ELECTRIC POWER RESEARCH INSTITUTE

Basis for National and International Low Activity and Very Low Level Waste Disposal Classifications

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Basis for National and International Low Activity and Very Low Level Waste Disposal Classifications

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This document does **NOT** meet the requirements of
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PRODUCT DESCRIPTION

In order to determine whether the Very Low Level Waste (VLLW) category would be a viable option in the United States, European and U.S. experiences were reviewed in detail.

Background

Over the last 15 years, there has been a growing awareness of the need for alternative disposal options for low activity waste (LAW) in the United States and internationally. LAWs are generated by operating nuclear power plants (NPPs) and to a much greater degree by decommissioning NPPs. To address the disposal of LAW, in 2009 the International Atomic Energy Agency (IAEA) published Safety Guide No. GSG-1, adding VLLW to its radioactive waste categories. In the same report, IAEA stated that this waste category does not need a high level of containment and is suitable for near-surface landfill disposal facilities. France's and Spain's regulators incorporated the VLLW category into their nuclear programs and have successfully operated VLLW facilities since 2003 and 2006, respectively.

In the United States, a number of utilities have received approval for alternative disposal procedures under 10 CFR 20.2002 for disposal of their waste at Resource Conservation and Recovery Act (RCRA) hazardous waste disposal facilities (RCRA Subtitle C facilities). Usually the next approval required is from the state agency responsible for regulating the RCRA facility being used. In Tennessee, five radioactive waste material processors have successfully used the state's Bulk Survey for Release (BSFR) Program to dispose of BSFR in four municipal disposal facilities (RCRA Subtitle D facilities).

Objectives

- To research waste disposal categories and methods both domestically and internationally
- To compare the different approaches
- To perform a cost-benefit analysis of a domestic VLLW application

Approach

The project reviewed European and U.S. reports, regulations, and papers to determine how VLLW, LAW, and BSFR materials were defined, managed, and disposed of. This was done by addressing nine specific questions aimed at gathering information that was pertinent to understanding how each of these approaches was undertaken and, finally, whether any one approach or a combination of approaches would best meet the needs of the nuclear industry. In addition, VLLW activity limits were applied to decommissioning waste volumes to determine the potential cost impact of implementing a VLLW category in the United States.

Results

The cost-benefit analysis shows that the U.S. nuclear power industry would benefit from adding a VLLW category to the current waste categorization system because the potential cost savings are conservatively estimated to be over $6 billion (in 2011 U.S. dollars) through 2056 without escalation and assuming that no new reactors are built.

Implementing a VLLW category is within the bounds of what is technically and operationally feasible at operating and decommissioning NPPs in the United States. There are some differences in the operation of hazardous waste disposal facilities used in Europe for VLLW and RCRA facilities expected to be used in the United States:

- European VLLW facilities accept only VLLW that may or may not be hazardous as well. In the United States, commercially operated RCRA Subtitle C facilities accept LAW and hazardous wastes.
- During the construction and operation of a VLLW disposal trench, both France and Spain use a moveable tent to preclude the infiltration of water. In the United States, tenting devices are not used at a RCRA site, but a soil cover is applied at the end of each day.
- France has located its VLLW disposal facility approximately 2 kilometers from its operating low and intermediate level waste (LILW) disposal facility. Spain has located its VLLW facility on the same property as its operating LILW disposal facility. RCRA facilities in the United States (Subtitle C facilities for hazardous waste) are usually remotely located.

Applications, Value, and Use

The need for cost-effective disposal of LAW was identified in the 1990s, and that need continues to grow as cost-effective disposal options are limited. Since10 CFR 20.2002 was issued, the Nuclear Regulatory Commission (NRC) has received an increasing number of applications to dispose of LAW at RCRA hazardous waste disposal facilities. The review of how Spain and France have implemented a VLLW category supports the conclusion that hazardous waste facilities (RCRA Subtitle C) in the United States can safety accept VLLW with the assurance of regulatory safeguards that are sufficient for both hazardous and VLLW. Additionally, the analysis of the operating and decommissioning waste volume reductions that would occur and the resulting cost savings, more than $6 billion (net present value), reflect the potential benefit to the industry of implementing a VLLW category.

There is significant evidence to support implementing a VLLW classification in the United States, supported by data gathered from European VLLW facilities as well as the U.S. NRC's 10 CFR 20.2002 application process for an alternative disposal procedure for LAW disposal at RCRA Subtitle C hazardous waste disposal facilities.

Keywords

10 CFR 20.2002
Bulk Survey For Release (BSFR)
Low activity waste (LAW)
Low level waste
Resource Conservation and Recovery Act (RCRA) disposal facility
Very low level waste (VLLW)

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INTRODUCTION AND CONCLUSIONS

Overview

Countries with active nuclear power programs started to grapple with significantly increasing volumes of low activity radioactive waste from decommissioning projects starting in the early 1990s. The International Atomic Energy Agency (IAEA), tasked with facilitating peaceful applications of nuclear technology develops nuclear safety standards and guides by bringing together nuclear experts from Member States. These safety guides are developed for the benefit of all nations using nuclear power. In 2009 the IAEA published their fourth and most comprehensive radioactive waste classification system, Safety Guide No. GSG-1.

This new waste classification system defined six waste categories,

1. Exempt Waste (EW),
2. Very Short Lived Waste (VSLW),
3. Very Low Level Waste (VLLW),
4. Low Level Waste (LLW),
5. Intermediate Level Waste (ILLW), and
6. High Level Waste (HLW).

In addition, Safety Guide No. GSG-1 provides a link between each waste class and the specific disposal options that would ensure the safety of the public. This is key because Very Low Level Waste was defined as waste that is not necessarily Exempt Waste and does not need a high level of containment. It could be disposed of in a near surface landfill. The IAEA defined the activity concentrations for Exempt Waste (EW) and suggested that VLLW could have from 10 to 100 times greater activity concentrations than Exempt Waste. This provided countries with a starting point for defining Exempt Waste and Very Low Level Waste. In the U. S., Exempt waste is comparable to wastes destined for RCRA Subtitle D facilities such as the Tennessee BSFR program whereas VLLW is more comparable to 10 CFR 20.2002 wastes disposed of in RCRA Subtitle C hazardous waste disposal facilities.

VLLW Handling by Other Countries

For countries like France where there is no universal “clearance” [1], and Spain where “clearance” is approved on a case-by-case basis [2], VLLW is a reasonable alternative for handling large volumes of low activity waste. France and Spain ultimately incorporated VLLW into their radioactive waste classification system and by 2003 and 2006 respectively were disposing of VLLW in hazardous waste disposal facilities. These facilities accept VLLW from the nuclear industry, medical and research facilities. The hazardous disposal facilities are located on the same site or within about two kilometers from each country's LILW disposal facility.

Alternative Disposal Requests

Prior to 2000, the majority of NRC's 10 CFR 20.2002 requests for alternate disposal were requesting disposal of the wastes onsite or at another property location. Between 2000 and 2006, the NRC received 20 requests for alternate disposal. 85% of those requests were for off site disposal. Increasingly, RCRA Subtitle C hazardous waste facilities were the off site, alternative requested. In those instances state regulators responsible for regulating the RCRA disposal facility were involved.

Tennessee's Bulk Survey for Release Program

Tennessee's Bulk Survey for Release (BSFR) Program is offered by four radioactive material processors in Tennessee. After processing, operating and decommissioning power plant wastes are disposed of in Tennessee Class 1 municipal landfills (equivalent to RCRA Subtitle D) at activity levels the Tennessee Department of Radiological Health's describes as extremely low activity waste. The individual radionuclide levels are derived by the regulator for each disposal facility using a 0.01mSv/y exposure scenario for the maximally exposed individual - the Resident Farmer.

Review and Analysis

This report reviewed what was being done in France, Spain, Tennessee and at RCRA disposal facilities in the U. S. under 10 CFR 20.2002 to better address what the similarities and differences were between European and U.S. programs for handling wastes with low activity concentrations. An analysis was made of the volumes and the utility industry savings that could be achieved if a VLLW category were adopted in the U.S.

Estimated Savings in U.S. Dollars

Based on the projected reduction in LLW volumes and the cost differential between LLW disposal and VLLW disposal, it is conservatively estimated that the industry could save over six (6) billion dollars in disposal costs expressed as 2011 dollars through 2056 without escalation and assuming no new reactors are built. A VLLW category could also provide significant savings in establishing the decommissioning fund basis for new reactors.

Purpose of the Report

The purpose of this report is to:

- propose a definition of the activity limits of VLLW
- make a case for pursuing the disposal of these wastes in operating RCRA Subtitle C hazardous waste facilities.

Supporting Sections Summary

Section 2 presents a summary of the activity concentration limits for VLLW as suggested by IAEA and currently used by France and Spain, the range of BSFR limits for Tennessee and the activity limits for wastes approved for alternative disposal (10 CFR 20.2002) in a RCRA hazardous waste disposal facility in Idaho.

Section 3 reviews the IAEA's latest characterization program, the linking of waste types to disposal types, and their basic guidance for developing a new disposal facility, including the need for dose modeling to develop a site's Waste Acceptance Criteria and successfully meeting the Waste Acceptance Criteria for a disposal facility.

Section 4 reports on France's VLLW program including their definition of VLLW, their process for calculating activity levels in individual and batches of packages, including operation of the VLLW facility, site verification requirements, packaging and transportation requirements, etc.

Section 5 reports on Spain's VLLW program including the definition of VLLW, the process for calculating activity levels in individual and batches of packages, the onsite waste verification process, operation of the VLLW facility, packaging and transportation requirements, etc.

Section 6 reports on the use of NRC's 10 CFR20.2002 request for alternate disposal of Low Activity Waste in RCRA hazardous waste disposal facilities. The section addresses some of the process for using this regulation, Waste Acceptance Criteria for disposal of LAW in RCRA facilities, and site specific design of these facilities along with safety assessments that are performed for LAW disposal.

Section 7 reports on Tennessee's Bulk Survey for Release (BSFR) Program. The section addresses the responsibilities of the regulators (Tennessee's Dept. of Radiological Health and the State's Division of Solid Waste Management), the waste processors who hold the permits and the landfill operators where the waste is ultimately disposed.

Section 8 summarizes the findings of Sections 2 through 7.

Section 9 estimates the volumes of LLW that potentially qualifies as VLLW. Estimates of the total volume of Class A, B and C LLW and VLLW were prepared for the remaining operating life and/or decommissioning period for plants currently operating and plants permanently shutdown.

Section 10 estimates the waste disposal cost savings that could result should a VLLW classification become available.

Conclusions

Development and Use of the VLLW Category

IAEA

IAEA published a new waste categorization system in 2009[1], It address the need for a more comprehensive radioactive waste classification system, linked each waste category to the appropriate form of disposal based on the hazard of the waste and provided specific activity limits for the category of VLLW.

1. VLLW's activity level does not require a high level of containment or isolation and it usually has very limited concentrations of longer lived radionuclides.
2. The VLLW activity concentrations could be one or two orders of magnitude greater than the EW concentrations.
3. Engineered "near surface" landfills can have limited regulatory control and may also contain other hazardous waste
4. Note that radiation protection for workers and general public will be greater than those for Exempt Waste but less protection is needed compared to higher classes of radioactive waste (i.e., LLW).

VLLW Management in France and Spain

The similarities between both countries approach to VLLW management:

5. Accept only VLLW at the facilities.
6. Construct and operate disposal cells/trenches under a movable tent.
7. Require VLLW be submitted for disposal in batches.
8. Characterizing the waste is key and relies heavily on process knowledge.

BSFR and LAW Disposal in the U.S.

- Both programs are working successfully.
- Tennessee's BSFR Program is in place and case-by-case reviews are no longer required.

The activity concentrations associated with BSFR materials are closer to what is typically defined as "free release" in the U.S. However the dose limit to the maximally exposed individual, which the activity concentrations are based on, is the same as those used for VLLW in Spain (0.01mSv/y).

The NRC's 10 CFR 20.2002 request for alternate disposal, to dispose of LAW, in a RCRA hazardous waste disposal facility uses a slightly higher exposure limit (0.05mSv/y is the maximum exposure limit) when evaluating the acceptability of the waste for RCRA disposal.

The Impact of Adding a VLLW Category in the U.S.

