4. SOLID MEDIUM AND LOW LEVEL WASTE

Medium and low-level waste (LLW) is frequently defined and described on the basis of what it is not rather than what it is. The Low-Level Radioactive Waste Policy Act of 1980 (Public Law 96-573) defines LLW as radioactive waste not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or by-product material as defined in Section 11(e)(2) of the Atomic Energy Act of 1954. This last exclusion refers to the waste (or tailings) produced by the mining and milling of uranium and thorium. In establishing its regulations for land disposal of radioactive waste (10 CFR Part 61), the NRC essentially repeated the definition of the act but specified that such waste must be "acceptable for land disposal." This qualification referred to a series of concentration limits for selected isotopes contained in the waste.

LLW is produced-or potentially produced-as a result of any action in which radioactive material is used. Information on sources of LLW, the chemical and physical form, and the radioisotope concentrations is needed to develop adequate handling procedures at the site of generation, during shipment, and at the disposal facility. LLW characterization is a moving target requiring periodic review and update. This is very different from the case of HLW the characteristics of which remain relatively stable even though the amount of waste produced in a given time may vary. The variation in LLW characteristics reflects the very large number of applications to which radioactive materials may be put, the many individual and independent users, and the development of new applications and packaging and treatment procedures. Regulatory requirements and economic viability further influence the amount and characteristics of LLW produced and requiring disposal at any given time. For clarification, it is important to note that there is a substantial difference, in volume and radioisotopes contained, between LLW "produced" and that "requiring offsite disposal." There are several management options available to an LLW generator and the mix of actions used depends on the isotopic distribution and the chemical and physical form of the waste. There are allowable limits on releases to air and water from a licensed facility and some material is handled in this manner. Storage onsite for decay and eventual disposal of the material as nonradioactive waste is a practice that is particularly applicable to short half-life isotopes such as those often found in medical waste.

Table 28 Typical LLW Streams by Generator Category produced by Power Reactors (R), Medical, Academic and Institutional generators (M), Industry (I) and Government (G).

Source	R	M	I	G
Compacted trash or solids	X	X	X	X
Dry active waste, dewatered ion-exchange resins	X			
Contaminated bulk and plant hardware	X		X	X
Liquid scintillation wastes, absorbed liquids		X	X	X
Biological waste+ animal carcasses		X		

Low-level radioactive waste policy amendments act (Berlin 1989)

The U.S. nuclear industry, particularly the utilities that generate large volumes of LLW, are increasingly using volume reduction treatment techniques to reduce processing and disposal costs. However, there are significant uncertainties for industry as to what techniques to develop because of the inability to predict evolving federal regulations and state LLW compact disposal requirements for waste parameters. At the same time current legislative and regulatory requirements are imposing increased LLW disposal costs requiring that the industry now employ volume reduction techniques. Under the Low-Level Radioactive Waste Policy Amendments Act of 1985, in 1993 the compacts containing the three commercial disposal sites will be able to exclude waste from generators in other compact regions and independent states. Until then, the amount and cost of LLW sent to these sites from outside their compact region are regulated by a volume limit and imposed surcharge. In addition to a cumulative limit for each of the existing sites, there is an allocation system limiting the volumes that may be shipped by individual power reactors. The current allocation limits will be further decreased in 1990, and the surcharge progressively scaled up from $10 in 1986-1987 to $40 in 1990-92 for waste shipped from non regional generators. There is, however, a development that could act to reduce to some extent the amount of volume reduction performed by nuclear facilities. If the NRC enacts a generic rule for some LLW to be considered as "below regulatory concern " and disposed as routine industrial waste, less waste would have to be processed. In the U.S., LLW is currently defined as all waste that is not high-level, that contains less than 10 nCi/g (370 Bq/g) transuranic nuclides and that is not mine or mill tailings.

Collection and segregation of waste (Berlin 1989)

The collection and segregation of LLW prior to either processing or packaging of the waste are driven by the acceptance criteria established by the NRC and DOE at commercial or government LLW disposal facilities, respectively. These criteria, which are essentially comparable, are in turn based on a waste classification system and a set of corresponding waste form acceptance criteria for land burial of LLW initially defined in 10 CFR Part 61. This

classification system defines three categories of waste: Class A, Class B, and Class C. In each class, maximum concentration limits are set for individual radionuclides, with the limits increasing from Class A to C. Class A places only minimum requirements on waste form and characteristics; Classes B and C set more stringent requirements on waste form and characteristics to ensure physical stability after disposal, mainly requiring that the waste must remain stable and recognizable for 300 y. This is accomplished by measures such as stipulating that less than 1% of the volume as free water is permitted for waste packaged in a disposal container designed to ensure stability. Where radionuclide concentrations exceed the Class C values, the waste would not be accepted for land disposal unless specific approval is granted by the NRC for an exception.

Table 29 Classification for radwaste based on 10 CFR Part 61, maximum activities in Ci/m^3

Class	T<5 y	Co-60	Ni-63*	Sr-90	H-3	Cs-137
A	700	700	3.5	0.04	40	1
B	-	-	70	150	-	44
C	-	-	700	7000	-	4600

* in activated metal 10 times higher

In addition to the classification system, 10 CFR Part 61 also imposes other minimal prescriptive acceptance criteria on LLW disposers for all waste classes, which in turn result in actions on the part of generators and processors to segregate the waste prior to packaging. These include : waste must contain less than 1% of the volume as free liquid, wastes must not be capable of deterioration, explosive decomposition or reaction, or explosive reaction with water, wastes must not contain or generate toxic gases, vapours, or fumes that might be harmful, wastes must not be pyrophoric, and hazardous, biological, pathological or infectious waste material must be extracted to minimize the non radiological hazard.

Sources of solid LLW (Berlin 1989)

LLW characteristics are described in the following sections for the several major categories of waste generators. This discussion is substantially based on material compiled in support of the NRC rulemaking on land disposal of radioactive waste.

Nuclear fuel cycle (NPWT 1978)

The term nuclear fuel cycle encompasses extraction of U (mining and milling), chemical purification and conversion (refining), preparation of proper mixture of fissile and fertile

material (enrichment), fuel fabrication, spent fuel reprocessing, treatment and storage and disposal of RW.

Uranium mine and mill tailings (Berlin 1989)

The NRC estimated in 1979, that premature deaths due to radon releases from 1978 to 2000 (some 20 million Ci) from uranium mining and milling operations would total 181 fatalities. Of this total, 60% was estimated to be due to mining operations. This value is considered an upper bound on potential effects because domestic uranium production has not increased as quickly as projected in this analysis. Because uranium is a very concentrated energy source, it is economic to process even very low concentration ore. Average ore grades of approximately 0.113% U_3O_8 are currently (1983) processed. The corollary to that statement is that substantial amounts of waste material are produced at the mine and mill for every pound of uranium recovered. Most regulatory and analytical attention to date has focused on the milling operation rather than the mine because the tailings material is more geographically concentrated and in a physical form (crushed stone and sand-like fines) that is more readily dispersible by wind and water. Management of uranium mine and milling wastes is needed because approximately 6% of the original uranium content is not recovered in the milling process and is retained in the waste. Two members of the uranium decay chain, radium-226 and radon-222, have received particular attention in terms of potential health impact. Both are alpha emitters with slow biological turnover once they enter the body. Radium settles preferentially in bone and the mobile gas radon produces daughter products that lodge in the lung. The actual amount of tailings generated per pound of yellow-cake produced depends upon the initial uranium content (grade) of the ore, and the recovery rate achieved in the milling process. Facility throughput then determines the total amount of tailings produced per year. The chemical and radiological constituents of reference uranium mill tailings are:

Table 30 Reference Uranium Mill Tailings.
Dry solids (pCi/g) : U_3O_8 : 63, 226-Ra : 450, 230-Th : 430; Total = 4400 pCi/g
Liquids (g/l) : ammonia : 0.5, Ca : 0.5, Cl : 0.3, Fe : 1, Mn : 0.5, Hg : 7E-5, Mo : 0.1, Na : 0.2, sulphates : 30;
Liquids (pCi/l) : U : 5400, 226-Ra : 400, 230-Th : 1500, 210-Pb, 210-Po, 210-Pb, 210-Bi : 400.

The tailings (produced at a rate of 1800 MT/day * 16E+17 Bq/MT = 1E+14 Bq/y) are considered to exist as sand (solids larger than 75 pm), slimes (solids smaller than 0.075 mm), and liquid solutions of chemicals from the ore and process reagents. The slimes are estimated to contain 35% (by weight) of the tailings. They also contain 85% of the radioactivity of the tailings. Physically, the tailings are slurried to an onsite impoundment or tailings pond. During mill operation, the tailings in the pond are initially kept

wet, reducing the potential for wind transport. The earthen embankments used to form the tailings pond are increased in height and thickness as operation continues to permit additional tailings to be stored as produced. The final tailings pond for the model mill analyzed by NRC is 1 km (3100 ft) long at the centerline. The embankments are 10 m (33 ft) high, 13 m (43 ft) broad at the crest, and 52 m (174 ft) broad at the base. The pond is estimated to contain 1 630 000 m^3 of tailings to a depth of about 8 m (26 ft) within approximately a 80 ha (200 acre) area (30 ha or 75 acres are wet during operation). Once operations cease, the tailings pond will dry out. Stabilization and closure procedures are intended to minimize the dispersion of particulate material primarily through wind erosion and the release of radon gas. In estimating the amount of tailings waste to be managed and methods, responsibilities, and schedules for such management, distinction is made among uranium mill tailings sites as being either active or inactive. Active sites are those at which milling operations are still being conducted. This means that the tailings are generally wet to a large extent and the retention walls are being monitored for stability. There is also an ongoing income source that can be used for necessary waste management actions and personnel to accomplish them. Inactive facilities, on the other hand, must first be analyzed to determine the amount of material present, and the condition of the impoundment and surrounding area that may have elevated uranium levels because, of dusting from the tailings impoundment. Many of these sites are being managed by the U.S. DOE in accordance with federal law (Uranium Mill Tailings Radiation Control Act).

Tailings sites

Active mill tailings sites exist in seven western states in which most of the U.S. uranium production has occurred to date. DOE estimates that some 9E+7 m^3 (3.48 billion ft^3) of commercial tailings had been accumulated through 1982. This material contains over 15E+15 Bq (408 thousand Ci) of radioactivity and produces about 9000 W of thermal power as a result of radioactive decay. New Mexico accounts for about half of this total amount and Wyoming almost 30%. The five other states in which commercial tailings exist are Colorado (6.1 %), Utah (5.8%), Texas (4.4 %), Washington (2.2 %), and South Dakota (1.2%). Future tailings production depends on the rate of growth of nuclear power capacity, the extent of uranium impacts, and the efficiency of recovery processes. In the 1987 report, DOE estimates that over 14E+7 m^3 (5 billion ft^3) of tailings will exist by 2000. In addition to wastes existing at active mill sites, tailings exist and must be managed at DOE sites and at inactive sites, many of which were originally operated in support of government weapons production programs. Several other sites were operated to recover radium from uranium ores. These facilities predated any of the currently existing regulatory agencies and requirements and the details on waste amounts, characterizations, and sometimes locations have had to be carefully

recreated from a variety of sources and with mixed success. DOE's 1987 waste inventory report identified over 17E+6 m^3 (608 million ft^3) of mill tailings subject to remedial action programs. Another 1E+6 m^3 (340 million ft^3) of soil and stabilization material subject to contamination by windblown tailings may also need to be handled. Most (91% by volume) of the material was produced at some 24 sites in 10 states, mostly in the west. It is being managed by DOE under the Uranium Mill Tailings Remedial Action Program. The one non-western site is in Canonsburg, Pennsylvania where radium was extracted from carnotite ore from 1911 to 1922. Operation from 1930 to 1957 produced radium and/or uranium from ores and scrap. The balance of the material consists of about 1E+6 m^3 (40 million ft^3) of tailings generated at a government-owned mill in Utah and about 57000 m^3 (2 million ft^3) of material removed from structures in the Grand Junction, Colorado area where mill tailings were used as construction material from 1952 to 1966.

