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Comparison of Low-Level Waste Disposal Programs of DOE and Selected International Countries

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ABSTRACT

This document is a result of the Secretary of Energy's response to Defense Nuclear Facilities Safety Board Recommendation 94-2. The Secretary stated that the U.S. Department of Energy (DOE) would "address such issues as...the need for additional requirements, standards, and guidance on low-level radioactive waste management." The authors gathered information and compared the disposal programs used by the U.S. DOE, France, Sweden, Canada, and the United Kingdom to dispose of low-level radioactive waste. The study identified many similarities in practices but also identified some differences in disposal practices and national policies.

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Comparison of Low-Level Waste Disposal Programs of DOE and Selected International Countries

1. INTRODUCTION

The purpose of this report is to examine and compare the approaches and practices of selected countries for disposal of low-level radioactive waste (LLW) with those of the United States (U.S.) Department of Energy (DOE). The report addresses the programs for disposing of wastes into engineered LLW disposal facilities and is not intended to address in-situ options and practices associated with environmental restoration activities or the management of mill tailings and mixed LLW. The countries chosen for comparison are France, Sweden, Canada, and the United Kingdom. The countries were selected as typical examples of the LLW programs which have evolved under differing technical constraints, regulatory requirements, and political/social systems. France was the first country to demonstrate use of engineered structure-type disposal facilities. The UK has been actively disposing of LLW since 1959. Sweden has been disposing of LLW since 1983 in an intermediate-depth disposal facility rather than a near-surface disposal[a] facility. To date, Canada has been storing its LLW but will soon begin operation of Canada's first demonstration LLW disposal facility.

The percentage of LLW resulting from electrical power reactor operations is 90, 77, 50, 43, and 0 percent for France, the UK, Sweden, Canada, and the DOE respectively. The remaining wastes of the various countries come from traditional activities such as defense activities, medical applications, research and development, and industrial applications. U.S. electrical power reactor operations are controlled by the U.S. Nuclear Regulatory Commission (NRC), which is not included in this study. The DOE does, however, have wastes resulting from specialty reactor operations, which include naval vessel propulsion, test reactors, and production of weapons materials. Other typical DOE operations include high enrichment of uranium; weapons research, development and fabrication; nuclear fuel reprocessing; and other forms of basic nuclear research and development.[1] A larger amount of DOE wastes are generated from activities other than electrical power reactors, than from the programs of other countries studied. However, as stated later in this report, the waste classification systems used by the DOE and the countries studied are quite similar, so it is appropriate to compare disposal activities to identify different practices of other nations. For the programs studied, wastes falling into the category of LLW are deemed as being suitable for near-surface disposal. These wastes have the general characteristics that they do not contain large concentrations of long-lived radionuclides or beta/gamma activities which generate amounts of heat significant enough to affect the disposal facility design.

Table 1 shows the nuclear fuel cycle activities performed by each of the nuclear programs included in this study. The fuel cycle activities shown for DOE are associated with the fabrication and reprocessing of defense-related reactor materials. Sweden has the least number of fuel cycle-related nuclear activities. The waste-types and the average isotopic make-up of the wastes may vary somewhat from one program to another due to differences in origins of the wastes. These facts may affect site-specific disposal facility performance assessment, waste classification considerations, waste acceptance criteria, and waste handling considerations.

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a. As defined in 10 CFR 61.7(a), "near-surface disposal" involves disposal in the uppermost portion of the earth, approximately 30 meters. Near-surface disposal includes disposal in engineered facilities that may be built totally or partially above-grade provided that such facilities have protective earthen covers.

This report should be used as an overview document to identify differing and similar practices of waste disposal programs. Summary considerations associated with LLW policy, disposal facility design, performance assessment, and operation are addressed. It should be noted that subtle and important differences between published information and actual practice may exist. Some of these important differences have been revealed during the preparation and review of this report and are presented. Further detailed investigation of national practices may be necessary to fully understand the advantages and disadvantages of other nation's waste management programs.

The appendix to this document contains a description of the LLW disposal programs and additional details for each of four countries and the U.S. DOE.

Much of the information used to assemble this report was identified using librarian-assisted searches of commercially-kept literature databases and from literature obtained from personnel who have toured or have employment ties with the foreign facilities. Time and resources did not allow visits to the various countries or personal contact with individuals at the various disposal facilities.

The authors have avoided drawing conclusions or making judgments concerning differences in approaches and practices of the various nations. In addition, no recommendations on certain international aspects that could be considered for implementation in the DOE program have been offered.

Table 1. Nuclear activities performed by the nations under study.

Program	Uranium production	Uranium ore conversion	Uranium enrichment	Fuel fabrication	Fuel reprocessing
U.S. DOE	*	*	X	X	X
United Kingdom		X	X	X	X
France	X	X	X	X	X
Sweden				X	
Canada	X	X		X	

X Indicates that the column topic is performed within the indicated program.

* These activities are performed by the U.S. commercial industry, but not by the DOE.

2. REGULATORY APPROACH AND APPROVAL REQUIREMENTS

In all cases studied, the federal government of each country establishes the policies and regulations for the design, construction, and operation of waste management facilities. Table 2 lists the federal agencies of each country that establish policies for waste disposal and that give formal approval to construct and operate a LLW disposal facility in each of the programs included in this study. All programs studied, except the U.S. DOE and Canadian programs, have a federal agency that is administratively independent of the waste disposal activities to approve disposal facility activities and to regulate and oversee the operation of the facilities. A recent federal advisory committee for DOE

Table 2. LLW disposal facility approval requirements and public involvement.

Program	Policies established by	Final approval agency	Public involvement	Regulator	Operator
U.S. DOE	Federal Government (Department of Energy)	DOE-HQ	Public hearing associated with the programmatic or site-wide Environmental Impact Statement.	Federal Government Agency (DOE-HQ)	Government Agency (DOE) Subcontractors
United Kingdom	Federal Government (Department of the Environment)	Health and Safety Executive	Local planning authority (county or borough) must review the proposal and address written public comments. If it refuses to approve the proposal, the applicant may appeal to the Secretary of State who may overrule the local authority after a public hearing.	Federal Government Agency (Health and Safety Executive)	Government-Owned Corporation (NIREX)
France	Federal Government (Atomic Energy Commission)	French Prime Minister	A "Public Inquiry" is held, making all information available to the public. Written public comments are evaluated by an "Inquiry Commission" appointed by the local administrative court. After a report is completed by the Commission and circulated to the public, communities within a 5-km radius of the project are allowed to vote for or against the project. A negative vote can be overruled by the national Parliament.	Federal Government Agency (Ministry of Industry)	Government Agency (ANDRA) Subcontractors
Sweden	Federal Government (Ministry of Environment and Energy)	Nuclear Power Inspectorate	The local municipal council, as well as the Federal Government licensing agency (Nuclear Power Inspectorate), must approve the licensing of the disposal facility.	Federal Government Agencies (Nuclear Power Inspectorate and National Institute of Radiation Protection)	Private Company (Nuclear Fuel and Waste Management Company)
Canada	Federal Government (Atomic Energy Control Board)	Atomic Energy Control Board	Most new nuclear facilities in Canada are referred to the Federal Minister of Environment for a formal public review by an independent panel, with full opportunity for public hearings, and funding for intervenors.	Federal Government Agency (Atomic Energy Control Bureau)	Generator [Government (Atomic Energy of Canada Limited) owns 90% of existing waste]

investigated the DOE approval policy and has recommended that the DOE be subject to control by an independent regulatory agency.[2]

Political and social considerations are recognized by most countries as crucial to gaining acceptance and approval of a LLW disposal facility. All programs studied allow the public to have input during the approval process for the facility. Table 2 describes how each country involves the general and local public in the approval process for LLW disposal facilities.

France is the only country studied that requires a public vote before licensing a new nuclear waste disposal facility. A negative vote can only be overturned by the national Parliament. The UK and Sweden require approval by a local representative council before a disposal project can be authorized. In the case of a negative vote, the Secretary of State of the UK may overrule the local authority after a public hearing. No overrule of the local municipal council is currently allowed by Sweden. Both the U.S. DOE and Canada hold public meetings to solicit public comment on the facility, but no formal local or public approval is required for those governments to approve the facility and begin construction and operation of a LLW disposal facility.

3. WASTE CLASSIFICATION

An initial step in comparing and discussing the LLW disposal approaches of various programs is to determine if the programs use similar criteria to identify LLW. Table 3 identifies the basic criteria used by each program to define LLW. The table also shows the classification criteria recommended as typical characteristics of waste classes by the International Atomic Energy Agency (IAEA)[3]. (Note: Use of IAEA criteria is not mandatory for any nation). As noted in the table, the recommended IAEA criteria have been adopted for use by Sweden.

Although there are differences in classification of LLW, all programs recognize that LLW disposed in near-surface disposal facilities should not contain significant quantities of the longer half-lived radioactive isotopes. All programs plan deep geologic disposal facilities for spent fuel and long-lived wastes (excluding mill-tailing type wastes). Since the alpha-emitting radionuclides typically have long half-lives, many programs have chosen to limit their concentrations in waste classified as LLW, as shown in Table 3. The alpha-emitting isotopes with half-lives greater than 20 years are listed in Table 4. The following differences are noted regarding acceptable maximum alpha-emitter concentrations:

> The UK, France, and Sweden all limit alpha-emitting radionuclide concentrations to approximately 100 nCi/g. France implements this requirement differently than other programs in that the long-lived alpha-emitter limit is as calculated 300 years after time of acceptance. France also allows up to 500 nCi/g per container, on a limited exception basis.
>
> DOE Order 5820.2A limits only alpha-emitting transuranium elements (having half-lives greater than 20 years) in LLW to concentrations less than 100 nCi/g. Alpha-emitters which are not transuranium isotopes (and have half-lives greater than 20 years) are identified in Table 4. These lighter alpha-emitter elements include uranium, thorium, and radium. DOE Order 5820.2A authorizes Heads of Field Elements to specify that other alpha-contaminated wastes, peculiar to a specific site, must be managed as transuranic wastes.
>
> Canada acknowledges that the LLW must be isolated for a period up to 500 years but has not yet assigned quantitative values on the long-lived elements. This undefined classification is meant to include those wastes that will be acceptable for near-surface disposal in the proposed Intrusion Resistant Underground Structure (IRUS) discussed later. Technical

Table 3. Comparison of low-level waste classification systems.

Program	LLW classification parameters[a]	Alpha limits	Disposal
U.S. DOE	Low-level waste - Not transuranic (<100 nCi/g transuranium alpha-emitting radionuclides with half-lives greater than 20 years); not high-level waste; not spent fuel; and not mill tailings.	<100 nCi/g transuranium alpha-emitters (half-life >20 years), Field Elements can further limit alpha activity.	Near-surface disposal.
United Kingdom	Low-level waste - <4 GBq/MT (108 nCi/g) alpha-emitting and <12 GBq/MT (324 nCi/g) beta-gamma.	<108 nCi/g alpha-emitters	Presently near-surface disposal; with deep geologic disposal planned in the future.
	Very low-level waste - <0.0004 GBq/MT (10.8 pCi/g) alpha-emitting and <0.02 GBq/MT (540 pCi/g) beta-gamma.	<10 pCi/g alpha-emitters	Land-disposed with ordinary domestic wastes.
France	Category A - Long-lived alpha-emitters with half-lives >31 years (as calculated 300 years after time of acceptance) <100 nCi/g (0.1 Ci/MT) per container; <10 nCi/g (0.01 Ci/MT) average per container; 100-500 nCi/g per container are accepted on a limited exception basis.	<100 nCi/g alpha-emitters per container <10 nCi/g alpha-emitters average	Near-surface disposal.
Sweden	IAEA recommended waste classifications are used. (See IAEA)	<108 nCi/g alpha-emitters per container <11 nCi/g alpha-emitters average.	Crystalline rock, 60 meters under the Baltic Sea floor.
Canada	Low-level waste - Not high-level waste, spent fuel, or mill tailings. Wastes that require isolation for up to 500 years. (Quantitative parameters for this requirement are in development and not yet well defined - see Section 3 discussion).	Not specified. TBD by Safety Analysis Report.	Near-surface disposal.
	Releasable waste - Decided on a case-by-case basis based on a de minimis dose to individuals of 0.05 mSv/yr (5 mrem/yr).	Based on calculation.	Near-surface disposal.
IAEA	Short-lived low-level and intermediate level wastes[b] - alpha-emitting radionuclides <108 nCi/g (4,000 Bq/g) per container; <11 nCi/g (400 Bq/g) average in the disposal facility; thermal power <2 kW/m^3.	<108 nCi/g alpha-emitters per container. <10.8 nCi/g alpha-emitters average.	Near-surface or geological disposal facility.
	Exempt wastes - Activity levels at or below clearance levels, which are based on an annual dose to the public <0.01 mSv (1 mrem). Suggested radionuclide-specific clearance levels are proposed and are currently issued for comment (Reference 4).	Calculated based on allowable exposure.	No radiological restrictions.

a Each disposal site may further limit the LLW characteristics allowable for disposal in their waste acceptance criteria, based on the facility performance assessment documentation.

b Under the proposed IAEA classification system, low-level waste and intermediate level wastes have the same radioactivity concentration limits but are distinguished by the shielding requirements typically imposed when handling the wastes. Intermediate level waste typically requires shielding during handling activities, whereas low-level waste does not. A contact dose rate of 2 mSv/hr (200 mrem/hr) is typically used to as the quantitative limit to separate the two categories of waste.

Table 4. Alpha-emitting radionuclides with half-lives greater than 20 years.

Radionuclide	Isotope	Transuranium	Half life (yrs)
Americium	Am-241	Yes	4.32e+02
Americium	Am-242m	Yes	1.52e+02
Americium	Am-243	Yes	7.38e+03
Californium	Cf-249	Yes	3.51e+02
Californium	Cf-251	Yes	9.00e+02
Curium	Cm-243	Yes	2.85e+01
Curium	Cm-245	Yes	8.50e+03
Curium	Cm-246	Yes	4.75e+03
Curium	Cm-247	Yes	1.56e+07
Curium	Cm-248	Yes	3.39e+05
Neptunium	Np-237	Yes	2.14e+06
Plutonium	Pu-238	Yes	8.78e+01
Plutonium	Pu-239	Yes	2.41e+04
Plutonium	Pu-240	Yes	6.54e+03
Plutonium	Pu-242	Yes	3.76e+05
Plutonium	Pu-244	Yes	8.26e+07
Neodymium	Nd-144	No	2.10e+15
Protactinium	Pa-231	No	3.28e+04
Radium	Ra-226	No	1.60e+03
Samarium	Sm-147	No	6.90e+09
Samarium	Sm-148	No	7.00e+15
Thorium	Th-229	No	7.34e+03
Thorium	Th-230	No	7.70e+04
Thorium	Th-232	No	1.40e+10
Uranium	U-232	No	7.20e+01
Uranium	U-233	No	1.59e+05
Uranium	U-234	No	2.45e+05
Uranium	U-235	No	7.04e+08
Uranium	U-236	No	3.42e+06
Uranium	U-238	No	4.47e+09

issues, including defining more detailed waste classification limits, are currently being reviewed and resolved by the Canadian authorities. More definitive limits will be established in the facility safety documentation.

Comparison of the above quantitative values should not be taken strictly at face value since some programs implement the limits more conservatively than others. The most notable differences in practices are best illustrated by comparing certain practices of France and DOE. Requirements found in French LLW disposal facility waste acceptance criteria allow the following practices in meeting the concentration limit:[5]

> The mass of the waste container and any solidification agent is included in calculating alpha concentrations.
>
> Alpha concentrations are determined as projected 300 years after the time of acceptance, not at the time of disposal.
>
> Higher alpha concentrations, up to 500 nCi/g, are allowed on a limited basis by special permission by ANDRA and the safety authorities.

In comparison, the U.S. DOE does the following:

> The mass of the waste matrix (excluding the disposal container) is used to calculate alpha-emitter concentrations for purposes of determining if the waste is LLW or transuranic waste. Varying practices are prescribed in site-specific facility waste acceptance criteria concerning how the weight of the waste container is used in calculating radionuclide concentration for purposes of disposal.
>
> Alpha concentrations at the time of initial characterization are used, rather than a projected value to a date in the future.
>
> Exceptions are not allowed to the maximum concentration limit for alpha-emitters in LLW. Those wastes which exceed this limit are classified as transuranic waste.

These practices can make an appreciable difference in allowable waste that may be called LLW. For example, allowable concentrations for Pu-238, which has an 86 year half-life, could be approximately a factor of 10 higher under the French compared to DOE requirement for reporting radionuclide concentrations.

Another difference in classification of LLW is the criteria used by the various programs and the IAEA to generically limit <u>beta-gamma</u> activity in the waste. The following differences can be noted from Table 3:

> The DOE, France, and Canada define this beta-gamma limit by exclusion. That is, if a waste is radioactive, but it is not high-level waste, transuranic or greater-than-Class C waste (applicable to the U.S. DOE only), uranium mill tailings, or spent fuel, then it is classified as LLW.
>
> The UK specifies quantitative activity concentration limits for beta-gamma activity in its LLW classification scheme.

Sweden uses the IAEA system, which defines a maximum heat generation criteria (2 kW/m^3) for the waste, which is related to the beta-gamma activity.

The UK and Sweden have defined generic quantitative limits for beta-gamma activity in their LLW as part of their waste classification schemes. Waste disposal facilities sometimes further limit beta-gamma emitters, on an isotope specific basis, in their waste acceptance criteria, based on the analytical results of the site-specific performance assessment. The IAEA classification guideline, which is not mandatory for any country to use, suggests no quantitative limits for beta-gamma activity but does note that limits on some radionuclides may be imposed on a disposal facility, site-specific basis. As already noted, the heat generation criteria suggested by the guideline is related to the beta-gamma activity. Variations in allowable beta-gamma activity may affect the shorter-term waste handling considerations since these isotopes are largely the source of the penetrating radiation that drives remote handling and shielding considerations while handling the waste containers during storage and disposal operations. Long-lived beta-gamma emitters, such as C-14, I-129, Tc-99, Ni-59, and Nb-94, are also important considerations in the long-term institutional control considerations of the disposal facility.

Compared to the other nations studied, Canada s LLW waste program is still in its infancy. Up to now, Canada has been storing all of its radioactive wastes and is now in the process of developing its first near-surface LLW disposal demonstration facility. Canada's disposal efforts are still in the demonstration phases and acceptable waste loadings are established as a part of the facility final safety assessment report.[6] This prototype demonstration facility is scheduled to begin operation in the 1998-99 time frame. The country will likely establish quantitative disposal limits, applicable to all LLW, as the disposal program matures.

