. The feed make up process will generally require accurate measurement of all species significant to melter and possibly waste form behavior. The feed make-up, as discussed above, is generally a batch process; the vessel in which the feed is being prepared should be well mixed to avoid composition errors that could damage the melter.

2.14.6 Feed Make-up and Chemical Addition. Feed make-up and chemical addition may require concentration and then addition of chemicals needed to meet target melter feed compositions. If the feed material is very well characterized and homogeneous, the chemicals can be added in advance and provided in the form of glass frit. If there is potential for variation, a chemical batching system that can handle specific recipes is required. The system may need to handle solids, liquids, and slurries, depending on the target feed composition. Generally, the chemical additions should be made up in a separate tank and batch-transferred into the feed tank. This recommendation derives from unique requirements for contamination control and batch quality control for high-level waste vitrification. There may be more innovative approaches for other waste types.

Accurate measurements of tank volumes and fluid density have proven very important in the feed make-up process. Special attention should be given to these areas for batch make-up tanks.

2.14.7 Feed Holding. Melters generally perform longer and best when operated as close to steady-state as possible. Enough feed should be prepared to support that objective while the next batch of feed is being prepared. Vessel sizing should consider the potential throughput, as well as the time necessary to prepare batches. In estimating those times, analytical turnaround times for process samples should not be discounted. They can and have been a large part of the time required to make feed.

Mixing the feed holding tank is a necessary design feature to maintain consistent feed material to the melter.

Re-suspension should be considered when developing the mixing system.

Vessel Erosion allowance should be factored into the hold vessel design, based on the feed material and the mixing system potential to abrade the tank wall.

2.14.8 **Feed to Melter.** Like the sampling system, the melter feed system moves homogeneous feed from the hold tank to the melter. Also as noted earlier, selecting a pumping system similar to the one from the sampling system will provide some degree of consistency between the samples and the actual feed to the melter. The feed system should consider the need for flushing and plug-clearing capabilities, particularly at the melter discharge point. Experience has shown the potential for high temperatures to dry out and accumulate feed material at this point.

2.14.9 **Acceptable Glass Compositions.** Acceptable glass compositions with respect to melter behavior are those compositions exhibiting properties amenable to proper melter operation and longevity. These properties include glass viscosity at operating temperature, liquidous temperature, glass electrical resistivity at operating temperature, and glass transition temperature. The actual values for these parameters depend on the melter technology employed. In addition, the following compositions should be avoided: those that exhibit behaviors such as foaming in either the cold cap during drying or in the glass melt pool and those with unfavorable Redox behavior (i.e., precipitation of metals during melting) or explosive gas generation (i.e., hydrogen evolution).

Acceptable glass properties and behaviors should be generally identified early in the project and promulgated through the process development phase.

2.14.10 **Melters and Melter Behavior.** Melters vary in size, configuration, and heating source. The large-scale, high-level waste melters in the DOE system are joule heated, by passing current through the actual glass melt. RF and gas-fired melters have also been proposed. The melter's configuration usually requires two chambers.

The first is a refractory chamber surrounding a glass melt "pool," where some mixing occurs either by natural convection or as enhanced with air bubblers; feed material is supplied from the top. This feed material usually forms a "cold cap" on top of the molten glass, melting into the glass pool as it is heated by convective currents rising from below. The melter feed will off-gas as the cold cap dries out and heats and as volatile elements are driven out of the glass itself. Off-gas will vary depending on the makeup of the feed material; it should be accommodated and treated in some type of off-gas system. The second chamber is a heated discharge cavity through which the molten glass can flow into a canister.

2.14.11 **Melter Life/Keeping Melters Hot/Not Cycling.** Melter life is dependent largely on how the melter is operated during its productive life. Operating temperatures of vitrification melters range between 1050 and 1250^{o}C. These temperatures present special design considerations for any metals used in melter fabrication.

The melter is usually heated through infrared heaters in both melter chambers. The heaters in the glass "pool" chamber are for startup only and are shut down once the glass becomes molten and joule heating can be activated. Melters should be brought up to operating temperature in a controlled manner at an average heat-up rate of approximately 10^{o}C per hour.

Exact heat-up rates are a function of the materials used and will vary accordingly. Refractories can be heated at much faster rates than the metals (50^{o}C per hour). Heat-up rates should be developed around stress/strain curves plotted as a function of temperature for the specific metal used. Once the operating temperature is reached, it is good operating practice to keep the melter at temperature and eliminate thermal cycling. For maintenance or other considerations that cannot be avoided, it will become necessary during the melter operating life to reduce the temperature of the melter. Design planning should accommodate these instances to minimize the number of occurrences and the amount of cool down and reheat required. When reheating, a slow, controlled reheat is recommended.

Glass Pouring is usually accomplished by air lift over a weir into a heated discharge chamber with a pour trough that directs the flow into the top of a canister.

Design Considerations in High-Temperature Devices include methods to accommodate the expansion and movement of materials during heat-up, the possibility of auxiliary cooling of the outer structural shell, selection of materials for low expansion coefficients, and the selection of materials for corrosive resistance at operating temperatures the component will actually experience.