Uranium conversion facilities

Source and special nuclear material wastes contain the isotopes of uranium, 235-U and 238-U, and small amounts of their daughter products (56Ci/y = 2E+12 Bq/y). They are produced in the early steps of the nuclear fuel cycle, at the conversion facilities where milled uranium dioxide is processed into gaseous uranium hexafluoride, at the enrichment facilities where the 235-U concentration is increased from the naturally occurring 0.7 % to approximately 3-4 %, and at the fuel fabrication facilities. Non-fuel-cycle producers of these wastes are primarily industrial facilities that process depleted uranium. Both economics and health and environmental impact considerations have resulted in fuel cycle facilities being designed to recycle and recover as much uranium as possible from the process streams. It is estimated (NRC 1981) that about 1400 m^3 (50 000 ft^3) of process waste is shipped for disposal as solid waste annually from existing conversion facilities. This is less than 1% of the plants throughput. The concentration of uranium isotopes in the process waste is estimated to be 0.4 mCi/m^3 (1.1E-5 Ci/ft^3), making this a very low-activity waste stream. The gaseous diffusion uranium enrichment complex in the United States is a government run process. Small amounts of uranium are contained in liquids from equipment cleanup that is routed to settling ponds onsite where it precipitates as sludge. This material is retained onsite rather than shipped for commercial disposal. Should site operations and uses change in the future, these areas would be analyzed to determine what recovery actions would be appropriate, if any. Fabrication of fuel produces LLW in the form of dry solids of CaF_2 or MgF_2 containing low concentrations of enriched uranium and other low-activity waste that is shipped offsite for disposal. Other uranium-bearing waste in the form of liquids and sludges are being stored onsite pending decisions on the timing and extent of waste treatment and uranium recycle operations or decontamination of the storage area and removal of the material fer offsite disposal.

Nuclear power reactors (Berlin 1989)

Control of radioactive materials in a power reactor is generally achieved by removing material from process streams, concentrating it in a relatively small volume and disposing of that volume as LLW. Projected volumes for new power plants were estimated by NRC in 1982 for licensing evaluation purposes to be 480 m^3 (17 000 ft^3) for PWRs and 820 m^3 (29 000 ft^3) for BWRs. Actual waste generation rates have decreased substantially over the past 5 y at many power reactors. This reduction was a response to both the rising costs of disposal services and the uncertainty of their availability. Small amounts of radioactive material are present in the coolant of a power reactor from the fuel as well as from corrosion of the system's metallic components and impurities in the coolant that have been activated by neutron bombardment. The radioactive waste treatment systems are designed to remove these materials on an ongoing basis through filtration and ion-exchange resins in both the primary system (in contact with the fuel) and secondary or auxiliary systems that treat liquids with which primary coolant may have come in contact (e.g., through steam generator or valve leakage).

Ion-exchange resins use small (about 1 mm diameter) organic beads or granules to remove radioisotopes from liquids. The resins may be specifically designed to remove anions or cations or may contain both cation and anion removing resins (this is called a deep bed or mixed bed resin or demineralizer). The resins are generally packed in cylindrical containers through which the liquid streams flow. As the waste flows through the resin bed, ions present in the resin selectively exchange with those in the waste at rates and in amounts that are dependent on differences in charge on the ions and concentrations in the waste and resin. Once the ion-exchange capacity of a resin bed has been exhausted, the resins may be either replaced or regenerated. Resin regeneration is accomplished by washing the resin with a concentrated solution of the ion originally present (generally H_2SO_4 for cation resins and NaOH for anion resins). Regenerant solutions may be further concentrated by evaporation and solidification prior to shipment. Spent resins are transferred as a slurry to shipping containers where they are dewatered (to 42-55% water absorbed in the resin) prior to shipment. Resin densities have been reported to range from 0.67 to 0.91 g/cm^3. Some facilities may solidify the resin in cement or a polymer prior to shipment. Radionuclide concentrations in spent resins are generally sufficiently high that shielded shipping containers are required. Gas generation (CO_2, NO_x, SO_x) due to chemical, radiolytic, and biological decomposition may occur in the resin. It must be considered in designing the waste package and disposal unit because it may provide a transport mechanism for radionuclides from the waste to the biosphere.

Filters used in nuclear power reactors are generally either cartridge filters or precoat filters. The primary difference is that cartridge filters contain disposable filter elements made of woven or wound fabric, or pleated or matted paper supported by a

stainless-steel mesh. Precoat filters have filter aids such as diatomaceous earth, powdered mixtures of cation and anion exchange resins (POWDEX resins), and high-purity cellulose fibres deposited as a thin cake on the initial, reusable filter medium. Once exhausted, these filter aids are backflushed from the filter and disposed as a dewatered but unsolidified sludge with an average density of 0.86 g/cm^3. For comparison, the average cartridge filter density is 0.6 g/cm^3. Cartridge filters are more commonly used in PWRs and precoat filters in BWRS. Concentrated liquids are produced at some power reactors using evaporators to reduce the volume of liquid waste to be disposed. The concentrated liquids are also known as evaporator bottoms. They have a high solids content and an average density of 1.0 - 1.2 g/cm^3 prior to solidification.

Dry active waste is the term generally applied to a wide variety of waste products such as cleaning materials, glass, filters, concrete, miscellaneous wood, and metal. It may be compactible (such as wood, glass, fibre) or noncompactible and combustible (such as paper) or noncombustible (pipe or hardware). The extent to which such material is segregated so that distinctions can be made between the waste types identified as compacted trash or solids, dry active waste, contaminated bulk, and contaminated hardware will vary with operations at a given facility and the volumes of waste being produced. For example, during a refuelling outage or facility modification there will probably be greater than normal numbers of workers using protective clothing and/or cleanup solutions and rags. They will produce sufficient amounts of waste of a given type (e.g., concrete block) that waste packages may contain only one waste type. During normal operations a single package (drum, crate, or liner) may contain a mix of several of these waste types. Similarly, compaction may be performed routinely on all waste generated or may be contracted on an as-needed basis for periods of high-volume production. Accurate characterization of this waste stream, therefore, must be based on facility and shipment-specific data rather than a generic model. Non fuel reactor components such as fuel channels, control rods, and in-core instrumentation are relatively low volumes of waste that require special handling (remote operation and shielding) onsite, in-transit, and at the disposal facility because of the high radiation levels (primarily due to 60-Co) and long half-lives of contained radionuclides (such as 63-Ni).

Decontamination

Decontamination of plant vessels and equipment may be performed periodically during the operating life of a reactor facility to reduce the occupational exposure that would otherwise be incurred during major equipment repair and/or replacement operations. Chemical decontamination has been performed on the primary cooling system of a BWR in 1980 and repair by sleeving or capping of steam generator tubes in PWRs has included prior removal of buildup on metal surfaces in contact with primary coolant water. LLW is produced by decontamination operations primarily as spent

ion-exchange resins through which the decontamination fluid is processed. The resins are expected to be similar to resins produced during plant operations with higher concentrations of activation products (e.g., iron, nickel, cobalt, chromium) found in reactor steel and fuel components. The resins may also contain large quantities of chelating agents. There are strict limits on the amounts of chelates permitted in LLW because of their tendency to concentrate and mobilize radioisotopes. Special packaging and disposal requirements may be imposed on wastes exceeding the regulatory limits for chelate concentration.

Decommissioning (Berlin 1989)

Decommissioning nuclear power reactors at the end of their useful life will produce large volumes of LLW, much of it as contaminated concrete and metal vessels and piping. Decommissioning waste will contain the same isotopes as plant hardware and noncompactible dry active waste produced during plant operation. Several of the small, early power reactors have already ceased operation because changes in regulatory requirements on issues such as seismic protection or emergency core cooling would require modifications so costly that producing power is no longer economic. Larger units more representative of current technology, however, are being studied to determine if life extension (beyond 40 y) is feasible with some equipment repair and/or replacement. Initial estimates of the volumes and activities of decommissioning waste were based on studies undertaken for the NRC in 1980. The studies considered three alternative decommissioning scenarios: immediate dismantlement (extending over several years) sufficient to enable the site to be released for unrestricted use; delay of up to 100 y after shutdown to permit decay of shorter lived nuclides prior to dismantlement; and encasement of radioactive materials in place in concrete or some other material and release of the site for unrestricted use once the contained radioactivity has decayed to acceptable levels. The volumes and activities estimated for immediate dismantlement of the reference facilities are:

Table 31 Summary of LLW from Immediate Decommissioning Reference Nuclear Power Reactors.

	PWR		BWR	
	Volume	Activity	Volume	Activity
Waste Stream	(ft^3)	(Ci)	(ft^3)	(Ci)
Activated metal	97100	4841300	4900	6552300
Activated concrete	25000	2000	3200	200
Contaminated metal	192000	900	549200	8600
Contaminated concrete	374700	100	59200	100
Dry solid waste (trash)	50600	-	119500	-
Spent resins	1100	42000	1500	200
Filter cartridges	300	5000	-	-
Evaporator bottoms	4700	-	18300	43800
Total	745800	4889300	754800	6605200

All Figures are rounded to the nearest hundred.

Deferring dismantlement for 100 y after shutdown results in approximately a 10-fold reduction in the volumes of decommissioning waste requiring disposal.

Institutions (Berlin 1989)

Medical and academic institutions use radioactive materials in research, diagnosis and therapy and employ a variety of radioisotopes and material forms to achieve their purposes. Surveys of institutional waste generators identified specific waste streams that were contributing significant waste volumes. These are biological wastes and trash. Smaller volumes of accelerator targets and sealed sources were also produced. The isotopes generally contained in these wastes were 51-Cr, 192-Ir, 35-S, 125-I, 32-P, 14-C, 90-Sr, 3-H, 57-Co, 99m-Tc and 60-Co. Isotopes with shorter half-lives, such as 8 day 131-I, are generally stored onsite until sufficient decay has occurred that the wastes may be discarded as nonradioactive (typically after 10 half-lives). Biological wastes are produced by research programs at hospitals and universities and consist of animal carcasses, tissues, animal bedding and excreta, vegetation, and culture media. Small amounts of pathogenic and carcinogenic substances may also be included in these wastes depending on the initial research problem. The materials are generally of very low specific activity. Care must be taken in packaging, storage, and disposal of these materials because the potential gas buildup due to biodegradation of the waste can overpressurize the containers and cause them to fail. If this happened in storage or in transit, time-consuming cleanup with additional occupational exposure would be required. After disposal it is possible that the gas may provide a transport mechanism for waste from the disposal cell. General practice is to ship carcasses packed with absorbent material and lime in a 30-gallon drum within a 55-gallon drum and place absorbent material between the two drums. Incineration rather than direct burial would reduce the volume of biological waste requiring disposal and preclude the problem of gas generation within the waste. Questions of offgas control for radioactive and other materials as well as the economics of operation at different volume levels are major determinants of when incineration is used for this waste stream as well as other combustible materials such as trash.

Trash produced at institutional facilities differs from that produced at power reactors in that it is primarily composed of materials such as paper, rags, glassware, packaging, and mops that are compactible and/or combustible. When treated onsite, it is generally compacted although some larger volume generators have installed LLW incinerators. Volumes of LLW produced by medical and academic institutions have decreased substantially since 1979. Much of this reduction has resulted from the regulatory changes exempting some liquid scintillation fluids from the need for disposal as LLW. Increased onsite storage for decay, more careful waste segregation to remove nonradioactive waste previously disposed as LLW for convenience and conservatism, and compaction of

trash are the other major reasons for this reduction. A 1986 study of medical waste generators anticipates small continued reductions in the overall volumes of medically related waste requiring disposal. With the exception of a few specific instances (e.g., use of 111-In rather than 99m-Tc for antibody labelling), this study does not anticipate that the characteristics of the waste will be significantly different from current experience.

Accelerator targets are used to produce radionuclides through direct interaction with charged particle beams or indirectly through the interaction of induced radionuclides and other materials. The targets are generally titanium foils containing absorbed tritium. Sealed sources have radioactive materials in the form of foils or beads encapsulated to prevent leakage during use. They may be relatively low in activity and used as calibration or reference sources for radiation detectors and analytical instrumentation used in research or clinical laboratories. High-activity sources may be used for medical radiotherapy or for research such as investigation of radiation effects on materials. These targets and sources may be disposed directly by the institution or they may be returned to the manufacturer at the end of useful life. Depending on the levels of activity and isotopes in a given source, the manufacturer may recycle the contained material or simply dispose of the used sources.