Table 3 also identifies that the UK, Sweden, and Canada have defined criteria to allow non-licensed disposal of radioactive wastes that have sufficiently low-levels of radioactivity.[7,8] These countries have defined that these "exempted" or "very low-level wastes" have limited health risk to the public and can be disposed with little or no radiological restrictions. Britain designates very low-level wastes as having an alpha content below 0.0004 GBq/MT (10 pCi/g) and a beta/gamma content of less than 0.02 GBq/MT (540 pCi/g).[7] Canada approves exempted wastes on a case-by-case basis using a maximum allowable de minimus radiation dose rate to individuals of 0.05 mSv/yr (5 mrem/yr) and provided that the radiological impact will be localized and the potential for exposures to large populations is small.[9] Sweden subscribes to the proposed IAEA exempt waste criteria which defines maximum radionuclide activity clearance levels which are based on a dose rate to members of the public of less than 0.01 mSv/yr (1 mrem/yr).[3,4]

France has not established criteria for exempt wastes. DOE does not have generic release criteria for wastes with volume contamination (such as activated material or smelted contaminated metals), but does allow release of such materials if criteria and survey technics are approved by EH-1.[10,11] Before this released material can be disposed in a DOE or non-DOE landfill, it must meet the acceptance criteria of that facility.

4. DISPOSAL FACILITIES DESIGN

4.1 Design Considerations

All programs studied recognize that the basic objective of the siting process is to select a suitable site for disposal and to demonstrate that the site has characteristics that, when combined with the facility design and waste package, provide adequate isolation of radionuclides from the biosphere for desired periods of time. All countries acknowledge that successful site selection involves many factors, not all of

which are technical. Public opinion and receptiveness to the proposed site is a key factor that has, at times, resulted in the abandonment of technically acceptable disposal sites.[12,13] Proximity to large populations and to facilities generating LLW are other key factors. Climate and surface hydrology are also primary considerations in site selection since water is a primary radionuclide transport mechanism. Canada utilizes a voluntary siting process to minimize the public outcry upon site selection. This is the result of opposition encountered when siting was done entirely on a technical basis.

DOE has developed each of its disposal sites on a facility-specific basis as done by individual countries; however, all DOE sites use common performance objectives established in DOE Order 5820.2A. DOE disposal facility locations are constrained to the boundaries of the DOE reservations. Although this is a constraint, it should be recognized that many of the sites were originally chosen with emphasis on favorable characteristics for nuclear activities.

Excluding the disposal program of the U.S. DOE, each national disposal program included in this study currently has only one major LLW disposal facility. For this reason, sites have been chosen and developed on a case-by-case basis by each country, with the common design goal of avoiding corrective actions during the facility operation period and after closure of the facilities. All countries except Sweden currently use near-surface disposal facilities for LLW. Sweden uses intermediate-depth disposal for its LLW, primarily due to the imposition of a requirement that post-closure performance should not be dependent on control or corrective actions.[14] Institutional control requirements for this facility are presented in Section 5.

The UK is currently planning to use deep geologic disposal for LLW when the existing near-surface disposal facility capacity is reached. This decision dates to 1987 when a House of Commons Select Committee recommended that all intermediate level waste (ILW) should be disposed in a deep geologic repository and that the same facility should be extended to take LLW also.[15,16] The Committee believed that putting all ILW in a deep repository would result in a gain in public acceptability. The Secretary of State was advised that a near-surface disposal facility for LLW alone would be uneconomic. The recommendation was accepted, while acknowledging that there is no technical requirement to dispose of short-lived wastes in a deep geologic repository.[13]

Approval of a LLW disposal facility, in any country, involves proving that the disposal system will perform acceptably. Various barriers are used to assure that radionuclide migration is controlled to acceptable levels. These are:

Site geology

Waste form

Engineered structures (e.g., vaults and tumuli)

Engineered surface barriers (e.g., surface caps).

Many factors such as climate, geologic makeup, depth to the water table, and surface water conditions are used in evaluating the need for engineered structure, surface barrier, and waste form requirements. Demographic, economic, socio-political, and institutional factors also play a significant role in defining acceptable disposal solutions in the more populated areas found in Europe and in some areas of the United States.

In the early years of LLW disposal, the site geology was generally regarded as the primary barrier for both short- and long-term facility performance. Gradually, as some disposal sites have experienced evidence of radionuclide migration through the geologic barriers, more emphasis has been placed by the European and some DOE facilities (Savannah River and Oak Ridge) on waste form and engineered structures and barriers as the primary migration deterrents during the earlier phases of the facility existence. Geological barriers assume a more important role as the engineered structures and barriers degrade with time and when human maintenance is no longer provided to inspect and care for barriers such as drainage collection systems. The long-term performance of the facility must rely on the site geology as a primary migration barrier.

Water is normally the primary vehicle supporting radionuclide migration at a LLW disposal site. Table 5 shows the average precipitation for the LLW disposal sites included in this study. As can be seen from the table, the humid disposal sites are those utilizing engineered structures (e.g., vaults and tumuli) in their facility designs. This is primarily due to the increased amounts of water available as a radionuclide migration transport medium. The arid DOE sites do not utilize engineered structures and instead rely on surface barriers (such as covers and caps) as the primary engineered deterrents to keep precipitation from reaching the waste.

Table 5. Climate and precipitation at LLW disposal facilities.

Program	Disposal Facility	Engineered Structure Feature	Average Annual Precipitation (inches)	Climate
United Kingdom	Drigg Site	Concrete Vault	40-42	Humid
France	Centre de l'Aube	Concrete Vault	27.6-33.5	Humid
Sweden	Swedish Final Repository (SFR)	Rock Cavern	18-20	Humid
Canada	Intrusion Resistant Underground Structure (IRUS)	Concrete Vault	20-30	Humid
U.S. DOE	Hanford	None	6.3	Arid
	Idaho National Engineering Laboratory	None	8.7	Arid
	Nevada Test Site	None	4.9	Arid
	Los Alamos National Laboratory	None	13.2	Arid
	Oak Ridge Reservation	Concrete Tumulus	54	Humid
	Savannah River Site	Concrete Vault	48.8	Humid

Each country s program and each facility approaches the design of its disposal system differently, with different emphases placed on site geology, waste form, and engineered structures and surface barriers. The facility technical evaluation (performance assessment) is the basis by which each facility justifies that its system design will prevent radionuclide migration sufficiently to prevent public radiation exposure beyond allowable limits.

4.2 Disposal Practices

Table 6 lists the current LLW disposal site designs being used by the programs included in this study. Designs include near-surface disposal with engineered structures and/or surface barriers, intermediate-depth geologic disposal, and deep geologic disposal. Although none of the countries is currently using deep geologic disposal for LLW, the UK plans to use it in the future when existing near-surface facility capacity is exhausted. The rationale for this decision was discussed earlier in Section 4.1.

The DOE is the only entity studied that currently has multiple LLW disposal sites. Canada may have multiple sites at some future date since its waste generators are responsible for disposal of their own wastes. All other foreign countries studied currently have no intention of developing multiple sites.

As already noted, those DOE facilities located in the more humid areas of the United States (Savannah River and Oak Ridge) use engineered structures (such as vaults and tumuli), as well as surface barriers, to assure that the facility will meet the performance objectives established in DOE Order 5820.2A. The more arid DOE disposal sites (Nevada, Los Alamos, Hanford, and Idaho) do not use engineered structures in their facility design, but do use surface barriers such as water-repellent layers (clay, asphalt, concrete), capillary barriers (hydraulic breaks), and rock layers (riprap, gravel) to minimize water contact with the wastes. Fewer surface barriers are sometimes used at the same DOE disposal site for low activity wastes than for high activity waste due to the limited radionuclide migration potential of the lower activity waste.

Sweden has elected to use intermediate-depth geologic disposal for its LLW. This crystalline-rock facility has been in operation since 1983 and is located 60 m below the Baltic Sea floor. Sweden is relying on the geology of its intermediate-depth, crystalline rock cavern repository as the primary migration barrier, although it does use waste form requirements and engineered barriers to augment the geological isolation.

The French l'Aube disposal facility relies heavily on waste form, the engineered concrete vault design, and its liquid collection system to ensure that radionuclides do not reach the biosphere during the institutional control period. If the French design performs as expected, the geology will play no role, through the institutional control period, in preventing radionuclide migration. As already discussed, the long-term performance (post-institutional control) of the facility reverts to primary reliance on the geologic barriers of the facility.

France began disposing of its wastes in 1969 at the La Manche disposal facility in shallow unlined disposal trenches over a layer of gravel in the bottom, backfilled with soil, covered with a plastic sheet, and topped with another layer of soil. Regional monitoring revealed migration of radionuclides from the La Manche site to a nearby water stream. As a result, the facility was redesigned to employ engineered structures and more stringent safety criteria. The La Manche site was chosen largely for reasons of convenience and the geology of the site is not deemed to be ideal for near-surface LLW disposal. The site is now closed and replaced by the l'Aube facility.

Table 6. Disposal facility designs.

Program	Disposal facility	Current disposal method
United Kingdom	Drigg Site	Near-surface concrete vaults (since 1988).
France	Centre de l'Aube	Near-surface concrete vaults.
Sweden	Swedish Final Repository (SFR)	Intermediate-depth crystalline rock cavern.
Canada	Intrusion Resistant Underground Structure (IRUS)	Near-surface concrete vaults.
U.S. DOE	Hanford Low-Level Burial Grounds	Near-surface V-trenches and wide-bottom trenches
	Idaho National Engineering Laboratory	Near-surface pits, trenches, soil vaults.
	Nevada Test Site	
	Area 3	Near-surface disposal in subsidence craters from underground nuclear tests.
	Area 5	Near-surface pits, trenches, boreholes.
	Los Alamos National Laboratory	
	MDA G	Near-surface pits and 20-m deep disposal shafts.
	Oak Ridge Reservation	
	Solid Waste Storage Area 6	Above-grade tumulus.
	Savannah River Site	
	Saltstone	Grout in above-grade vaults (covered with soil, clay, and gravel earthen cap).
	E-Area Vault	Above-grade concrete vaults (covered with soil, clay, and gravel earthen cap).

Britain used trench disposal in a largely clay medium at Drigg until 1988. While maintaining that the risk assessment assured that this method of land disposal was radiologically acceptable, in 1987 BNFL announced a program to improve disposal practices and enhance the visual impact and perception of the Drigg site.[7,17] Trench disposal at the Drigg facility was phased out in preference to more engineered disposal placing containerized, conditioned wastes in concrete vaults.

4.3 Engineered Structures, Surface Barriers, and Waste Form Requirements

Engineered structures, surface barriers, and waste conditioning requirements are increasingly being used by all the disposal programs to help ensure that radionuclide migration is maintained at acceptable levels and to minimize the need for active maintenance of the facility. The decision to impose waste form conditioning requirements and/or engineered structures and surface barriers acknowledges that either greater confinement or additional facility safety associated with redundant migration barriers is desired beyond that afforded by the geology alone.

Tables 7 and 8 briefly summarize the waste conditioning requirements and engineered structures and surface barriers in place at the various disposal facilities included in this study. The table shows that all disposal programs have waste conditioning requirements, although some are more rigorous than others. The waste conditioning requirements for DOE wastes include both generic waste form requirements established in DOE Order 5820.2A and requirements established in the facility waste acceptance criteria, based on site-specific performance assessments. The tables also show that a variety of engineered structures and barriers are used in the programs. Vaults are the most commonly used engineered structure to prevent radionuclide migration, and all facilities plan to employ a cap of some type to limit precipitation infiltration into the wastes.

Volume reduction and physical and chemical stabilization techniques are used to some degree by all the disposal programs. France, Sweden, Canada, and the U.S. DOE use incineration as a treatment option. The UK uses high-force compaction and grouting within the final containers as its primary means of volume reduction and stabilization.[17] Bitumen waste forms are also used extensively in France, Sweden, and Canada, but not commonly by the DOE.

As noted in Section 4.1, all of the programs studied, except the U.S. DOE, currently have only one LLW disposal facility. Thus, the waste conditioning and engineered barrier requirements shown in Table 7 were developed specifically for the single national site, not as a generic national requirement for LLW disposal facilities. As with those of the countries studied, the DOE sites have developed different approaches and designs in accordance with the site-specific geological, hydrological, climatic, and demographic conditions. Some of the DOE sites located in the more arid regions of the United States have determined that the national performance objectives (in DOE Order 5820.2A) for LLW disposal can be met without employing the use of engineered structures. These sites utilize various combinations of reliance on site geology, engineered surface barriers (e.g., earthen cap), and waste form stability requirements. Other DOE sites, in the more humid climates, require waste form stabilization (e.g., grout and high-integrity containers), engineered structures (concrete vaults or tumuli), and engineered surface barriers (such as caps to keep precipitation from the waste).

Table 7. Disposal facility waste conditioning, engineered structure, and surface barrier requirements.

Disposal facility	Waste conditioning	Engineered structures and surface barriers
Drigg Site (UK)	As far as reasonably practicable, waste forms must be insoluble in water and not readily flammable. The UK has two major treatment facilities, the Waste Monitoring and Compaction Facility (WAMAC) and the Drigg Grouting Facility. Volume reduction by high-force compaction is performed at the WAMAC Facility and the compacted containers are filled with grout at the Drigg Facility to fill internal voids.	Concrete vault covered with a water-resistant cap and a soil layer planted with a vegetative cover. Engineered clay base beneath the concrete floor slab.
Centre de l'Aube (France)	Waste must be physically stabilized and radionuclides immobilized for specified concentration thresholds. Waste forms must pass strict tests for physical and chemical stability before the waste form will be accepted for disposal. Generators choose from approved treatment methods including incineration, bitumenation, cementation, polymerized resins, etc. Waste must be in ANDRA-approved containers.	Concrete vault covered with a concrete cap, sealed with a polyurethane and multi-layer cap (clay, bitumen, soil, and a vegetative cover). Space between waste containers is filled with grout or gravel (dependent upon the waste-specific activity). Each vault has a drain system which routes any liquids from the vault to a collection tank. The drain system is located in a concrete tunnel which provides access for inspection and repair.
Swedish Final Repository (SFR)	Each type of waste package must be approved by the Nuclear Power Inspectorate (SKI) and National Institute of Radiation Protection (SSI). Ion-exchange resins are solidified with cement or bitumen. Other processing includes incineration, melting, decontamination, and super-compaction to reduce volumes. Waste must be in solid form; have good chemical, thermal, and mechanical stability; have good immobilization properties; and have a low leach rate. The waste container must be grouted inside steel or concrete drums or boxes.	Crystalline host rock of under-sea caverns, fitted with concrete-walled cells. Each filled cell is backfilled with concrete grout. High-activity waste is disposed in a special cell containing a concrete silo-shaped cavern equipped with internal walls to divide the silo into square shafts. The silo is built on a bed of sand/bentonite (90/10 percent) and the space between the silo wall and the rock is filled with pure bentonite. Once emplaced, waste is surrounded with grout.
Intrusion Resistant Underground Structure (Canada)	Waste is characterized and processed at the Chalk River Waste Treatment Center before disposal. Typical treatment includes incineration, compaction, and solidification. Most drums contain a bitumen waste form produced from liquid-solidification or ash immobilization.	An underground concrete vault with a permeable floor consisting of a 0.3 m thick layer of sand (90%) and clinoptilolite (10%) and a 0.3 m thick layer of sand (90%) and Dochart clay (10%). The clinoptilolite and clay have the capacity to sorb nuclides. The vault is covered by a 1 m thick concrete cap and 1.5 m of sand and soil with a vegetative cover.
U.S. DOE (See Table 8 for specifics of U.S. DOE facilities.)	Requirements are established on a site-specific basis from analytical results of the disposal site performance assessment.	Some sites use near-surface disposal techniques with surface barriers. Other sites use concrete vaults or tumuli in conjunction with surface barriers, if deemed necessary by the disposal site performance assessment.

Table 8. DOE disposal facility waste conditioning, engineered structure, and surface barrier requirements.

Disposal facility	Waste conditioning	Engineered structures and surface barriers
Hanford Low-Level Burial Grounds	No free liquids are accepted and void space must be minimized (generally less than 10 percent of the package volume). All Category 3 (higher activity inventory) waste must be stabilized. Waste may be stabilized by enclosing it in a high-integrity container (HIC), by processing into a stable waste form, or may be shown by analysis to be inherently stable. Processed waste must satisfy performance testing criteria of the NRC Technical Position on Waste Form.	Near-surface disposal in V-trenches and wide-bottom trenches. The waste is backfilled with soil and a final cover, designed to limit the infiltration rate to less than 0.5 cm/yr, is applied to the parts of the disposal facility containing Category 3 (higher activity inventory) wastes.[19]
INEL Radioactive Waste Management Complex	No free liquids are accepted and void space must be minimized. Combustible wastes are incinerated and the ash is stabilized in cement. Large metal shapes are cut down and some wastes are compacted for volume reduction.	Near-surface disposal in pits, trenches, and soil vaults. An earthen cover is placed over the waste during the operational period. Upon closure, a thick soil barrier with a vegetative cover will be emplaced over the operational cover, giving a total soil cover of 5 m.[20]
Nevada Test Site Area 3	No free liquids are accepted and void space must be minimized. Containers must meet stacking strength specifications. Fine particulates must be immobilized. Where practical, waste must be crushed, shredded, and configured to promote waste minimization and to provide a more structurally and chemically stable waste form. Chemical stability must be documented.	Near-surface disposal in subsidence craters from underground nuclear tests. Wastes are disposed using conventional landfill techniques where each layer of waste is covered with 1 m of fill before additional wastes are disposed in the pit.
Area 5		Near-surface disposal in pits, trenches, and boreholes. An earthen cover is placed over the waste during the operational period. Upon closure, a final cap (not yet designed) will be emplaced to enhance facility performance.[21]
Los Alamos National Laboratory MDA G	No free liquids are accepted and void space must be minimized. Fine particulates must be immobilized.	Near-surface disposal in pits and 20 m deep disposal shafts. Waste is placed in the pits and shafts in lifts and crushed tuff is placed in void spaces, between the lifts, and on top of the waste. Filled pits are covered with at least 3 ft. of crushed tuff and 4 inches of top soil and planted with native grasses. Shafts are topped with 1 ft. of concrete shaped to promote drainage away from the shaft.[22]
Oak Ridge Reservation Solid Waste Storage Area 6	No free liquids are accepted and void space must be minimized. Non-compactible wastes are segregated and compactible waste is compacted. The waste form must be stable under the presence of moisture, microbial activity, and internal factors such as radiation effects and chemical changes.	Above-grade tumulus uses concrete rectangular vaults filled with waste, annular spaces are filled with concrete, pre-cast concrete lid is placed on the vault and sealed with bitumen. The vault is subsequently loaded and stacked onto a curbed concrete pad and capped with natural materials. Surface drainage channels divert surface runoff away from the pad.[23]

Table 8. (cont.) DOE disposal facility waste conditioning, engineered structure, and surface barrier requirements.