Adding a VLLW category in the U.S. is technically possible and would have the following positive effects.

Change in Volume

It is projected that a total of 70.5 Million ft^3 of Class A waste (Table 9-3) could be reclassified to Very Low Level Waste with the institution on this new waste classification. As discussed in Section 9 this total is likely below what the actual reduction would be due to the conservatisms used in determining the estimate.

From these potential volumes, that can be disposed at a lower cost, we can calculate the expected Disposal cost saving.

Cost Savings Potential of $6 Billion

From this estimated reduction in volumes, a projected total savings of $6.2 Billion (2011 dollars without escalation through 2056) will result from the addition of a U.S. version of a VLLW classification. (Figure 10-1) The assumptions used to determine the waste volume and cost savings are conservative and likely an underestimate.

Summary

There are European countries currently using the Very Low Level Waste category. Their approaches are based on sound practices that specifically address the continuing safety of the public. The equivalent of VLLW disposal in RCRA facilities has been approved in the U. S. under 10 CFR 20.2002. If the U.S. were to implement a VLLW category where the wastes were disposed of in RCRA Subtitle C Disposal Facilities, the nuclear industry and the public would benefit from lower costs and the public would remain adequately protected.

References for Section 1

1. French Nuclear Safety Authority (ANS), Research facilities and waste department, *Release of radioactive materials from regulatory requirements – Provisions for exemption and clearance*, French policy for exemption and clearance, Wiesbaden, Germany, 21-23, September, 2009 (French_policy_for_exemption_and_clearance_-_C._Clemente.pdf) [publication]
2. International Atomic Energy Agency (IAEA) Regional Workshop on the Release of Sites and Building Structures, *Management of very low level radioactive waste in Europe-application of clearance* (and the alternatives), F. Borrmann, sat science GmbH, Karlsruhe, Germany, September 27, - October 1, 2010 (management-of-low level waste) [workshop]

2
PROPOSAL FOR NEW CLASSIFICATION: VLLW

Brief

In subsequent Sections of this report the experiences with the disposition of VLLW and LAW are described. In order to illustrate the radionuclide concentration limits that have applied for those disposals, the following is a summary of those limits.

Summary of Radionuclide Concentration Limits

Table 2-1 shows a summary of the radionuclide concentration limits for:

- The VLLW classification currently utilized in certain countries in Europe
- VLLW limits based on guidance from the International Atomic Energy Agency
- Release limit utilized for the disposal of radioactive waste (called LAW in this report) at certain RCRA Subtitle D landfills in Tennessee
- Limits for disposal under at NRC 20.2002 exemption for decommissioning waste from the Connecticut Yankee and Yankee Rowe plants (From Appendix B)
- Concentrations below which the waste is exempt from Department of Transportation shipping regulations

Table 2-1
Summary of Limits on the Transportation and Disposal of VLLW

Radionuclide	IAEA Bq/g	France Bq/g	Spain Bq/g	Tennessee Processors Bq/g	Yankee Rowe 20.2002 Exemption for WCS Bg/g	Connecticut Yankee 20.2002 Exemption for US Ecology Idaho Bg/g	U.S.DOT Exempt. Limits Bq/g (pCi/g)	Possible U.S. VLLW Limits Bq/g (pCi/g)
H-3	10,000	1,000	1,000	0.074 to 208	N/A	48,000	1,000,000 (27,000,000)	1,000 (27,000)
C-14	100	1,000	1,000	0.02 to 1	N/A	16.2	10,000 (270,000)	100 (2,000)
Co-60	10	10	10	0.05 to 3	20	1	10 (270)	10 (270)
Ni-63	10,000	1,000	1,000	6 to 136	N/A	Note 2	100,000 (2,700,000)	1,000 (27,000)
Sr-90	100	1,000	1,000	0.01 to 0.5	N/A	Note 2	100 (2,700)	100 (2,700)
Cs-137	10	10	30	0.01 to 1	100	3.4	10 (270)	10 (270)
Eu-152	10	10	N/A	0.3 to 1.8	N/A	Note 2	10 (270)	10 (270)
Pu-241	1,000	1,000	1,000	3 to 134	N/A	Note 2	100 (2,700)	100 (2,700)
Pu-239,	10	10	10	0.1 to 4	N/A	Note 2	1 (27)	1 (27)
Pu-240	10	10	10	0.1 to 4	N/A	Note 2	1 (27)	1 (27)
Am-241	10	10	10	0.1 to 4	N/A	Note 2	1 (27)	1 (27)

Notes: 1. N/A - Not Available
2. These radionuclides have very low dose consequences but due to their ratio to Co-60 and/or Cs-137 would be limited by the concentrations of those radionuclides present

Basis of Limits for U.S. VLLW Possible Classification Criteria

The final column in Table 2-1 are possible limits that could apply if a new VLLW classification was established in the US based on approved limits in other countries depicted in this report. These possible limits have been determined so as to meet all of the following criteria:

- Are no higher (lower for some of the radionuclides) than the French and Spanish VLLW limits and the IAEA guidance on VLLW limits
- Are no higher (lower for some of the radionuclides) than the U.S. DOT exempt limits for transport of radioactive material

For the remainder of this report the values in the final column of Table 2-1 will be referred as the "possible VLLW limits proposed in this report".

3
INTERNATIONAL ATOMIC ENERGY AGENCY'S (IAEA'S) DEFINITION OF VLLW

Introduction

The IAEA was set up in 1957 as the world's center of cooperation in the nuclear field within the United Nations. The Agency serves as an intergovernmental forum, working with its Member States and multiple partners worldwide to encourage peaceful applications of nuclear technology. In particular the IAEA develops nuclear safety standards that promote high levels of safety in the application of nuclear power [1].

Over the past 39 years the IAEA has developed radioactive waste classification systems through its Safety Standards Series. Briefly, the IAEA safety standards reflect an international consensus on what constitutes a high level of safety for protecting people and the environment from harmful effects of ionizing radiation. The Safety Standards Series has three categories:

> *"**Safety Fundamentals** (blue lettering) present basic objectives, concepts and principles of safety and protection in the development and application of nuclear energy for peaceful purposes.*
>
> ***Safety Requirements** (red lettering) establish the requirements that must be met to ensure safety. These requirements, which are expressed as 'shall' statements, are governed by the objectives and principles presented in the Safety Fundamentals.*
>
> ***Safety Guides** (green lettering) recommend actions, conditions or procedures for meeting safety requirements. Recommendations in Safety Guides are expressed as 'should' statements, with the implication that it is necessary to take the measures recommended or equivalent alternative measures to comply with the requirements"* [3]. Reproduced with permission from the International Atomic Energy Agency, Near Surface Disposal of Radioactive Waste Safety Requirements, IAEA Safety Standards Series WS-R-1, IAEA, Vienna (1999).

The first three classification systems, developed between 1970 and 1994, had fewer waste categories. Each waste category was defined based on its activity content. Member States felt these classification schemes needed to cover all types of radioactive waste and provide a direct link to potential disposal options. Member States felt this would expand the use and application of the resulting waste classification system [4]

International Atomic Energy Agency's (IAEA's) Definition of VLLW

In 2009 IAEA published Safety Guide No. GSG-1 with a classification system that addressed all types of radioactive waste. The result was generic waste classes that could be defined in terms of the specific properties of the waste . The wastes were classified according to general concepts. From these general concepts, the IAEA, developed specific criteria for the different types of waste [4]. The outcome provides regulators with a standard way of managing radioactive waste. In addition the 2009 Safety Guide No. GSG-1 provides the generic linkage between different classes of waste and disposal options. The IAEA points out, that *"…notwithstanding such generic linkage, the suitability of waste for disposal in a particular disposal facility is required to be demonstrated by the safety case and supporting safety assessment for the facility [IAEA5],"* [4]. Reproduced with permission from the International Atomic Energy Agency. *Classification of Radioactive Waste: General Safety Guide*, IAEA Safety Standards Series GSG-1, IAEA, Vienna (2009).

The specifics of this characterization system, which includes Very Low Level Waste (VLLW), are quoted directly from the IAEA Safety Standards, No. GSG-1 [4].

> *2.1 … "A comprehensive range of waste classes has been defined and general boundary conditions between the classes are provided. More detailed quantitative boundaries that take into account a broader range of parameters may be developed in accordance with national programmes and requirements. In cases when there is more than one disposal facility in a State, the quantitative boundaries between the classes for different disposal facilities may differ in accordance with scenarios, geological and technical parameters and other parameters that are relevant to the site specific safety assessment.*
>
> *2.2. In accordance with the approach outlined in the Appendix, six classes of waste are derived and used as the basis for the classification scheme:*
>
> ***(1) Exempt waste (EW):*** *Waste that meets the criteria for clearance, exemption or exclusion from regulatory control for radiation protection purposes as described in Ref. [IAEA6]. The term 'exempt waste' has been retained from the previous {1994} classification scheme for consistency; however, once such waste has been cleared from regulatory control, it is not considered radioactive waste.*
>
> ***(2) Very short lived waste (VSLW****): Waste that can be stored for decay over a limited period of up to a few years and subsequently cleared from regulatory control according to arrangements approved by the regulatory body, for uncontrolled disposal, use or discharge. This class includes waste containing primarily radionuclide's with very short half-lives often used for research and medical purposes.*
>
> ***"(3) Very low level waste (VLLW):*** *Waste that does not necessarily meet the criteria of EW, but that does not need a high level of containment and isolation and, therefore, is suitable for disposal in near surface landfill type facilities with limited regulatory control. Such landfill type facilities may also contain other hazardous waste. Typical waste in this class includes soil and rubble with low levels of activity concentration. Concentrations of longer lived radionuclide's in VLLW are generally very limited."* [4] Reproduced with permission from the International Atomic Energy Agency.

Classification of Radioactive Waste: General Safety Guide, IAEA Safety Standards Series GSG-1, IAEA, Vienna (2009).

(4) Low level waste (LLW): *Waste that is above clearance levels, but with limited amounts of long lived radionuclides. Such waste requires robust isolation and containment for periods of up to a few hundred years and is suitable for disposal in engineered near surface facilities. This class covers a very broad range of waste. LLW may include short lived radionuclides at higher levels of activity concentration, and also long lived radionuclides, but only at relatively low levels of activity concentration.*

(5) Intermediate level waste (ILW): *Waste that, because of its content, particularly of long lived radionuclides, requires a greater degree of containment and isolation than that provided by near surface disposal. However, ILW needs no provision, or only limited provision, for heat dissipation during its storage and disposal. ILW may contain long lived radionuclides, in particular, alpha emitting radionuclides that will not decay to a level of activity concentration acceptable for near surface disposal during the time for which institutional controls can be relied upon. Therefore, waste in this class requires disposal at greater depths, of the order of tens of metres to a few hundred metres.*

(6) High level waste (HLW): *Waste with levels of activity concentration high enough to generate significant quantities of heat by the radioactive decay process or waste with large amounts of long lived radionuclides that need to be considered in the design of a disposal facility for such waste. Disposal in deep, stable geological formations usually several hundred metres or more below the surface is the generally recognized option for disposal of HLW.*

2.3.Quantitative values of allowable activity content for each significant radionuclide will be specified on the basis of safety assessments for individual disposal sites (which is outside the scope of this Safety Guide)." [4] Reproduced with permission from the International Atomic Energy Agency. *Classification of Radioactive Waste: General Safety Guide,* IAEA Safety Standards Series GSG-1, IAEA, Vienna (2009).

Paragraph 2.3 above refers to the fact that allowable activity content for each significant radionuclide, in the waste, can be derived using either the generic scenarios used in IAEA Safety Guide No. RS-G-1.7 [5]. Or a pathway analysis designed specifically to take into account the natural and engineered features of the disposal facility. This is done using a site specific pathway analysis model. The model determines how much of each specific nuclide can be placed in the disposal facility, while keeping exposure limits to the worker, the public and the inadvertent intruder to activity levels specified by the regulator. Based on projected quantities of waste to be disposed, the acceptable activity limit per waste package is determined.