Industry (Berlin 1989)

Industrial processes that result in the production of LLW (36 000 m^3/y, 390 000 Ci/y) include the production and distribution of radioisotopes for medical, academic, or industrial use, manufacture of materials containing radioisotopes, and the use of radioisotopes for research or testing and in gauges or instrumentation. Medical isotope production is achieved through irradiation of highly enriched uranium fuel and the separation and purification of the resulting fission products. LLW generated from medical isotope production includes solidified aqueous liquids and trash produced in the separation, cleanup, and shipping of the radioisotopes. The solidified aqueous liquids contain a radioisotope distribution similar to spent fuel with several isotopes (particularly shorter lived isotopes such as 90-Mo 131-I, 133-Xe, and 125-I) being selectively removed. Isotopes present in the solidified aqueous liquids include uranium, transuranics at concentrations less than 100 nCi/g, and fission products such as 90-Sr, 137-Cs, and 3-H. Compaction is used to reduce volumes generated and storage is used to reduce activity shipped. The solidified solids are stored onsite for approximately 9 months after production. The final waste package shipped for disposal consists of a small metal container of the solidified aqueous salts packed within a drum containing low specific activity trash. Although technically an industrial source, manufacture of clinical and research radiopharmaceuticals produces waste that is very similar in isotopic distribution and concentrations to that present in institutional waste streams. Testing of the biological uptake and transport of new

pharmaceuticals frequently involves use of radioisotope tracers, often 3-H because it follows general bodily fluids.

Table 32 Volumes, Specific Activities (SA) and Activities of Industrial LLW Streams (1980).

Waste Stream	Volume (ft^3)	SA (Ci/ft^3)	Activity (kCi)
Medical isotope production	6,800	16	109
Industrial tritium	3,500	66	231
Sealed sources	200	160	32
Other (A > 1 Ci/ft^3)	2,600	6	16
Source and special nuclear material	1,103,500	8E-5	0.1
Other (A < 1 Ci/ft^3)	162,700	8E-5	0.01
Total	1,279,300	-	388

All Figures rounded to nearest hundred cubic feet.

High specific activity industrial waste (defined as greater than 0.1 Ci/ft^3, 3.5 Ci/m^3) is the name used by NRC to describe activated metal and equipment produced by accelerators, research reactors, and neutron irradiation capsules. The isotopes expected in this waste stream are primarily the activation products 14-C, 55-Fe, 59-Ni, 60-Co, 63-Ni, and 94-Nb produced by the interaction of neutrons with trace elements in the metal. The distribution of these isotopes is expected to be similar to that observed in decommissioned non-fuel-reactor components. Tritium is widely used in biological research and medicine as well as in commercial products such as paints and dials because of its luminescent properties. Because of its many applications there are relatively large volumes of tritium wastes, from the production of tritium (which results in tritium, fluoride, and trash), from incorporation of the tritium into biological compounds (labelling), and fabrication of luminous products such as watch dials. This is the only one of the 25 generic waste streams characterized and evaluated by the NRC in the LLW rulemaking (10 CFR Part 61) that contains only a single, isotope. Based on surveys of previously disposed tritium waste, the NRC estimated that the average concentration of tritium in the waste was 2300 Ci/m^3 (66 Ci/ft^3).

Government (Berlin 1989)

Government operations produced 2% (1600 m^3/y) of the LLW disposed at commercial facilities in 1984. These wastes are similar to wastes being shipped by other generator categories. For example, Veterans Administration Hospital waste is substantially the same as other medical waste, and government-produced used luminous equipment dials are the same as those from industrial facilities. LLW produced by "atomic energy defense activities" and "federal research and development activities" are not subject to the provisions of the Low Level Radioactive Waste Policy Act requiring the states (or compacts) to provide disposal capability. These wastes are produced by naval reactors development and propulsion, weapons activities, verification and control technology, defense

materials production, inertial confinement fusion, defense waste management, and defense nuclear materials security and safeguards.

Phosphogypsum wastes

Phosphate rock contains relatively high concentrations of uranium. When the phosphate ore is mined, the uranium-bearing material is brought to the earth's surface, crushed, and processed by screening and flotation. The waste products include slimes and sands similar to uranium mill tailings. Phosphate mines have operated in the United States in Florida, Tennessee, Idaho, Montana, Utah, North Carolina, and Wyoming. The Florida deposits occur near the surface and strip mining and return of the tailings as rill in the mined-out areas are common practices. Treatment of the phosphate rock to produce phosphoric acid for use in fertilizers produces insoluble gypsum (calcium sulfate), which is also generally deposited with the mine tailings. The National Council on Radiation Protection and Measurements reports that the slimes contain about 50% of the radioactivity present in the original ore. An additional 12% is retained in the sand. Radium concentrations of 1.1 - 1.5 Bq/g (30-40 pCi/g) have been measured in the gypsum by-product. Several facilities have been operated to process the phosphoric acid stream to remove the contained uranium for commercial sale. Use of this by-product production for uranium reduces the amount of activity present in the tailings piles and mitigates the waste management demands. It also reduces the uranium concentration of the ammonium phosphate fertilizer product. The phosphogypsum wastes represent a large volume of material with relatively high concentrations of radionuclides easily accessible to humans. Homes have been built over reclaimed phosphate mine areas in Florida and the gypsum piles are estimated to be accumulating at a rate of 11 million tons per year. Cumulative gypsum piles contained about one billion MT through 1983. Assuming specific activity of 400 mcCi/MT one obtains total activity of 1000 Ci. NRC estimated in 1979, that 36 000 Ci of radon are released annually from land reclaimed after phosphate mining.

Waste categories (NPWT 1978)

The dry radwaste generated at nuclear power plants can be classified as compactable or noncompactable, combustible or noncombustible, and as combinations thereof. Although the treatment of these wastes varies in detail from plant to plant, there are just a few practices in general use. The dry wastes under consideration in this discussion are either noncompactable and noncombustible (e.g., spent cartridge filters) or compactable and combustible (e.g., paper, rags, plastics, etc.).

Noncompactable/noncombustible

Spent cartridge filters which probably cannot be strictly classified as "wet" or "dry" are in the category of noncompactable

and noncombustible wastes. They are routinely used in rather large numbers (i.e., estimated as high as 175 per year) in a modern twin-unit PWR power station, but seldom used in BWR plants. The cartridges are changed out on the basis of either high pressure drop or a limiting radioactivity level. Spent cartridge filters at PWR plants are highly radioactive, with contact dose rates of several R/h being common. Because of their high levels of radioactivity, they are put into portable lead shields immediately upon removal from their in service cubicles. This operation has usually been done with custom design equipment including remote control apparatus or special purpose long-handled tools furnished by the utility itself. The necessity arose because the cartridge filters throughout the plant normally have not been standardized. Recently, a 20 cm (8 in) I.D. lead filter-transfer cask has been offered commercially by PPI which features an internal hoist with a grapple operated by a simple pendant switch and a drip pan at the bottom. The assembly will accommodate the filters presently in common use. The massive shields containing the spent filters are transported by overhead crane to the packaging station. Here they are placed into shielded storage or shipping containers and either imbedded in some solidification agent or packed in sorbent materials. The storage or shipping casks containing the packaged cartridges are then moved to an on site storage pit to allow for radioactive decay, or they may be shipped immediately for off-site disposal.

Compactable/combustible

Large volumes of compactable and/or combustible wastes are generated at nuclear power plants. This is especially true during refuelling and maintenance operations. The most popular way of preparing these wastes for off-site shipment in the United States has been to compact them in 210 l drums. However, at least one older nuclear power plant merely collects the bulk of these wastes in standard 0.13 m^3 (4.5 ft^3) fibreboard boxes which are shipped off site with no further treatment. One of the older PWR plants (Yankee Rowe) is the only operating reactor in the United States that uses incineration to treat some of these wastes, whereas this practice has been widespread in Europe for many years.

Special classes of wastes (Murray 1989)

"Mixed wastes" are those containing both hazardous chemicals and radioactive substances. Hazardous wastes are defined as materials that are toxic, corrosive, inflammable, or explosive. They contain specific elements such as lead and mercury, pesticides such as DDT, and cancer-producing compounds such as PCBs and dioxin. The disposal of hazardous wastes is regulated by the Environmental Protection Agency under the Resource Conservation and Recovery Act while radioactive wastes are controlled by the Nuclear Regulatory Commission under the Atomic Energy Act. Thus it has been necessary to establish consistent dual rules by agreement between

agencies. The EPA and NRC provide a formal procedure by which one can decide whether a certain material is mixed waste. They also give written suggestions for the design of a disposal facility that meet both agencies requirements. Included are double liners and leachate recovery equipment for EPA and waste isolation and intruder barriers for NRC. It is estimated that only a few % of low-level wastes are in the category of mixed wastes. Their disposal is not believed to pose a major problem. Greater than Class C (GTCC) wastes are in the category of low-level wastes but differ in two ways. They either have a higher activity than the upper limit of the NRC's Class C or contain larger amounts of transuranic materials than allowed. Examples are activated metal wastes from reactor decommissioning and TRU wastes from nuclear fuel testing or uranium-plutonium mixed oxide fuel fabrication. Such materials cannot be given near-surface disposal. The Department of Energy is committed to accept GTCC wastes for storage and disposal along with high-level wastes. The volume of such wastes generated through the year 2020 is estimated to be relatively small.

Dry solid treatment technologies (Berlin 1989)

Technologies used for processing dry solid LLW can be classified as transfer concentration and conditioning technologies. Transfer technologies such as decontamination will transfer activity from one waste form (e.g. contaminated equipment) to another waste form (radiactive cleaning solution). Concentration technologies include such procedures as compaction and shredding and they will concentrate activity without changing its physical or chemical form. Conditioning technologies (packaging, impermeabilisation) will confine and immobilise radioactivity and prepare waste forms for transport and disposal. Some technologies (combustion) combine concentration and transfer of radioactivity.

Dry solids are treated to reduce their volume rather than to meet transportation and/or disposal requirements as is the case with liquid and semisolid wastes. Volume reduction for dry solids is considered when the high disposal costs resulting from shrinking disposal site capacity make volume-reduction technologies an economically attractive alternative. The selection of the appropriate volume-reduction technology is initially a function of facility waste generation rate, since treatment is most appropriate where high waste volumes are produced. Other factors to consider in selecting a technology are the radioactive characteristics of the waste (extent of surface contamination and activity levels), shreddability, combustibility, and metal content. It is also important to note that the chosen dry solids treatment technology should not change the disposal classification (A, B, or C) of the waste by increasing the specific activity beyond the class limit, nor should it increase package surface radiation levels to those that require shielded transport or cause handling and/or storage problems.

Compaction and compactors (NPWT 1978, Berlin 1989)

Compaction is a concentration technology in which a press is used to compress the dry solid waste into the final disposal container or into reusable shipping containers. The volume reduction factor achieved during compaction is a function of the void space in the waste, the force applied by the press, the bulk density of the material, and its spring back characteristics. Compactors vary in size, design, and capacity and often are custom designed for the facility's floor space and waste characteristics. A typical compactor system (F 13) will contain a power unit, a drive system (hydraulic or mechanical), a platen, a base plate, supporting members, a platform on which the package is positioned, and a control panel. A system to control dispersion of airborne radioactive particulates is also usually incorporated into the compactor system. The control system may include a hood, a shroud placed around the package (i.e., drum) opening, a high-efficiency particulate (HEPA) filter, and an exhaust blower. Volume reduction factors of between 2 and 6 can be achieved for these types of systems when used to compact dry solids such as clothing, laboratory equipment, paper, and plastics. Compaction should not be used with dense or bulky articles where minimal volume reduction would be achieved, with wastes containing free liquids, or with wastes containing explosives.

Nearly all LWRS in the United States have some type of compactor for compressing dry compactable radwaste into 210-l drums (so called drum compactors). Problems most often encountered in this operation are in-building dust releases and occasional bent or broken platens due, usually, to poor waste segregation. A compactor system consists essentially of a hydraulic system with a ram operating vertically downward, a contoured support plate, and frame. Most compactors used at power plants have been designed with a 9 100 kg (20 000 lb) maximum force. In an attempt to circumvent the most prevalent problems with commercial drum compactors, the S-E Co. has designed a compactor with 13 600 kg (30 000 lb) force which features a hinged loading table door and a hinged enclosure door for extension of the space above the drum. This facilitates loading and accommodates waste stacked to as high as 1.5 m (60 in.). For example, rolled up paper generated during refuelling, placed endwise in the drum, can be compacted with ease. The drum enclosure is equipped with a filter system, exhaust fan, air filter, gages, and controls. Filled drums can be removed by overhead crane or lift truck.