Disposal facility	Waste conditioning	Engineered structures and surface barriers
DOE Savannah River Site E-Area Vault	No free liquids are accepted and void space must be minimized. Waste packages must not contain greater than 15% void volume. Fine particulates must be immobilized.	Above-grade concrete vaults covered with soil, clay, and a gravel/earthen cap with a vegetative cover. The vaults have a concrete cover to divert surface runoff away from the vaults. The floor of the vault slopes to a drain which runs to a collection sump, which is monitored for radionuclides during the operational period of the facility.[24]
Saltstone	Decontaminated salt solution from the In-Tank Precipitation and Effluent Treatment Facilities is treated by a grouting facility and permanently disposed in above-grade vaults.	Above-grade concrete vaults covered with soil, clay, and a gravel/earthen cap. The saltstone is poured into the vault, leaving approximately 0.3 m from the top of the vault wall to be filled with uncontaminated grout. After all cells are filled, a permanent concrete roof is installed. On closure, soil is placed between the vaults and clay/gravel drainage system with earthen and vegetative cover installed to route precipitation.[25]

5. POST-CLOSURE AND INSTITUTIONAL CONTROL CONSIDERATIONS

The planned institutional control period[b], during which public access to the facility will be controlled, are listed in Table 9 for the programs studied. Sweden is notable in the comparison table since it has no planned institutional control period.

Table 9. Plans for institutional control and use of the facility after institutional control.

Program	Facility name	Institutional control period	Use of facility after institutional controls
U.S. DOE	All DOE sites	Minimum 100 years.	Not specified, TBD on a site-specific basis from technical analysis.
United Kingdom	Drigg Site	Up to 300 years, TBD by regulatory bodies.	Not specified.
France	Centre de l'Aube	300 years.	Unrestricted use.
Sweden	Swedish Final Repository (SFR)	0 years.	Unrestricted use.
Canada	Intrusion Resistant Underground Structure (IRUS)	Not specified. TBD prior to final closure.	Not specified, TBD on a site-specific basis from technical analysis.

The Swedish Final Repository (SFR) facility is located 60 meters beneath the Baltic Sea in crystalline bedrock (gneiss and granite) and is about one kilometer offshore from the harbor at Forsmark. The location has a very low hydraulic gradient and thereby the ground water is almost stagnant. Designers consider that there is no risk of a well being drilled so long as the repository is covered by seawater. Due to the land uplift in Sweden (about 6 mm/yr), the sea bottom above the SFR will become dry land in 1,500 to 2,000 years and the hydraulic considerations will change.[18] This time frame is beyond the typical period of concern for implementing institutional controls.

The SFR facility safety assessment calculates the radiation exposure to the most exposed individual to be 0.0001 mSv/yr (0.01 mrem/yr) during the period the SFR is covered by the Baltic Sea and to be 0.01 mSv/yr (1 mrem/yr) for the period thereafter.[26] Some radiological monitoring of the region will be done but the need for active maintenance based on this monitoring is not anticipated or planned.

France states that it plans to allow unrestricted use of its facilities after the planned 300 year institutional control period. France acknowledges that long-lived radionuclides will still be present upon release of the facility to unrestricted use. Calculations in the l'Aube disposal facility performance assessment include postulated intrusion scenarios during the period when site access is no longer restricted.[27] These scenarios, although called off-normal accident scenarios, include road construction,

b. DOE Order 5820.2A defines "institutional control" as a period of time, assumed to be about 100 years, during which human institutions continue to control waste management facilities.

housing construction, and use of a water well, which are plausible events in an unrestricted access area. The reported external exposure to a housing resident on the site is estimated at 260 mrem/yr from external radiation sources and 23 mrem/yr from inhaled radioactive dust during early years of the free access period. The l'Aube performance assessment uses an allowable exposure limit of 0.005 Sv/yr (500 mrem/yr) to the public during the post-institutional control phase of the facility.[27]

The above allowable dose exceeds the current U.S. DOE guidance given in DOE Order 5400.5, which states that doses from unrestricted use of a disposal facility should not exceed the primary dose limit (100 mrem/yr, except for radon isotopes and progeny). Use of the as low as reasonably achievable (ALARA) process will likely further reduce allowable exposures for DOE properties released for unrestricted use. Proposed regulations (10 CFR 834 and 40 CFR 196) reportedly will deal with these limits, and the allowable exposure limits are expected to be near the 15 mrem/yr range.

The DOE, UK, and Canada have not specified any intended or allowable use of their disposal facilities upon relinquishing institutional control. It has been suggested that restrictions on use of these facilities will be imposed indefinitely, although this would constitute at least a form of administrative institutional control. Allowable future use of the facilities will ultimately be dependent on results of technical analysis in the facility performance assessment that is maintained and updated over time.

The DOE and Canada currently have significant amounts of LLW which contain uranium, thorium, and radium. As discussed in Section 6.2, uranium, thorium, and radium wastes will influence institutional control considerations due primarily to the radon gas, a daughter product of these wastes.

France has imposed isotope-specific concentration limits for Ra-226 and Th-232, which are long-lived (half-life greater than 31 years) alpha-emitters, having radon daughters. The l'Aube limit was established primarily on the results of a pathways analysis during the free access period, which occurs after the 300 year institutional control period.[5] No restrictions related to this concern were found in the Drigg waste acceptance criteria.[28]

Radon emission is not a significant problem for the intermediate-depth Swedish disposal facility since radon gases have a short half-life (Rn-222 has a half-life of 3.8 days) and will decay to lead before escaping the geologic barriers of the intermediate-depth (60 m) facility. The considerations are different in a near-surface disposal facility since the gas can escape the ground and be inhaled by a local inhabitant before it decays.

Canada's regulatory policy statement for radioactive wastes recognizes that it has a significant volume of contaminated equipment and debris originating from uranium mining and milling activities (this is waste other than actual mill tailings) that will require long-term institutional control considerations due to the daughter products of long-lived uranium components.[29] A program separate from the IRUS is being developed to handle the large existing inventory of these "historic wastes." Disposal of these wastes is being planned as a separate effort from currently-produced radioactive wastes and is scheduled to be operational about the year 2000.[7]

Canada is conceptually planning to dispose of newly-generated (non-historic), longer-lived wastes containing radionuclides such as uranium, thorium, carbon-14, and plutonium in some form of rock cavern, possibly in conjunction with a nuclear fuel waste disposal facility, rather than in near-surface disposal facilities. Only a small fraction of the newly-generated waste will not qualify for disposal in a disposal facility like the IRUS. This quantity of Canadian wastes is so small as to not require attention at this time. Instead, Canada plans to store these wastes in engineered facilities for the indefinite future.[6,30]

6. PERFORMANCE ASSESSMENT

6.1 Assessment Performance Objectives

All programs included in this study require technical assessment of the radiological performance of the disposal facility before a proposed LLW disposal operation can be approved. Several different types of performance objectives are used by the various programs as acceptable performance parameters for these assessments. As shown in Table 10, these objectives are expressed as follows:

> Statement of acceptable doses to the general public and to hypothetical inadvertent intruders. (U.S. DOE, Sweden)
>
> Statement of acceptable risk of fatal cancers and genetic defects. (Canada, United Kingdom)
>
> Requirement that fundamental intrinsic safety provisions (waste form and engineered barriers) are present, precluding the possibility of significant release during the facility operational and institutional control periods. After the institutional control period, acceptable radiological exposure to the most exposed member of the public for unrestricted access activities is required. (France)

DOE and Swedish designers base disposal facility radiological performance design considerations largely on calculated exposures of an all-pathways analysis for the most exposed individual. In contrast, Canada and the UK base their radiological performance designs on likelihood of fatalities to the general population (e.g., risk of fatal cancer to public $<10^{-6}$ per year). Approaching facility performance from a perspective of risk to the general population, rather than to the most exposed individual, allows likely events affecting large numbers of people to drive the design instead of a postulated event that might affect and involve relatively few people.

France designs the facility to provide radionuclide migration barriers that confine the nuclides so that essentially no exposure is received by the public during the operational and institutional control periods of the facility. The waste form, engineered barriers, and a liquid collection system are designed to prevent any migration from the disposal vault area, so long as human operators maintain the facility.

It should be noted that the French designers are forced to revert to a different protection scenario at the end of the institutional control period. As faced by all disposal programs, the institutional control period allows significant decay reduction of the radioisotopes with relatively short half-lives, but the longer half-lived isotopes are still a potential exposure radiation hazard that must be addressed. After the 300 year institutional control period, human maintenance and monitoring of the disposal system will cease and reliance can no longer be placed on the integrity of the vault or the liquid collection system. France bases long-term exposure estimates on the conservative assumption that all of the man-made engineered structures and barriers have failed and that the natural geology is the primary mechanism to retard radionuclide migration to members of the public. The allowable exposure limit for this disposal area, returned to unrestricted use, is 0.005 Sv/yr (500 mrem/yr) for the public.

A brief discussion of the Canadian program is provided to illustrate the principles used in a risk-based criteria system. This policy and the basis for the policy is detailed in Canada's Regulatory Policy

Table 10. Time period of compliance and criteria for performance assessment.

Program	Facility name	Time of compliance	Objectives for performance assessment
U.S. DOE	Multiple facilities at DOE sites.	Unspecified by DOE 5820.2A; 10,000 years has been used to-date.	25 mrem/yr to most exposed member of the public. 100/500 mrem/yr (chronic/acute) to inadvertent intruder. Atmospheric releases shall meet the requirements of 40 CFR 61. Groundwater resources shall be protected consistent with Federal, State, and local requirements.
United Kingdom	Drigg Site	Unspecified.	Risk of fatal cancer to public $<10^{-6}$ per year.
France	Centre de l'Aube	Unspecified. PA calculations exceed 10^{6} years.	Designed so that it possesses intrinsic safety (effectively no release to the environment) through the institutional control period, based on the reliability of its first two systems of containment. Reliance is placed on geological barriers after the institutional control period with 500 mrem/yr as the maximum allowable exposure to the public.
Sweden	Swedish Final Repository (SFR)	10,000 years	0.1 mSv/yr (10 mrem/year) to public.
Canada	Intrusion Resistant Underground Structure (IRUS)	10,000 years	Risk of fatal cancers and serious genetic effects $<10^{-6}$ per year.

Statement on disposal of radioactive wastes.[29] In this policy, Canada establishes the following general requirement:

> "The predicted radiological risk to individuals from a waste disposal facility shall not exceed 10^{-6} fatal cancers and serious genetic effects in a year, calculated without taking advantage of long-term institutional controls as a safety feature."

The Canadian policy document explains that this level of risk, 1×10^{-6} in a year, was chosen because of other activities that consider this level of individual risk to be insignificant in daily lives. Since the probability of fatal cancers and serious genetic effects is approximately 2×10^{-2} per Sv (2×10^{-4} per rem), the probability of these health effects associated with a dose of 1 mSv (100 mrem) is 2×10^{-5}. (These reported health effect values are from the ICRP Publication 26 and are a factor of approximately 2 less than more recent values given in ICRP Publication 60). To put this in perspective, a risk of 10^{-6} (1 in a million) is the risk associated with a dose of 0.05 mSv (5 mrem) in a year. Where it is reasonable to assume that the probability of the scenario approximates unity, the risk is simply the product of dose and the probability of the health effect per unit dose. On the other hand, if an event is highly unlikely, such as an inadvertent intrusion into an undesirable location, the probability of the event will allow designers to discount the unlikely event and to use a more probable scenario as the basis for designing the facility.

Canada also provides guidance for the case where the above risk criteria cannot be met. If there is no practicable method of fully meeting the above health risk, an optimization study must be performed to determine the preferred option. A disposal facility built under these circumstances shall be: 1) compatible with the results of such a study and 2) such that the predicted risk to individuals does not exceed that which is presently accepted from current operations involving the same wastes.

The approach taken by DOE is briefly discussed to illustrate the principles used in a dose-based criteria system. DOE established the following criteria as the basis for determining acceptable performance of its LLW disposal facilities:

(1) The effective dose equivalent to any member of the public shall not exceed 25 mrem/yr.

(2) Releases to the atmosphere shall meet the requirements of 40 CFR 61.

(3) The committed effective dose equivalent received by individuals who inadvertently intrude into the facility after the loss of active institutional control (100 years) will not exceed 100 mrem/yr for continuous exposure or 500 mrem for a single acute exposure.

(4) The groundwater resources shall be protected consistent with federal, state, and local requirements.

Designers must postulate and analyze potential credible scenarios that may impact the above criteria. In practice, the likelihood of the events are normally determined by an analyst's judgement rather than formal numerical calculation. It is possible for relatively unlikely events or events that affect relatively few people to drive the facility design.

6.2 Time of Compliance

The time of compliance is defined as a specific time period during which the performance of a LLW disposal facility or the disposal system (disposal facility and environmental conditions) must be shown to satisfy the performance objectives. Table 10 (above) shows the time period of compliance for

each of the programs. All the disposal programs recognize that the wastes contain at least a small amount of long-lived radioactive isotopes requiring long-term consideration. Canada and Sweden both have a time period of compliance of 10,000 years. France and the UK do not specify a specific time of compliance, although they are analyzing the long-term performance of their facilities.

The performance objectives in DOE Order 5820.2A do not specify a time period over which they are to be applied. Facility-specific performance assessments written to date have used 10,000 years and recognized the time of peak dose. Times from 1,000 to 10,000 years have been proposed as an official time of compliance but no formal policy has been approved.[31]

As discussed in Section 3, France, Sweden, and the UK limit the concentration of alpha-emitting isotopes in their LLW wastes to approximately 100 nCi/g. DOE imposes the limit that only transuranium alpha-emitting isotopes (with half-lives greater than 20 years) must be at concentrations less than 100 nCi/g, although Field Offices are allowed to specify other alpha-contaminated wastes that must be managed as transuranic waste. This allows the DOE to dispose of wastes having >100 nCi/g uranium, thorium, and other non-transuranium alpha-emitting isotopes (shown in Table 4), assuming that the performance assessment shows acceptable exposures over the time period of compliance. Most DOE sites limit inventories of these wastes through their facility waste acceptance criteria.

The time period of compliance used in the performance assessment is significant when considering the impacts of uranium isotopes that require significant amounts of time to reach equilibrium with their radioactive daughter products. The time required for radium and its decay products (principally Rn-222) to reach equilibrium with initially pure uranium is approximately 10^5 years for U-238 plus U-234 and 10^6 years for U-238 only.[31] Therefore, the radiological exposures from uranium daughters are increasing with time, for hundreds of thousands of years. If a disposal site uses a time period of compliance of 10,000 years or less, the impact of uranium and other long-lived alpha-emitters on exposures to distant-future inhabitants is still increasing at the end of the time period of compliance.

Some believe that this approach is reasonable since projections of impacts that far into the future are too uncertain to be considered realistic or useful for making decisions. Many factors such as proximity to inhabitants and migration and solubility of the wastes will affect the magnitude of the future exposure from the daughter products of long-lived isotopes.

Canada and DOE, in particular, have considerable quantities of these uranium, radium, and thorium wastes associated with their early uranium enrichment and defense activities. Canada has not yet decided what disposal option will be used for its large quantities of historic wastes. Potential solutions include prescribing a flux limit for discharge of radon gas, implementing very long-term institutional control of the disposal facility, and/or burying the waste deeper to prevent the radon gas from escaping the geology before decaying.

7. SUMMARY

This study revealed that notable differences exist between the programs of the U.S. DOE and the other countries concerning how they dispose of LLW. Table 11 briefly summarizes many of the differing practices of the disposal programs included in this study. The reasons for these differences are complex and dependent on many variables both technical and political.

Even though each disposal program studied utilizes a different LLW classification system, it is interesting to note that all programs recognize the need to limit long-lived radionuclides in the disposal facility. The UK, France, and Sweden all limit alpha-emitting radionuclide concentrations to approximately 100 nCi/g. DOE Order 5820.2A limits alpha-emitting transuranium radionuclides (having half-lives greater than 20 years) to concentrations less than 100 nCi/g. The lighter alpha-emitter elements (including uranium, thorium, and radium) are thus allowable unless restricted by a site-specific waste acceptance criteria based on the results of the facility performance assessment. Canada acknowledges that the waste must have a hazardous lifetime of less than 500 years but has not yet put generic quantitative values on the alpha-emitting isotopes.

Various criteria are used to limit allowable beta-gamma activity in waste classified as LLW. The DOE, France and Canada impose classification limits, related to beta-gamma emitting radionuclides, by exclusion. That is, if a waste is radioactive, but it is not high-level waste, transuranic waste or greater-than-Class C (applicable to the U.S. DOE only), uranium mill tailings, or spent fuel, then it is classified as LLW. The UK specifies quantitative activity concentration limits for beta-gamma activity in its LLW classification scheme. Sweden uses the IAEA system, which defines a maximum heat generation criteria (2 kW/m^3) for the waste, which is related to the beta-gamma activity. Waste disposal facilities sometimes further limit both alpha and beta-gamma emitters, on an isotope specific basis, in their waste acceptance criteria based on the analytical results of the site-specific performance assessment.

Significant differences were found when investigating how facilities implement their limits. The French use calculated alpha-emitter concentrations 300 years after time of acceptance, allow up to 500 nCi/g per container (on a limited exception basis), and include the mass of the waste container and any solidification agent when calculating alpha concentrations. In contrast, the U.S. DOE uses alpha concentrations at the time of waste form assay, rather than a projected inventory at some date in the future. DOE does not allow exceptions to their 100 nCi/g limit and uses only the mass of the waste matrix (excluding the disposal container) to calculate alpha concentrations, for purposes of determining if the waste will be classified as LLW or transuranic waste. These practices can make appreciable difference in allowable waste forms that may be called LLW.

Sweden, the UK, and Canada have established accepted definitions of "exempt" or "very low-level" waste which can be disposed in non-licensed facilities with minimal or no disposal constraints. Sweden has adopted the IAEA recommended classification system that establishes acceptable dose consequences to the general public resulting from exempt waste at less than 0.01 mSv/yr (1 mrem/yr). The IAEA recently published suggested radionuclide-specific clearance levels for comment by IAEA member countries.[4]

All programs studied require waste conditioning before land disposal in near-surface disposal facilities. These conditioning requirements vary but universally no sites accept free liquids, and hazardous components must typically be eliminated or stabilized. Waste conditioning requirements vary from one disposal site to another based on the site-specific needs established in the facility performance assessments and on national policy requirements.

Table 11. Practices of national programs for disposal of low-level waste.