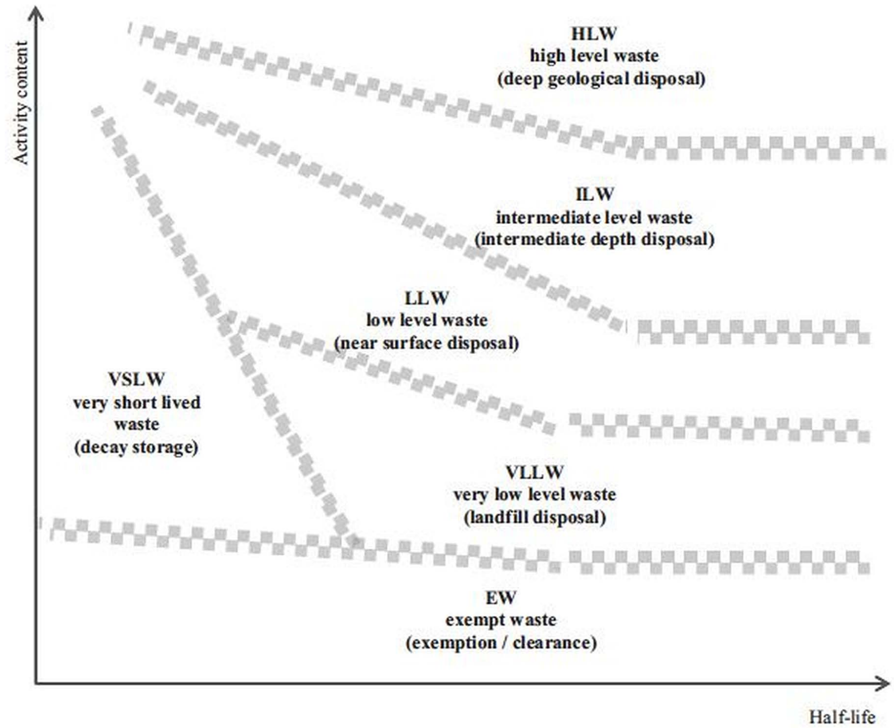

Figure 3-1
Illustration of IAEA's 2009 Waste Characterization System [4]. Reproduced with permission from the International Atomic Energy Agency. *Classification of Radioactive Waste: General Safety Guide,* IAEA Safety Standards Series GSG-1, IAEA, Vienna (2009).

Specific numbers for each category of waste are not provided in the body of the IAEA Safety Guide No. GSG-1. [4]. In total the report addresses VLLW in six relatively short paragraphs and three figures. The following discussion uses all of the information contained in those paragraphs and more often than not quotes them directly. However the following sub sections have been addressed using relevant information from other IAEA reports that apply to all radioactive wastes. These include elements of a waste characterization program, relevant disposal site pathway analysis, waste acceptance criteria development and waste package verification.

How is VLLW Defined in Terms of Radioactivity and Chemical Content?

Radioactivity

> *Very low level waste (VLLW): Waste that does not necessarily meet the criteria of EW, but that does not need a high level of containment and isolation and, therefore, is suitable for disposal in near surface landfill type facilities with limited regulatory control. Such landfill type facilities may also contain other hazardous waste. Typical waste in this class includes soil and rubble with low levels of activity concentration. Concentrations of longer lived radionuclides in VLLW are generally very limited* [4].

The IAEA Safety Guide No. GSG-1 [4] on page 10 reiterates that wastes with a limited hazard, where the hazard is *"...above or close to the levels for exempt waste, is termed very low level waste."* [4] Reproduced with permission from the International Atomic Energy Agency. *Classification of Radioactive Waste: General Safety Guide*, IAEA Safety Standards Series GSG-1, IAEA, Vienna (2009).

As stated above, it was not possible to provide generally valid criteria for VLLW in the IAEA Safety Guide No. GSG-1 [4], because of the need to conduct a safety assessment using either the generic scenarios used in their development of the exemption and clearance levels [5] or conducting a safety assessment for a specific disposal facility.

Chemical Content

The chemical content of VLLW is not addressed in the IAEA Safety Guide No. GSG-1, although it does note the need to understand the chemical content of the waste and the fact that the waste may be disposed of with hazardous wastes. IAEA's Safety Guide No. GS-G-3.3, [6], addresses potential hazardous components of radioactive waste. Hazardous properties identified include: a) Flammability; (b) Corrosivity; (c) Reactivity; (d) Pyrophoricity; (e) Rapid oxidation promotion; and (f) Biodegradability. The report goes on to say that these properties should be immobilized, stabilized or have more stringent container qualifications. [6]

Does the IAEA Have an Operating VLLW Disposal Facility?

As discussed in the Introduction to this Section, the IAEA's mandate encourages the agency to develop technical reports and publish workshop proceedings, etc., that address a wide variety of radioactive waste issues. IAEA's mandate does not include operating a radioactive waste disposal site. In the event that the IAEA were to develop a disposal facility for VLLW with a Member State, the IAEA Safety Guides would have to be followed. IAEA is required to follow its Safety Guides anytime it participates in a Member State project.

IAEA Safety Standard

IAEA SAFETY STANDARD No. GSG-1 illustrates how the report links waste class to disposal options using Figure 3.2.

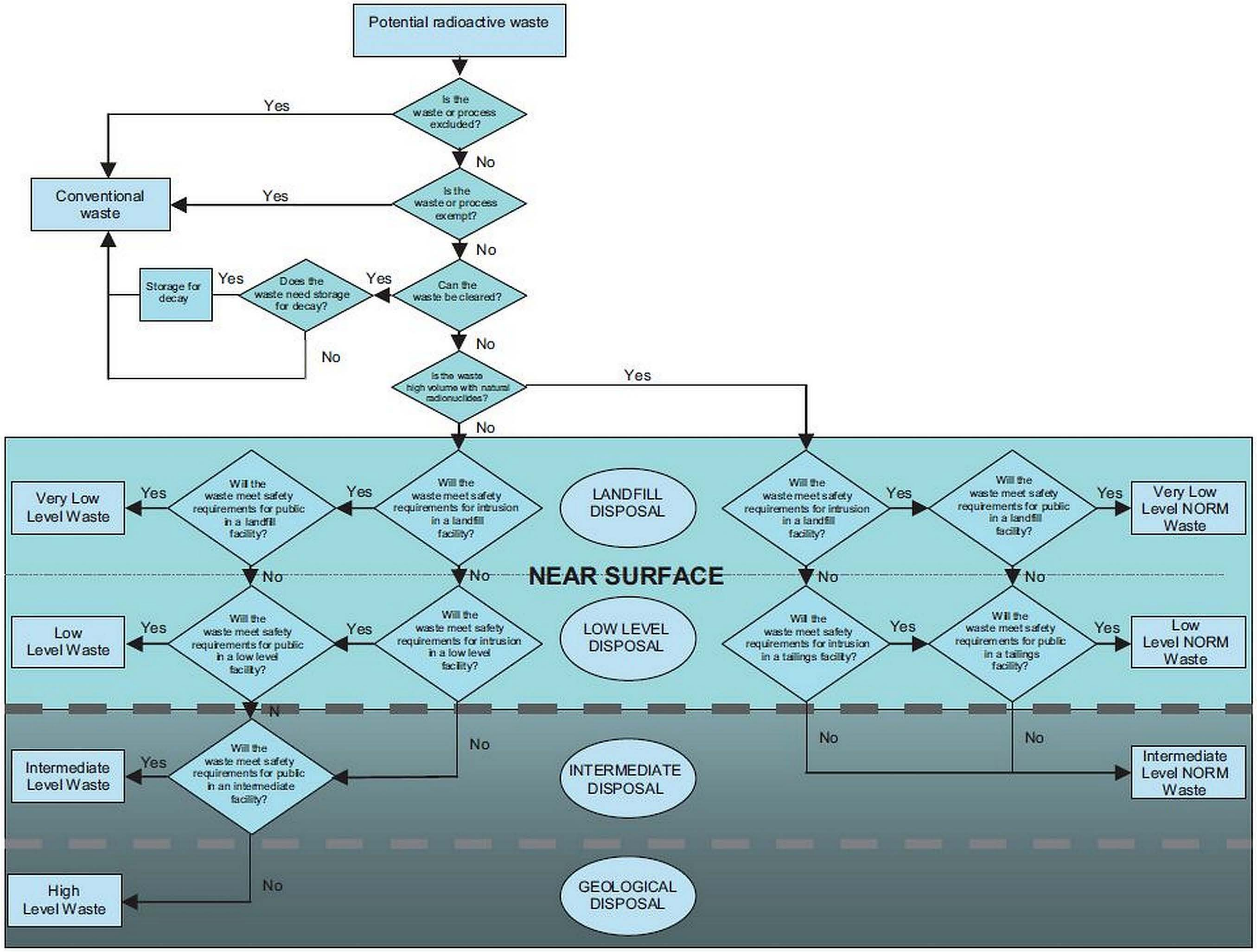

Figure 3-2
Illustrates how IAEA's Classification Scheme can be used [4]. Reproduced with permission from the International Atomic Energy Agency. *Classification of Radioactive Waste: General Safety Guide*, IAEA Safety Standards Series GSG-1, IAEA, Vienna (2009).

In the definition of VLLW, the IAEA identifies the suitability of near surface landfills for the disposal of these wastes. The report goes on to say what type of landfill is suitable for these wastes. "An adequate level of safety for VLLW may be achieved by its disposal in engineered surface landfill type facilities." These facilities can range from simple covers to more complex systems and usually require both active and passive institutional controls [4]. Even though the U. S. and South Africa have used near surface LLW disposal facilities (in arid zones), the use of near surface landfills for VLLW is particularly relevant to non-arid conditions, such as those found in the North Eastern U. S. and Europe.

The report finishes this discussion with the following statement:

> *Nevertheless, it is expected that with a moderate level of engineering and controls, a landfill facility can safely accommodate waste containing artificial radionuclides with levels of activity concentrations one or two orders of magnitude above the levels for exempt waste, for waste containing short lived radionuclides and with limited total activity. This applies as long as expected doses to the public are within criteria established by the regulatory body*. [4] Reproduced with permission from the International Atomic Energy Agency. *Classification of Radioactive Waste: General Safety Guide*, IAEA Safety Standards Series GSG-1, IAEA, Vienna (2009).

Which Nuclear Power Plant, Operating and Decommissioning, Waste Streams Qualify as VLLW?

IAEA SAFETY SERIES No. GSG-1 points out that it is not unusual for VLLW to consist of low activity soil and rubble [4]. Later in the report's Waste Classes discussion, it points out that, substantial amounts of waste are generated at operating and decommissioning nuclear facilities. The activity levels of these wastes are in the region of or slightly above those for the clearance of material from regulatory control. The management of VLLW does require radiation protection and safety be considered for potential worker and general public exposures. However the extent necessary is limited compared to waste in the higher classes such as Low Level Waste (LLW), Intermediate Level Waste (ILW) [a classification often used internationally] or High Level Waste (HLW) [4].

In Annex III of the report it goes on to say that VLLW is one of the categories, along with LLW and ILW, generated from nuclear power plants as a result of reactor operations, decontamination and decommissioning, among other fuel cycle activities. The report points out that the largest volumes from dismantling of nuclear installations will mainly be Very Low Level and Low Level Waste [4].

What is the Waste Acceptance Criteria (WAC) for VLLW?

As noted in "How is VLLW defined in terms of radioactivity and chemical content?" above, where possible using site specific safety performance analyses is an important way to develop site specific waste acceptance criteria for VLLW. However, there are generic assumptions that can be use, as IAEA did in developing the concentrations for Exempt Waste in the IAEA SAFETY GUIDE No. RS-G-1.7 [5]. Furthermore the IAEA has suggested that VLLW activity levels are on the order of 10 to 100 times higher than Exempt Waste. These VLLW concentrations can be applied when developing the WAC for a VLLW disposal facility.

It is not the intent of this section to include every element used in developing WAC but rather to attempt to present basic information the IAEA has provided, pointing out some of the more significant elements.

Overview

In the simplest terms there are three entities involved in the disposal of waste on a day- to-day basis. These are the regulator(s), the repository operator and the waste generator. In the IAEA report, the term "repository" covers the range of disposal options from a Low Level Waste disposal. In Figure 3.1 the IAEA outlines, in general terms, the responsibilities of each of these entities. Basically, the regulatory body defines the safety regulations and general waste acceptance criteria. The repository operator defines site-specific waste acceptance criteria and is responsible for authorizing the acceptance of waste packages for ultimate disposal. The waste generator follows its waste package quality assurance program thereby meeting the disposal facility's waste acceptance criteria. [7].