Baling (Berlin 1989)

Baling is a concentration technology operating on the same principle as compaction, but in which the waste is compressed into generally rectangular bales and secured (banded) to maintain the reduced volume. The bales are then usually placed into disposable containers. Compaction and baling are often accomplished as a sequence of operations. The use of this technique has been

pioneered at federal laboratory facilities. Bales come in a range of sizes, processes, and design configuration (hydraulic, electric, and hand-operated platen models). Design variations include continuous extrusion typically for paper and cloth, multiple platen stroke for compressing a variety of wet or dry wastes including scrap metal, and two-stage multiple platen operating at right angles used generally for scrap metal. The units are constructed to operate either in a horizontal or vertical position. In using bales with radioactive waste it is necessary to install contamination control systems to contain radioactive particulates released during the baling process.

Shredding (Berlin 1989)

Shredding is primarily used as a pretreatment operation for dry solid waste prior to incineration or compaction. Shredding can be used on paper, cloth, and plastics achieving a volume reduction factor of about 3. Shredders operate through the intermeshing of a number of motor-driven counter rotating shafts. Upgraded versions will permit reversal of the direction of motion of the motor and shaft to clear jamming, and replacement of cutting teeth to handle different waste forms.

Sectioning (Berlin 1989)

Large metallic and nonmetallic waste objects containing significant void volume are capable of substantial overall volume reduction through sectioning with cutting equipment. Tanks, reactor components, boxes, and contaminated vehicle bodies are among the objects that can be sectioned prior to packaging. The cutting equipment can be operated directly in a hands on fashion when radiation levels are low, or remotely when radiation levels are high.

Decontamination (Berlin 1989)

Decontamination is a transfer technology that involves the removal of surface radioactivity from equipment and structural components using chemical or physical techniques. Decontamination permits reuse of the equipment or buildings and, when used prior to disposal, may enable the equipment to be disposed of as nonradioactive. The decontamination process typically produces waste cleaning fluids containing the radionuclides removed from the cleansed surface, which then must be disposed of as liquid LLW. The decontamination project can involve the application of one or more surface treatment techniques including chemical decontamination, manual decontamination, ultrasonic cleaning, and electropolishing. There are industrial firms that specialize in performing decontamination operations that bring their specialized mobile equipment and trained personnel to a site for the duration of the operations. Although the use of equipment such as high-pressure water and steam cleaning systems, electropolishing systems, wet and

dry vacuuming systems, ultrasonic cleaners, and degreasing systems has become more common as the technology has become more sophisticated, much simple surface decontamination is still accomplished by hands-on manual and chemical processes.

Concentration technologies (Berlin 1989)

Increasing pressure to reduce the volume of waste shipped to the disposal sites is resulting in greater use of concentration technologies (CT) and increasing expenditures on the part of both the DOE and private industry for development of new CT concepts and improvement of existing technology. Although CT has been primarily used at DOE facilities in which economics was not an overriding consideration, the emphasis on volume reduction technologies driven by the increased disposal costs resulting from a growing scarcity of existing disposal space, and potentially from the more demanding siting and disposal requirements imposed by 10 CFR Part 61 on the compacts, is making the comparative economics of CT of commercially generated waste more attractive and spurring the adaptation of CT technology in the private sector. A further impetus to the use of CT for commercial wastes results from the increasing volumes of wastes generated and the additional radiological safety personnel required to manage the wastes at these facilities.

Combustion (Berlin 1989)

The combustion process of incineration has been used as a volume-reducing technique for LLW, and transuranic waste since the early stages of the nuclear industry. Incineration converts the combustible waste into radioactive ashes, residues that are chemically inert, nonflammable, and homogeneous, and gaseous effluents containing entrapped radioactive particulates. It is therefore essential to equip the incineration system with highly efficient, multistaged "off-gas" control systems to remove the radioactivity to below environmental release standards before emission of the cleansed gases. Thus, in addition to the ash and other solid residue generated, the off-gas controls will produce a secondary waste stream (e.g., filters) that requires disposal. At least 50% of the solids generated at nuclear fuel cycle facilities are combustible with dry active waste (DAW) being the waste form most suitable for volume reduction by incineration. Certain organic liquids and the semisolid spent ion-exchange resins can also be incinerated. Solid waste forms that are not suitable for incineration, and that need be sorted out prior to treatment, are primarily those that have a radioactivity content sufficiently high as to cause excessive doses to operating personnel or to raise the radioactivity content of the ash above prescribed limits. In addition, wastes with a high rubber or PVC content, containing large metal objects or a high content of other noncombustibles, or having an explosion potential are also excluded from incineration. Since the incinerators are designed to operate with wastes having total heating values between specific limits, wastes with high

heating values may have to be excluded whereas those with low heating value will require the addition of supplementary fuel to achieve complete combustion.

An incineration system designed to process radioactive wastes should achieve complete combustion of the wastes while providing radiological protection to operators and the public under normal and accident conditions, and containing the radioactivity within the incineration system and the off-gas treatment system. The generic incineration system for radioactive waste treatment consists of waste feed pretreatment and loading facilities (e.g., shredding), single or multiple combustion chambers, ash collection and unloading equipment, ash transfer and/or immobilization equipment, off-gas treatment system including particulate removal components, gas scrubber, fans, a stack, and a system for recycling certain secondary waste streams through the combustion chambers, and process equipment instrumentation to monitor critical operating parameters, instrumentation to monitor health and safety-related limits, and controls to enunciate radioactive releases and initiate systems shutdown if required. Incinerators designed for treatment of radioactive wastes use a variety of combustion approaches either singly or in combination.

Combustion techniques, which transform the waste to an inert or less reactive form and reduce volume and weight, include commercially available and developmental incineration technology and promising developmental concepts such as acid digestion, molten salt combustion, and pyrolysis. The concepts discussed below have been developed under federal government sponsorship at DOE laboratory facilities. The various combustion techniques will achieve volume reduction factors between 20 and 100, when applied to combustible wastes that are essentially organic materials such as paper, cloth, plastic, resins, solvents, and rubber. The residue remaining after the combustion process consists primarily of the inorganic constituent in the waste. These residues are generally of a higher activity level than the original wastes, and must be packaged in the appropriate container for shipment to a disposal site.

Acid digestion

Acid digestion is a combustion process in which the organic materials are converted to gaseous end products (CO_2, H_2O, etc.) and insoluble sulfate or oxide residues by digestion in hot concentrated H_2SO_4 in the presence of an oxidant. The concept was originally conceived at the DOE Hanford Engineering Development Laboratory. The gaseous stream produced is scrubbed to remove the end products of the reaction to permit release to the environment, and the contaminated liquid scrubber material becomes a secondary waste stream. The major advantages of this system are the wide range of waste that it can process when the waste is sorted and pre-shredded, the low temperature, single stage operation, and the ability to process high levels of radioactivity. Developmental problems with the system include the need to use glass or

Teflon-lined containment vessels to achieve corrosion resistance, relatively low processing rates (5 kg/h maximum), handling and disposal of the acid after depletion, and the need to remove radionuclides from the acid after digestion.

Molten salt

Molten salt combustion is a process by which organic materials are rapidly and completely oxidized by the molten salt medium containing an oxidizing agent to produce CO_2 and H_2O, and the ash and other combustion products are trapped in the molten salt. Thus, the molten salt serves as a heat-transfer agent, as a source of oxygen to accelerate combustion, and as a scrubbing agent to react with the gases generated (except CO_2) and entrap ash in the furnace. This is accomplished by having the waste and air fed below the surface of the salt causing the combustion gases to pass through the melt before release to the environment stripped of everything except CO_2 and H_2O. Periodic removal of the ash and other inorganic materials from the melt is required.

Pyrolysis

Pyrolysis is a combustion process in which the organic combustibles are gasified in an oxygen-deficient atmosphere. This developmental concept is being applied to the slagging-pyrolysis incinerator. While the conditioning process of solidification is primarily required for liquids and semisolids prior to packaging and shipping, the highly dispersible ash produced during combustion may require solidification to meet disposal site constraints.

Combustion techniques and types of incinerators

The principal combustion techniques are categorized by process temperature ranges. Controlled-air incineration limits the air supply in the primary combustion phase and requires a secondary combustion phase to achieve complete combustion (800-1100 C). In excess-air incineration an excess of oxygen is fed into the primary combustion-phase to permit both the solid and gaseous components to burn directly (800-1100 C). In pyrolytic or thermal decomposition the organic waste material is essentially distilled in a highly reducing atmosphere (absence of oxidation) generating combustible liquids and gases. A carbon residue (char) remains after pyrolysis (500-600 C). In fluidized bed incineration an inert bed of particles is suspended by air flow through the bed and the pre-shredded waste is burned upon introduction into the self-sustaining bed (800 C). Slagging incineration at high temperature (1400-1600 C) burns the organic material releasing sufficient heat energy to convert the non combustibles to a molten slag residue. In electromelt incineration the waste is burned and sorbed into a glass melt maintained by electromelt joule heating (1200 C). A variety of incinerator designs has been developed incorporating the combustion techniques described above. These designs, originally

developed under DOE sponsorship, have, in a number of instances, been adapted for commercial use.

Agitated-Hearth

The most recent agitated-hearth incinerator developed at the RFP is a 70 kg/h production scale unit modeled after a small-scale 5 kg/h unit. It is a stationary, refractory-lined circular steel vessel 2.6 m in diameter by 4.6 m high with rotating agitator arms. The arms move the ram-fed shredded waste through the combustion zone of the hearth. The waste is processed on a batch basis using a semi-continuous feed and the ash is discharged on a batch basis. The off-gas produced in the after burner is then treated with a series of scrubbers, a gas-liquid separator, and a HEPA filter.

Controlled-Air

The controlled-air incinerator (F 5.10) is a multiple-chamber unit in which combustion takes place in both the primary chamber and secondary chamber. Presorted wastes are ram fed into the primary chamber, and the resultant gases and entrained solids are further oxidized in the secondary unit. The offgas treatment system is, in sequence, a water spray for quenching, a venturi scrubber to remove particulates, a moisture removal system, and a series of roughing and HEPA filters. As a result of the successful performance of the 45 kg/h unit, the private sector has adapted the design for commercial use.

Excess-Air

An excess-air cyclone incinerator (F 5.11) was developed to process solid TRU wastes. This unit has a combustion chamber consisting of a fixed upper section, and a lower removable combustion vessel. Batch quantities of waste are fed into the combustion chamber concurrent with the combustion air that spirals down from the upper section of the chamber. The ignited wastes burn downward and gases pass through scrubbers and filters prior to release.

Fluidized-Bed

The fluidized-bed incinerator (F 5.12) is a demonstration unit for the treatment of LLW and TRU wastes. This 80 kg/h production size incinerator is has a primary chamber holding a bed of sodium carbonate granules that is fluidized by an air-nitrogen gas flow. The waste undergoes partial combustion and pyrolysis (flameless) and becomes self sustaining from the heat generated from the pyrolysis process. The off-gas from the primary chamber passes through a separator to remove entrained solids and into the secondary chamber where complete combustion is achieved using added air in a bed of chromium oxide. The off-gas stream from the secondary chamber containing fly ash, dust, and gases phases

through a separator and a series of HEPA filters to remove these constituents. The system (F 5.13) is capable of incinerating solids, resins, heavy sludges, and liquids and has been installed at a number of utilities including to process DAW, waste oil, and evaporator bottoms.

Rotary-Kiln

A 40 kg/h rotary-kiln incinerator (F 5.14) consists of a horizontally rotating, cylindrical primary chamber and an afterburner. The primary chamber is designed to tumble the wastes to improve combustion efficiency and move the incinerated waste in a continuous, gradual flow toward the ash discharge where it then fails into the collecting container. Off-gases are sequentially treated in an afterburner, wet scrubber, and HEPA filter systems. The RFP system was designed to treat TRU wastes for plutonium recovery, and can also process a variety of solid and liquid wastes including HEPA filters, sludges, and ion-exchange resins. There are no commercially available rotary-kiln incinerators to process radioactive wastes.

Slagging-Pyrolysis

The slagging-pyrolysis incinerator, a vertical furnace with a primary-stage gas fill and a secondary-stage combustion chamber, was evaluated at INEL for processing soil-contaminated TRU waste. This incinerator type provides a large throughput, has a high volume reduction factor, produces a stable residue suitable for disposal, and is thus suitable for treating contaminated soil. The INEL concept, which represents an adaptation of a commercial slagging-pyrolysis unit exposes the waste to treatment in three stages as it passes down through the gasifier drying, pyrolysing, and combustion/melting ranges. Supplementary heat is added to the gasifier to generate the molten slag that is removed and quenched with water. The gases pass from the gasifier to an afterburner, then a regenerative column, spray dryer, cyclone particulate separator, and sintered metal filter.