Topic	U.S. DOE	France	Sweden	United Kingdom	Canada
Working Definition of Exempt Waste (i.e., below regulatory concern) Is In Use			X	X	X
Waste Conditioning Performed	X	X	X	X	X
Disposal Practices					
Near-surface disposal utilizing an engineered structure (e.g., vaults, tumuli)	X	X		X	X
Near-surface disposal without an engineered structure (e.g., vaults, tumuli)	X				
Intermediate-depth geologic disposal (60 meters below the Baltic Sea floor)			X		
Cap Design					
Solid Concrete	X (Oak Ridge and Savannah River)	X	NA		X
Earthen Cover	X	X	NA	X	X
Active Institutional Controls Planned After Closure	X	X		X	X
Time Period of Compliance	Unspecified 10,000 years is precedence	Unspecified	10,000 years	Unspecified	10,000 years
Performance Criteria					
Based on most exposed individual	X	X (free access period)	X		
Risk-based chance of cancer				X	X

X Indicates that the row topic is applicable to the indicated national program.

All programs studied that conduct near-surface LLW disposal use engineered structures (such as vaults) for LLW disposal. DOE facilities located in the more humid areas of the United States (Savannah River and Oak Ridge) use a similar system. The more arid DOE disposal sites (Nevada, Los Alamos, Hanford, and Idaho) do not have engineered structures in their facility design, but do use surface barriers such as water-repellent layers (clay, asphalt, concrete), capillary barriers (hydraulic breaks), and rock layers (riprap, gravel) to minimize water infiltration.

The variety of combinations of engineered structures and barriers found at the various LLW disposal facilities studied is an indication that the technical performance of each near-surface disposal site is based on many factors including geology, climate, and hydrology. An analysis (performance assessment) of the disposal system is a universally accepted method of judging the technical adequacy of a disposal system.

Demographic, economic, socio-political, and institutional factors also play a significant role in defining acceptable disposal solutions. Design conservatism is often reflected in disposal facility designs to gain public acceptance of the facility.

Unlike the other disposal programs, Sweden has elected to dispose of their LLW in an intermediate-depth disposal facility, located in crystalline rock, approximately 60 meters beneath the Baltic Sea floor. Sweden uses intermediate-depth disposal primarily due to the requirement that active institutional controls and corrective actions after closure of the facility should be minimized or eliminated, if possible.

With the exception of Sweden, all of the countries and DOE utilize some form of engineered earthen covers to shed water and protect waste from the direct weather elements. Concrete caps are utilized in France and Canada, and at the humid DOE sites (Oak Ridge and Savannah River). Sweden has chosen an alternative design approach of intermediate-depth disposal in crystalline rock, 60 m below the Baltic Sea floor. The UK vault design does not use a concrete cap beneath its multiple-layered earthen cover.

Institutional control periods are planned for all the disposal programs studied, with the exception of Sweden, which plans no active institutional control measures. The undersea location of the Swedish disposal facility eliminates the need for physical institutional control measures after the entrance is sealed.

All of the disposal programs studied recognize the need for analysis of long-term periods of performance for waste analysis efforts. Sweden and Canada use 10,000 years as the time period of compliance. DOE, France and the UK do not specify the time period of compliance for their facilities. Existing DOE facility-specific performance assessments written to date have used 10,000 years and generally determine the time of peak dose.

Several approaches to defining criteria for assessment of acceptable disposal facility performance are utilized by the various nations. The U.S. DOE and Sweden specify criteria for the most exposed individual as the criteria to establish performance and design requirements. The UK and Canada have adopted risk-based criteria based upon the likelihood of cancer or genetic defects. France requires features intended to eliminate any significant radionuclide migration from the vault area during the operational and institutional control periods. France places no reliance on the geological barriers until after the institutional control period. At that point, the facility changes to performance criteria of allowable exposure to the most exposed individual.

The various programs studied have established different approaches to LLW disposal. These include near-surface disposal with engineered structures and/or surface barriers, intermediate-depth geologic disposal, and deep geologic disposal. All programs studied utilize varying degrees of a multi-barrier approach to isolating radionuclides from the environment and all implement engineered barriers and/or waste conditioning requirements beyond those of traditional shallow land disposal in trenches. The policies of each nation are influenced by geographic, climatic, demographic, economic, socio-political, and institutional factors.

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Appendix A

Low-Level Waste
Disposal Program of France

Low-Level Waste
Disposal Program of France

1.0 Regulatory Background

The Atomic Energy Commission (CEA) was established in 1945 to control nuclear matters in France. In 1946, the energy-supply industries were nationalized and the CEA was assigned the responsibility for nuclear development in France. The Decree of 1970 permitted the CEA to extend its activities, particularly fuel-cycle support, to industrial firms. This began France's current policy of relying on industrial firms to conduct fuel-cycle operations, as well as bear the associated costs. Implementation of the nation s nuclear policy is now left to industrial firms, which are heavily subsidized by the federal government.

The federal government provides regulatory oversight, including licensing, of all nuclear activities. The nation's nuclear power program strategy/policy is to develop a domestic capability in all aspects of the nuclear fuel-cycle except the supply of uranium. Uranium is mined domestically at six sites, however, the majority of uranium used by France is imported.[1] Management of radioactive waste, including transportation and disposal, is the responsibility of the National Radioactive Waste Management Agency (ANDRA), a federal agency set up by an interministerial order in 1979.

Near-surface engineered facilities are considered adequate for disposal of short-lived radioactive wastes, while geologic disposal is considered necessary for longer-lived radioactive wastes. National policy on LLW disposal requires:

Co-mingling commercial and defense radioactive wastes at planned disposal facilities.

The waste generator to pay associated waste conditioning, storage, transportation and disposal costs.

The federal agency, ANDRA, to subcontract transportation and disposal activities to industrial firms, while allocating costs back to the waste producers.

ANDRA is responsible for recovering the costs associated with transportation and disposal operations from the waste producers.

2.0 Approval Requirements

Licenses are required for construction and operation of major nuclear installations, except those classified as secret by the Prime Minister.

The major licensing requirement (Licensing Decree) was initiated in 1963. The licensing decree fixes the perimeter of the installation, the conditions imposed upon the operator, and details of the commissioning procedure. Licensing applicants must submit a license application to the Central Service for the Safety of Nuclear Installations (SCSIN). This agency solicits reviews from appropriate groups of the federal government and at the local level. The federal government review is conducted by the Permanent Group, with assistance from the Institute for Protection and Nuclear Safety, which is part of the CEA.

Once the reports from the Permanent Group, the local level, and appropriate Ministers have been received, the SCSIN prepares a draft licensing decree that is sent to the Interministerial Committee for Major Nuclear Installations and to the Secretary for Health for approval. If the approval is not given by the Secretary for Health within three months, the decree may be approved by the Council of Ministers. Final approval by the French Prime Minister is required after these agencies have approved the licensing decree. Final commissioning is contingent upon approval by the Minster of Industry of the final safety report and general rules. The licensing decree also includes plans and a schedule for decommissioning the facility.

Statutes enacted in 1985 expanded the role of the public in the licensing of nuclear installations. Construction authorization is subject to a "Public Inquiry" which entails making all information concerning the proposed project available to the public. A Commissioner of Inquiry is appointed to interact with the public during the inquiry. The commissioner is responsible for notifying the public within 10 km of the site of a "Public Inquiry" using newspaper announcements, official public notices, etc. During the inquiry, all interested parties are allowed to examine the license application and may record and submit written comments.

An "Inquiry Commission" is appointed by the local administrative court to evaluate the comments and prepare a report, that is circulated to the public and appropriate governmental agencies. After circulation of this report, communities within a 5 km radius of the proposed project are allowed to vote for or against the project. A negative vote can be overruled by the national Parliament.

The Ministry of Industry has jurisdiction and regulatory authority over all nuclear activities. The SCSIN, a department of this Ministry, develops and enforces safety regulations, and issues construction permits and operating licenses. The Ministry of Health has primary jurisdiction over protection of public health, including the effects of radioactivity in the environment. A department of this Ministry, the Central Protection Service Against Ionizing Radiation (SCPRI), monitors radioactivity in the environment, monitors and controls radioactive effluents, and issues permits for radioactive releases.

3.0 Siting

ANDRA is responsible for finding and evaluating LLW disposal sites. The Ministry of Industry is responsible for choosing a site from the list and final approval of the site selection is approved by the Prime Minister.

The l'Aube facility was sited based on geologic and hydrogeological requirements[1] that include:

a) Well-known system hydraulic limits unchanging with time.

b) Lithologically homogeneous medium, easy to characterize and model, with good radionuclide retention capacity.

c) Medium to low regional seismicity.

d) Site protected against landslides, rockfalls, subsidence, and erosion.

e) Good topography, with limited water flowing across the site, water runoff downhill, protected from floods, and near a river or the sea or other water source with a high dilution capacity.

4.0 Past, Existing, and Planned Disposal Facilities

Beginning in 1967, France, along with several other European countries, disposed of LLW by dumping it into the north Atlantic ocean. This practice was discontinued during 1969 in favor of the lower-cost near-surface disposal initiated at La Manche.

4.1 Centre de La Manche Disposal Facility

The Centre de La Manche disposal facility is located at the tip of a peninsula in Normandy, 25 km west of Cherbourg near the English Channel and east of the La Hague reprocessing plant. The Centre de La Manche is a sparsely-populated region with only 1,500 inhabitants within a radius of 10 km of the site. Fishing, agriculture, and cattle ranching are the primary industries of the area. The climate is temperate with an annual rainfall of approximately 100 cm/year. The site was developed and sited largely for reasons of convenience, since it is close to the La Hague reprocessing plant. The geology of the site is complex and is not deemed to be ideal.

The site began operations in 1969 and has a capacity of 500,000 m^3. The site capacity was reached in 1995 and site closure operations began in 1991. The original disposal concept at La Manche, when the site was opened, was to dispose of LLW in shallow unlined disposal trenches. These trenches were plain earth with a layer of gravel placed at the bottom of the trench. LLW disposed in the trenches was required to be solid but was not necessarily stabilized. Waste was disposed in various forms including metal drums or boxes and some plastic bags. The trenches were backfilled with soil, covered with a plastic sheet, and topped with another layer of soil. Approximately 700 m^3 of LLW were disposed of by this method. Intermediate-level waste (waste normally requiring remote handling) was disposed in trenches modified by the addition of concrete vaults made of prefabricated slabs joined in place, backfilled with sand or concrete, and covered with a concrete cap. Four trenches of this type were filled at La Manche.[1]

In 1976, regional monitoring revealed leaching of radionuclides from the La Manche site to a nearby stream. In response, the facility was redesigned to employ engineered structures and more stringent safety criteria. The new design required short-lived wastes with relatively high external radiation levels to be disposed in rectangular concrete pits buried a few meters underground. The pit foundation was prepared by excavating a large trench lining the bottom with a concrete floor having a drainage channel. Layers of waste packages were placed on the concrete, with the more radioactive wastes towards the bottom and middle. These wastes were then completely encapsulated by backfilling with concrete to form a monolith, that was completed by pouring a concrete slab on the upper layer of the trench. Lower-activity LLW containers were stacked approximately 6 m above the monolith and all spaces were filled with gravel. The waste was then covered with a layer of impervious clay and topsoil and seeded with grass. The final result was a grassy mound about 10 m high, commonly referred to as the "tumulus" concept.[1]

4.2 Centre de l'Aube Disposal Facility

The French Centre de l'Aube near-surface LLW disposal site began operation in January 1992, and is located in Soulaines-Dhuys, near the border of the Aube and Haute Marne departments, about 40 km east of the city of Troyes. The 95-hectare site is situated among three small villages in a rural area with a declining and aging population structure. The site is connected to the public road system by a 4 km access road. A dedicated rail terminal for waste shipments is located in the nearby town of Brienne-le-Chateau.

The general area is forested and slightly elevated. The site is located on a hill with a sand formation on top of a 30 m deep clay formation. The clay formation prevents surface water from reaching the subterranean water table, while the sand filters rainwater from the site and drains it into a nearby stream that is monitored. The disposal site is in the southern Champagne region, known to have stable geology with very little seismic activity.

The design of the site draws on the operating experience of the Centre de la Manche site. The site matches the simple hydrogeological model established for site selection, and was chosen following a site investigation program conducted in 1984 and 1985 in several areas of the Aube, Indre, and Vienne departments and in the town of Cholet. The geology and hydrology of the area around Soulaines was studied intensively, with geophysical surveys of the land and approximately 500 boreholes drilled at varying depths.

The site has a capacity of 1,000,000 m^3 and is expected to accommodate the LLW from French sources for at least the next 40 years. Waste packages are placed in engineered disposal structures resembling concrete vaults.[1] These vaults are 25 m square and about 8.5 m high with walls and floors of 30 cm thick reinforced concrete. Each vault rests on another concrete pad which is above the groundwater level. A water collection system beneath the disposal structures collects any water from around or inside the vaults for monitoring.

The vaults are covered by movable buildings during loading operations. After the waste has been loaded in the disposal unit, the structure is stabilized by filling the voids with concrete or gravel (depending upon the activity level of the drums). The filled disposal structure is covered with a concrete slab and sealed with a polyurethane coating. As disposal units are filled and sealed, the spaces between units are filled with earth. A final cap is placed on top of the disposal structures, which is made up of multiple layers of clay, bitumen, and seeded top soil. The completed disposal facility will look like gently sloping hills covered with vegetation.

The facility uses mobile hangars to shelter the disposal structures and delivery trucks during operations. With no rain contacting the waste packages, rainwater runoff does not require elaborate monitoring systems. The hangars contain an automated overhead handling system with a waste package tracking system utilizing scanners to read the container bar code labels.

Galleries of piping in a concrete trench underlie the disposal vaults. This piping network collects rainwater from empty disposal structures and any infiltrated water from full disposal structures. The concrete trench allows inspection and maintenance of the piping. Any leaks into the disposal cap, signaled by the presence of infiltration water, can be pinpointed and repaired.

5.0 Design Considerations

The primary criteria used for the design of the l'Aube disposal facility are[1,2,3]:

(1) Radionuclides must be kept isolated from the biosphere until they have decayed sufficiently and the residual potential risk can be considered negligible. For LLW Category A wastes, an isolation period of 300 years is considered adequate.

(2) A multi-barrier system coupled with a drainage collection and treatment system will be relied upon to prevent release of radionuclides into the biosphere by infiltrating waters.

(3) ANDRA will maintain institutional control of the disposal facility for 300 years to prevent release of radionuclides into the biosphere through human intrusion.

These performance criteria have resulted in three primary waste acceptance criteria[2]:

(1) Physically stabilize the waste form.

(2) Contain or immobilize the radionuclides in the waste.

(3) Limit the specific activity of short-lived (0.5-6 year half-life) and long-lived (greater than 30 year half-life) radionuclides in the waste.

The LLW generators sending LLW for disposal are required to meet a waste acceptance process that ensures that the waste form meets the first two criteria (stabilization and immobilization). The third criterion resulted in the development of maximum values for specific activities of the waste that may be disposed. The limit for each radionuclide is different and depends on the radio-toxicity of each radionuclide. Pathways analysis for the unrestricted access period resulted in the specification of a maximum activity of long-lived alpha-emitters to less than 0.1 Ci/MT (100 nCi/g) for individual packages and to less than 0.01 Ci/MT (10 nCi/g) for the average of all waste packages.[2] Packages with 0.1-0.5 Ci/MT (100-500 nCi/g) are accepted on a limited exception basis.

Near-surface disposal facilities are designed based on the triple barrier disposal concept, comprised of: 1) grout + waste + container (first barrier); 2) the engineered waste disposal facility (second barrier); and 3) the site itself (third barrier).[1,3]

6.0 Closure/Post-Closure

Site closure operations began at the La Manche facility in 1991. Closure involves the inspection and reinforcement of monitoring galleries (concrete trenches) and drainage systems built under the disposal structures, and the construction of the final disposal cap consisting of multiple layers of natural materials, from bottom to top, including:

A base layer of compacted local schist
A drainage layer of sand
A bitumen membrane
A drainage layer of sand
A layer of loose gravel
A layer of planted topsoil.

France has defined the institutional control and monitoring period as 300 years, for the currently operating l'Aube disposal facility, during which the waste activity naturally decays, the integrity of the waste isolation is monitored, and the site is protected from human intrusion.[3] After this period is completed, current French plans state that free access period to the disposal site will be allowed and all restrictions concerning the use of the site will be raised. Residual long-lived radionuclides will still be present during this free access period and the maximum allowable exposure to the public is 0.005 Sv/yr (500 mrem/yr). (See discussion under Section 10, Safety Assessment).

7.0 Waste Classification

France divides its radioactive wastes into three categories based on their characteristics.[1]

Category A - is the equivalent of United States LLW. This category is the only category deemed suitable for near-surface disposal. The content of long-lived alpha-emitters with half-lives greater than 31 years is limited to 100 nCi/g (0.1 Ci/MT) per container with the average alpha-emitter concentration of the containers limited to less than 10 nCi/g (0.01 Ci/MT) average per container. Containers with 100-500 nCi/g alpha-emitters are accepted on a limited exception basis. All activity limits are as calculated 300 years after time of acceptance. The containers are further classified as contact-handled (<200 mrem/hr at the package surface) or remote-handled (>200 mrem/hr at the package surface).

Category B - includes wastes which are not Category C and which have radionuclide contents greater than those allowed in Category A. This category includes long-lived, alpha-emitter contaminated wastes. Future deep geologic disposal is planned for these wastes following conditioning.

Category C - includes the high-level wastes generated in the reprocessing of spent nuclear fuels. These wastes are destined for deep geologic disposal.

8.0 Waste Generation

The major sources of Category A LLW in France are nuclear power reactors, fuel-cycle facilities, and users of radioisotopes. Nuclear reactor operations produce 90 percent of the Category A wastes. ANDRA estimates that the volume of Category A wastes that will be generated through the year 2000 is 800,000 m^3.

9.0 Waste Stabilization

In France, the waste generator bears the responsibility (including costs) for waste conditioning and packaging, interim storage, and transportation (except to the disposal site) of radioactive waste residues. The waste generators are allowed to choose the method of conditioning their wastes for storage and disposal from ANDRA-approved methods, that include incineration, bitumenization, cementation, resins, etc. LLW is normally conditioned within 70 days after generation.

LLW must be treated and conditioned in accordance with criteria established by ANDRA prior to temporary storage and shipment to the disposal facility. Pathways analysis has been used to establish waste acceptance criteria concerning waste form radionuclide immobilization thresholds.[1] Waste with a specific activity greater than the immobilization threshold determined for each radionuclide must be immobilized, while waste that is below the threshold may simply be stabilized with a medium such as grout. The principal test used by France for stabilization is resistance of the final waste package to a load of 0.35 MPa (50 psi) with little or no deformation (less than 3 percent) and with little or no liquid release.[2]

The waste must pass an inspection by the Institute for Protection of Nuclear Safety (IPSN) and be accepted by ANDRA. Waste containers must be one from an ANDRA-approved list. These containers include steel 110-L drums, metal boxes, and high-integrity concrete drums (2 m^3 capacity).