The regulatory body has overall responsibility for ensuring the safety of the waste disposal facility. In the broadest terms, this includes overseeing the main aspects of facility construction, operation and closure to ensure worker safety, public safety and protection of the environment. The regulator develops the overall waste acceptance criteria for each category of waste and when the facility becomes operational conducts periodic reviews to ensure that those waste acceptance criteria are being met.

The repository operator (in the case of VLLW the disposal facility operator), will use its site specific safety assessment to issue a site specific WAC that is based on the requirements of, and is in compliance with, the regulator's WAC. The site specific WAC is developed within the framework of the waste and waste packages to be accepted. Ensuring that the WAC is met will depend heavily upon the quality assurance program the generator has developed, which the disposal facility operator has reviewed and accepted.

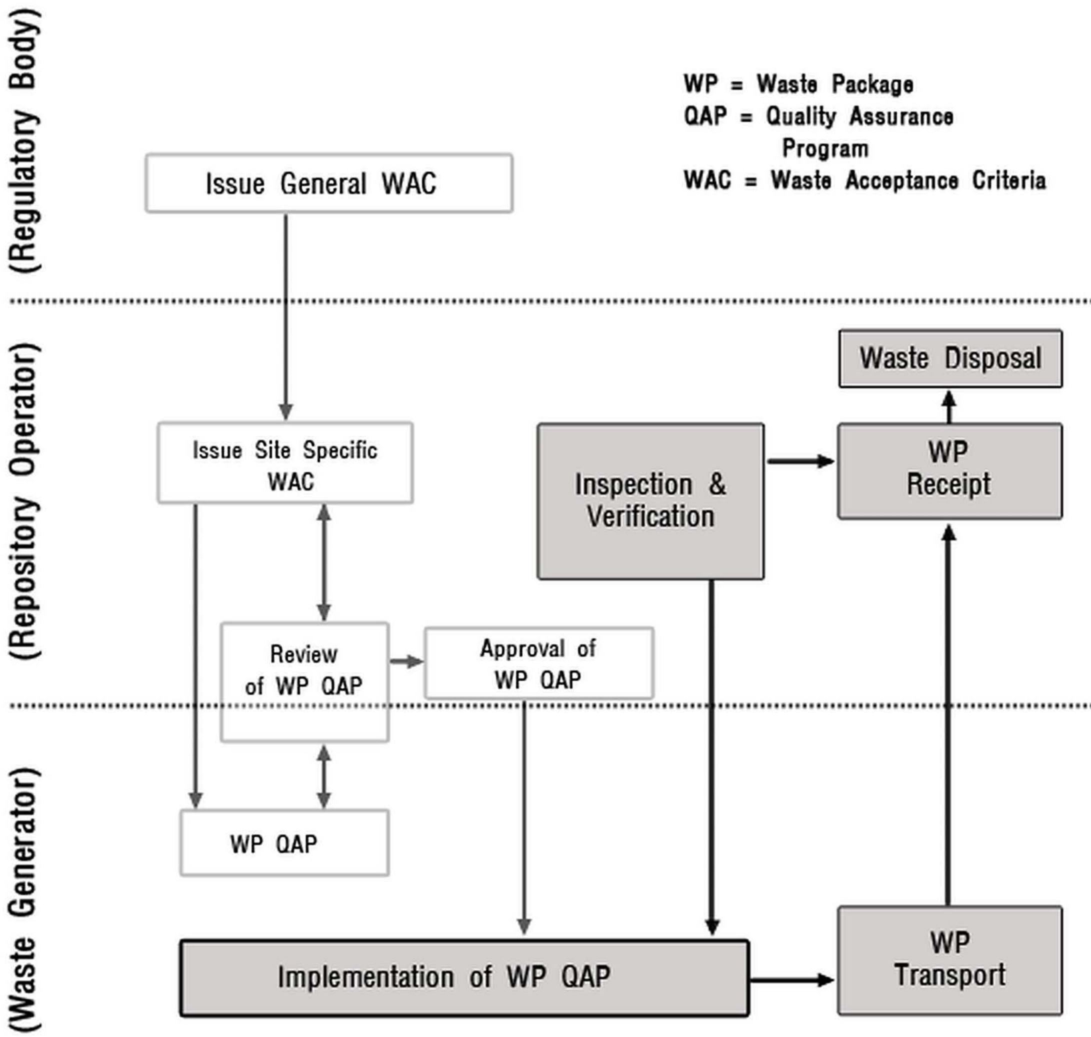

Figure 3-3
Major responsibilities of entities involved in the disposal of waste packages [7]
Reproduced with permission from the International Atomic Energy Agency, Inspection and Verification of Waste Packages for Near Surface Disposal, IAEA-TECDOC-1129, IAEA, Vienna (2000).

The waste verification portion of this figure will be discussed in "How does the disposal site verify the waste packages' radioactivity and chemical content?"

Waste Acceptance Criteria that are Site Specific

Building on the information provided in the introduction to this Section, we see that there are a number of elements that make up site specific Waste Acceptance Criteria (WAC). Some of the major elements are discussed below.

Safety Assessment

Safety assessments are used to determine what concentrations of individual nuclides can be accepted by the disposal facility to ensure that exposures to workers, the public and the inadvertent intruder will not exceed limits set by the regulators. The pathway analyses are based on the geochemical characteristics of the specific facility as well as other factors such as liners, trench caps, disposal containers, etc., used at the facility [9]. In this report IAEA does not suggest a model or models to be used in this type of analysis. The RESRAD model, developed by the US Department of Energy, is typically used for safety assessments in the United States. The calculations resulting from the RESRAD and other safety analysis models define the total activity content, for each radionuclide that can be disposed of at the site during the operation of the disposal facility. Based on the anticipated volume of waste to be disposed of at the facility, the activity limits for waste packages can be determined.

In a number of reports the IAEA provides information on types of exposure pathways to consider for safety assessments.

> *"Radiological impacts may need to be assessed for different types of exposure scenario. A typical scenario assumes that water enters the disposal units and that radionuclides are mobilized and transported by the water to a point of possible utilization by humans or other organisms.*
>
> *However, other release scenarios, for example releases due to transport in gas or erosion of the ground surface, may also be important. An additional scenario that is particularly relevant for facilities where the waste is emplaced at a limited depth and that is generally considered in the assessment of safety after the end of institutional control is inadvertent human intrusion. Various intrusion scenarios have been defined for this type of assessment, including construction activities at the repository site or a farming community becoming established above the disposal units, Assessing the radiological consequences of all these scenarios requires assumptions about the conditions of the biosphere and an adequate modeling capability.* [IAEA73, IAEA74, IAEA75, IAEA125]." [8] Reproduced with permission from the International Atomic Energy Agency, Scientific and Technical Basis for the Near Surface Disposal of Low and Intermediate Level Waste, IAEA Technical Reports Series 412, IAEA, Vienna (2002).

There are generic, accepted methodologies for developing Waste Acceptance Criteria applicable to all waste categories and the IAEA provides guidance on these, in other documents [7, 9, 10]. For this reason, there are no specific requirements for developing WAC for Very Low Level Waste.

Because the Waste Acceptance Criteria from other countries is discussed in the following Sections, a quick list of IAEA, Waste Acceptance Criteria items is provided below. These items are taken from pages 14 and 15 of IAEA-TECDOC-1537, Strategy and Methodology for Radioactive Waste Characterization, March 2007 [9].

Elements of WAC and Where They Originate

> *The waste acceptance criteria will define the requirements that the waste package must meet in view of transport, interim storage and final disposal. The requirements will be based on the following:*
>
> - *limitations of conditioning processes or facilities conditioning*
> - *process control parameters worker safety at all phases*
> - *legal requirements*
> - *transportation limits*
> - *interim storage requirements*
> - *integrated performance assessments*
> - *disposal facility performance assessments*
> - *overall quality assurance requirements (independence, testing, etc.)*[10] Reproduced with permission from the International Atomic Energy Agency Predisposal Management of Radioactive Waste: General Safety Requirements Part 5, IAEA Safety Standards Series GSR Part 5, IAEA, Vienna (2009).

When developing Waste Acceptance Criteria, certain characteristics of the waste are key to safety and environmental protection. [6].

Waste Characteristics

Elements of waste characterization important to understand are: 1) Waste Composition – including radiological and toxic/hazardous constituents. This data can be gathered either by analytical means or process knowledge. Where appropriate this includes leachability data for the radionuclides, toxic materials and gas generation rates from volatile organic compounds and other hazardous gases. 2) Chemical Instability – should be addressed either through stabilizing the waste or using appropriate containers. Properties of interest are: a) Flammability; (b) Corrosivity; (c) Reactivity; (d) Pyrophoricity; (e) Rapid oxidation promotion; and (f) Biodegradability. In addition, chemically incompatible waste forms should be carefully controlled. 3) Immobilization and/or Stabilization – hazardous constituents in waste, that are mobile should be immobilized or stabilized. "…the chemical and physical properties of the waste should be consistent with the assumptions made about the modeling of contaminant migration and transport after containers fail in the environment of the disposal facility [6]. 4) Respirable Fraction – for non monolithic waste forms, controlling the respirable fraction is important for alpha emitting waste because of the potential inhalation pathway. 5) Distribution of Activity – it is important not to use dilution to achieve the activity distribution derived through the safety assessment. [6].

These data describe the wastes, assist in quality control and ensure waste processes such as conditioning and packaging are kept within compliance levels of the Waste Acceptance Criteria and backup the quality assurance program.

Scaling Factors

Due to the mix of radionuclides in some waste streams, determining the activity content may require more sophisticated methods. Scaling factors are used to determine the concentration of difficult to measure (DTM) and impossible to measure (ITM) radionuclides by using known correlations between the DTMs and IMTs and key nuclides, such as Co-60 and Cs-137, which are easy to measure. Developing scaling factors requires the gambit of sampling, destructive analysis, modeling, non- destructive analysis, and ultimately calculation. [9].

Mixtures of Nuclides

The IAEA also provides guidance on how to analyze waste that has a mixture of radionuclides. Though the guidance is not specifically developed for VLLW it can be an integral part of the waste acceptance criteria.

> *4.6. For mixtures of radionuclides of natural origin, the concentration of each radionuclide should be less than the relevant value of the activity concentration given in Table I (Table I is contained in Safety Guide No. RS-G-1.7).*
>
> *4.7. For material containing a mixture of radionuclides of artificial origin, the following formula should be used:*

Equation 3-1
Activity Concentration

$$\sum_{i=1}^{n} \frac{c_i}{(activity \quad concentration)i} \leq 1$$

i *is an individual radionuclide [3,9,10,11]*

C_i *is the concentration (Bq/g) of the i_{th} radionuclide of artificial origin in the material,*

(activity concentration)i *is the value of activity concentration for the radionuclide i in the material and*

n *is the number of radionuclides present.*

> *4.8. For a mixture of radionuclides of both natural and artificial origin, both conditions presented in paragraphs 4.6 and 4.7 should be satisfied.* [5] Reproduced with permission from the International Atomic Energy Agency, Application of the Concepts of Exclusion, Exemption and Clearance Safety Guide, IAEA Safety Standards Series RS-G-1.7, IAEA, Vienna (2004).

New Waste and Historic Waste

New and Historic Wastes are included here because they are a consideration in the Waste Acceptance Criteria at a VLLW facility. How the wastes are verified can be very different and therefore noteworthy for inclusion in the WAC.

New Waste

With new waste, the key factors are the:

- History of the waste is known,
- Characterization is robust, and
- Process knowledge is sufficient.

This includes the step-by-step management of the waste and custody of the waste from origin through shipment.

Historic Waste

Historic wastes may not have a traceable characterization program, making it difficult to identify the process that generated the waste. Finally, the waste may consist of more than one waste stream. When the waste was generated is not necessarily the significant factor. As long as the wastes lack a sufficient characterization program and information on how they were managed, they are considered historical waste [9].

The IAEA provides additional detail on these waste types including how to properly fill in characterization gaps through historical sources, sampling, etc., in IAEA-TECDOC-1537 [9].