Molten glass

The Penberthy Electromelt is a commercial incinerator (F 5.15) adapted for developmental treatment of LLW at DOE laboratories. The process renders combustible and noncombustible wastes into glass. Waste glass is initially fed into the system, is melted, and becomes electronically conducting as the furnace heats up. An immersed electrode provides the current necessary to maintain self-sustaining joule heating. The liquid and shredded solid wastes are oxidized in the furnace and incorporated into the glass. The final product is removed as a liquid and then solidified. The offgas system consists of a flue-gas cooler, spray for acid removal, demister, reheater, and charcoal and HEPA filters. The developmental testing of the Penberthy units at DOE laboratories

has encompassed a demonstration of vitrification of defense waste (Pacific Northwest Laboratory) and immobilization of combined waste sludges (Savannah River Laboratory).

Packaging and container handling of solidified wastes (NPWT 1978)

Solidified radioactive wastes from nuclear power plants are packaged in various kinds of containers ranging from 210-l (55 gal) drums to large disposable steel cask liners, having volumes of 1.4 to 5.7 m^3 (50 to 200 ft^3), and in some instances disposable concrete casks have been used. In usual practice within a nuclear plant, filled waste containers are moved by means of a conveyor, crane, forklift, or air pallet. Some solidification system vendors prefer disposable liners, but others prefer drums. However, of all the solidification systems discussed in this chapter, only the one is limited to a single style container. The remotely operated system is integrated from beginning to end and is designed throughout to accommodate only drums. The salient feature of this system is the trolley-mounted bridge crane which has a 7.5-ton hoist equipped with a specially designed grab for handling 210 l (55 gal) drums. The drum grab has a clamping jaw grip with a motor-operated actuator to ensure positive load release control. The crane is monitored and operated remotely with the aid of TV cameras and lamps mounted on both the trolley and the grab mechanism. A locating grid visibly mounted on the ceiling, together with a device which indicates elevation of the grab, allows precise drum placement and retrieval.

Regardless of the solidification system used, there are some wastes that may require shielding and/or remote handling during the solidification and shipping operations if minimum personnel exposure is to be attained. Good general housekeeping practice is important, but it may also be necessary to incorporate special design features for adequately cleaning waste mixing and transfer equipment, for surveying external radiation on containers and providing for their decontamination, and for placing waste containers inside additional shielding (e.g., storage or shipping casks). All nuclear power plants should provide adequate storage for processed waste containers awaiting off-site shipment. Ideally, the storage area would be capable of accommodating at least 30 days of normal waste generation. This allows capacity for abnormal plant operation or for backlog storage in the event of a transportation strike.

Disposal of solid low-level wastes (Murray 1989)

The management of low-level radioactive wastes is in a state of evolution. Early disposal practices were inadequate; stricter regulations have been developed; and designs of facilities other than shallow land burial have been proposed. The problem of finding acceptable locations for disposal sites is generally believed to be more social than technical. Only after World War II did the problem of low-level wastes arise. New radioisotopes for medicine,

research, and industry became abundant through irradiation in nuclear reactors. Their residues formed wastes, and continue to do so. The second and much larger source was nuclear power production, which produced contaminated materials and equipment. The Atomic Energy Commission (AEC) maintained control of all nuclear activities for a number of years. The AEC disposed of its wastes by near-surface burial. In 1963 the first commercial disposal site was established, and by 1971 there were six sites, located at West Valley, New York; Sheffield, Illions; Maxey Flats, Kentucky; Richland, Washington; Beatty, Nevada; and Barnwell, South Carolina. The total volume of LLW that have been placed in the six commercial sites is 42.58 millions ft^3 (12E+5 m^3).

Table 33 Volumes of Low-Level Wastes at Commercial Burial Sites (in millions of cubic feet)
Barnwell, South Carolina : 18.76, Beatty, Nevada : 3.58, Richland, Washington : 9.87, Maxey Flats, Kentucky* : 4.78, Sheffield, Illinois* : 3.12, West Valley, New York* : 2.47; *inactive sites.

At all of these sites, the technique basically was to dig a trench, fill it with boxes and drums of waste, replace the excavated earth, apply some compaction, and form an earthen cap above the trench. As such, they were a cut above sanitary landfills, but below modern standards for disposal facility design.

Shallow land burial (Murray 1989 and Chapman 1987)

Burial of low-level radioactive wastes in trenches or pits at shallow depths has been practised for the last 40 years, principally by those countries with early development of nuclear programmes. The volumes of low activity wastes currently being generated are many orders of magnitude greater than high-level wastes, but because of their short-lived nature and relatively low potential hazard, a simple technique such as shallow burial has generally been considered an appropriate solution to their disposal. Burial in simple trenches up to 10 m deep with an earth or clay cap is a widespread practice. By 1980 about 760 000 m^3 of waste had been disposed of in this way in six of the principal commercially operated sites in the USA. By 1980 the USA was generating more than 90 000 m^3 of LLW a year, most of which was being buried at the Barnwell facility in South Carolina. It is estimated that more than 10 million m^3 of LLW will have been produced in the USA by the late 1980s. In the UK the Drigg site in Cumbria currently disposes of up to 1E+5 m^3 a year in shallow trenches and additional sites are being sought for disposal of these wastes. The wastes buried in shallow trenches are now subject to stringent controls in terms of their activities and radionuclide content, and in many countries this applies also to their physical and chemical form and packaging.

Although some operational sites still dispose of unpackaged and unconditioned mixed LLW in a rather haphazard fashion, there is more impetus now towards standardized packaging or compaction,

allowing for better engineered disposal. Such packages are carefully emplaced in a trench, void spaces filled with earth, and the trench compacted and capped.

The sites at Oak Ridge National Laboratory, Tennessee, West Valley, New York, Sheffield, Illinois, and Maxey Flats, Kentucky, have presented particular problems since they were developed before the importance of detailed geological characterization and groundwater surveys was appreciated. However, most of the regulations now governing trench disposal were in force by the mid 1970's and present site selection and operation procedures for shallow land disposal are the subject of more adequate methodology and safety precautions. Clearly the history of shallow land disposal is long and complex, and it is our intention here simply to review the current practices and proposals for shallow and deep (up to 30 m) trench disposal, the techniques for selecting and developing sites, and the factors that have to be considered when carrying out safety assessments.

The disposal of LLW in a trench slightly below the surface is called shallow land burial, in contrast to deeper mined-cavity disposal as practised for high-level wastes. The conventional arrangement is with the wastes placed a suitable distance above an aquifer and the site fenced to prevent entrance. There has been mixed success with this disposal technique. Three of the commercial sites, (West Valley, Sheffield, and Maxey Flats) developed leaks and were closed, while the remaining three have operated satisfactorily. Three principal types of failure have been noted. The first type is simple erosion by surface water, which exposes waste containers to the elements. The second type occurs when the wastes are loosely packed and will compress under the weight of dirt. Then the cap subsides, pockets of water appear, and water percolates into the waste, eventually leaching out the radioactive material. The third type is the "bathtub effect," in which a leaking cap lets water into an excavated cavity with nearly impermeable walls. The wastes are immersed in water for long periods, the containers corrode, and the wastes dissolve. The bathtub fills and overflows, carrying contaminated water to the environment. Accounts differ widely as to the amounts of release of radioactivity and resultant hazard to the public. It is a fact, for example, that tritium and strontium-90 have been measured at Maxey Flats. One estimate is that the resulting dose was less than 1% of that from natural background. Statements by others suggest that the hazard was much greater.

The design of such earlier facilities was clearly inadequate. The problems were due partly to insufficient investigation of geologic features before sites were selected. Contributing factors were the loose packing of wastes, the presence of liquids initially, and the poor design of caps to exclude water. Operating companies claim that they followed the federal regulations that existed at the time. Hindsight reveals the primitive nature of regulations on site selection, waste packaging, and facility design. The remaining sites (Richland, Beatty, and Barnwell) had problems of a different type. Poor packaging of wastes by shippers

resulted in contamination of the area on arrival at the sites. At one location, contaminated equipment was stolen. The states were concerned with the injustice of having to receive the bulk of the wastes from the entire nation. The governors of the states threatened to close the facilities, leaving no place for LLW to go. This prompted Congress to pass in 1980 the Low-Level Radioactive Waste Policy Act. This Act placed responsibility for waste disposal on the states producing the waste. The Act recommends that regional facilities be established. As a result, several interstate compacts have been arranged.

Site selection (Murray 1989)

Experience at the closed disposal sites led the NRC to develop a new set of siting rules (F 5.2). The result after several years of work was the regulation entitled, "Licensing Requirements for Land Disposal of Radioactive Waste," Title 10 of the Code of Federal Regulations, Part 61. The regulation requires LLWs to be classified by the generator into one of 3 categories A, B, or C, taking account of half-lives and concentrations of the nuclides in the wastes. Class A wastes are of lower activity than Class B or Class C. Regulation 10 CFR 61 stresses the need for careful assessment of geology, hydrology, and other features. Some key requirements are the site geology should be simple enough to admit mathematical modelling and computation, location of the site should be far from housing or commercial development that would affect underground water flow, there should be no significant underground resources whose extraction would affect site performance, the sites should be well drained, to prevent entrance of water, there should be little chance of runoff that would erode the surface, the water table at the site should be deep enough to prevent immersion of wastes, there should be no springs in the site that could bring contamination to the surface, the area should not be subject to volcanic action, earthquakes, landslides, or excessive weathering, and nearby activities should not affect performance or monitoring.

The process by which a site is selected consists of three parts. The first is a general survey of the region or state, examining existing records on the geology, hydrology, meteorology, seismology, population distribution, land use, environmental features, and cultural aspects. The second is an interaction between the siting agency or company with communities that may wish to serve as host to the facility or at least will find it acceptable with appropriate compensation. The third is a series of measurements, including a limited number of test borings, leading to a site characterization, which is a catalog of features that relate to the siting regulations. The site characterization study following NRC guidelines must be carried out before a license to operate can be issued. Such an investigation must take at least a year to complete. Once a license is issued, the disposal site can be opened to receive wastes, and will operate for about 30 years, during which regulatory surveillance is provided by the NRC or the cognizant state agency. At the end of the operating period, the

site may be closed, meaning that no additional wastes are received. However, institutional control must continue for at least an additional 100 years. This involves measurements of waste migration and remedial action as needed. After institutional control is released, the site may be used for any purposes. The design is expected to protect the public for at least 500 years after closure. The performance specification of no more than 25 mrems per year to any member of the public must be continued.

Design of trench structures (Chapman 1987)

The detailed design of trenches varies considerably depending on the types of waste emplaced and the geology and hydrology of the host rocks. Essentially, two types of trench can be distinguished: simple trenches which are used for LLW containing primarily short-lived radionuclides, and engineered trenches for short-lived ILW or LLW with higher contents of long-lived or alpha-emitting radionuclides.

Simple trenches are generally only up to about 10 m deep, often excavated in soils or poorly consolidated sediments and constructed without any sophisticated lining or backfilling material. Simple systems of drainage may be installed in the trenches, or they may be mounded over and capped by an impermeable material such as puddled clay or bitumen, to direct rainwater away to surface drains. Waste may be emplaced in simple packages (e.g. steel drums or casks) but lower-activity wastes or, for example, large items of contaminated equipment, may simply be dumped without any form of packaging.

Engineered trenches are usually somewhat deeper (some tens of metres), and are more complex. The trench is lined with a material carefully chosen for its mechanical and hydraulic properties (e.g. compacted clay, concrete), which may be specifically modified to minimize degradation due to interaction with the host rock. The packaged wastes are embedded in cement in a concrete structure, situated below a separate concrete raft designed to protect against inadvertent intrusion. Waste packaging may be quite sophisticated, involving an embedding matrix for the waste material, a container and backfill. A range of waste types may be incorporated into standard packages (e.g. concrete casks or chests) both to ease handling and to optimize use of available space. Trench backfills, caps and seals may involve a range of materials to ensure desired properties of mechanical strength, low permeability and chemical stability, and may use clay, concrete, bitumen and plastics, either individually or in specific combinations. France, for example, has a well-developed facility for both engineered deep trenches and shallow burial at Centre Manche (F 9.3). It is not possible to give a comprehensive guide to which types of trench are used for which wastes. In general, simpler and cheaper designs will be used where possible, but the deciding factor will always be that the disposal option selected should satisfy radiological protection criteria when a performance analysis is carried out.