Waste generators normally compact their dry, solid LLW and most perform further treatment up to and including the placement in an approved container. The wastes generated by the hundreds of small generators are usually compacted and transported to the CEA's facilities at La Hague, Cadarache, or Marcoule for further processing, which is paid for by the waste generator.

Category A liquid wastes are converted to solids and placed in an ANDRA-approved container before disposal. Liquid volume reduction is normally done at the generator's site using standard filtration, ion-exchange, refiltration, and evaporation techniques to concentrate the radionuclides. Some liquid wastes are treated chemically to precipitate the sludge, which is then encapsulated in bitumen. Some radioactive spent resins are immobilized in polymer. Organic liquids are typically incinerated after phase separation from aqueous layers and the residue ash is then further treated by encapsulation into concrete, bitumen, or polymer. Category A solid wastes typically undergo volume reduction by compaction or incineration. Incineration ash is encapsulated into concrete or bitumen.

10.0 Safety Assessment

French regulations require the facility operator to demonstrate how safety requirements will be met before permission is given to proceed with the project. In particular, the following must be demonstrated:

Protection of operating personnel, of the public, and of the environment under operating conditions.

Low probability of accidents, taking both socioeconomic and technical factors into consideration.

Accident consequences, if any, below regulatory limits.

Safety studies are documented in the form of a performance assessment and safety analysis and submitted to regulatory authorities before facility construction and again before operation. Licenses are granted based on a thorough review of the applicant's safety-related documentation. The documentation must contain (1) general safety criteria, (2) general technical requirements, (3) status of design studies, safety analyses, (4) description of the site, and (5) equipment design and operating conditions.

By law, an Environmental Impact Study (EIS) must be submitted with the license application and provided during public hearings. At a minimum, this document must contain (1) reference conditions of the site and its surroundings, (2) analysis of potential environmental impacts, (3) reasons for the proposed project, (4) environmental protection measures planned by the facility operator and their cost, and (5) construction and operation schedule.

The l'Aube disposal facility is designed with a "zero release concept" during the operational and institutional control periods of the facility. The l'Aube performance assessment[4] addresses radiological impacts under abnormal situations involving failure of containment systems. The cause of the failure may be man-made (e.g., poor materials, errors in the design and/or construction of barriers) or natural (landslides, tectonic shifts, earthquakes, etc.). The reference accident scenario chosen for the institutional control period is collapse of the disposal cap over a 100 m^2 area combined with total loss of the effectiveness of the water collection system. Whole body exposure limits are 0.05 Sv/yr (5 rem/yr) for a worker and 0.005 Sv/yr (0.5 rem) for the public.

The l'Aube analyzes potential release scenarios during the "free access" period (period after the release of institutional controls) with the conservative assumption that containment systems have lost all integrity by the end of the institutional control period and no longer offer intruder protection for purposes of radioactive water and air pathway analyses. These scenarios, although called off-normal accident scenarios, include road construction, housing construction, and use of a water well, that are plausible events in an unrestricted access area. The reported external exposure to a housing resident on the site is estimated at 260 mrem/yr from external radiation sources and 23 mrem/yr from inhaled radioactive dust during early years of the free access period. The exposure for this postulated scenario is below the allowable exposure limit of 0.005 Sv/yr (500 mrem/yr) for the public.

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Appendix B

Low-Level Waste Disposal Program of Sweden

Low-Level Waste
Disposal Program of Sweden

1.0 Regulatory Background

The Swedish "Act on Nuclear Activities" of 1984 requires owners of nuclear power reactors (four nuclear utilities) to assume primary responsibility for all aspects of radioactive waste management, including R&D and decommissioning. To fulfill the obligation, the four owners of the utilities have set up the jointly owned Swedish Nuclear Fuel and Waste Management Company (SKB). The SKB is a private company composed of representatives from the nuclear utilities. The SKB has responsibility to plan, design, build, and operate facilities for the disposal of LLW from the power plants. However, the ultimate and long-term responsibility for disposal safety is acknowledged to lie with the federal government.

In 1980, the Swedish Parliament voted to build no more reactors and to shut down the existing 12 reactors by the year 2010 (Act on Nuclear Activities and Ordinance on Nuclear Activities). This phaseout was originally scheduled to begin in the 1995-1996 time frame with the shutdown of reactors at Barsbac and Ringhals. Phaseout of the reactors has been delayed due to recent pressure from environmentalists to ban new major hydroelectric projects and the government's pledge under the Toronto agreement to freeze emissions of greenhouse gases.[1] This pressure has caused authorities to reconsider the commitment to eliminate nuclear power. In 1991, it was decided that the nuclear phaseout shall be determined by the success of efforts to increase energy efficiency, by the availability of new, environmentally-acceptable methods of power production, and by the goal of maintaining internationally-competitive electricity rates.[2]

Sweden's existing waste disposal facilities are designed and sized based on the assumption that the majority of the country's LLW production will cease upon decommissioning of its reactors in the year 2010. Sweden has the following policies regarding LLW disposal:

Manage all radioactive wastes of Swedish origin in Sweden.

Dispose of short-lived LLW and intermediate-level waste (ILW) in an underground cavern repository. This intermediate-depth repository is designed to isolate the wastes for 10,000 years.

Dispose of limited amounts of "very low-level wastes" (exempt waste) in a limited number of shallow land burial facilities. These disposal facilities (located at Studsvik Center, Oskarshamn, Forsmark, and Ringhals NPP) are licensed facilities designed to limit calculated public radiation exposures to less than 1 mrem/yr.

2.0 Approval Requirements

The "Radiation Protection Act," as amended in 1988, regulates the handling of radioactive substances in Sweden and requires that licenses be obtained for all activities involving ionizing radiation. The local municipal council, as well as the federal government, must approve the licensing of a waste management facility. To obtain a license for a nuclear facility, each license applicant is required to secure:

A site permit from the municipal administration.

A permit from the federal government under the Nuclear Activities Act of 1984, after favorable recommendations from the Nuclear Power Inspectorate (SKI) and the National Institute of Radiation Protection (SSI).

A permit from the federal environmental protection agency.

A building permit from the municipal employment board.

During the licensing process, the public interest is represented by the local municipal council and the federal government representatives. The National Board for Spent Fuel (SKN) is responsible to provide the public with information concerning disposal of radioactive waste.

The SKI licenses, supervises and controls the safety of design, construction and operation of the nuclear facilities in Sweden. The agency also reviews and subcontracts R&D needed to assess nuclear safety issues. The SSI establishes and enforces basic radiation protection standards. It oversees radiation protection measures and reports. Both agencies report to the Ministry of Environment and Energy.

3.0 Siting

Studies of options for LLW disposal facilities were performed during the mid-70s to establish the requirements for the nation's only LLW disposal facility, the Swedish Final Repository (SFR). The study recommended that an underground location should be chosen and that, if possible, no institutional control or corrective actions should be necessary after closure of the facility.[3] These criteria drove the selection of the SFR site.

The SFR facility is located 60 meters beneath the Baltic Sea in crystalline bedrock (gneiss and granite) and is about one kilometer off-shore from the harbor at Forsmark. The location has a very low hydraulic gradient and, thereby, the ground water is almost stagnant. Designers consider that there is no risk of a well being drilled so long as the repository is covered by seawater. Due to the land uplift in Sweden (about 6 mm/yr), the sea bottom above the SFR will become dry land in 1,500 to 2,000 years and the hydraulic considerations will change.[7] This time frame is beyond the typical period of concern for implementing institutional controls.

4.0 Past, Existing, and Planned Disposal Facilities

Short-lived low-level and intermediate-level wastes are disposed in the SFR. The application to construct and operate SFR was submitted to the Government in March 1982, and the license was granted in June 1983. No other LLW disposal facilities are planned in Sweden due to the national plan to abandon use of nuclear power (currently being reconsidered).

The license requires the operators of the facility to: 1) keep characterization data on each package; 2) maintain a surveillance program during repository operation; 3) carry out periodic safety analyses; 4) obtain approval of each waste package by the Nuclear Power Inspectorate (SKI) and National Institute of Radiation Protection (SSI); 5) obtain a permit from SKI before sealing off any part of the repository; and 6) maintain the maximum radioactivity inventory of the SFR facility below 270,000 Ci (only a small fraction of which can be long-lived).

The SFR began operation in 1988 and will meet all needs until 2013. Limited amounts of "very low-level wastes" (exempt waste) are disposed in shallow land disposal facilities near the waste-generating site (at Studsvik Center, Oskarshamn, and Forsmark). These facilities have been licensed for

very low-level wastes with the criterion that calculated dose consequence must be less than 0.01 mSv/yr (1 mrem/yr) to the public.

The SFR facility is expanded as needed to accommodate Sweden's wastes. The first construction phase of the SFR facility began in 1988 and included surface buildings and rock caverns for 60,000 m^3 waste. The second phase will start operating in the late 1990s and will include additional disposal capacity for reactor waste. The final construction phase will provide 100,000 m^3 for wastes when reactors are decommissioned after 2010.

During the operations and disposal period, the facility is drained by pumping. After all waste has been disposed and the repository has been sealed, the pumping will stop and the repository will be water-filled by natural inflow of groundwater.

The SFR contains four 160 meter long caverns with a width of 14-18 meters. One of the rock caverns will contain only contact-handled LLW that is emplaced using forklifts. The other three caverns will contain intermediate-level waste which requires radiation shielding or remotely-operated equipment while handling waste containers. These caverns are fitted with concrete-walled cells in which waste is remotely emplaced using an overhead track-mounted crane. After emplacement of waste, each cell is backfilled with concrete grout.

The highest activity waste will be disposed in a silo-shaped cavern, that will contain 90 percent of the repository radioactivity. This cavern consists of a 53 meter high x 27.5 meter diameter concrete silo built by a "slipform" technique within a 70 meter high x 30 meter diameter cavern. The concrete silo is equipped with internal walls to divide the silo into square shafts. It is built on a bed of sand/bentonite (90/10 percent) and the space between the wall and the rock is filled with pure bentonite. Waste is brought into the silo area in shielded containers and placed by remote handling devices into the silo. Once emplaced, the remote handling equipment is used to surround the waste with a low viscosity, porous grout.

Future metal drums of bitumenized LLW/ILW will be placed in caissons made of concrete constructed in a large tunnel. After filling, each caisson will be remotely sealed with concrete. A thick slab of concrete will then be poured over all filled caissons and other spaces will be backfilled with a bentonite clay and sand mixture.

5.0 Design Considerations

The primary criteria used for the design of the SFR are[4]:

(1) The facility will provide long-life containment for the waste packages.

(2) The SFR repository will be situated in bedrock under the Baltic Sea, where any leakage would be into the sea rather than into the groundwater.

(3) The safety analysis of the Swedish disposal system must consider the entire system, but analysis of performance of specific subsystems must be performed.

a) Safety analyses should play a major role in defining R&D.

b) Deterministic and probabilistic models should be used and should be complementary.

(4) The safety of the final repository should not rely on post-closure monitoring, but monitoring may be valuable for other reasons.

(5) No wastes from other countries will be accepted for disposal in Sweden.

Long-term safety of the facility is ensured by the combination of the 1) waste packages, 2) engineered barriers, and 3) the isolation provided by the rock. The primary barrier for the facility is considered to be the host rock.[5]

The escape mechanisms for radionuclides are through moving groundwater or by a well intruding the repository. The SFR was built under the sea bottom because the groundwater at that point is practically stagnant due to the fact that no topographical driving forces are present. The facility location eliminates the risk of a water well being drilled into the waste horizon, as long as the repository is covered by seawater. Approximately 1,500 years after facility closure, geologists predict that the land uplift will raise the shallowest sea bed formations above the sea level. Thus, until past that time, there is no reason to believe that any one will drill a water well and the location beneath the sea is a deterrent to intruders.

6.0 Closure/Post-Closure

As stated in Sections 3.0 and 5.0, Sweden has adopted rather unusual criteria and requirements for closure and post-closure of the facility. On closure, the SFR entrance tunnels will be plugged with concrete to seal the caverns and prevent future access. Some radiological monitoring of the region will be done but the need for active maintenance based on this monitoring is not anticipated or planned.

The country's intent is to leave no legacy for future generations to inherit. The country has established the clear policy that no institutional control should be necessary after closure of the facility and that monitoring should not be relied upon to assure post-closure safety. Commercial off-shore efforts such as invasive geological exploration and drilling (which are not considered likely)[7] will be administratively controlled, and restraints to ordinary use of the sea above the facility are not currently planned.

A surveillance program is maintained during the operational phase of the facility and periodic safety analyses are performed for the operating facility based on the latest data. A final safety evaluation is required before permission will be granted to seal the facility.

7.0 Waste Classification

International Atomic Energy Agency (IAEA) waste classifications[6] are used for Swedish nuclear wastes. Characteristics for the IAEA waste classifications are shown in Table B-1. Exempt wastes are disposed in shallow land disposal facilities near the waste-generating site.

8.0 Waste Generation

Most LLW/ILW results from the Swedish nuclear power program reactor wastes and decontamination and decommissioning (D&D) wastes. Resins and filters from the water cleanup

systems are the source of the majority of the radioactivity. The estimated LLW/ILW through the end of the planned Swedish nuclear power program in 2010 is approximately 208,000 m^3.

Table B-1. IAEA Waste Classification System

Waste Classes	Typical Characteristics	Disposal Options
1. Exempt Waste	Activity levels below clearance levels based on an annual dose to the public of less than 0.01 mSv (1 mrem).	No radiological restrictions.
2. Low and Intermediate-Level Waste* (LILW)	Waste between the categories of exempt and high-level waste.	
2.1 Short-lived waste (LILW-SL)	Restricted long-lived radionuclide concentrations (limitation of long-lived alpha-emitting radionuclides to 4,000 Bq/g in individual waste packages and to an overall average of 400 Bq/g (108 nCi/g) per waste package.	Near-surface or geological disposal facility.
2.2 Long-lived waste (LILW-LL)	Long-lived radionuclide concentrations exceeding limitations for short-lived waste.	Geological disposal facility.
3. High-Level Waste (HLW)	Thermal power above about 2 kW/m^3 and long-lived radionuclide concentrations exceeding limitations for short-lived waste.	Geological disposal facility.

* The LILW waste classification includes both low-level waste (LLW) and intermediate-level waste (ILW). LLW is contact-handled and ILW is remote-handled. A contact dose rate of 2 mSv/hr (200 mrem/hr) is used to distinguish between the two classes.

9.0 Waste Stabilization

Each type of waste disposed in the SFR must be approved by the SKI and SSI, as dictated in the license requirement. Sweden conditions reactor wastes for disposal at the generating reactor sites. The Ringhals and Oskarshamn power stations have the capability to solidify wastes with a cementation process and the Barsback and Forsmar power stations have bitumen matrix processes available. Used ion-exchange resins and sludge are solidified by mixing with cement or bitumen before transport to the SFR.[5] Processing for volume reduction is done at Studsvik Nuclear Center. Processing options include incineration, melting, decontamination, and super-compaction. The SFR facility includes a surface receiving and handling facility that consolidates small waste packages into larger ones.

Concrete boxes 1.2 meters square (typically used for concreted wastes) and 200 liter drums (typically used for bitumenized waste) are the most frequently used waste containers. Some concrete tanks are used for dewatered resins and filter material.

10.0 Safety Assessment

A preliminary safety analysis report was prepared and presented to the government in 1982 after siting and design studies were completed. A construction permit was issued in 1983 by the federal government. Research and detailed design studies were performed while the construction work was performed. The post-closure safety of the repository has been analyzed in detail as part of the licensing process. Calculated doses during the first 1,000 years after sealing of the repository were found to be

insignificant. Additional calculations for the period up to 10,000 years after sealing, showed doses were found to be below the design goal of 0.1 mSv/yr (10 mrem/yr).[3] A final safety analysis report was prepared in 1987 and the license for operation was issued in March 1988 by the SKI and SSI.

The safety assessment was based on a systematically performed scenario analysis. All possible situations (features, events, and processes) were described in the form of "event trees" and the most significant branches were used to formulate scenarios for further analysis. A base case was formulated for each part of the repository and for each time period of concern. Sensitivity analysis was performed for selected parameters.[3]

The operational license includes some requirements for further research and analysis that must be done during the facility operational life. These requirements include maintaining a surveillance program during repository operation and carrying out periodic safety analyses on the facility. A safety evaluation is required before permission will be granted to seal the facility.

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Appendix C

Low-Level Waste Disposal Program of the United Kingdom

Low-Level Waste Disposal Program of the United Kingdom

1.0 Regulatory Background

The nuclear industry in the United Kingdom (UK) is regulated by the British federal government with responsibilities assigned to various government-supported agencies and corporations reporting to different government ministries or departments. Responsibility for the development of a national strategy for the management of radioactive waste in the UK lies with the Secretary of State for the Environment. The Department of Environment assumed the responsibility for management of radioactive wastes from the Department of Energy/UKAEA in 1977. This responsibility includes development of a national waste disposal strategy and the performance of related research. Waste generators have the responsibility for safety and costs associated with radioactive waste management.

In July 1985, the Nuclear Energy Radioactive Waste Executive (NIREX) was set up as a government-owned corporation to implement the government s strategy for the disposal of most LLW and ILW produced by the UK. This agency is responsible for implementing government policy for nuclear waste disposal. NIREX is jointly owned by the four main organizations involved in nuclear power operations in the UK (BNFL, Nuclear Electric, Scottish Nuclear, and the UK Atomic Energy Authority). In addition, one "golden" share is held by the Secretary of State for Energy to represent the British government. Waste generators [e.g., Central Electricity Generating Board (CEGB) and British Nuclear Fuels (BNFL)] implement the disposal strategy of NIREX.

Disposal of radioactive wastes in the UK is governed by the Radioactive Substances Act of 1960. Under the Act, Her Majesty's Inspectorate of Pollution (HMIP) with the Department of the Environment and the Ministry of Agriculture, Fisheries and Food (MAFF) regulate radioactive waste disposal. Facility licenses are issued by the Nuclear Installations Inspectorate (NII).

Current United Kingdom (UK) policy calls for disposal of radioactive LLW in both near-surface facilities and deep geological formations. NIREX's initial aim was to develop a deep repository for disposal of long-lived ILW and a near-surface disposal facility for disposal of short-lived ILW and LLW. In 1987, government policy shifted. A House of Commons Select Committee recommended that all ILW should be disposed in a deep geologic repository and that the same facility should be extended to take LLW also.[1] The Committee believed that putting all ILW in a deep repository would result in a gain in public acceptability. With ILW out of the equation, NIREX advised the Secretary of State that a near-surface disposal facility for LLW alone would be uneconomic. The Secretary of State accepted the recommendation, while acknowledging that there is no technical requirement to dispose of short-lived wastes in a deep geologic repository.[2]

2.0 Approval Requirements

All nuclear installations in the UK must be licensed. Licenses are granted by the Health and Safety Executive (HSE) through the Nuclear Installations (NII) after receiving approval from all appropriate agencies. The HSE, which is an independent federal govemment agency, has the authority to enforce statutory requirements regarding health and safety after granting the license for operation of a nuclear facility. The HSE obtains advice from the Radioactive Waste Management Advisory Committee (RWMAC), which reports to the Department of the Environment.