These are just some of the basic concepts that are addressed when formulating Waste Acceptance Criteria at the regulatory level and at the site specific (disposal facility) level. For additional details see IAEA Inspection and verification of waste packages for near surface disposal, IAEA-TECDOC-1129, Vienna, 2000 [7] and IAEA-TECDOC-1537, Strategy and Methodology for Radioactive Waste Characterization, Austria, March 2007 [9]. Keep in mind, the information in these reports was developed for Low Level and Intermediate Level Waste and may contain items that would not necessarily be required for VLLW disposed in a landfill.

How Does the Disposal Site Verify the Waste Packages' Radioactivity and Chemical Content?

The IAEA suggests that waste packages go through the generator's Quality Assurance Program process, prior to being transported to the disposal facility. As pointed out in the Waste Acceptance Criteria discussion above, the generator's Quality Assurance Program is reviewed and approved by the disposal facility operator. The disposal operator may inspect the waste packages at the site of generation. As such, this becomes the first step in the disposal facility's waste package verification process.

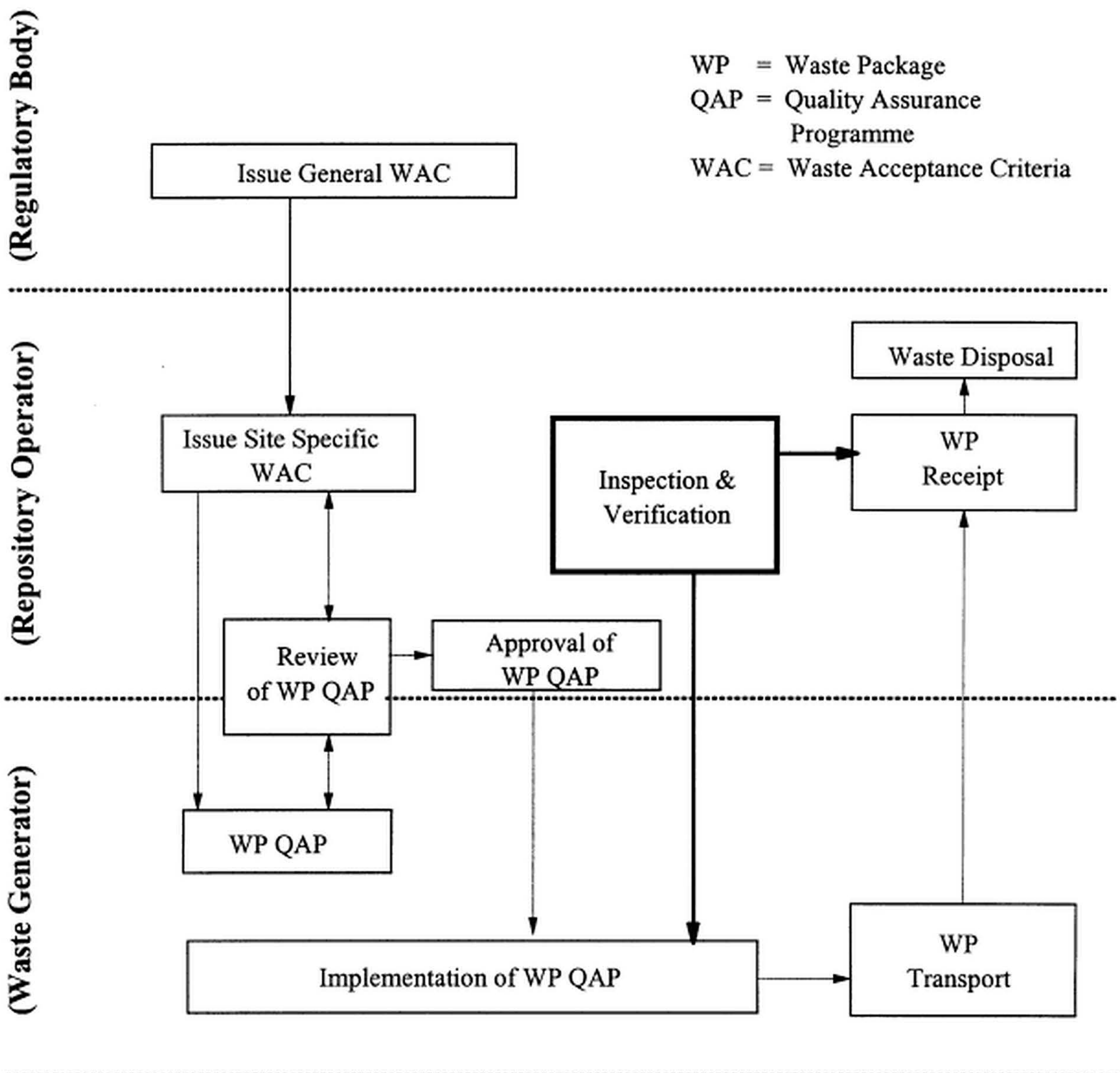

Figure 3-4
Major Responsibilities of Entities Involved in the Disposal of Waste Packages: Inspection and Verification [7] Reproduced with permission from the International Atomic Energy Agency, Inspection and Verification of Waste Packages for Near Surface Disposal, IAEA-TECDOC-1129, IAEA, Vienna (2000).

The second step is on site inspection and verification that the waste packages meet the WAC. There are three techniques used to achieve on site inspection and verification: Administrative checks, Visual checks and Direct measurements. The following items, as they relate to the three categories, will highlight those most suited to VLLW packages.

Administrative checks include verifying that the information in the shipping record is accurate for the shipment of waste received. This includes verifying unique package numbers and making sure that surface contamination levels and activity levels per package meet the Waste Acceptance Criteria for the site.

Visual checks focus on the package and include basic items such as checking individual packages for integrity and verification that all required labels are affixed.

And finally, conducting direct measurements that include weighing selected packages and conducting a radiation field survey to verify the waste generator correctly reported radiation field limits. These types of surveys are typically done with hand held or portable radiation monitors. One should note that VLLW is not always packaged in the same type of containers as LLW.

Destructive Testing

Direct measurement may include destructive testing. This can be a useful tool because it allows the disposal facility operator to verify a variety of aspects of the waste package.

> *"Destructive testing can be used to verify the physical contents of a waste package, the presence/absence of prohibited or restricted materials, the homogeneity of conditioned waste, properties of the conditioning matrix, the types and quantities of radionuclides present (notably those that do not emit gamma rays) and to determine the presence of hazardous chemical constituents in the waste.*
>
> *Examples of situations where destructive testing may be needed include historical waste packages for which limited data exist, examination of waste packages that are suspect and cannot be returned to the generator, or as part of a routine quality assurance programme.*
>
> *A sampling and analysis plan needs to be developed to document the type of sampling to be performed, the frequency of sampling and the selection method for waste packages, the analytical measurements to be performed, the data quality objectives, the process for reviewing and validating analytical results, the reporting of results and record keeping requirements."* [7] Reproduced with permission from the International Atomic Energy Agency, Inspection and Verification of Waste Packages for Near Surface Disposal, IAEA-TECDOC-1129, IAEA, Vienna (2000).

Process Knowledge

Process knowledge consists of the knowledge and documentation on the process(es) that generated the waste. Though process knowledge is more likely to be used in the front-end classification of wastes. It is also an important element when a disposal facility has received historic waste packages to verify prior to disposal. Adequate process knowledge will allow the site operator to determine the best way to verify package contents. This may include requesting samples from the generator or collecting and analyzing its own samples. [10].

For additional details see Section 5, Inspection and Verification by the Repository Operator, of the IAEA Inspection and verification of waste packages for near surface disposal, IAEA-TECDOC-1129, Vienna, 2000 [7]. Keep in mind that the information in that report was developed for Low Level and Intermediate Level Waste and may contain items that would not necessarily be required for VLLW disposal in a landfill.

What are the Packaging and Transportation Requirements for VLLW?

Packaging Requirements

Again, IAEA does not specify the packaging requirements of VLLW. However, in discussing the general containment and isolation requirements of LLW, the 2009, Classification of Radioactive Waste, General Safety Guide No. GSG -1 says VLLW does not require "… shielding or particularly robust containment and isolation [4]."

Transportation Requirements

The IAEA has not identified any specific requirements for VLLW transportation. IAEA Regulations for the safe Transport of Radioactive Material, 1996 edition as amended 2003, Safety Standards Series No. TS-R-1, IAEA, Vienna (2004) lists the transportation requirements for radioactive waste. This document is periodically updated and used directly, or is the referenced basis of the transportation regulations for countries transporting radioactive waste [11].

What are the Disposal Site's Design Requirements?

As discussed in the introduction, IAEA Member States felt previous classification systems were not comprehensive because they did not cover all types of radioactive waste, nor did they provide a direct link to potential disposal options. The following paragraph, from the 2009 Classification of Radioactive Waste, General Safety Guide No. GSG -1 has the most details on potential disposal site design provided to date.

> *2.18. An adequate level of safety for VLLW may be achieved by its disposal in engineered surface landfill type facilities. … "Some States also use this disposal method for waste with low levels of activity concentration arising from nuclear installations* [IAEA8] [IAEA9]. *The designs of such disposal facilities range from simple covers to more complex disposal systems and, in general, such disposal systems require active and passive institutional controls. The time period for which institutional controls are exercised will be sufficient to provide confidence that there will be compliance with the safety criteria for disposal of the waste."* [4] Reproduced with permission from the International Atomic Energy Agency. *Classification of Radioactive Waste: General Safety Guide,* IAEA Safety Standards Series GSG-1, IAEA, Vienna (2009).

The reader should note again that the disposal site design depends heavily on the climatic conditions.

What Scenarios are Used to Analyze the Potential for Radiation Exposure and What are the VLLW Exposure and Dose Limits?

IAEA's context for analyzing the potential for radiation exposure from a disposal facility are based on the typical scenario that assumes water entering the disposal trench/cell, the water mobilizing and transporting the radionuclides to a place of potential use by humans or other organisms.

The inadvertent human intruder scenario is believed to be particularly relevant, for waste disposed at a limited depth, and after the institutional control period. Two scenarios of particular relevance for the inadvertent intruder are 1) Construction at the disposal site and 2) Establishment of a farming community above the disposal facility [8].

How is the Disposal Site Monitored and How Long is the Institutional Control Period?

In the 2009, Classification of Radioactive Waste, General Safety Guide No. GSG -1, the IAEA provides the following guidance on disposal facility monitoring and duration:

> *The designs of such disposal facilities range from simple covers to more complex disposal systems and, in general, such disposal systems require active and passive institutional controls. The time period for which institutional controls are exercised will be sufficient to provide confidence that there will be compliance with the safety criteria for disposal of the waste.* [4]
>
> *…compliance with the safety criteria for disposal of the waste.*

Refers to the safety assessment. The safety assessment includes the pathway analyses, and applicable regulatory requirements for safe disposal of the waste.

Key Conclusions

Ultimately the IAEA provided the foundation for implementing a VLLW category when it defined Exemptions (Exempt Waste) in 2004 in terms of an acceptable exposure limit to the public of 1 μSv/y. With this number they were able to calculate the specific activity limits for each radionuclide. With this in place the next steps were to:

1. Define VLLW as a waste with activity levels that are not low enough to meet the criteria for Exempt Waste (EW) but do not require a high level of containment or isolation. Furthermore VLLW generally has very limited concentrations of longer lived radionuclides.
2. Provide a concentration limit for each radionuclide. IAEA suggested the VLLW activity concentrations could be one or two orders of magnitude greater than the EW concentrations. Besides Exempt Waste, VLLW is the only other waste the IAEA attempts to numerically define.
3. Define the exposure limit for Exempt Waste as no more than 1 μSv/y (0.1 mrem/y) gives regulators a starting point for determining an acceptable exposure limit for VLLW waste disposal.
4. Link disposal options with waste hazard. IAEA's disposal options are commensurate with the hazard of the waste being disposed. In the case of VLLW, engineered "near surface" landfills having limited regulatory control and may also contain other hazardous waste.
5. Point out that each site proposed for the disposal of VLLW will require a safety assessment, from which the site specific activity concentrations for the waste can be derived along with relevant Waste Acceptance Criteria.
6. Note that radiation protection for workers and general public will be above those for clearance from regulatory control (as defined by IAEA [5]), but the extent of protection is limited compared to higher classes or radioactive waste.