It is generally recommended practice to construct simple trenches so that their bases are above the water table, taking into account any seasonal fluctuations. Consequently any movement of infiltrating meteoric waters is downwards past the trench, through the unsaturated zone, and into the water table. Hence any flow paths to the surface are lengthened. Depending on the physical properties of the sediments in which the trench is constructed, and in particular their permeability, it may also be acceptable to dispose of wastes below the water table. This would normally be acceptable when host-rock permeabilities are low enough to ensure that advective flow is negligible and hence diffusion is the dominant transport mechanism. Disposal in a zone through which a water table would move seasonally is to be avoided, since this leads to flooding and drying of the trenches, high potential leach rates, and high rates of degradation of the trench structure. Most of the problems in existing sites, mentioned earlier, result from flooding of the trenches, either through fluctuation of the water table or through failure of the trench cap to deflect rainwater. Unpredictable changes in water table can be produced if extensive earth-moving during construction changes the properties of the site.

There are two approaches to trench siting (F 9.4). For disposal above the water table the host sediments should not be too impermeable as this may lead to pounding of rainwater in the trenches if an adequate cap is not provided. The ratio of cap permeability to underlying sediment permeability is thus very important, and the latter should be capable of draining percolating waters down and away from the trench while at the same time not being so permeable as to allow rapid migration of the leachate. Disposal below the water table generally means trenching in an impermeable clay. Obviously a reliable impermeable cap is essential to prevent rapid recharging and possible overflowing of the trenches in periods of high rainfall, the so-called 'bathtub' effect.

Selecting a site for simple trenches is by no means straightforward and climatic factors are clearly important, particularly when siting above the water table in humid regions. The several instances of leakages mentioned earlier clearly demonstrate the problems. However, suitable sites can certainly be found. At such sites some water is inevitably going to pass into the trenches from the unsaturated zone, so a system of gravel-pack French drains may be installed to gather this water and disperse it to underlying sediments. This prevents the wastes becoming saturated. It is possible to install piezometers and monitoring boreholes into the drain system and sediments below and around a trench to detect any migration of leached radionuclides.

Engineered trenches, being deeper, are often below the water table. Containment is ensured by a system of impermeable barriers, and any eventual release of radionuclides is controlled by diffusion in these barriers. The near-field properties are thus of considerable importance, and the models used in safety assessment must take into account gradual changes in chemistry and the

hydrogeological properties of the near-field barriers as they degrade with time.

Sites for trench disposal are thus selected on the basis of local and regional hydrogeology and climate. Sites overlying exploitable aquifer units at shallow depth would generally be avoided, as would areas with high hydraulic gradients or nearby small surface water bodies into which local discharge of shallow groundwaters occurs. Low-lying areas prone to flooding must be avoided and long-term (1000 years) geomorphological changes should be taken into account. Areas of possible slope instability, high seismic risk or rapid erosion are unsuitable. When considering the long-term behaviour of a trench site, the type of waste must also be considered. Most countries no longer bury liquid wastes of any type since rapid release may result. Consequently, some means of conditioning must be used for liquid wastes eventually bound for geological disposal. Explosive, unstable and pyrophoric materials are also excluded. Many wastes are chemically or biologically degradable and eventual gas production and settling in the trenches must be taken into account when modelling their long-term behaviour. In many older sites, where packaging of the wastes was inadequate, settling has led to problems with cracking of the trench cap and even small cave-ins. The release of biogenic gases such as methane, and gases such as hydrogen from anoxic corrosion of metals must also be considered, and for some waste types it is practice to install vents in the trench caps, or use backfills with high gas permeability. Proper packaging, emplacement, backfilling, compaction and post closure maintenance during the early life of a trench are thus essential.

Safety assessment

As for deep disposal, the principal mechanism whereby wastes buried in trenches may be returned to man's environment is groundwater transport. As radionuclide transport paths to the surface are relatively short, hydrogeological factors are of great importance in selecting a site and designing the engineered barriers of a shallow disposal facility. In addition, because the wastes are buried close to the earth's surface, a number of other release routes become important.

Release mechanisms from shallow burial sites (Chapman 1987)

For modelling potential doses to man a detailed assessment of local groundwater flow must be coupled with models of the leaching and release rates of radionuclides from the near-field. The latter is very difficult to assess since, unlike HLW, the wastes are highly variable and inhomogeneous. Given the great variability of the waste, complex leaching models are not generally justified for such trenches. For simple trenches, the near-field release model may simply assume a constant fractional release rate, with the entire inventory being mobilized over a relatively short period of tens or hundreds of years. Such release rates may be derived

empirically from simple laboratory experiments. If the near-field chemistry can be defined (for example in engineered trenches where it may be buffered by concrete or cement), then a somewhat more sophisticated model, taking into account limiting radionuclide solubilities, can be used. If thick backfill or linings are present, diffusive transport through these can be included in the near-field model, using appropriate laboratory measured retardation data.

Evaluation of transport in the far-field is effectively identical to that for deep disposal, except that the transport paths are shorter, and possibly better defined, as are the relevant time scales. Problems in modelling near-surface transport arise primarily from complexities involved in handling the unsaturated zone, and the potentially large significance of colloids, biodegradation products and other organic compounds. For some simple trench designs, indeed, it may be possible to ignore far-field transport as such, and consider a model chain which links nearfield releases directly to a biosphere model. In addition to releases into groundwater, safety assessments must also examine other potential exposure routes which arise owing to the shallow burial depths involved. Since these wastes are of low activity, and predominantly of short half-life, they will decay to relatively innocuous levels close to the natural background within a few hundred years. However, their proximity to man's environment may necessitate some period of institutional control and restriction of access to the sites during the early, high hazard period. After this period the land may be returned to normal use, with some possible restrictions. Potential exposure mechanisms during this period must be taken into account, and indeed risk analysis in a safety assessment is a valuable technique for prescribing requisite lengths of time for institutional control to minimize hazard to man.

The mechanisms which are considered can be classed as direct exhumatation of the waste, for example during borehole drilling or the construction of wells or deep foundations for buildings, or surface disruption of trench seals by agricultural practice, or by civil engineering operations (e.g. road construction, pipe laying and so on). This naturally assumes a breakdown in long-term control of land use and loss of records of previous activities on the site. Some engineered trench designs now incorporate a separate 'intrusion barrier', such as a thick concrete pad located above the top of the waste emplacement vault. While this cannot prevent intrusion, it acts as a 'stop-and-think' barrier. The likelihood of occurrence of any of these exposure routes can be modelled on a probability basis. In the case of shallow trenches, agricultural use of the land and other potential biological release routes must be considered for the post institutional control period. The mechanisms involved here are possible erosion by repeated deep-ploughing and penetration of the trench cap by deep tree roots and burrowing animals.

Alternative designs (Chapman 1987)

We have described above only the most common form of shallow land burial of radioactive wastes. There are many variations on this theme, including the use of shallow 'burial' in an engineered silo as retrievable storage for certain types of waste, which utilizes the radiation shielding properties of the ground in a cheap and convenient handling technique. Operator exposure must always be taken into account when burying wastes, especially as the types considered here are usually only thinly shielded during transfer from transport vehicle to disposal hole. Since it may take some time to fill and cover a trench, low dose rates to those emplacing the waste containers are expected. For some types of waste the extensive surface areas involved in a large trench are unacceptable in this respect and other burial techniques are used. These include slit trenches, wide boreholes, shallow pits or concrete caissons. In each case, the waste can be emplaced so as to take immediate advantage of the radiation shielding of the earth, and the surface area of radiation 'shine' is minimized.

The NRC (Murray 1989) has suggested several techniques by which the 10 CFR 61 requirements might be achieved, starting with conventional shallow land burial with sloped trench walls. Use of a narrow trench with vertical walls cuts down on the radiation exposure to operators as they place the wastes. Containers can be stacked in a regular pattern to reduce the amount of potential void space. Cubes or hexagons leave essentially zero voids. Square 55-gallon drums are now available commercially. Layered waste disposal puts material of high activity in the bottom of the trench, to protect the inadvertent intruder. The NRC recommends several moisture barriers as part of the cap system. It also suggests a design for engineered barriers against intrusion.

The three disposal facilities in operation at Richland, Beatty, and Barnwell have not experienced difficulty with waste migration and are regarded as successful, even though they are examples of shallow land burial. Those in the West are in dry regions. Barnwell is in a humid region with heavy rainfall but is located where the soil is clay, which prevents water intrusion and filters contaminants effectively. Despite these favourable conditions, publicity about the now-closed LLW disposal sites has given shallow land burial a bad reputation. The public is concerned about reactors, radioactivity, and radiation in general, and shallow land burial in particular. Consequently, some states and compacts have prohibited its use or have required an "alternative waste disposal technology" that involves "greater confinement disposal." Several design concepts that incorporate "engineered barriers" have been proposed. Generally, these provide additional protection against waste migration.

Tumuli

The use of earth-covered mounds (tumuli) to dispose of waste on the surface, while still taking advantage of underlying geology, is also a practical solution for some very short-lived wastes. In

France this technique is used to cap-off deep engineered trenches. Above-ground disposal gets around the problem of saturating the wastes with water, since they are well above the water table and rainfall can be deflected by a suitable earth cap and drain system. Any leachate which does arise would permeate down into underlying sediments. Clearly such tumuli are very vulnerable to long-term geomorphological and climatic processes, so would be suitable only for the shortest-lived categories of waste.

Below-ground vault

The below-ground vault as sketched (F 9.2) provides a barrier to migration in the form of a wall such as concrete. It has a drainage channel, a clay top layer and a concrete roof to keep water out, a porous backfill, and a drainage pad for the concrete structure. A certain amount of shielding against gamma rays from the waste is available, and intrusion is minimized. Concern has been expressed, however, about the life of concrete. Although testing of concrete is under way, it is difficult to predict behaviour for 500 years.

Above-ground vault

The above-ground vault makes use of slopes on the roof and surrounding earth to aid runoff. The roof substitutes for an earthen cover. To some, the above-ground vault appears desirable since the wastes are accessible and readily retrievable, but these features raise the issue of protecting the inadvertent intruder. The structure is in plain sight and might be viewed as a nuisance. The uncertainty about the life of concrete is accentuated because of direct effects of the weather, including freeze-thaw cycles and erosion, along with acid rain. Analysis shows that wastes eventually would be transferred by surface water, giving a dose much higher than that from other systems.

Shaft disposal

Shaft disposal uses concrete for a cap and walls. The method is seen to be a variant on the underground vault that conceivably could be easier to build. It appears, however, that costs would be higher for the same volume of disposed waste because of the amount of concrete needed. Modular concrete canister disposal consists of placing individual waste containers within concrete canisters, which are then disposed of in a shallow land site. An earthen cover for the array of canisters would also be provided. Mined-cavity disposal consists of a vertical shaft going deep in the ground, and with radiating corridors at the bottom. This is very similar to the planned disposal system for spent fuel and high-level wastes from reprocessing. It is applicable only to those low-level wastes that are comparable in activity to high-level wastes.

Intermediate depth disposal

Intermediate depth disposal is similar to shallow land disposal except for the greater trench depth and thickness of cover. The earth-mounded concrete bunker, used by the French, combines several favourable features (well below grade a concrete bunker is constructed, intended to hold Class B and C wastes with their higher activity). A mound above, called a tumulus, is formed from Class A waste containers of low activity. The rounded cap prevents water from standing. Little water can get into the structure if the soil does not erode and the concrete lasts. The steepness of the slope must be limited and a rock cover or suitable vegetation should be used. The earth-mounded concrete bunker is clearly the most secure, but also the most expensive, short of having individually monitored retrievable waste containers.

Safety features

The Department of Energy provides technical information on the design and expected safety features of the various concepts. A DOE conceptual design study of alternatives reports several significant conclusions. First, all designs will probably meet the usually accepted limit for an inadvertent intruder of 500 mrems/yr. Second, any system that involves concrete gives a higher radiation dose to personnel because of the time spent in placing the concrete. Although the concrete enhances stability and retards waste migration, its integrity cannot be guaranteed. Third, there is only a modest improvement in protection provided by intermediate depth disposal, below-ground vaults, modular concrete canisters, and earth-mounded bunkers. Benefits may well be largely psychological. Fourth, installation requires anywhere from 60 months to 84 months.