In October 1983, the British government announced a public consultation on the principles of assessing disposal facilities, and procedures for dealing with possible sites identified by NIREX. This

effort resulted in the government establishing written requirements for assessing disposal sites entitled, *Disposal Facilities on Land for Low and Intermediate-Level Radioactive Wastes: Principles for the Protection of the Human Environment.*[3] The document includes detailed guidance for safety assessments that NIREX is required to produce and gain approval of, prior to licensing.

The site disposal license issued by the NII is principally associated with the safe operation of the site as a nuclear facility throughout its operating phase. The licensing process includes:

Preparation of a Preliminary Safety Report (PSR) and a Pre-Construction Safety Report (PCSR) by the applicant.

A decision on proceeding with licensing based upon review and approval by the licensing departments mentioned above.

Conduct of a public inquiry by NII.

Approval of construction following revision of the PCSR by the applicant to incorporate information/decisions developed in the review process.

Preparation of a Pre-Operational Safety Report by the applicant reflecting the knowledge gained during the construction period.

Approval of operation followed by continuing inspection and regulation during the lifetime of the project.

The license specifies criteria for management arrangements, training, operating conditions, inspection and maintenance requirements, dose assessment, record keeping, and emergency procedures.

Any proposal to extend an existing UK nuclear site or create a new one must be sent to the relevant local planning authority (borough or county) under the Town and Country Planning Acts. The proposal is reviewed by the local planning authority, made up of locally-elected representatives, who will consider public comments submitted in writing. If the local planning authority refuses to approve the proposal, the applicant may appeal to the Secretary of State who can set up a public, local inquiry to help him determine the case. The Secretary of State can overrule the local authority after the public hearing.

3.0 Siting

Britain has disposed of most of its radioactive waste at the Drigg near-surface disposal site since 1959. The Drigg site is located 4 miles southeast of Sellafield, the center of BNFL's reprocessing and waste management operations. Major programs at the Sellafield site include operation of the four Calder Hall Magnox reactors, reprocessing plants for Magnox and oxide fuels, a spent fuel storage facility, and a high-level waste vitrification facility.

The Drigg site has a total area of 270 acres and runs parallel to the coast, about 1 km from the sea. The geology consists of a complex heterogeneous sequence of glacial sediments overlying an irregular surface of red sandstone bedrock. The glacial deposits include compacted clays, silts, coarse sand, and gravels. Only 88 acres are presently authorized for use. Within this area, there is an essentially continuous clay layer at about the 5-8 m depth.[4]

Unsuccessful attempts have been made in England to site other LLW disposal facilities. The government began investigation of a clay site at Elstow in 1983. The siting proposal met public

resistance and, in 1985, the government asked NIREX to select at least two more sites for investigation. Test drilling was done at four locations in the clay geology of Eastern England. Significant local controversy and some mild civil disobedience were experienced at all four sites when they were announced.

As discussed in Section 1.0, the British government, in May 1987, established the policy that a dual-purpose deep geologic facility would be sited with the mission to dispose of both ILW and LLW. The facility is slated to take LLW once the final disposal capacity of the Drigg site is exhausted. This decision led to the abandonment of investigation of the four potential near-surface disposal sites.[2]

Sellafield has been chosen as the proposed location for construction of a deep geologic repository for both low- and intermediate-level waste disposal. The Sellafield site was selected because it was deemed a technically acceptable site and because of its location in relation to the BNFL reprocessing operations which offered transportation advantages.[9] Deep borehole and surface geophysical testing is being performed to characterize the site.[2,5,6,7,8]

4.0 Past, Existing, and Planned Disposal Facilities

The UK disposed of low- and intermediate-level radioactive waste by sea disposal between 1949 and 1982. This practice was discontinued due to international pressure, although the UK maintains that the practice is both safe and practical.

Low-level wastes have been disposed of in the United Kingdom in near-surface disposal facilities since 1959. The British Nuclear Fuels (BNFL)-owned Drigg site in Cumbria is the principal site in England for LLW disposal. A small amount is disposed on-site at Dounreay. The Drigg site began operation in 1959 and is projected to accept LLW until approximately 2054.[4]

The Sellafield deep geological disposal facility is scheduled to be available in 2005.[5] It is expected that limited amounts of higher activity LLW will initially be sent to the deep geological repository for disposal, once available, but the majority of LLW will continue to be disposed of at the Drigg site until the disposal capacity of that near-surface facility is exhausted.

Until 1988, all wastes were disposed of by tumble tipping into trenches cut into an essentially continuous clay layer which is 5 to 8 meters deep. While maintaining that risk assessments showed that near-surface trench disposal was radiologically acceptable, BNFL announced, in 1987, a major upgrade program aimed at improving disposal practices and enhancing the visual impact and perception of the Drigg site.[4] Trench disposal at the Drigg facility was phased out in preference to more engineered disposal using containerized, conditioned wastes in concrete vaults.[9]

The first engineered vault was introduced in 1988. This vault had a capacity of 180,000 m^3 and consists of three bays each of about 60 m width, 200 m length, and 5 m depth. The vault consists of a concrete base and walls and an underlying drainage layer. A clay layer is present beneath the drainage layer. Surface runoff and drainage beneath the concrete vault base are collected and monitored independently prior to discharge to the sea.

Wastes are received principally in either 18 or 38 m^3 gross volume International Standards Organization (ISO) containers and emplaced and stacked in the vault. The larger containers typically contain drummed waste and the smaller ones contain compacted or non-compactable wastes. When the vault is filled, it is covered by a layered cap which uses a low-permeability clay layer.

5.0 Design Considerations

Both the existing vault disposal operations and the planned deep geologic disposal facility adopt the multi-barrier approach to ensure that radioactive materials are initially contained and that their eventual dispersion and return to the biosphere is minimized.[3] The first barrier is the packaged and immobilized waste. The waste form is surrounded by a backfill of cement to provide a conditioned near-field environment for the second barrier. The concrete disposal vault and the natural geological environment form the final barriers to radionuclide migration.

In the deep geologic repository, the waste will be packaged into containers which will be sealed into vaults using cement-based grout. The vaults will form part of the cement-based structure of the repository which will be located in a geological formation with low hydraulic conductivity. The design provides isolation from man by a combination of chemical and physical barriers.[5]

6.0 Closure/Post-Closure

The 1987 BNFL upgrade program of the Drigg disposal site involved the placement of design provisions to better control radionuclide migration before and after facility closure. The program included the installation of groundwater cut-off walls of a cement/bentonite mix to prevent the lateral migration of groundwater and installation of an interim cap over existing trenches. The interim cap was water resistant and was dome-shaped with surface slopes of 1 in 25 to ensure that surface water is directed to perimeter drainage channels. This runoff water from the cap is collected in perimeter drainage channels and routed to the sea, after collection and monitoring.[9] The cap is completed with a soil layer and has been planted with a mixture of grass and shrubs.

Another feature of the capping program was the installation of vertical perforated pipes through the cap into the wastes themselves to allow monitoring of the water levels, to sample water and gases and to allow venting of the small quantities of gases produced by waste decomposition. The cap is monitored for settlement and maintenance is carried out as necessary.

For current disposal operations, as each vault is filled, it is covered by a layered cap that uses a low-permeability clay layer. All trench and vault facilities will be capped with a thicker and more durable cap that will incorporate low-permeability clay before final closure of the site is completed.[10]

The institutional control and monitoring period is expected to last up to 300 years. At a future, appropriate time, regulatory bodies will firmly establish how long institutional control and site surveillance will be continued.[11]

7.0 Waste Classification

Waste is defined by the UK as follows[4]:

<u>High-Level Waste (HLW)</u> - Those wastes in which the temperature may rise sufficiently as a result of radioactivity that it must be taken into account in designing storage or disposal facilities.

<u>Low-Level Waste (LLW)</u> - Those wastes with sufficient radioactivity to be subject to control but which do not exceed 4 GBq/MT (108 nCi/g) alpha-emitting radionuclides and less than 12 GBq/MT (324 nCi/g) of beta-gamma activity.

Intermediate-Level Waste (ILW) - Those wastes between HLW and LLW.

Very LLW - Those wastes containing less than 0.0004 GBq/MT (10.8 pCi/g) alpha or 0.02 GBq/MT (540 pCi/g) beta-gamma activity.

8.0 Waste Generation

Approximately 75 percent of the LLW in the UK originates from the nuclear power industry. The remaining 25 percent results from defense activities, medicine, industry, and research and development.[4] The UK currently generates approximately 44,000 m^3 of LLW per year.[7] The estimated LLW cumulative generation for the UK over the period from 1986 to 2000 is 490,000 m^3 and from 1986 to 2030 is 980,000 m^3.

9.0 Waste Stabilization

The only waste which is acceptable for disposal at Drigg is "relevant waste," which is defined by the Radioactive Substances Act of 1960 as solid radioactive waste that has been treated or packaged in such a way as to render it, as far as reasonably practicable, insoluble in water and not readily flammable.[4,9]

The UK has been developing and implementing LLW/ILW conditioning processes since the start of its nuclear program. A variety of processes such as evaporation, ion-exchange, precipitation, compaction, incineration, bitumenization, and cementation are employed. The UK has constructed two major treatment facilities, the Waste Monitoring and Compaction (WAMAC) Facility and the Drigg Grouting Facility.

Wastes are sent to the WAMAC Facility for volume reduction. Prior to high-force compaction, drums and boxes of wastes can be diverted for monitoring. Monitoring is normally performed on a representative fraction of the waste containers. Non-destructive monitoring is performed using real-time radiography, a high-resolution gamma spectrometer, and passive and active neutron counting systems. Facilities are also provided that allow waste containers to be sampled for hazardous chemicals and radionuclides.

Containers are high-force compacted and any liquid generated during the compaction is absorbed in a suitable matrix. The compacted waste containers are loaded into half-height ISO containers and sealed prior to monitoring and shipment by rail to the Grouting Facility at Drigg. At the Grouting Facility, the containers are filled internally with grout to fill internal voids. The grouting of the containers is to prevent subsidence within the waste stack within the concrete vault.[9]

10.0 Safety Assessment

The fundamental aim of the UK is to ensure that, once in place, the waste will remain effectively isolated from the human environment. To accomplish this, the UK has established performance criteria for the disposal facilities. These criteria require that an annual risk of a fatal cancer to an individual in the critical group will not exceed one in a million. In radiological terms, the target is currently expressed as 0.1 mSv/yr (10 mrem/yr).[3,6]

Government guidelines which NIREX must meet when performing environmental assessments are given in the government document, *Disposal Facilities on Land for Low and Intermediate-Level*

Radioactive Wastes: Principles for the Protection of the Human Environment.[3] This document requires the developer to prepare and submit an environmental assessment which shows that:

The radiological impact of disposal has been thoroughly assessed.

The facility will be designed, constructed, and operated to the required standards of safety.

The site is suitable for the use proposed in the light of all the material planning considerations, including radiological and other safety considerations.

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Appendix D

Low-Level Waste
Disposal Program of Canada

Low-Level Waste
Disposal Program of Canada

1.0 Regulatory Background

The Atomic Energy Control Board (AECB) is the Canadian federal regulatory authority administering the Atomic Energy Control Act, which covers the siting, design, manufacture, construction, commissioning, operation, and decommissioning of nuclear facilities and the production, possession, use and disposal of prescribed substances. AECB regulations are based on objectives and general requirements presented in the *Regulatory Objectives, Requirements, and Guidelines for the Disposal of Radioactive Wastes - Long-Term Aspects*.[1] This document establishes the regulatory basis for judging the long-term acceptability of radioactive waste disposal options and establishes the basic objectives of radioactive waste disposal and the regulatory requirements that must be satisfied to achieve these objectives. The objectives of radioactive waste disposal are to:

Minimize any burden placed on future generations.

Protect the environment.

Protect human health.

The basic regulatory requirements include:

1) The burden on future generations shall be minimized by:

a) Selecting disposal options for radioactive wastes, that, to the extent reasonably achievable, do not rely on long-term institutional controls as a necessary safety feature.

b) Implementing these disposal options at an appropriate time, technical, social, and economic factors being taken into account.

c) Ensuring that there are no predicted future risks to human health and the environment that would not be currently accepted.

2) Radioactive waste disposal options shall be implemented in a manner such that there are no predicted future impacts on the environment that would not be currently acceptable and such that the future use of natural resources is not prevented by either radioactive or non-radioactive contaminants.

3) The predicted radiological health risk to individuals from a waste disposal facility shall not exceed 10^{-6} fatal cancers and serious genetic effects in a year, calculated without taking advantage of long-term institutional controls as a safety feature.

a) If there is no practicable method of fully meeting the above health risk, an optimization study shall be performed in order to determine the preferred option. A disposal facility under these circumstances shall be: 1) compatible with the results of such a study and 2) such that the predicted risk to individuals does not exceed that which is presently accepted from current operations involving the same wastes.

Canadian policies on the management of LLW place the primary responsibility for management of LLW, including disposal, with the producers of the wastes. In the case of major LLW producers, such as Ontario Hydro and AECL, this includes the development of their own disposal facilities.

The federal government accepts the responsibility for cleanup and disposal of "historic wastes" and for post-operational management of disposal sites.[2] Historic wastes are wastes that were managed in the past in a manner no longer considered acceptable and for which the producer cannot be held responsible. These long-lived wastes are largely a result of the radium industry and the early days of the uranium industry.[3] Approximately 90 percent of the existing inventory of LLW in Canada consists of "historic wastes" located primarily at old sites in the Port Hope area of Ontario.[4]

The federal Low-Level Radioactive Waste Management Office (LLRWMO) is the agent of the federal government responsible to ensure that LLW is properly managed within the country. The agency is responsible to provide public information on LLW management, to resolve historic waste disposal problems that are the responsibility of the federal government, and to ensure that adequate disposal facility capacity is planned and developed by waste generators to accommodate the LLW produced on an ongoing basis. The LLRWMO maintains a comprehensive database of LLW and publishes an annual report on LLW production rates and inventories in Canada.

Licensees are responsible to apply for a license and to prove to the AECB that the safety and radiation protection systems proposed will meet the regulations and conform to the ALARA principle.

2.0 Approval Requirements

AECB licensing is required for the construction, operation, and decommissioning of all nuclear facilities inclusive of those related to waste management. Licenses are renewed at periodic intervals, provided that the required conditions continue to be met.[3]

The licensing of a LLW disposal facility is a three-step sequential process in Canada.

1) Licensing is initiated by application for siting and construction to the AECB by the responsible waste generator. The AECB awards a license following an extensive review by AECB staff, outside experts, provincial and local government staff, and public reviews.

2) An operating license is issued after 1) final information is provided on questions which arise during the review of the application and 2) operating procedures and limits are developed.

3) Prior to closure of a LLW disposal site, the operator must submit a final package, updating all information relative to closure, including the need for institutional controls.

Licensing of proposed prototype/demonstration facilities requires a four-stage safety assessment approval process.[5] The steps of this process are shown in Table D-1.

In addition to the regulatory requirements of the AECB, it has become the practice in Canada to carry out independent and public processes of environmental review and consultation for major new activities and major new initiatives related to existing activities. Most new nuclear facilities in Canada are referred to the federal Minister of Environment for a formal public review by an independent panel, with full opportunity for public hearings, and with funding for intervenors.[3]

Table D-1. Four-Stage Safety Assessment Approval Process

DOCUMENT	DOCUMENT SEEKS APPROVAL OF	APPROVAL REQUIRED TO START
Concept Safety Assessment Report	Site selection and conceptual design	Detailed design
Preliminary Safety Assessment Report	Detailed design	Construction
Final Safety Assessment Report	Construction	Waste loading
Updated Final Safety Assessment Report	Waste inventory in disposal unit	Closure of the disposal unit

3.0 Siting

Although Canadian regulations do not require public approval of nuclear facilities, the responsible agencies have recognized the need for public support. Public reviews/hearings are held when potentially controversial nuclear issues arise such as siting of nuclear facilities. Public opposition to the siting of new LLW disposal facilities has led to a community-oriented, cooperative siting process that provides a method for addressing social concerns.[2]

In 1980, the AECB directed Eldorado Nuclear Ltd., the Crown Corporation, to develop plans for decommissioning the major waste management sites. Attempts by Eldorado Nuclear to find a permanent site for the wastes were based primarily on technical, scientific, and economic considerations. The sites chosen under these criteria were greeted with a public outcry by local residents. The community opposition reached such a level that, in 1986, the Canadian government cancelled the siting process and initiated an independent task force with a mandate to come up with suggestions for a less confrontational process that would lead to the resolution of the siting problem.[3]

Two independent siting initiatives are currently underway in Canada to find acceptable disposal sites for historic LLW.[3,4] The initiatives (described below) are technically supported by the LLRWMO.

The federal Siting Task Force on LLW management was established in late 1988 to implement a cooperative process based on voluntary participation of communities that are potentially interested in hosting a new facility for the approximately 1,000,000 m^3 of historic wastes in the Province of Ontario. Sites in two potential volunteer communities in Ontario, Geraldton and Deep River, will be evaluated as to their technical and social suitability. In Deep River, AECL's Chalk River Laboratories' property is being evaluated for the transfer and disposal of the historic wastes. AECL is cooperating in the site evaluation and as a technical consultant. The Community Liaison Group in Port Hope, Ontario, has suggested that Port Hope might consider becoming a volunteer community for a disposal facility for its own wastes.

The Surrey Siting Task Force launched a similar cooperative siting process in Surrey, British Columbia, in the fall of 1989 to find a site for a small quantity of contaminated soil and niobium slag currently located on two industrial properties in Surrey. Preliminary province-wide consultations to identify possible volunteer communities for the wastes were undertaken and a Community Liaison Group was established at Surrey. A sampling program to assess the composition of the waste in order to

determine its status under provincial waste regulations was completed. This information has allowed the task force to better define the potential disposal options.

Ultimately, any proposed facility will have to be approved by the AECB. It is expected that the AECB will consider a variety of technological options and rationales to find an acceptable long-term management solution for disposal of the low-level wastes.

4.0 Past, Existing, and Planned Disposal Facilities

Canada currently stores most of its LLW until a suitable disposal technology can be proven. The inventory of approximately 1,200,000 m^3 is stored at several locations. Most of this inventory is historic wastes which resulted from uranium refinery operations. This inventory volume does not include the approximately 200 million tons of uranium mill tailings that have been generated since the mid-1950's.