Penetrations/Sleeves through the melter shell will usually encounter the largest temperature differentials during normal operation. Special consideration should be afforded the selection of materials for corrosive-resistant properties and strength characteristics at the operating temperatures involved. In some cases, the use of a sacrificial inner liner material may be worthwhile, keeping in mind that the details of remote replacement of the component will have to be a part of the design.

Warping and Alignments are critical concerns in melter design. Because of the operating temperatures involved, materials are going to distort. Selection of materials, innovative use of refractories, and the ability to cool areas of the melter that are not integral to the glass melting function can be used to minimize warping. Movement of materials during heat-up is a reality that should be factored into the design.

2.14.12 Off-Gas Processing. Off-gas from melter- and canister-filling operations should be processed through some type of off-gas collection and treatment system. The collection portion of typical systems includes a ducted suction arrangement to maintain a slightly negative pressure on the melter "pool" chamber and the discharge cavity, which in turn interfaces with the canister filling operation. The amount of actual airflow from these areas should be balanced against loss of heat from both melter chambers and effects on the glass discharge stream while maintaining adequate collection of the off-gas stream. For example, excess airflow from the discharge chamber can affect the glass pour stream by causing wavering, a less viscose glass stream, and the formation of glass "bird nests" in the canisters being filled.

The treatment portion of the off-gas system typically consists of air scrubbers and HEPA filtration for particulate removal plus nitrous oxide reduction through addition of ammonia in the presence of a catalyst.

2.14.13 Material Considerations. The chemical make-up of the off-gas stream requires special attention. Typically, resistance to the corrosive characteristics of the off-gas will dictate the types of materials used in the system. Material selections typically include the 300 series of stainless steels, Inconel, TM and Hastalloy.TM

2.14.14 Process Development. Vitrification development is the process by which a recipe and protocol are developed for the manufacture of an acceptable high-level waste product. In particular, this activity focuses on developing and refining the process for manufacturing the vitrified high-level waste into a glass matrix exclusive of the canister. The development process focuses primarily on two areas:

- identification and development of the final waste glass product composition and
- development of the process for achieving the final waste glass product given the available technology.

Development of the final waste glass product composition, frequently referred to as the “target composition,” is driven by a number of factors, including regulatory performance specifications, behavior of the target composition in the melter, and compatibility with the melter technology.

Regulatory requirements are issued through various DOE Offices including the Office of Civilian Radioactive Waste Management (DOE-RW) and Office of Environmental Management (DOE-EM). Those offices, respectively, promulgate their requirements in the Waste Acceptance System Requirements Document and Waste Acceptance Product Specifications for Vitrified High-Level Waste Forms. Each of these documents contains the technical specifications for acceptance of their vitrified high-level waste into the repository. As such, among other characteristics, target compositions exhibit the characteristics necessary for meeting these technical specifications. With respect to the glass waste form, these specifications address the chemical composition, radionuclide inventory, product consistency, phase stability, and hazardous waste. Additionally, the specifications provide requirements for the high-level waste Canisters and Canistered Waste Form.

The behavior of the target composition in the melter and its compatibility with the technology addresses the issues of foaming, melt rate, precipitation of metals, viscosity, mixing behavior, temperature, and corrosiveness. Because these issues relate to melter technology, they represent the practical aspects of preparing an acceptable glass in compliance with regulations, given the technology.

Development of the process for manufacturing the final waste glass product involves generating data associated with the behavior of the various feed streams. Data required to develop predictive models can aid in product and production control. Data for models include pH, viscosity, rheology, off-gas, and redox.

Typically, the development program should follow a strategy of—

- identifying various sets or families of potential glass target compositions that meet regulatory specifications;
- identifying and further developing a subset of glass target compositions that will be compatible and processable through a melter;
- choosing a preferred or reference composition on which front-end process development and control schemes are developed.

The strategy uses a combination of laboratory crucible testing, small melter scale-up, and full-process scale-up to obtain data and develop empirical relationships necessary for predicting process behavior and glass product performance.

ADDITIONAL REFERENCES

DOE Orders and Standards

DOE-STD-1020-2012	*Natural Phenomena Hazards Analysis and Design Criteria for DOE Facilities*
DOE STD-3014-2006	*Accident Analysis for Aircraft Crash into Hazardous Facilities*
DOE-STD-3025-2007	*Quality Assurance Inspection and Testing of HEPA Filters*

Other Government Documents

NRC R.G. 3.54	*Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation*
NRC R.G. 8.8	*Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable*

Non-Government Documents

ACI 349-06	*Code Requirements for Nuclear Safety Related Concrete Structures*
ASHRAE Handbook	*Vol 1-4: Fundamentals, Refrigeration, HVAC Applications, and HVAC Systems and Equipment*
ASHRAE Std-62-2010	*Ventilation for Acceptable Indoor Air Quality*
NCRP Report No. 151	*Structural Shielding Design and Evaluation for Megavoltage X- and Gamma-Ray Radiotherapy Facilities*
ANSI/HPS N13.56-2012	*Sampling and Monitoring Releases of Airborne Radioactivity in the Workplace*