Waste disposal costs (Murray 1989)

The total life-cycle costs for a typical facility range from $196 million to $434 million (in 1986 dollars). They depend on whether the facility is publicly or privately owned. For the former, costs range from $33/ft^3 to $69/ft^3 and a few dollars more for the latter. As expected, unit disposal costs drop rapidly with increased volume disposed. Using a reference design of 235,000 ft^3/yr, if the waste volume increased 50%, the unit cost would be reduced 75%; if the volume decreased by 75%, the unit cost would increase by more than 300%. The NRC believes that the regulation 10 CFR 61 covers all the alternatives and does not plan to issue revised regulations. Instead, the NRC will provide extensive guidance as to acceptable designs.

Uranium mill tailings

Mill tailings are a very large volume source of radioactive material existing in a very accessible and dispersible condition. They are, therefore, a potentially large radiation exposure source. The milling process physically consists of crushing and grinding the ore, leaching the uranium with either an acid or alkaline

solution (depending on the lime concentration of the original ore), and processing the uranium-bearing liquid through a series of tanks that permit settlement of suspended solids (source of the tailings) followed by solvent extraction and precipitation of the concentrated U_3O_8. The U_3O_8 is dried and packaged for shipment as yellow-cake.

Surface impoundments (Berlin 1989)

Surface impoundments are employed for the permanent disposal of uranium mill and phosphate tailings. They may also be used for the disposal of other semisolid or liquid radioactive wastes from processing operations. Tailings impoundment design has undergone significant revision in the last 15 y to improve the isolation and stability characteristics of the impoundment. The primary objectives of the impoundment function are to eliminate or minimize radiochemical leachate migration and resultant impacts on groundwater, eliminate or minimize airborne radioactive emissions, including radon gas and particulates, and resultant environmental dispersion, and ensure long-term stability and isolation of the tailings without the need for continued active maintenance. These objectives are achieved by a combination of tailings treatment steps to remove radioactive constituents prior to emplacement in the impoundment (e.g., barium chloride precipitation of radium-226) and the careful selection of disposal site locations and impoundment design parameters incorporating the use of natural and artificial migration barriers. In general, the location of tailings disposal has been, and continues to be, above grade in traditional or upgraded surface impoundments. While the NRC has expressed a preference in 10 CFR Part 40 Appendix A for consideration of below-grade, near-surface, disposal, this approach has not been used in any significant manner primarily because very few new uranium projects have been undertaken in the 1980s since adoption of the Appendix A criteria, and because the NRC is willing to consider above-grade options if properly designed.

Major components

A tailings impoundment can be divided into four major components : a physical containment basin and structures (i.e., dams or dikes), natural or synthetic liner, singly or in combination, and seepage control measures, tailings management system including tailings distribution and water decant facilities, and natural or artificial cover, singly or in combination, and other stabilization and reclamation features including vegetative or overburden layers and sealants. The extent of incorporation of specific aspects of these four components is a function of site-specific characteristics such as topography, hydrogeology, climate, and soils parameters. The four components function in concert to produce a desired tailings management program, and therefore a disposal facility design must combine them to produce the desired tailings management program. For example, the seepage

potential of the underlying soil in the selected containment basin establishes whether a liner is needed or whether an impermeable bottom of the impoundment can be created from the natural soils.

The containment basin and associated structures comprise the design features that provide the physical volume of the impoundment to hold both the permanent solids fraction of the semisolid discharge and the temporary volume of fluids in the discharge. The surface impoundments can be constructed as ringlike impoundments that are four-sided structures in relatively flat areas. They can also be formed as valley dam impoundments by constructing a dam in an existing natural drainage area such as a valley or canyon. In the latter case, which has been the predominant practice to date, an impoundment basin is formed by placing a dam wall across a valley and using the natural basin sides to provide containment. Below-grade (in-pit) disposal (F 6.15) can be achieved by using existing open pit mines, or excavations to function as the basin.

In preparing the bottom and sides of the basin prior to introduction of tailings, a number of options are available to ensure that leachate migration does not occur and the tailings remain isolated from the groundwater. If the underlying soil or rock is generally impermeable, or if there is no danger of affecting the groundwater, the preparation could be limited to soil compaction to increase soil density and thus reduce permeability. However, in most cases the use of natural clay liners or synthetic liners is required to obtain the necessary isolation characteristics.

Liners

Clay liners, generally between 1 and 3 ft in thickness (0.3-0.9 m), can be used as a sealant over compacted soil to inhibit seepage, and are a superior ion exchange medium. Permeabilities of 1E-7 cm/sec and lower can be attained. Although a number of different kinds of clay exist, bentonite clay is a favoured type because of its availability in the Western states. The bentonite clay is attractive because it has a high content of montmorillonite, an expanding-lattice-clay mineral that swells when wet. Synthetic liners, in the form of sheeting or membranes, are being increasingly used, typically in combination with clay or other natural materials. Although synthetic liners are capable of obtaining permeabilities of from 1E-9 to 1E-10 cm/sec they have historically posed problems in this type of application. To avoid mechanical failures, they require careful preparation of the base to eliminate rocks and sudden slope changes; they tend to lose their flexibility when exposed to sunlight or chemicals; and it is questionable whether they will retain their integrity over the long term when subjected to the chemical and physical environment in the impoundment. Failure would tend to be catastrophic, resulting in a sudden release of contaminants. However, recent advances in the development and application of reinforced liners give promise of overcoming these shortcomings. Hypalon, a nylon-reinforced elastomer, is considered to be preferable for use with uranium

tailings. Other materials under consideration include polyvinyl chloride (PVC), Neoprene, gunite, cement grout, and asphalt or asphaltic concrete.

Management program

Tailings management encompasses, in addition to the storage and isolation capability provided by the basin and liner, the system for distributing the tailings within the impoundment, the liquid removal system(s) by means of decantation or drainage under the tailings, and the dam operating procedures and constraints. A tailings management program should successfully control the volume and locations of the fines and slimes through distribution and segregation of the tailings solids within the impoundment, the amount of water stored in the impoundment to minimize the seepage head, and the amount of tailings under water or kept damp to minimize airborne releases. Stabilization of the tailings with a cover, after removal of the water, is accomplished either as an ongoing measure or after the impoundment is full. A variety of stabilization techniques is employed typically involving multiple layers of material as is done with covers at LLW disposal facilities. A layer of clay, 1-3 ft (0.3 - 0.9 m) in thickness, is generally used to reduce water infiltration into the tailings and to reduce airborne radon emissions. The clay, however, should be covered to keep it damp and prevent wind or water erosion of the fine clay particles. A natural soil cover may be used over the clay, or an artificial cover or sealant used as an additional barrier. Where rainfall is sufficient, native flora can be planted on top of the soil cover. In semiarid regions, a layer of coarse gravel or crushed rock (rip-rap) can be used to stabilize the surface and prevent soil erosion.

Alternative tailings disposal siting (Berlin 1989)

A number of alternative tailings disposal siting and design concepts have emerged from the variety of possible combinations of locations, impoundment types, and engineered barriers that can be used on tailings impoundments. The following represent a range of tailings management alternatives.

Above-grade with continued active care

An earthen berm (dike) is constructed on the sides of the surface impoundment. Tailings are conveyed to the impoundment by slurry pipeline and the water recycled to the mill. As the beach areas dry out, they would be covered with compacted clay and a native soil cover. A vegetative cover would then be developed in a layer of top soil. Active care and maintenance would be required indefinitely to prevent erosion of the cover.

Below-grade mines or pits

Tailings are emplaced (F 6-15) in an open pit mine or excavated pit and then covered with a clay cap, natural soil (overburden), and a vegetative cover developed. In one variation of this concept, the tailings slurry is deposited in a mine pit lined with clay and partially backfilled to the level of the water table. Other variations include dewatering the tailings prior to emplacement eliminating the need for extensive liners on the sidewall, and creation of an impoundment by digging a specially excavated pit in an isolated area with relatively impermeable soils, eliminating the need for a liner. In each case the tailings would be permitted to dry before the cover materials were emplaced. For these below-grade, near-surface disposal alternatives, the tailings are isolated from erosional forces thus eliminating the need for ongoing care. These alternatives are representative of the tailing disposal concepts being encouraged by the NRC.

Specially excavated below-grade trench

For this below-grade, near-surface disposal concept (F 6-16), a pit is excavated in the form of a trench in sections, with construction, filling with tailings, drying, sealing, backfilling, and restoration moving sequentially along the length of the trench. This approach permits the reclamation of the tailings to be phased, reduces the area of exposed tailings prior to reclamation, and allows for segregation of the slimes from the sands with the slimes being covered by the sands during the deposition stage, This design approach is considered to be quite promising for future tailings disposal concepts.

Upgraded conventional above-grade impoundment

In this case (F 6-17), the intent is to incorporate design and siting features that make the upgraded impoundment alternative essentially equivalent to below grade burial from the standpoint of resisting erosion and achieving long term isolation. Among the siting and design features that would, taken in combination, help achieve this objective are siting of the impoundment where the upstream drainage area is small and where the topography shelters the face of the tailings dam from the wind, constructing the dam incorporating features accepted as standard geotechnical engineering practice (e.g., earthen dam with clay core, appropriate slope), incorporating features to cause deposition of sediment on the tailings area from any runoff and creating gradually sloped embankments during final reclamation, using appropriately thick cover layers, and stabilizing the surface with a continuous vegetative cover or rip-rap as appropriate to minimize erosion potential. This approach incorporates those features that would make above-grade disposal reasonably equivalent to that provided by the below grade concepts (alternatives 2 and 3).

Sea dumping of radioactive waste (Bewers 1987)

The use of the ocean for waste disposal (F 2) is a subject of some controversy. Some regard the ocean as a legitimate receptacle for wastes arising from human and industrial activities; others wish to preserve the ocean in as pristine a state as possible and therefore oppose any deliberate use of the oceans for waste disposal. Debates over the use of the ocean for waste disposal have intensified since the end of the Second World War, both within national jurisdictions and in the international community. Increased public awareness of environmental damage, hazards to human health and the desirability of improving the level of environmental protection from the adverse effects of human and industrial activities on the one hand, and, on the other, of the need to dispose of a variety of wastes arising from anthropogenic activities has contributed both to the polarization and intensity of these debates. The two main routes of deliberate disposal of radionuclides into the ocean being practice now are the direct discharge into the sea of low-level liquid wastes from the reprocessing of nuclear fuels for the recovery of plutonium, and the dumping of packaged low-level radioactive waste into the deep ocean. A third route of deliberate disposal being considered for future use is the emplacement of high-level radioactive waste within, or on, the seabed. Use of this latter option currently seems unlikely and, in any event, it is at least a decade distant. The word 'deliberate' is used here to discriminate between these activities and the incidental introduction of radionuclides into the ocean through fallout from nuclear weapons explosions. This latter fallout has both increased the marine concentrations of certain natural nuclides, such as tritium and radiocarbon, and introduced a variety of predominantly artificial (fission-product and activation-product) nuclides into the marine environment. The particular avenue of radioactive waste disposal that has been the subject of most international debate is the dumping of packaged low-level radioactive waste into the deep ocean, which has been practice since the end of the Second World War.

History of dumping

Dumping of low-level radioactive waste in the ocean has been carried out since 1946. Between 1946 and 1967, the United States dumped approximately 4000 TBq of radioactive waste into the Pacific and Atlantic Oceans and the Gulf of Mexico. This includes about 1200 TBq of activation products in the reactor pressure vessel of the Seawolf submarine propulsion unit. About 90% of this total activity was dumped in the North Atlantic at the '2800 m site' located at 38'30'N, 72'06'W. Packaged radioactive waste has also been dumped at ten sites in the northeast Atlantic in the vicinity of 46'N, 17'W (F 1) by seven western European countries since World War II. The dumped low-level wastes come from nuclear power plants, other nuclear fuel cycle operations, medicine, research, industry and the decontamination and decommissioning of plant and equipment. The waste is of a similar nature to that arising from non-nuclear industrial, medical, and research facilities, except that it

includes items having radionuclide contamination in surficial and chemically-incorporated forms and induced radioactivity. Accordingly, this material requires a range of special handling, treatment, and disposal arrangements. The aggregate radioactive waste dumping in the Atlantic Ocean 1948-1982 is as follows : gross mass 142275 tonnes, alpha activity 680 TBq (18.4 kCi), beta/gamma activity 38000 TBq (1 027 kCi) and tritium 15000 TBq (405 kCi). The composition of the wastes dumped has varied year by year. Plutonium isotopes and 241-Am account for over 96% of the aggregate alpha activity and tritium and 241-Pu account for over 87% of the aggregate beta-gamma activity dumped. The remainder of the long-lived beta-gamma activity is composed principally of the fission products 90-Sr and 137-Cs and the activation product 60-Co. The average dumping rates of a number of individual nuclides during the period 1978-1982 varied from 0.03 TBq for U-238 to 1490 TBq for Pu-241. The waste packages are designed to provide shielding and containment of the waste during handling and transportation, and to ensure that the packages reach the seabed at depths equal to, or greater than, 4000 metres, without losing their integrity. The integrity of the packages after descent to the seabed is not assumed or required in the development of regulations. However, some types of package can maintain their integrity, and restrict the release of contained radionuclides, for several decades after dumping.