The LLW generated by the nuclear power plants in Canada is being stored at the generating sites.[6] LLW from all other Canadian sources (hospitals, universities, and industry) is stored at AECL's Chalk River Laboratories (CRL). The storage activities at CRL are considered as waste management, with little formal distinction made between storage (a situation which is intended to be temporary) and disposal (a situation which is intended to be permanent). Currently wastes with only low levels of activity (less than 0.1 Ci/m^3) are placed in a large unlined trench, while wastes with higher levels of activity are put into concrete bunkers and steel-lined concrete pipes, embedded in the ground, known as tile holes. Even though the waste placed in trenches is deemed as being stored , there is little distinction between this technique and traditional shallow land disposal, other than acknowledgment that the storage configuration is temporary. It is anticipated that, with some additional upgrading of the cover, the trench facility will eventually be classified as disposal, whereas wastes going into bunkers and tile holes will be recovered for disposal in some other type of facility.[5,11]

AECL decided, in the early 1980s, to establish a demonstration program to make the transition from storage to permanent disposal of LLW at CRL. The current objective of the program is to demonstrate disposal technology by establishing appropriate prototype/demonstration facilities.[5] The CRL site is currently in the process of demonstrating a modular near-surface disposal system at the CRL site for wastes produced by the AECL, and those received on a commercial basis from small-volume producers who have no interest in developing their own facilities for long-term LLW management.

The demonstration unit under development is called the Intrusion Resistant Underground Structure (IRUS). The Canadian government is currently dealing with the political and financial issues of implementing the IRUS system.[6] Detailed design of the facility is completed and detailed cost estimates have been produced.[4]

IRUS is an underground concrete vault designed to contain about 3,900 m^3 of LLW per module. It is designed for wastes with hazardous lifetimes of up to 500 years. The IRUS module is 32 m long, 22 m wide and 9 m deep, and is divided into 6 cells that provide a total usable volume of about 3,900 cubic meters. The walls are 0.61 m thick and the reinforced concrete roof is 1 m thick. The floor is permeable to avoid a "bathtub effect," and is composed of two buffer layers: a 0.3 m thick mixture of sand (90 percent) and clinoptilolite (10 percent), and a 0.3 m thick mixture of sand (90 percent) and Dochart clay (10 percent). The clinoptilolite and clay have the capacity to sorb many critical nuclides from aqueous solutions, and thus reduce radionuclide escape from the vault.

While operating, the vaults will be covered by an unheated, weather-resistant metal frame building equipped with an overhead gantry crane. When filled, the vault will be covered by the 1 m thick concrete cap and 1.5 m of sand and soil. Vegetation will be planted to prevent erosion. The crane, building, and other equipment will be moved to another IRUS vault when the initial vault has been filled.

A separate program from the IRUS development is being developed to handle the large existing inventory of LLW from the refinery operations and is expected to have a demonstration facility about the year 2000.[2] These wastes pose different disposal post-closure considerations than typical reactor wastes due to the long-term buildup of radon and other decay daughter products. Disposal of these wastes is being planned as a separate effort from currently-produced radioactive wastes. Canada states the following in its regulatory policy statement for radioactive wastes:[1]

> "The practical disposal options presently being studied usually involve containment of wastes and their isolation from the biosphere for extended time periods. For some waste types, though, such as the large-volume wastes from uranium mining and milling, the ideal type of disposal may sometimes not be practicable. In such instances where there are no practical disposal options for achieving the ideal goal, there may be a long-term need for continued institutional controls to guard against particular exposure scenarios after the facility has ceased receiving waste and has been closed."

Thus, Canada recognizes that its historic wastes and associated mill tailings will require long-term institutional controls due to the daughter products of long-lived uranium components.

Canada is planning to dispose of newly-generated (non-historic), longer-lived wastes containing radionuclides such as uranium, thorium, carbon-14, and plutonium in some form of rock cavern, possibly in conjunction with a nuclear fuel waste disposal facility, rather than in near-surface disposal facilities. Only a small fraction of the newly generated waste will not qualify for disposal in the IRUS unit. This quantity of Canadian wastes is so small as not to require attention at this time. Instead, Canada plans to store these wastes in engineered facilities for the indefinite future.[4,5]

5.0 Design Considerations

The development/demonstration program is designed to deal with three categories of LLW according to the radiological decay properties of the wastes.

1) Wastes containing radionuclides with short half-lives (residual hazard for less than 150 years) such as radiopharmaceuticals containing iodine-125, activated corrosion products such as zinc-65 and cobalt-60, and tritium. It is envisioned that an improved sand trench-type disposal facility may be appropriate.[4]

2) Wastes containing fission products and other isotopes that must be isolated for up to 500 years. These wastes will be disposed of in an intrusion-resistant underground structure (IRUS) facility.[4]

3) Wastes containing long-lived radionuclides (residual hazard is greater than 500 years) such as uranium, thorium, and carbon-14. It is envisioned that these wastes will be disposed of in some form of rock cavern, possibly in conjunction with a nuclear fuel waste disposal facility.[4]

The general rule-of-thumb is that the hazard associated with a radionuclide will decay to an insignificant level after a period of ten half-lives. Thus, a waste with a residual hazard for less than 500

years would contain radioactive isotopes with a half-life of less than 50 years. Only trace amounts of longer half-lived elements would be allowed in such a waste.

A design concept called Improved Sand Trench (IST) is being developed by the Canadians at CRL to contain LLW with a hazardous lifetime of less than 150 years. The concept uses a water shedding membrane cap supported on a panel structure of lean concrete. Infiltrating water is intercepted by the cap and drains laterally to channels at the boundaries of each panel. A free-flowing unconfined aquifer in a sand layer well below the waste dilutes any escaping nonreactive radionuclides, but provides a reasonable retardation for most other nuclides.

Multiple barriers are employed in the design of the IRUS demonstration module to minimize water entry and to retard the release of radionuclides. The IRUS is being developed at Chalk River Nuclear Laboratories. The primary design criteria are[5]:

The disposal unit should deter intrusion for at least 500 years.

The roof and cover of the disposal unit should minimize the infiltration into the disposal unit.

The disposal unit should not rely on active intervention, such as leachate collection and treatment, for its long-term safety.

The disposal facility should not result in a risk of a serious health effect to the most exposed individuals of more than 10^{-6} per year to comply with the Canadian regulatory criteria.[7]

The IRUS concrete vault is placed underground to provide attenuation of radiation and because the relatively stable underground environment, free from the extremes of Canadian winters and summers, will be more conducive to ensuring the durability of the structure. The structure will be located in a free-draining sand deposit with its foundations at least one meter above the highest recorded water table to avoid flooding within the waste horizon. The structure relies on the durability of concrete to provide the required 500 years of service life. The concrete formulation has been selected based upon an extensive research and development program. The substantial thickness of the roof (1 m) serves two purposes: 1) to deter inadvertent intruders and 2) coupled with the drainage features above it, to minimize infiltration of water. The unit has been designed with seismic resistance appropriate for the region.[5]

The IRUS design is modular and can be constructed on an as-required basis. This avoids the large up-front capital investment required for some other disposal concepts.

The program to design a concrete vault for the IRUS repository that will last 500 years has focused on major concrete degradation agents, rate of diffusion of corrosive ions, rate of advancement of the corrosion front into the concrete, and test methods and extrapolation procedures to predict long-term durability. Different forms of concrete were tested to determine their performance under repository environmental conditions. Various degradation mechanisms were considered including alkali-aggregate reactions, chemical deterioration by chloride and sulfate attack, carbonic acid corrosion, freeze-thaw scaling, leaching and dissolution of lime.[8]

6.0 Closure/Post-Closure

As discussed in Section 2.0, an updated Final Safety Assessment Report must be completed and approved before closure of the disposal unit can be performed.[5] Final closure of the IRUS facility will include a multi-layer earthen cover system about 2 meters thick.[9]

Prior to closure of a LLW disposal site, the operator of the site must submit a final package updating all information relative to closure, including the required institutional control period.

7.0 Waste Classification

In Canada, LLWs are defined by exclusion. That is, if a waste is radioactive, but it is not high-level waste, nor uranium mill tailings, then it is classified as LLW. In terms of the U.S. classification, all wastes from the very lowest of the Class A waste to greater-than-Class C are included. Three LLW/ILW categories, classified according to their hazardous lifetime, are used[2,4]:

1) Low radionuclide concentrations with short half-lives (duration of potential hazard is <150 years) improved sand trench-type facilities are considered.

2) Wastes that remain hazardous for 150-500 years (engineered facilities at shallow depth are considered).

3) Long-lived radionuclides or intermediate-level wastes that remain hazardous for >500 years (deep geologic disposal facilities are considered).

Quantitative limits for specific isotopes have not been assigned to the above categories. Technical issues, including defining more detailed waste classification limits, are currently being reviewed and resolved by the Canadian authorities. These details will logically evolve as the safety documentation for demonstration facilities, like the IRUS, develop and obtain approval.

Canada is currently designing programs to reduce the volume of LLW that must be stored and disposed in licensed radioactive waste management facilities. Volume reduction programs are being developed to exempt certain LLW from licensing requirements upon transfer for disposal. Some wastes with trace activity are designated as "releasable" based on meeting de minimis dose criteria. The design and construction of a pilot-scale study is now underway to demonstrate the cost-effectiveness and technical feasibility of segregation and unconditional release of solid wastes from AECL Research Chalk River Laboratories (CRL). The current position of the AECB on the exemption of radioactive materials from further licensing upon transfer for disposal is summarized in Regulatory Document R-85. R-85 states that when the circumstances of such disposal are considered to represent a negligible, or de minimis risk, expenditure of additional regulatory resources or continued licensing of the material is not justified. The AECB uses a de minimis dose of radiation to individuals of 0.05 mSv/yr (5 mrem/yr) for deciding exemptions on a case-by-case basis, provided that the radiological impact will be localized and the potential of exposure of large populations is small.[10]

The de minimis dose criterion of 0.05 mSv/yr follows from the acceptance of a corresponding de minimis health risk. The criterion represents an extrapolation from a fatality risk from cancer of 10^{-6} per year.[10] The secondary requirement, that the potential for exposure of large populations be small, is intended to restrict undue reliance on dilution as a means of attaining compliance with the de minimis criterion.

8.0 Waste Generation

The total volume of Canadian LLW, including contaminated soils, as of 1985 was 1,200,000 m^3. Estimated generation for the period between 1985 to 2025 is approximately 370,000 m^3, including decommissioning wastes.[2] The primary generators of LLW are Canada's three nuclear-electric utilities, Ontario Hydro, New Brunswick Power, and Hydro Quebec; the two national laboratories at Chalk River, Ontario, and Pinawa, Manitoba; and the uranium refining company Eldorado Nuclear Limited. Canada has 22 operating nuclear power reactors.

The total volume of ongoing LLW produced in Canada is currently in the range of 4,000 to 6,000 m^3/yr. LLW from nuclear reactors consists mostly of slightly contaminated garbage from operations and maintenance activities with a small volume of higher activity wastes in the form of filters and ion-exchange resins from purification systems, and irradiated equipment. Other sources of LLW are slightly contaminated garbage from radioisotope uses in research, medicine and industry, building materials from decommissioning of facilities where processing of radioactive elements was carried out in the past, and process residues with low-levels of radioactivity left over from the refining and conversion of uranium.[3]

9.0 Waste Stabilization

Canada has done extensive research on waste treatment. LLW/ILW treatment research and development is centered at the Chalk River Waste Treatment Center. Researchers have built a waste treatment center to demonstrate volume reduction techniques on a commercial scale and to improve the management of wastes. The waste treatment center integrates several processes with the specific aim of converting waste into a stable, leach-resistant form suitable for disposal. The facility is comprised of: a controlled air incinerator for combustible solid and liquid wastes; a baler for non-incinerable solid wastes; membrane filtration, reverse osmosis and evaporator systems for diluted aqueous wastes; and a blender and bituminizer to minimize the incinerator ash and liquid waste concentrates.

Before waste is sent to the IRUS for disposal, it will be characterized and processed at the Chalk River Waste Treatment Center.[2] The principle waste packages will be 0.4 m^3 bales of compacted waste and 200 liter metal drums. The bales will include fibrous materials, plastics, and small quantities of metals. Most drums will contain a bitumen waste-form produced from liquid-solidification processing or ash immobilization. Waste will occupy about 50 percent of the IRUS volume. Voids between the packages and layers of waste will be backfilled with sand (90 percent) and clinoptilolite (10 percent).[9]

The Bruce Nuclear Power Development site processes and stores all the LLW/ILW from Ontario Hydro reactors (Ontario Hydro operates 20 of Canada's 22 nuclear reactor power plants). Treatment of solid wastes depends on content and radioactivity. Solid wastes with contact dose rates less than 60 mrem/hr and that do not contain large quantities of halogenated material are incinerated. A low-force compactor/bailer is used for solid wastes that cannot be combusted. The remaining wastes are stored without size reduction. Liquids (mostly hydraulic and lubricating oils from fueling machines) are currently stored pending immobilization. Tritium-contaminated heavy water is treated at the Ontario Hydro tritium extraction plant. Solids contaminated with low-levels of tritium are processed like other LLW. Highly tritiated wastes are packaged to retain the tritium.

10.0 Safety Assessment

To be acceptable under Canadian regulations, a LLW disposal system must ensure that the serious risk to individuals from escape to the environment and from inadvertent intrusion is less than 10^{-6} per year. The period for demonstrating compliance using mathematical models need not exceed 10,000 years. The calculated individual risk is based on a risk conversion factor of 2×10^{-2} per sievert (2×10^{-4}

per rem) and the probability of the exposure scenario with either (a) the annual individual dose calculated as the output from deterministic pathways analysis or (b) the arithmetic mean value of annual individual dose from the distribution of individual doses in a year calculated as the output from probabilistic analysis.[1]

The AECB is primarily interested in radiological protection, both during operation and after closure, but will also ensure that the project complies with other relevant regulatory requirements. Non-radiological impacts are addressed by compliance requirements with the federal government's Environmental Assessment and Review Process. This is a self-assessment process involving, as the first step, an environmental screening to be followed, if necessary, by further assessment and public hearings.[5]

The long-term safety of the IRUS disposal units has been evaluated by use of pathways analysis. The key determinants of safety are the radionuclide inventory, the design of the disposal unit, the deterioration of the facility with time, and the geological and geographic setting of the disposal unit. The radionuclide inventory of each disposal unit is conservatively controlled to ensure acceptable performance results. The analysis has shown that no major constraints are imposed by inventory limits on the volume of waste which may be accepted in the IRUS units.[5]

The radionuclide inventory data, models of radionuclide release and migration and site-specific data were used in applying the COSMOS-3 safety assessment code to analyze the impact of the facility on the critical individual for the IRUS disposal facility. This individual is assumed to be living in a subsistence farming manner, on the shores of the lake that would receive any contaminated groundwater from the facility. The results of the analysis show that all risks are well below the regulatory limit. The greatest potential contributor to risk is tritium. However, the tritium peak occurs a few decades after closure, which is well within the period over which institutional control should be maintainable. The next highest peaks result from plutonium isotopes and occurs several thousand years into the future.[5]

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Appendix E

Low-Level Waste Disposal Program of the United States Department of Energy

Low-Level Waste Disposal Program of the United States Department of Energy

1.0 Regulatory Background

The first major defense-related use of radioactive material in the United States was by the Manhattan Engineering District whose single purpose was to develop and produce a useable nuclear weapon. The Atomic Energy Act of 1946 transferred Manhattan Engineering District facilities and responsibilities to the civilian-controlled Atomic Energy Commission (AEC). The Act stressed that the Commission's paramount objective remained "assuring the common defense and security." The AEC guidelines limited exposure of employees to the maximum permissible levels recommended by the National Committee on Radiation Protection.

AEC licensing and regulatory oversight of organizations outside the agency that possessed nuclear materials began with the growth in civilian uses of nuclear materials. Regulation was necessary to control the distribution of nuclear materials and to ensure that organizations outside the AEC that managed these materials adhered to the safeguards observed within the agency.

The Energy Reorganization Act of 1974 split the AEC into two organizations, the Energy Research and Development Administration (ERDA) and the NRC. The ERDA was directed to continue the federal government's program for management of nuclear-related programs for research and development and national defense. Congress assigned defense-related activities to ERDA regulation because it was viewed that national defense is the responsibility of the federal government. ERDA was later eliminated and its functions were absorbed by the Department of Energy, which was created by the Department of Energy Organization Act of 1977.

The DOE's policies and guidelines for managing the Department's LLW were formally established in February 1984 with the publication of DOE Order 5820.2, *Radioactive Waste Management*. This order replaced the policies of the AEC that had evolved over the years. In 1986, DOE initiated a revision of DOE Order 5820.2, Chapter III, Management of Low-Level Waste. DOE established a working group to draft a prescriptive or performance objective-oriented revision of the LLW chapter of the Order. DOE-HQ expanded this initiative and issued formal direction to rewrite the entire Order. The revision was intended to address the requests from disposal site operators that DOE Order 5820.2 should establish more definitive requirements, such as generation, characterization, acceptance criteria, treatment, shipment, storage, and disposal of waste, and disposal site closure, environmental monitoring, quality assurance, and records and reports. The revised DOE Order (5820.2A) was approved on September 26, 1988, and is currently in use as the primary criteria for designing and conducting LLW disposal activities for DOE wastes.

2.0 Approval Requirements

The DOE is a self-regulating entity which authorizes the design, construction, and operation of facilities under its direction. The DOE has established a set of orders that prescribe requirements that must be met before the DOE will give its management and operating contractors authority to operate any hazardous facility. The primary requirements of these Orders to obtain approval to operate a LLW disposal facility are discussed below.

DOE Order 5820.2A is the primary Order dictating DOE requirements for LLW disposal activities. The Order establishes the primary performance objectives and technical requirements for LLW disposal. The Order requires field organizations with disposal sites to prepare and maintain a site-

specific radiological performance assessment (PA) for the disposal of waste with the purpose of demonstrating compliance with the radiological performance objectives in the Order. For new DOE LLW disposal facilities, PAs are reviewed by the responsible field element and submitted to the Deputy Assistant Secretary (DAS) for Waste Management before construction begins. Recent DOE guidance[1] establishes policy for review and approval of disposal facility PAs, which are reviewed by a peer review panel (PRP) at the request of the DAS for Waste Management. The purpose of this review is to ensure consistency and technical quality throughout the DOE complex in the development and application of PA models that include site-specific geohydrology and waste composition. The PRP is selected by the DAS for Waste Management and is composed of DOE, contractors, and other specialists in PAs, with participation by representatives of the Office of Environment, Safety, and Health, and operations offices.

Documentation from the PRP review accompanies the PA, as well as other information, as needed, that assesses disposal facility performance (such as the closure plan and safety analysis report for the disposal facility). Waste management staff evaluate the PA and PRP reviews, consult with the Office of Environment, Safety, and Health; and make a recommendation to the Assistant Secretary for Environmental Management regarding compliance with the performance objectives of DOE Order 5820.2A. The Assistant Secretary for Environmental Management decides whether or not to authorize construction of the disposal facility. When construction is authorized, the DAS for Waste Management prepares a Disposal Authorization Statement that sets forth the conditions for design, construction, and operation of the disposal facility that are appropriate to ensure compliance with the LLW performance objectives.