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4
VLLW PRACTICES IN FRANCE

Introduction

The National Radioactive Waste Management Agency (ANDRA) was created in 1979. France's 1991 Waste Act established ANDRA as a State owned organization, independent of waste generators, in charge of the long-term management of all radioactive waste [1]. France's nuclear regulator is the Autorité de Sûreté Nucléaire or Nuclear Safety Authority (ASN) therefore ANDRA is regulated by the ASN for its nuclear facilities.

In 2000, France began moving toward establishing and operating a VLLW disposal facility. This was the result of the following decommissioning programs being developed,

EDF had decided to decommission nine (2) nuclear power plants over a 25 year period,

A number of nuclear facilities were going to be in various stages of dismantling by 2010 and

The Marcoule reprocessing plant was to be dismantled.

In addition France's Nuclear Safety Authority implemented a program where nuclear facilities were mapped out in zones. There are two types of zones in a nuclear facility [2, 3],:

- Conventional waste zones
- Nuclear waste zones in which materials are contaminated or activated, might be contaminated or activated or might have been contaminated or activated [4].

A nuclear waste zone can generate Very Low Level Waste, Low Level Waste, Intermediate Level Waste and/or High Level Waste [3].

The result of this exercise was the recognition that there are significant quantities of waste generated that have very low specific activity and a number of instances where waste is only presumed to be contaminated.

As a result France's generators and regulators realized that there was a very real need for safe and cost effective disposal for wastes that did not require the same degree of isolation as Low and Intermediate Level Wastes (LILW).

For ANDRA to develop such a facility, it had to issue Waste Acceptance Criteria for a VLLW disposal facility. In this instance ASN was not involved because the VLLW facility is not regulated by the ASN. Instead the VLLW facility, Centre de Morvillier is regulated by the local prefecture [4].

Figure 4-1
Location Centre de Morvilliers

In 2000 ANDRA submitted an application to develop a VLLW disposal facility near the village of Morvilliers, which is in the Aube district and about two kilometers from ANDRA's LILW disposal site (Centre de l'Aube). Because of the low inventory of radionuclides proposed for the facility, it was not subject to Nuclear facility (Installation Nucléaire de Base [INB]) licensing requirements. Instead, its license was classified as a "facility for the protection of the environment", i.e., a hazardous waste landfill. As such it would be evaluated and approved by the local prefecture [3].

An overview of the process included the following steps:

1. Characterizing the site,
2. Conducting performance assessment evaluations, developing Waste Acceptance Criteria, etc.
3. Conducting two public hearings, one to address the clearing of the forest and the second to address the facility itself.
4. Securing approval from the local prefecture, and
5. Building the facility.

The local prefecture approved the VLLW facility in 2002 with construction beginning in October of 2002. In October 2003 the first VLLW was delivered for disposal [3].

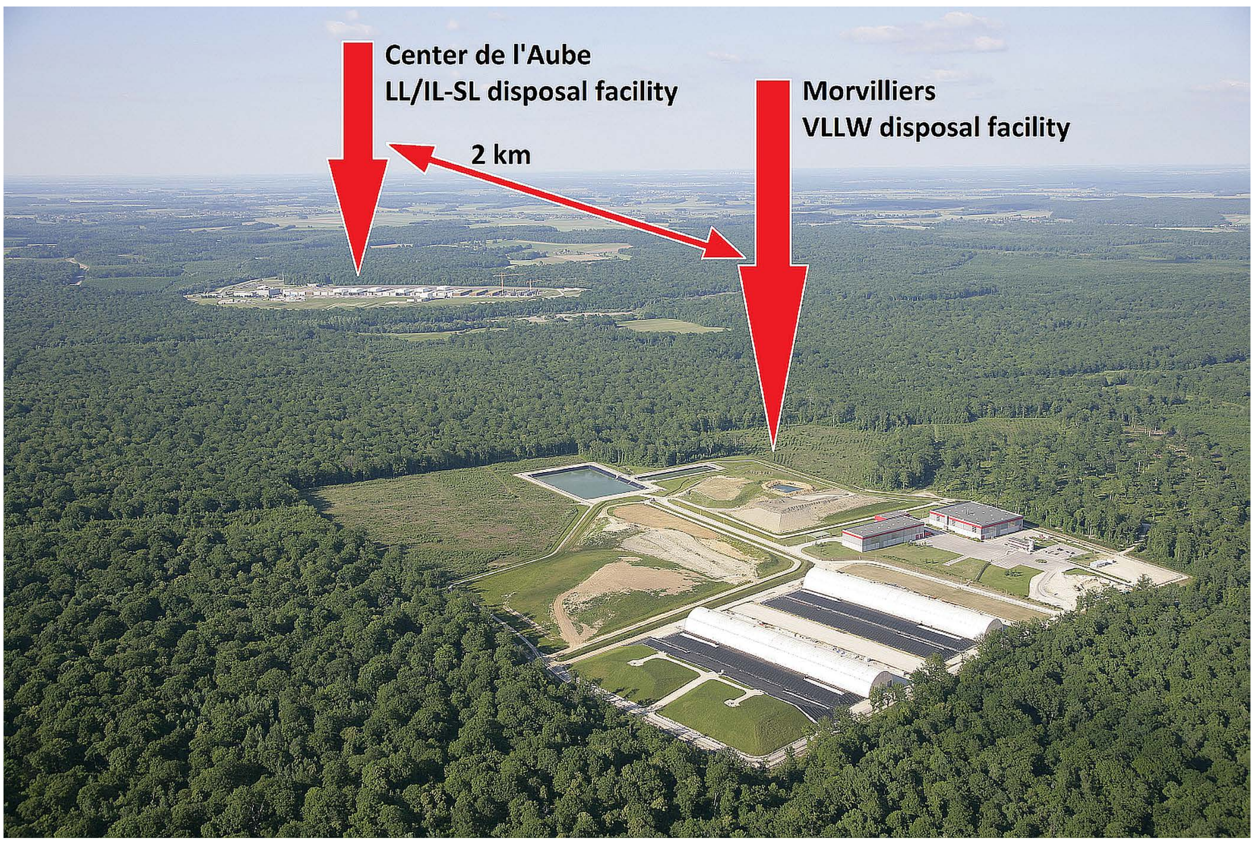


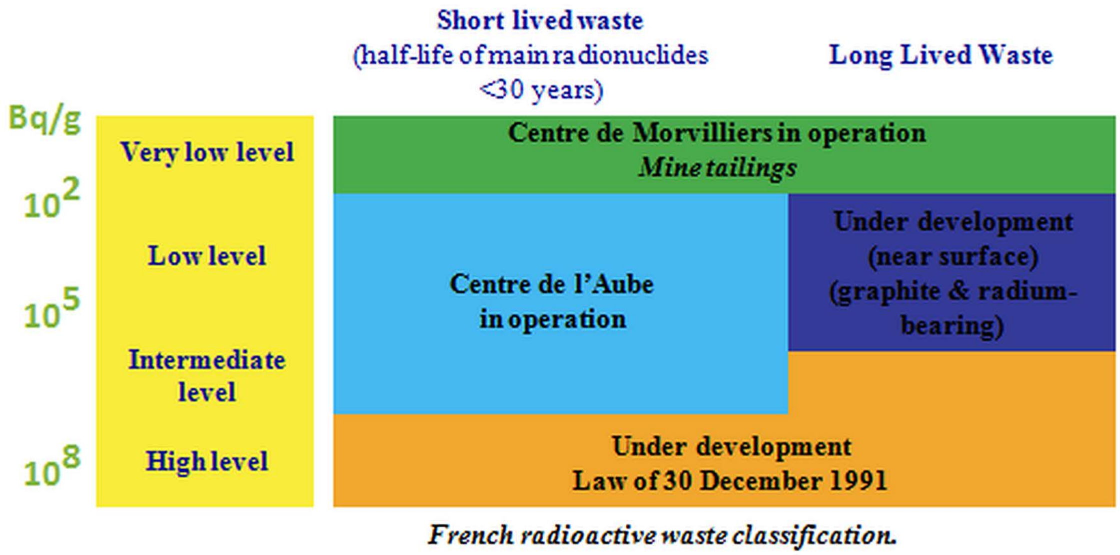
Figure 4-2
Centre de Morvilliers Relative to the LILW Disposal Facility at Centre de l'Aube (Reference C, Dutzer)

Because the French consider the very low level of activity of the waste, the result is reduced packaging, transportation, disposal site design, and monitoring requirements, compared to their LILW disposal facility. These requirements are consistent with a hazardous waste facility.

Radioactive Waste Characterization in France

The radioactive waste categories in France are based on waste management solutions commensurate with the waste's activity level and half-life. The management approach and disposal practices for each type of waste are summarized in Table 4-1 [4].

Table 4-1
Management solutions for each waste category [3, 5]



The following discussions address various aspects of ANDRA's VLLW program.

How does France Define VLLW in Terms of Radioactivity and Chemical Content?

Radiological

Simply put, France's VLLW is described as waste with very low specific activity, which is generally between 1 and 100 Bq/g [6, 7]. More precisely, the waste has to comply with the disposal facility's Waste Acceptance Criteria (WAC) as discussed in "What is the Waste Acceptance Criteria for VLLW" of this Section.

Non-Radiological

The different categories of waste to be received were identified as:

1. Inert Wastes show no significant changes with time. These wastes comprise about 52% of the waste received. About 25% of the forecast inventory is concrete rubble,
2. Metal Wastes have slow environmental release rates through time. These wastes comprise about 44% of the waste to be received and consist of the following: Wastes that do not

require preconditioning prior to disposal, wastes that have surface contamination and require grouting prior to disposal and small quantities of metallic residues, i.e., contaminated pipes that require stabilization prior to disposal,

3. Non-metal materials make up about 3% of the total waste and are compacted prior to disposal,
4. Sludges, make up about 1% of the waste and are solidified prior to disposal, and
5. Hazardous residues such as asbestos comprise less than 0.1% of the waste. [3]

The percentages reported here come from the initial forecast inventory developed by the major generators, see below, "Does France have an operating VLLW disposal facility?". Radioactive waste projections are updated periodically, by ANDRA, in France's National Waste Inventory [4].

Chemical

From its investigations ANDRA identified 12 "toxic chemicals" that are important when disposing of LILW. For VLLW, which has a significantly lower specific activity, ANDRA narrowed its chemical inventory of toxic elements to:

Cadmium (Cd)	Lead (Pb)
Arsenic (As)	Zinc (Zn)

As, Zn, Pb and Cd have been studied and are known to have a threshold for health effects. This prompted ANDRA to establish a hazard indicator, i.e., a hazard ratio in relation to the threshold level of each of these chemicals. However, for chemical carcinogens, such as As and Cd, ANDRA established "probability" indicators [3].

ANDRA has adopted a maximum value of 0.25mSv for the exposure of an individual member of the public in normal conditions and 5 mSv/y for a disposal site worker under normal conditions.

However, the scenario ANDRA has chosen was very conservative, assuming the cover became permeable after 30 years of institutional control, allowing water from the trench to reach the groundwater, and over a 160 year period, was discharged to a nearby stream. In this scenario a number of local residents are living near the stream, where they water their crops using the local water supply i.e., the stream and then consume the crops.

Using the projected volume of waste disposed at the Morvilliers facility and the conservative parameters outlined above, the projected annual dose to these residents was 0.1µSv/year (0.0001mSv/y). The chemical impact was also very low at 1.2E-05 for the hazard ratio and 5.5E-09 for the probability of carcinogenic effects [3].

Does France Have an Operating VLLW Disposal Facility?

In 2003, France constructed the first VLLW disposal facility: Centre de Morvilliers. It accepts wastes from nuclear facilities, including operating and decommissioning nuclear power plants (NPP), as well as medical, industrial and research facilities. The Morvilliers facility has a total capacity of about 1,000,000 metric tons or about 650,000 m^3 of waste over the 30 year life of the facility [3].