Regulation of sea dumping

The political and administrative framework within which sea dumping of radioactive waste is carried out involves two international bodies. The first of these is the London Dumping Convention (LDC) which was finalized in 1972 and entered into force in 1975. This is the major international instrument for the formulation of international regulations for sea dumping activities and has now been ratified by 61 States. The other international body is the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD), within which data on the actual amounts dumped are collated and the safety of such disposals assessed multilaterally on a quinquennial basis. The NEA created, in 1977, a Multilateral Consultation and Surveillance Mechanism for these purposes and, in all respects, the NEA activities are consistent with the intent and principles of the LDC. All countries involved in dumping in the Northeast Atlantic (Belgium, the Netherlands, Switzerland, and the United Kingdom) are parties to this agreement, while other non-dumping NEA countries (Canada, Denmark, Finland, France, the Federal Republic of Germany, Ireland, Italy, Japan, Norway, Portugal, Spain, Sweden, and the United States) have been willing to participate in associated site-suitability reviews and safety assessments carried out under the auspices of the NEA. However, it is within the forum of the LDC, or in connection with this Convention, that the major international negotiations respecting radioactive waste dumping at

sea have occurred. As will be shown, the debate on the future of this practice within the LDc has intensified since 1983.

The London Dumping Convention

The London Dumping Convention (formally referred to as the Convention for the Prevention of Marine Pollution from Dumping of Wastes and Other Matter, London, 1972) was created following a recommendation of the First Stockholm Conference on the Environment and has as its objective the prevention of marine pollution through dumping at sea. The Convention is composed of a series of principles or articles and three technical annexes. The first annex contains a list of substances that are proscribed for dumping in the ocean, except as 'trace amounts' in other materials, includes high level radioactive material - deemed unsuitable for dumping at sea because of the human health and other hazards associated with such disposals. Annex 2 lists materials for which special care must be exercised in respect to their disposal into the marine environment and includes radioactive wastes and all other radioactive matter not included in Annex 1. Annex 3 contains a list of criteria upon which an evaluation of the effects and permissibility of a proposal to dump material should be evaluated.

The Role of the International Atomic Energy Agency

The major role that the IAEA has played in the LDC as the 'competent international authority' for radioactive matters under the Convention has been to provide definitions of high-level radioactive wastes 'unsuitable for dumping at sea' (i.e. the definition of Annex I radioactive materials). This definition of the boundary between Annex I and Annex 2 radioactive matter is termed the IAEA Definition.'The IAEA also appends to the Definition, a set of Recommendations that contain its advice as to manner in which radioactive materials having radioisotope concentrations below those specified in the Definition can be dumped and how the safety of such dumping might be assessed and ensured. These periodic 'Definition and Recommendations' documents have been issued by the IAEA in 1975, 1978 and, most recently, in 1986. The Agency has also developed additional guidance on the subject of sea dumping of radioactive wastes as well as ancillary material relating to the administration of the LDC in respect of radioactive materials. Examples of such guidance are IAEA Safety Series Nos. 61 and 65 which deal, respectively, with the overall framework for the control of waste disposal into the marine environment and environmental assessment methodologies that can be applied to sea dumping of radioactive wastes. This, then, describes the role and responsibilities of the IAEA under the LDC.

Modelling of dumping

The basic process by which the Definition of 'wastes unsuitable for dumping at sea' is derived is composed of an

evaluation of the 'capacity' of a hypothetical ocean basin, about the size of the North Atlantic, to receive radionuclides without violating the appropriate dose limits established by the ICRP for members of the public. The introduction of radionuclides into the ocean through dumping is counterbalanced both by radioactive decay and by the removal of those radionuclides to ocean sediments within which the radionuclides eventually become isolated from the biosphere. These processes, namely introduction, removal, and decay, can be modelled in such a way as to relate the concentrations of individual radionuclides in various sectors of the ocean to the rate of waste dumping. The process of deriving the definition is thus one of calculating the rates of release (at the ocean floor) of each potential constituent radionuclide, which results in an equilibrium concentration field which, in turn, corresponds to a radiation exposure (dose) to individual members of critically exposed population groups (critical groups) equal to the ICRP individual dose limit. The oceanographic model relates the marine concentration fields to rates of release of individual nuclides. While equilibrium concentrations can be reached relatively quickly for short-lived nuclides, which decay before they can be transported great distances, such equilibria for the very long-lived nuclides are only obtained on time scales comparable with the half-life of the nuclides, which can be much longer than ocean mixing time scales. Therefore, the model has to predict maximum concentration fields that are obtained after some preconceived time of continued dumping practice, or assume that the practice continues indefinitely, and predict equilibrium fields that in some cases are only obtained on geological time scales. The oceanographic model is coupled to a radiological model that accounts for routes of human exposure from the marine environment, such as the consumption of seafood, recreational occupation of beaches, and the inhalation of marine aerosols. Other potential exposures associated with future activities like deep-sea manganese nodule extraction are also considered. The limiting rates of release that correspond, for each constituent nuclide, to the dose limit are referred to as 'release rate limits' and these constitute the basic values for the establishment of a definition of HLW unsuitable for dumping at sea.

Definition of HLW unsuitable for dumping at sea

High-level radioactive waste or other high-level radioactive matter unsuitable for dumping at sea are: irradiated reactor fuel; liquid wastes from the first solvent extraction cycle of chemical reprocessing of irradiated reactor fuel, or equivalent processes; and solidified forms of such waste; any other waste or matter of activity concentration exceeding: 5E-5 TBq/kg for alpha-emitters; 2E-2 TBq/kg for beta/gamma-emitters with half-lives of greater than 1 year (excluding tritium); and 3 TBq/kg for tritium and beta gamma emitters with half-lives of 1 year or less. The above activity concentrations shall be averaged over a gross mass not exceeding

1000 tonnes. The maximum dumping rate into a single ocean basin of volume at least 1E+17 m^3 shall not exceed 1E8 kg per year.

The IAEA recommendations pertaining to the dumping

The main features of the Recommendations are dictated by the need to ensure that dose limits are not exceeded and that optimization is carried out adequately for both individual and aggregate sea dumps of radioactive materials. The Recommendations first define appropriate individual dose limits for the practice. They stress that, since members of the public will be receiving doses from other sources and activities, it cannot be assumed that a dose limit of 1 mSv is intrinsically acceptable and that, for actual ocean dumping activities per se, an upper bound to the dose should be established. Since, however, no such bound has yet been internationally established, individual national authorities should use a dose limit that is substantially less than 1 mSv. The recommendations then outline the criteria relevant to environmental assessment and safety assurance of both individual and aggregate dumping operations. They define exclusionary criteria relating to the selection of dumping sites - these must be situated between latitudes 50 N and 50 S and have average water depths greater than 4,000 metres, be clear of continental margins, islands, mid-ocean ridges, ocean trenches, fracture zones, plate boundaries, and areas of volcanic activity. In addition, the use of dumping sites must not interfere with, or prejudice, other legitimate uses of the sea. Sites should therefore be situated away from spawning areas, fishing grounds, the paths of submarine communication cables, and potential ocean mining sites (for the recovery of mineral deposits). Finally, the number of sites should be minimized and their location strictly defined. Each site should be as small as practicable (and no greater than 10 000 km^2 in area) and should not be subject to undue navigational hazards during dumping (i.e. coverage by satellite navigation should be available and the site should not be situated in shipping lanes).

Safety assessment of dumping

Safety assurance procedures used for sea dumping of radioactive waste are based upon the use of predictive models to describe the results of various scenarios for ocean disposal of radioactive wastes. Since, in the main, the radionuclides released from previously dumped wastes are not detectable, even within the area of the present dumpsite, heavy reliance has to be placed upon the use of models that depict the processes controlling the transport and behaviour of analogue stable elements. In fact, the weakest aspect of the most recent predictive models is the reliability of representations of bio-accumulation and sediment-water partitioning processes for radionuclides that are vitally important to an appreciation of the rates at which radionuclides are able to enter exposure pathways for man, and the likely effects upon populations of organisms. Significant

individual exposures are many decades, perhaps centuries, distant, but the scenarios used for safety evaluation conceive of ocean dumping and direct discharges continuing for the life of the nuclear fission industry, currently projected to be 500 years. From the results of surveillance work, it was concluded that the incidence of radionuclides in biological samples obtained from the dumping site were generally consistent with those expected from fallout and could not be attributed to radionuclides released from dumped wastes. The radiological impact of dumping activities is predicted to be very low. The peak individual dose from past dumping is calculated to be 20 nSv/a. It arises 200 years after dumping starts and occurs by way of 239-Pu and 241-Am accumulation in molluscs. Moreover, this peak individual dose involves the assumption that molluscs from the Antarctic might be exploited for human consumption, which is currently not the case. Even if dumping is continued for a further five years at ten times the rates of previous years, the peak individual dose is only 100 nSv/a occurring 200 years after the commencement of dumping. The corresponding peak collective dose rates are predicted to be 4.2 manSv per year for aggregate past dumping, and 42 manSv per year for past dumping combined with continued dumping for five years at ten times previous rates. These collective dose rates are dominated by the radionuclide 14-C which, because of its long half-life, would need to be isolated and contained for very long periods in order to reduce the collective dose from this or other disposal practices. Sensitivity analyses indicated that the peak individual dose rates were most sensitive to changes in the numerical representations of particle scavenging of radionuclides, which confirms previous conclusions that this aspect of the modelling, namely the representation of particle scavenging processes and the manner in which water/particle partitioning is parameterized and numerically represented, is the most important for continued investigation and improvement. The NEA review [1985] concludes that the site is suitable for continued dumping for a further five years at rates up to ten times those dumped in recent years. If rates of dumping are proposed that would exceed ten times previous rates, the suitability of the site should be reconsidered before approval for these increased rates of dumping is given.

Recent developments of the London Dumping Convention

In 1983, at the Seventh Consultative Meeting of the LDC, Kiribati and Nauru, Pacific island Contracting Parties to the LDC, proposed an outright ban on the dumping at sea of any radioactive waste. After discussion, the meeting adopted a moratorium on further dumping pending a review, by an independent panel of experts, of the scientific and technical basis upon which dumping practices were regulated and their safety assessed. This panel, composed of experts nominated by the IAEA and the International Council of Scientific Unions, subsequently submitted its report to the Ninth Consultative Meeting in September 1985. The main conclusions of this report can be summarized as follows:

1. The present and future risk to individuals from past ocean dumping of radioactive waste is extremely small. The risk of developing a fatal cancer or severe hereditary defect is predicted to peak about 200 years in the future at a level of less than 10 per annum. The most potentially exposed individuals would be those consuming shellfish harvested in Antarctic waters.

2. Notwithstanding the very small risk to individuals, the aggregate exposure to the global population from long lived components of the dumped waste imply that the total casualties resulting from past dumping may be up to about 1 000 spread over the next 10 000 years or so. The dominant isotope responsible for this collective dose commitment is 14-C, with 239-Pu being the next most important isotope, giving rise to a few % of the total collective dose. If the radiocarbon, and a few other long-lived radionuclides, were to be removed from the waste before disposal in the ocean, the collective dose commitment from future dumping operations would be very much reduced. However, other means of disposal of these nuclides, other than very long-term containment, would result in comparable collective dose commitments.

3. The incremental dose from past dumping to individual marine organisms on the sea-floor at the dumpsite, or nearby, will be significantly less than the dose that the organisms receive from naturally-occurring radioactivity, and hence is not expected to cause any detectable effects on populations of organisms.

Despite these conclusions the Meeting adopted a Resolution which states that the LDC agrees to a suspension of all dumpings at sea of radioactive wastes and other radioactive matter. It requests also additional studies and assessments.