DOE contractors are also required to obtain the Program Secretarial Officer (PSO) approval of Preliminary Safety Analysis Reports (PSARs) prior to undertaking procurement of materials and components, construction, and preoperational testing of DOE nuclear facilities.[2] The PSAR is a document routinely prepared to document the adequacy of the safety basis for a new nuclear facility; it provides assurance that the facility can be constructed, operated, maintained, and shut down safely and in compliance with applicable laws and regulations. It differs from the PA in that it primarily deals with worker and public safety issues during routine and credible off-normal operational conditions, whereas, the PA is concerned with providing a reasonable estimate that the facility will meet the performance objectives established in DOE Order 5820.2A.

DOE contractors are required to submit the facility's Final Safety Analysis Reports (FSARs) to the PSO for approval and authorization to operate DOE nuclear facilities. This approval is required in addition to the approval of the PA and PSAR prior to the start of facility construction. FSARs document the adequacy of the safety basis and provide assurance that the facility can be operated, maintained, and shut down safely and in compliance with applicable laws and regulations.[2]

The National Environmental Policy Act (NEPA) Implementing Procedures (10 CFR 1021) require DOE to normally prepare an EIS for all major system acquisitions posing a potential threat to the environment. Thus, the LLW disposal facility operation will be addressed by a site-wide or a facility-specific EIS. The DOE EIS process requires public hearings and the document is ultimately approved by DOE after resolving public concerns.

The last step to beginning startup of new or significantly altered DOE LLW disposal facilities requires the successful completion of an operational readiness review (ORR) as outlined in DOE Order 425.1, *Startup and Restart of Nuclear Facilities* (supersedes DOE Order 5480.31, *Startup and Restart of Nuclear Facilities*). The ORR consists of both a contractor review and a DOE review of the facility's readiness to operate. Upon completion of the ORR, a final report documents the results of the ORR and

reaches a conclusion as to whether startup of the facility can proceed safely. Thus, the issuance of a Disposal Authorization Statement, approval of the FSAR, approval of an applicable EIS, and a favorable ORR are required before a DOE LLW disposal facility can begin operation.

Operating disposal facilities are subject to regulation and oversight by various DOE offices. The Deputy Assistant Secretary for Waste Management (EM-30) is charged with carrying out Assistant Secretary for Environmental Management (EM-1) responsibilities for managing DOE waste management activities, developing and interpreting waste management policy, and issuing guidance to the field. The Deputy Assistant Secretary for Environmental Restoration (EM-40) also has responsibility of environmental restoration waste disposal facilities. The Assistant Secretary for Environment, Safety, and Health (EH-1) provides oversight and independent assessments of waste management operations.

EM and EH offices are organizationally independent, but both report to the Secretary of Energy. EM-30 requirements are implemented in the waste management facilities by DOE management and operating contractors through written operating procedures and documented training programs. DOE field office representatives oversee the contractor operations and use a system of contractor incentive fees to encourage compliance with requirements. EH-1 and EM-1 have shutdown authority for waste management operations if environment, safety, and health risks are judged to be unacceptable.

A recent Federal Advisory Committee for DOE investigated the appropriateness of continued self-regulation of DOE and the need for external regulation. The committee recommended that the DOE be subject to control by an independent regulatory agency.[3] Details of implementation of this recommended policy have not yet been established.

3.0 Siting

DOE disposal siting options are constrained by the locations of current DOE reservations, which were deemed as "appropriate" sites for nuclear activities at the time they were selected from national candidate sites. DOE Order 5820.2A establishes the primary requirements for the siting of a LLW disposal facility. The Order requires:

(a) Disposal site selection criteria (based on planned waste confinement technology) shall be developed for establishing new LLW disposal sites.

(b) Disposal site selection shall be based on an evaluation of the prospective site in conjunction with planned waste confinement technology, and in accordance with the NEPA process.

(c) The disposal site shall have hydrogeologic characteristics which, in conjunction with the planned waste confinement technology, will protect the groundwater resource.

(d) The potential for natural hazards such as floods, erosion, tornadoes, earthquakes, and volcanoes shall be considered in site selection.

(e) Site selection criteria shall address the impact on current and projected populations, land use resource development plans and nearby facilities, accessibility to transportation routes and utilities, and the location of waste generation.

Multiple documents (discussed in Section 2.0) must be reviewed and approved to demonstrate that the chosen site and the planned waste confinement technology are adequate to protect the public. The primary documents which address the acceptability of the site are the EIS and the PA.

4.0 Past, Existing, and Planned Disposal Facilities

A variety of designs are in use for DOE LLW disposal facilities. Table E-1 describes disposal engineered structures and surface barriers used in existing DOE disposal facilities. As the table shows, disposal methods vary from open trench disposal with surface barriers to use of near-surface disposal methods that use engineered structures (such as vaults) to provide "greater confinement" for LLW disposal. Each DOE site uses engineered structures and surface barriers in addition to the migration barriers afforded by the waste form and site geologic feature of the site to assure that the performance objectives of DOE Order 5820.2A are met. Engineered structures are used at the more humid DOE disposal sites, Savannah River and Oak Ridge.

5.0 Design Considerations

The primary criteria used for the design of DOE LLW disposal facilities are given in DOE Order 5820.2A (recently canceled DOE Order 6430.1A was the basis for many design requirements of existing facilities). These requirements are functional performance objectives which must be met by all facilities as demonstrated in the approved facility PA. The facility is required to:

(1) Protect public health and safety in accordance with standards specified in applicable EH Orders and other DOE Orders.

(2) Assure that external exposure to the waste and concentrations of radioactive material which may be released into surface water, groundwater, soil, plants and animals results in an effective dose equivalent that does not exceed 25 mrem/yr to any member of public. Releases to the atmosphere shall meet the requirements of 40 CFR 61. Reasonable effort should be made to maintain releases of radioactivity in effluents to the general environment as low as is reasonably achievable.

(3) Assure that the committed effective dose equivalents received by individuals who inadvertently may intrude into the facility after the loss of active institutional control (100 years) will not exceed 100 mrem/yr for continuous exposure or 500 mrem for a single acute exposure.

(4) Protect groundwater resources, consistent with federal, state and local requirements.

The above performance objectives are the fundamental basis for the siting, design, and operation of all DOE LLW disposal facilities. DOE uses functional performance objectives and requires facility designers and analysts to establish disposal barrier requirements on a site-specific basis that ensure that the criteria are met. This includes waste classification requirements and any associated conditioning or stabilization requirements, packaging, special backfills, depth of burial and engineered barriers such as vaults and covers. Compliance with DOE performance objectives is documented in the facility PA.

6.0 Closure/Post-Closure

DOE Order 5820.2A establishes the following closure/post-closure requirements:

(1) Field organizations shall develop site-specific comprehensive closure plans for new and existing operating LLW disposal sites. The plan shall address closure of disposal sites within a five year period after each disposal site is filled and shall conform to the requirements of the NEPA process.

Table E-1. DOE low-level waste disposal facility descriptions.

DOE site		Current disposal method
Hanford		
	Low-Level Burial Grounds	Near-surface disposal in V-trenches and wide-bottom trenches. The waste is backfilled with soil and a final cover, designed to limit the infiltration rate to less than 0.5 cm/yr, is applied to the parts of the disposal facility containing Category 3 (higher activity inventory) wastes.[7]
INEL		Near-surface disposal in pits, trenches, and soil vaults. An earthen cover is placed over the waste during the operational period. Upon closure, a thick soil barrier, which includes a vegetative cover, will be emplaced over the operational cover giving a total soil cover of 5 m.[8]
Nevada Test Site		
	Area 3	Near-surface disposal in subsidence craters from underground nuclear tests. Wastes are disposed using conventional landfill techniques where each layer of waste is covered with 1 m of fill before additional wastes are disposed in the pit.
	Area 5	Near-surface disposal in pits, trenches, and boreholes. An earthen cover is placed over the waste during the operational period. Upon closure, a final cap (not yet designed) will be emplaced to enhance facility performance.[9]
Los Alamos		
	MDA G	Near-surface disposal in pits and 20 m deep disposal shafts. Waste is placed in the pits and shafts in lifts and crushed tuff is placed in void spaces, between the lifts, and on top of the waste. Filled pits are covered with at least 3 ft. of crushed tuff and 4 inches of top soil and planted with native grasses. Shafts are topped with 1 ft. of concrete shaped to promote drainage away from the shaft.[10]
Oak Ridge		
	Solid Waste Storage Area 6	Above-grade tumulus uses concrete rectangular vaults which are filled with waste, annular spaces are filled with concrete, pre-cast concrete lid is placed on the vault and sealed with bitumen. The vault is subsequently loaded and stacked onto a curbed concrete pad and capped with natural materials. The pad has a concrete curb around the perimeter. Surface drainage channels divert surface runoff away from the pad.[11]
Savannah River		
	Saltstone	Above-grade concrete vaults covered with soil, clay, and a gravel/earthen cap. The saltstone is poured into the vault leaving approximately 0.3 m from the top of the vault wall to be filled with uncontaminated grout. After all cells are filled, a permanent concrete roof is installed. On closure, soil is placed between the vaults and clay/gravel drainage system with earthen and vegetative cover is installed to route precipitation to a drainage system.[12]
	E-Area Vault	Above-grade concrete vaults covered with soil, clay, and a gravel/earthen cap having a vegetative cover. The vaults have a concrete cover (covered by the cap) to divert surface runoff away from the vaults and the floor of the vault slopes to a drain which runs to a collection sump which is monitored for radionuclides.[13]

(2) During closure and post-closure, residual radioactivity levels for surface soils shall comply with existing DOE decommissioning guidelines.

(3) Corrective measures shall be applied to new disposal sites or individual disposal units if conditions occur or are forecasted that could jeopardize attainment of the performance objectives of this Order.

(4) Inactive disposal facilities, disposal sites, and disposal units shall be managed in conformance with the RCRA, CERCLA, and SARA, or, if mixed waste is involved, may be included in permit applications for operation of contiguous disposal facilities.

(5) Closure plans for new and existing operating LLW disposal facilities shall be reviewed and approved by the appropriate field organization.

(6) Termination of monitoring and maintenance activities at closed facilities or sites shall be based on an analysis of site performance at the end of the institutional control period (normally 100 years).

Thus, a formal approved closure plan is required for each disposal facility. Approval of that plan will require that it can be demonstrated that the facility will meet DOE Order 5820.2A performance objectives and applicable existing DOE decommissioning guidelines. DOE takes the position that maintenance and monitoring of the site will be available, as necessary, for as long as it may be required.

7.0 Waste Classification

The DOE defines wastes as follows:

High-Level Waste (HLW) - Highly radioactive waste material that results from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid waste derived from the liquid that contains a combination of transuranic waste and fission products in concentrations requiring permanent isolation.

Transuranic Waste (TRU) - Without regard to source or form, waste that is contaminated with alpha-emitting transuranium radionuclides with half-lives greater than 20 years and concentrations greater than 100 nCi/g at the time of assay. Heads of Field Elements can determine that other alpha-contaminated wastes, peculiar to a specific site, must be managed as transuranic waste.

Low-Level Waste (LLW) - Waste that contains radioactivity and is not classified as HLW, TRU, or spent nuclear fuel or 11e(2) byproduct material. Test specimens of fissionable material irradiated for research and development only, and not for the production of power or plutonium, may be classified as LLW, provided the concentration of transuranic activity is less than 100 nCi/g.

Thus, in the United States, LLWs are defined by exclusion. That is, if a waste is radioactive and it is not HLW or TRU, it is LLW. Uranium mill tailings are also considered a separate category from LLW.

DOE Order 5820.2A also recognizes the greater-than-Class C (GTCC) waste category. This category includes wastes which exceed the maximum allowable limits for Class C waste as defined by the NRC in 10 CFR 61.55. The DOE equivalent of that waste must be handled as special-case wastes.

Disposal of these wastes in near-surface facilities must be justified by a waste-specific PA through the National Environmental Policy Act (NEPA) process with concurrence from designated DOE-HQ officials. Disposal of GTCC wastes in near-surface disposal facilities has not been done to date.

Waste disposal requirements (waste form stabilization, packaging, burial depth, and barriers) for specific waste types and for specific waste compositions (fission products, induced radioactivity, uranium, thorium, and radium) are developed through the performance assessment model as necessary to meet the performance objectives of DOE Order 5820.2A. In the course of this process, each DOE disposal site may develop site-specific waste classification limits, if they are found useful in determining how specific wastes should be stabilized and packaged for disposal. LLW classification requirements may be imposed by each DOE site in their waste acceptance criteria.

Each site implements waste acceptance requirements as deemed necessary to segregate the waste so that handling, stabilization, and disposal requirements can be imposed to meet disposal performance objectives. For instance, Hanford classifies its LLW as Category 1 or 3 wastes based on an activity limits table; Category 3 waste requires stabilization. The waste acquisition criteria for Savannah River E-Area vaults places isotope-specific limits on waste received by the facility. Oak Ridge SWSA-6 requires that generators identify and segregate waste into categories that include fissile waste material (based on a isotope limit table), very low activity waste, contact-handled solid low-level waste (SLLW), remote-handled SLLW, biological waste, asbestos waste, and naturally occurring and accelerator-produced radioactive material.

8.0 Waste Generation

A wide variety of radionuclides are found in DOE LLW. Uranium isotopes and their daughters dominate in the conversion, enrichment, and fuel fabrication steps of the nuclear fuel cycle. Reactor operations produce LLW containing mostly activation products and fission products. By the end of 1993, approximately 66 percent of the total cumulative volume of waste disposed in the United States resulted from DOE activities. The remaining 34 percent resulted from domestic commercial activities, governed by the NRC.[4]

The DOE has disposed of approximately 3,000,000 m^3 of waste through the year 1995. Total DOE projected waste volumes for upcoming years vary from 44,200 to 146,200 m^3 per year. Total cumulative disposal volumes from the programs inception are 3,579,000 m^3 through the year 2000 and 6,076,000 m^3 through the year 2030.[4]

Eighty-four percent of DOE's LLW volume is located at six defense nuclear sites with operating land burial facilities.[5] Listed in the order of decreasing volumetric inventory, these are: the Savannah River Site (SRS), Hanford, Nevada Test Site (NTS), Y-12 and the Oak Ridge National Laboratory (ORNL), Los Alamos National Laboratory (LANL), and the Idaho National Engineering Laboratory (INEL). The Fernald Environmental Management Project does not have an operating LLW disposal facility, however, the volume of its LLW waste stored on-site represents 12 percent of the DOE total. The waste mostly consists of material contaminated with uranium/thorium. The remainder of the waste volume is located at a number of smaller DOE sites.

9.0 Waste Stabilization

DOE Order 5820.2A states:

(1) Wastes shall be treated by appropriate methods so that the disposal site can meet the performance objectives established in the Order.

(2) Waste treatment techniques such as incineration, shredding, and compaction to reduce volume and provide more stable waste forms shall be implemented as necessary to meet performance requirements. Use of waste treatment techniques to increase the life of the disposal facility and improve long-term facility performance by improved site stability and reduction of infiltrating water is required to the extent that it is cost-effective.

Individual DOE sites establish waste form stabilization requirements based on site-specific technical analysis and PA. Thus, any waste acceptance criteria and associated waste form requirements found necessary to limit public or hypothetical inadvertent intruder exposure are established on a site-specific basis, and are based on calculations of dose under a credible, site-specific scenario in the facility site-specific performance assessment. Site-specific waste classification systems often go hand-in-hand with site-specific waste conditioning and stabilization requirements since the classification system is generally designed to group wastes with similar hazards.

The Order does impose some generic requirements on wastes for LLW disposal with the stated intention to improve stability of the disposal site or to facilitate handling and protection of the health and safety of personnel at the disposal sites. These include:

(a) Waste must not be packaged for disposal in cardboard or fiberboard boxes, unless such boxes meet DOE requirements and contain stabilized waste with a minimum of void space. For all types of containers, void spaces within the waste and between the waste and its packaging shall be reduced as much as practical.

(b) Liquid wastes, or wastes containing free liquid, must be converted into a form that contains as little freestanding and noncorrosive liquid as is reasonably achievable, but, in no case, shall the liquid exceed 1 percent of the volume of the waste when the waste is in a disposal container, or 0.5 percent of the volume of the waste processed to a stable form.

(c) Waste must not be readily capable of detonation or of explosive decomposition or reaction at normal pressures and temperatures, or of explosive reaction with water.

(d) Waste must not contain, or be capable of generating, quantities of toxic gases, vapors, or fumes harmful to persons transporting, handling, or disposing of the waste. This does not apply to radioactive gaseous waste packaged as identified in paragraph (e) below.

(e) Waste in a gaseous form must be packaged at a pressure that does not exceed 1.5 atmospheres at 20 degrees C.

(f) Waste must not be pyrophoric. Pyrophoric materials contained in waste shall be treated, prepared, and packaged to be nonflammable.

10.0 Safety Assessment

DOE established functional criteria (performance objectives) in DOE Order 5820.2A for each site to use as the basis for design and operation of LLW disposal sites. The performance objectives stated in Section 5.0 include public exposure and environmental release limits and allowable effective dose equivalent limits for a hypothetical inadvertent intruder. Primary among the DOE performance objectives is a requirement that the site not release radioactive material into the environment in concentrations that would result in an annual effective dose equivalent exceeding 25 mrem to any member of the general population. The Order also includes a performance objective for allowable exposure to a hypothetical inadvertent intruder. The inadvertent intruder performance objective is based on a hypothetical scenario, not an expected scenario. The scenario is intended to be used as a design mechanism to ensure that disposal facility designers provide defense-in-depth design considerations regarding long-term waste stability. The waste should provide acceptable characteristics under potential future environmental and administrative control conditions.

The performance objectives in DOE Order 5820.2A do not specify a time period over which they are to be applied. Facility-specific performance assessments written to date have used 10,000 years and recognized the time of peak dose. The DOE has considered specifying the period of performance for analyzing disposal facility performance. Times from 1,000 to 10,000 years have been proposed but no formal policy has been approved.[6] Recent DOE thinking is leaning towards specification of 1,000 years as the "time of compliance" to be addressed by DOE LLW disposal facility designs and facility performance assessment.

Each site is directed by DOE Order 5820.2A to prepare and maintain a site-specific radiological PA for the disposal of waste for the purpose of demonstrating compliance with the performance objectives. A formal review of the PA is performed by a DOE-HQ-established PA Peer Review Panel prior to being submitted to DOE-HQ for authorization for disposal. The Order requires sites to use monitoring measurements, where practical, to evaluate actual and prospective performance and to evaluate and modify the models used in the PA.

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