It is worth noting the approach ANDRA used to develop the Centre de Morvilliers site. First, the need and size of the facility was based on projections of annual waste quantities for disposal from the three largest generators in France. The three largest generators in France are Électricité de France (EDF), the French Atomic Energy Commission (CEA) and AREVA (French company that builds nuclear reactors). They estimated that over 30 years there would be a need to dispose of 800,000 tons of VLLW. This number was later increased to 1,000,000 tons to include small generators contributions. Next, contracts were put in place with these large VLLW generators. These contracts are based on their estimates of VLLW for disposal. These volumes determined the size of the Morvilliers site and led to the following agreements:

The site will operate for 30 years:

- A total of 25,600 metric tons per year of VLLW will be sent to the site by each of the major generators (this can vary between 20,000 and 30,000 m^3/year),
- The disposal fee in 2003 was 270 €/ton (~$0.20/lb) but is not a set fee through time,
- The disposal fee includes the cost of closure and long-term institutional control of the Morvilliers site [3,8]

Figure 4-3
ANDRA's Morvilliers VLLW Disposal Facility [2]

In the ANDRA literature the disposal structures are referred to as cells, trenches and vaults. For the purposes of this Section they will be referred to as trenches.

The design of Centre de Morvilliers disposal facility is based on France's and the European Union's (EU's) hazardous waste regulations. For example, the regulations prescribe not only a clay layer as a passive barrier, but also the addition of a membrane as an active barrier with drainage material above it to collect the leachate. This includes how the trenches are constructed, the liners used and design of the trench cap. It also includes items such as the post closure site monitoring program being required for 30 years. This requirement is based on the site safety analyses that show that the low levels of activity disposed of in the trenches will result in exposure levels that would not cause any radioactive hazard, after 30 years [4]. However it is understood that the local government may deem it necessary to continue monitoring the disposal facility after the 30 year, post closure surveillance period.

One exception to the site's being run strictly as a hazardous waste disposal facility is the use of a mobile shelter (cover) while constructing and operating the individual trenches. This is done to preclude rainwater from entering the trench prior to its permanent closure. Figure 4-4 shows the development of the trench under the mobile shelter.

Figure 4-4
Cell Excavation Under Mobile Shelter

It also shows the first step in disposal trench development, i.e., excavation of the disposal trench. Trenches are excavated in low permeability clay, whose permeability is 10 to 100 times lower than the required < 1.0E-09 meters/second [2] required by the EU and French hazardous waste regulations. Originally, each trench was 80 meters long, 25 meters wide and 18 meters deep (note, to accommodate an increase in the annual amount of waste received, the trenches are now 160 meters long). The sides and bottom of each trench are covered with a water tight membrane as required by hazardous waste repository regulations [4].

The Morvilliers site is 45 hectares (the disposal area is 30 hectares) and will consist of 65 trenches. The site has a waste "Conditioning Building" that houses the reception area along with compaction equipment, hazardous waste treatment facilities, and solidification equipment to solidify sludges [2,3].

Which Operating Nuclear Power Plant and Decommissioning Waste Streams Qualify as VLLW?

The majority of VLLW that will be sent to the disposal facility will come from dismantling decommissioned nuclear facilities. The majority of these wastes are concrete, rubble, earth [6] and metallic wastes. Waste generated from the decommissioning of nuclear facilities, includes NPPs. These wastes are regarded as common industrial waste and include scrap metal [9] and plastic. These wastes are generated from demolition work, such as the removal of structural steelwork, ventilation ducts, pipes etc.

There are some operating NPP wastes that will also be sent to the Morvilliers facility such as, secondary side, PWR blow down resins, and processed metallic wastes.

What are the Waste Acceptance Criteria (WAC) for VLLW?

Background

As discussed in the introduction ANDRA developed general Waste Acceptance Criteria for a VLLW disposal facility which. Next, taking the site specific information developed as part of the application for the proposed disposal facility, ANDRA developed a specific set of Waste Acceptance Criteria (WAC) starting with the concentration limits for each radionuclide.

Developing Morvilliers' Concentration Limits

ANDRA derived its limits for VLLW using two approaches. First it conducted detailed exposure pathway analyses for its proposed VLLW disposal site based on site characteristics, local geology, site meteorology, etc. which are discussed in "What scenarios are used to analyze the potential for radiation exposure and what are the VLLW exposure pathway dose limits?" Based on those exposure pathway analyses, ANDRA was able to determine the maximum activity content allowed in the Morvilliers disposal facility [3] (see Table 4-2).

ANDRA then converted the maximum radioactivity content allowed in the Morvilliers site that would protect the public, divided by the total volume and mass to be disposed of and derived the Declaration Threshold for about 150 radionuclides.

Considering the different safety scenarios to be investigated ANDRA developed IRAS classes (Index of acceptance in disposal facility). The IRAS Classes were developed using input the performance assessment safety analysis for Morvilliers. The following is an example of how different nuclides are assigned to the three classes of "reference activities" in Bq/g.

- Class 3 nuclides H-3, C-14, Ni-63, and Sr-90 have a "reference activity" of 1000,
- Class 2 nuclides U-232 to U-238 have a "reference activity" of 100, and
- Class 1 nuclides Pu-236 to Pu-240, Am -241, Pu-242, Pu-244, Co-60 and Cs-137 have a "reference activity" of 10.

These classes are a critical element of the process used to determine the activity level in waste packages and batches of packages sent to Morvilliers. The following is a brief description of how the process is implemented.

The generator uses the IRAS formula for two calculations. First the IRAS calculation is used to determine the acceptability of each individual package. For individual packages the sum of the IRAS value for individual nuclides for any individual package cannot exceed 10. Once the individual package IRASes are calculated the generator calculates the IRAS sum for the "batch" of packages that are to be sent for disposal. In this instance, the sum of the IRAS indexes for the entire batch must not exceed one [3, 10].

Equation 4-1
IRAS formula for tow calculations

$$IRAS = \sum_i \frac{Ami}{10^{Classi}}$$

Ami is the average specific activity of the waste batch for radioactive element i

Classi is the activity class for that radioactive element [3]

(ANDRA has defined four activity classes: 0, 1, 2, and 3.)

The following table shows the reference activity class, declaration threshold and upper limit for a subset of radionuclides disposed in Morvilliers [3].

Table 4-2
Activity Class and Limits for Subset of Morvilliers Radionuclides [11]

Nuclides	VLLW Reference Activity Class	Declaration Threshold Bq/g	France Upper limit Bq/g
H-3	**3**	**1**	**1,000**
C-14	**3**	**0.1**	**1,000**
Co-60	**1**	**0.1**	**10**
Ni-63	**3**	**10**	**1,000**
Sr-90	**3**	**1**	**1,000**
Cs-137	**1**	**0.1**	**10**
Pu-241	**3**	**10**	**1,000**
Pu-239,	**1**	**0.1**	**10**

Table 4-2 (continued)
Activity Class and Limits for Subset of Morvilliers Radionuclides [11]

Pu-240	**1**	**0.1**	**10**
Am-241	**1**	**0.1**	**10**
U-234	**2**	**1**	**100**
U-236	**2**	**1**	**100**
U-238	**2**	**1**	**100**

If a package contains radionuclides below the Declaration Threshold the generator does not have to report the concentrations for those radionuclides in the package.

Understanding what the waste limits are, the generator compiles a detailed description of the wastes including radiological and physio-chemical data to demonstrate how the particular waste, intended for disposal, meets the WAC. This information also includes “waste zoning” information on where the waste comes from, sampling data, knowledge of process, etc. as a way to verify the information in the waste description document. Finally, the generator is obliged to confirm how its quality control program will ensure that the characteristics of waste being shipped will match the description document developed for that waste.

ANDRA reviews this document and determines whether or not the waste qualifies as Very Low Level and can be accepted for disposal at the Morvilliers facility. Based on this generator developed document ANDRA issues the generator a waste certificate for the particular waste stream. Each waste stream will have its own waste certificate. Some waste certificates are general and apply to waste from similar facilities, while others are specific, for example the ion exchange resins from a pressurized boiling water reactor [11].

These certificates consider a comprehensive array of nuclides specific to the waste stream being approved and include a review of the non-radioactive substances listed in Table 4-3:

Table 4-3
Acceptance Certificates May Consider any of These Non-Radioactive Substances [12]

lead	boron	nickel	chrome	antimony	selenium,	cadmium
mercury	beryllium	arsenic	free cyanides	ammonia	Asbestos	-

These substances are reviewed in light of their applicability to the actual waste stream under consideration.

Remember, France’s VLLW waste acceptance specifications are consistent with regulations for non-radioactive wastes. Therefore, the hazardous waste requirements include three types of waste that are considered as hazardous according to conventional waste regulations [4]:

- Chemical characterization of the waste for each batch
- Compliance criteria for the leachable fraction of the waste, and
- Stabilization of the waste to reduce the leachable fraction to regulatory limits, as applicable.

Once a generator has the approved waste certificates from ANDRA they are able to begin scheduling and shipping VLLW to the disposal site.

How Does the Disposal Site Verify the Waste Package's Radioactivity and Chemical Content?

ANDRA's waste certificates are its first line of waste verification. These certificates are the basis for ANDRA's knowing what is expected and what is being received at the disposal facility. This in part is because of the detail with which the documents are put together. Before the waste leaves the generation site it is systematically checked against these criteria by the generator. When the generator's quality assurance program is satisfied the wastes are as described in the waste certificate the waste is shipped to the Morvilliers disposal facility [11].

There are two verifications that occur at the actual disposal facility, these are: 1. Gamma ray spectrometry of the trucks as they arrive on site and 2. Periodic checking of the actual packages for surface contamination and in some instances opening packages for visual verification. In 2011 a workshop dedicated to opening packages for visual inspection was implemented at the Morvilliers facility [4].

ANDRA uses a very sophisticated computer system for receiving and tracking each package of VLLW. Beside basic information of the generator and the contents of each package this system includes detailed information on the chemical and radiological characteristics of the waste along with details regarding the waste package dimensions. With the variety of odd sized materials that arrive at the site from decommissioning activities it is critical for logistics planning to know this level of detail on incoming shipments [11].

What are the Packaging and Transportation Requirements for VLLW?

Packaging Requirements [13]

There are no specific waste containment requirements for VLLW; therefore waste packaging requirements have been established in order to standardize handling equipment as much as possible. Since dose rates from the waste are extremely low, handling techniques are simple, meaning essentially any type of container can be used. However large bags are the most commonly used. They most often contain concrete, plastics, ceramics and other wastes. Steel boxes are also commonly used. These are 2m^3 metal containers and usually contain metallic waste. 200 liter drums are used and predominantly hold concrete dust.

Any conditioning of packages done at the generator site or by ANDRA is done to prevent dissemination of activity and to facilitate handling operations at the disposal site and prevent the dispersion of contaminants in the disposal trenches during waste loading. Some bulky large wastes may be disposed of without any preliminary packaging.

Transportation Requirements

Waste shipments have to comply with ADR. If the waste contains sufficient radioactivity then the radioactive material transportation requirements derived from IAEA TS-R-1 apply. In some cases the level of activity is so low that the radioactive material transportation requirements do not apply [4].

What are the Disposal Site's Design Requirements? [5, 10]

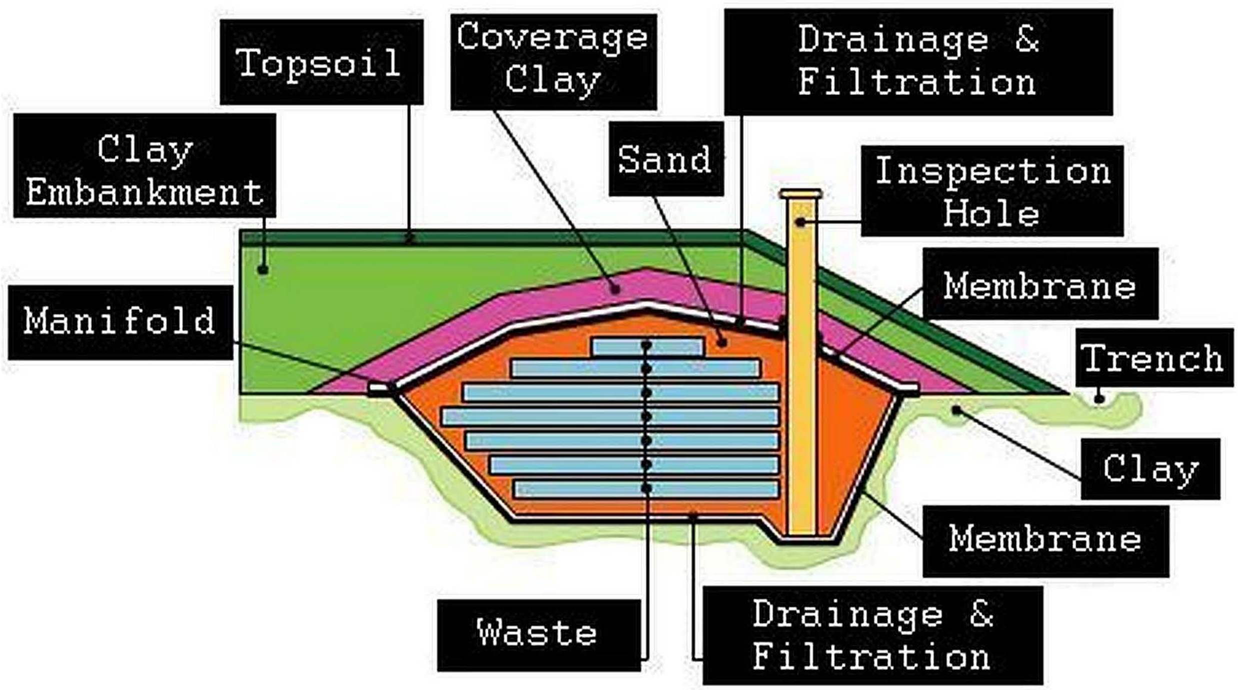


Figure 4-5
VLLW Waste Disposal Trench Design at Morvilliers [2]

Figure 4-5 shows how the trenches are constructed. It is important to note that this design, unlike those used for LILW relies on the use of clay surrounding the disposal units along with drainage and leachate collection systems. The waste is layered and backfilled in approximately 10 layers of waste per trench.

When the facility was opened in 2003 one trench at a time was actively filled. Since then, the increase in waste quantities has required that ANDRA not only operate one trench but simultaneously construct a second trench. Though the volume is within the range projected by the generators, 30,000m^3/year versus the 25,000 m^3/yr originally projected, the actual mass of the waste is greater than originally calculated by ANDRA. For this reason there is concern that the facility is not equipped to accept 30 years worth of VLLW. ANDRA is currently considering this issue.

The Centre de Morvilliers is designed to meet EU/French hazardous waste regulations, allowing it to accept both VLLW waste and toxic chemicals. The waste is placed in a low permeability surface clay layer with the following properties:

Table 4-4
Low Permeability Surface Clay Layer [2,6]

Clay Layer Properties	Regulatory Requirements	Repository Properties
Permeability	< 10E-09 meters/second	10E-10 to10E-11m/s (10-100 times lower [better] than required)
Thickness	>5m	15 to 25 m

Trenches are excavated in a clay layer, the sides are lined with a watertight membrane, and the waste is placed on top of the membrane. During trench excavation and filing, a mobile roof is used to protect operations and prevent rainwater ingress. The trenches are sealed with the same watertight membrane and finally backfilled with clay. An inspection pipe allows the site operator to check for water seepage around the waste.

The radiological capacity of the Centre de Morvilliers repository for key nuclides is shown below. These total quantities were derived from the exposure limits used in analyzing the different pathway analyses.

Table 4-5
Maximum Concentration Limits for Morvilliers VLLW Disposal Facility [3]

Nuclides	Total Activity in Morilliers Disposal Facility
Cl-36	64 GBq
I-129	31 GBq
Cs-135	1.8 TBq
Tc-99	130 GBq
Sr-90	37 TBq
C-14	1.9 TBq
Ag-108m	3.8 GBq

Table 4-5 (continued)
Maximum Concentration Limits for Morvilliers VLLW Disposal Facility [3]

Se-79	740 GBq
Sn-126	100 GBq
Pu-239	1.2 TBq
Ra-226	1.4 TBq
Th-232	11.6 GBq

Table 4-5 is included to give a sense of the difference in magnitude between the Morvilliers disposal facility with VLLW and the L'Aube facility that accepts LILW.

Table 4-6
Total Activity for Five Radionuclides for the Morvilliers and the Centre de L'Aube (LILW) Facilities [3]

Nuclide	Morvilliers Capacity of Facility	L'Aube Capacity of Facility [14]
		TBq
Cl-36	64 GBq	4.00 E-01
I-129	31 GBq	3.03 E-01
Cs-135	1.8 TBq	6.00 E+01
Tc-99	130 GBq	1.23 E+01
Ag-108m	3.8 GBq	2.49 E+01

What Scenarios are Used to Analyze the Potential for Radiation Exposure and What are the VLLW Exposure Pathway Dose Limits? [5]

The dose limits in effect at Centre de Morvilliers are the commonly accepted international standards set by the International Atomic Energy Agency (IAEA), and the International Commission on Radiological Protection (ICRP): 0.25mSv/y for public exposure and 5mSv/y for disposal site workers under normal conditions [15, 16].

Table 4-7
Exposure Pathway Dose Limits for Morvilliers [3, 15, 16]

Exposure Scenarios	Exposure Limits /Dose Constraints
Operating Phase:	
- Air Transfer – airborne internal exposure	Much less than < 0.25mSv/y
- Irradiation – external exposure	Much less than < 0.25mSv/y
- Accident – radioactivity release by fire	Much less than < 0.25mSv/y
- Public Exposure	0.25mSv/y
- Disposal Site Worker under normal conditions	5mSv/y
Long-term Impacts	
- Human Intrusion	Much less than < 0.25mSv/y
- Construction of Roads on the Site	Much less than < 0.25mSv/y
- Construction of Homes on the Site	Much less than < 0.25mSv/y
-“Bathtub” Effect	0.0001mSv/y

The chemical impact is also very small. ANDRA assessed four chemical elements with a threshold for health effects (As, Zn, Pb, Cd) and chemical elements with carcinogenic effects (As, Cd). ANDRA estimates the hazard ratio is 1.2 x 10^{-5} and the carcinogenic effect at 5.5 E-09 [3].

How is the Disposal Site Monitored and How Long is the Institutional Control Period?

The Morvilliers site will conduct radiological and physiochemical monitoring during its 30 year operating life. The following items will be monitored [7]:

1. Groundwater will be measured by 7 piezometers located around the disposal facility,
2. Surface water collection points collected prior to the building of the Morvillers site will be re-sampled. Sampling will include automatic samples collected from the site’s storm water collection pond, prior to an releases into a local brook that is downstream from the disposal facility, off-site runoff water, and streams located in the immediate vicinity of the facility,
3. Sediments from the sediment pond will be sampled,
4. Gaseous discharges,
5. Trench cell leachate collection system will be sampled for the possible presence of water, and

6. Dosimetry measurements will be taken at the facility's fence line from six separate film dosimetry devices. These measurements will be compared with measurements taken from the nearby community that are beyond the influence of the site. ANDRA also monitors elements of the food chain, i.e., milk and vegetables collected in the vicinity of the disposal facility. This includes measurements of moss growing in the nearby wooded area [7].

Regarding long-term monitoring, an order will be issued by the local prefecture (the local government where the disposal site is located) that will prescribe the conditions for monitoring the site for 30 years post closure [2]. However the following monitoring requirements are likely,

1. Leachate generation is not expected, should leachate generation occur, monitoring requirements will be defined,
2. Gas collection and monitoring – no significant gas generation is expected,
3. Groundwater will be sampled annually,
4. Other items to be sampled are the discharge from the storm water basin, vegetation, river sediment and general background radiation.

Key Conclusions

France's VLLW Program

1. Disposes of VLLW in a facility designed consistent with a hazardous waste disposal facility. It is owned and operated by the public company ANDRA, that is also in charge of France's LLW, ILW and future High Level waste facilities
2. Locates the VLLW facility within two kilometers of their operating LILW facility.
3. Accepts only VLLW at the facility.
4. Constructs and operates disposal trenches under a movable tent.
5. Require VLLW be submitted for disposal in batches.
   5.1. Individual package can have an IRAS sum $\leq$ 10.
   5.2. Batch of packages (same waste stream) can have an IRAS sum $\leq$ 1.
6. Characterize the waste upfront, rely heavily on process knowledge and includes chemical parameters required of any hazardous waste disposal facility.
7. Requires generators to complete a successful characterization, before ANDRA issues the generator a waste certificate for each waste stream.
8. Includes waste inspections by ANDRA within generators' facilities and checks on delivered waste.

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5
VLLW PRACTICES IN SPAIN

Introduction

In 1984 the Spanish Parliament established a public company—Empresa Nacional de Residuos Radiactivos, S.A. (Enresa)—to safely treat, condition, manage, store, and dispose of Spain's radioactive wastes. Enresa is also responsible for decommissioning and dismantling nuclear power plants when their service life has ended, and conduct environmental restoration of closed uranium mines and facilities. [1]

In 1986 the El Cabril site, an abandoned uranium mine site, located in the hills of the Sierra Albarrana Mountains in the province of Córdoba, was transferred to Enresa.

Figure 5-1
Location of El Cabril Disposal Facility

In October 1992 the Spanish Ministry of Industry and Energy issued an operating permit allowing for Near-Surface Disposal of Low and Intermediate Level Waste (LILW) at the El Cabril site.

In 1999, the European Union (EU) member Countries saw an opportunity to treat and dispose of low activity radioactive waste as a hazardous waste. This occurred when the European Union Directive on the landfill of waste [2], expanded its list of hazardous wastes. Based upon the European Union Directive (1999/31 EC of 26 April 1999), Spain developed regulations addressing hazardous waste disposal.

In 2003 Enresa submitted an application to modify the design of their El Cabril LILW disposal site to accept VLLW waste, and the design modification was approved in 2008. In October 2008 the first VLLW was accepted for disposal trench/cell 29.

Figure 5-2
El Cabril Disposal Facility (n° = number) [3]

An important factor in the acceptance of a VLLW facility is its estimated contribution of 1% of the total activity from LILW disposed at the El Cabril facility. The activity contribution is so small that it did not require any modification of the authorized radiological inventory at El Cabril. [4] The total activity of the El Cabril LILW site, for specific nuclides, on average, is < 3.70E+02Bq/g alpha. [5]

Finally, the disposal principles for VLLW are the same as those used for LILW. However, a different barrier design is used, based on the European Directive addressing the landfill of waste [2]. Which is more in keeping with the type of waste and risk associated with its very low activity content.

Radioactive Waste Characterization in Spain

Spain uses the following waste characteristics for their radioactive waste categories. [5,6]

Table 5-1
Radioactive Waste Characterization in Spain

Category	Limits	
Waste acceptable for near surface disposal	**VLLW** with very low specific activity. 1 ≤ 100 Bq/g	
	LILW with very low long-lived radionuclide content. Specific criteria for disposal facilities.	
	Level 1	Maximum activity per unit mass for different radionuclides at a disposal unit, some of the limits are: < 1.85E+02Bq/g per total alpha at 300 years < 7.40E+03 Bq/g tritium < 3.70E+04 Bq/g total beta/gamma activity
	Level 2	More detailed limits and limits per package for those nuclides in the Reference Inventory (≤3.70E+03Bq/g alpha per "disposal unit").
All other waste	**High level waste**, including heat generating or high long-lived radionuclide content or both.	

How is VLLW Defined in Terms of Radioactivity and Chemical Content?

Radiological [4, 7, 8]

Spain defines their Very Low Level Waste as waste with very low specific activity, generally 1< 100 Bq/g. Co60 and Cs137 are the two most important nuclides considered when characterizing a waste stream to determine whether it qualifies as VLLW.

Non-Radiological [3, 9, 10]

The waste may be considered: 1. An inert material, showing no significant changes with time, like minerals or substances in natural substrates, such as soil, or debris, 2. Non-hazardous materials with slow changes through time, including slow environmental release rates, such as scrap iron and non-ferrous materials (pipes, equipment, textiles, plastics, etc.), or 3. A hazardous material containing chemical contaminants that produce toxic risks in addition to the radioactivity present.

Does the Country Have an Operating VLLW Disposal Facility?

El Cabril site accepts Nuclear Power Plants (NPPs) operating and decommissioning VLLW, as well as medical, industrial and research VLLW. There will be four cells/trenches (29-32) for VLLW but currently only cell 29 is built and is accepting waste. The first trench/cell capacity is about 33,000m^3. The facility's total VLLW capacity is approximately 120,000 m^3 of waste. [10, 11]

In the U.S.A, what is commonly referred to as a disposal trench

is called a cell at El Cabril.

What is commonly called a disposal site, in the U.S.A,

is sometimes referred to as a storage site, in Spain. [3, 11]

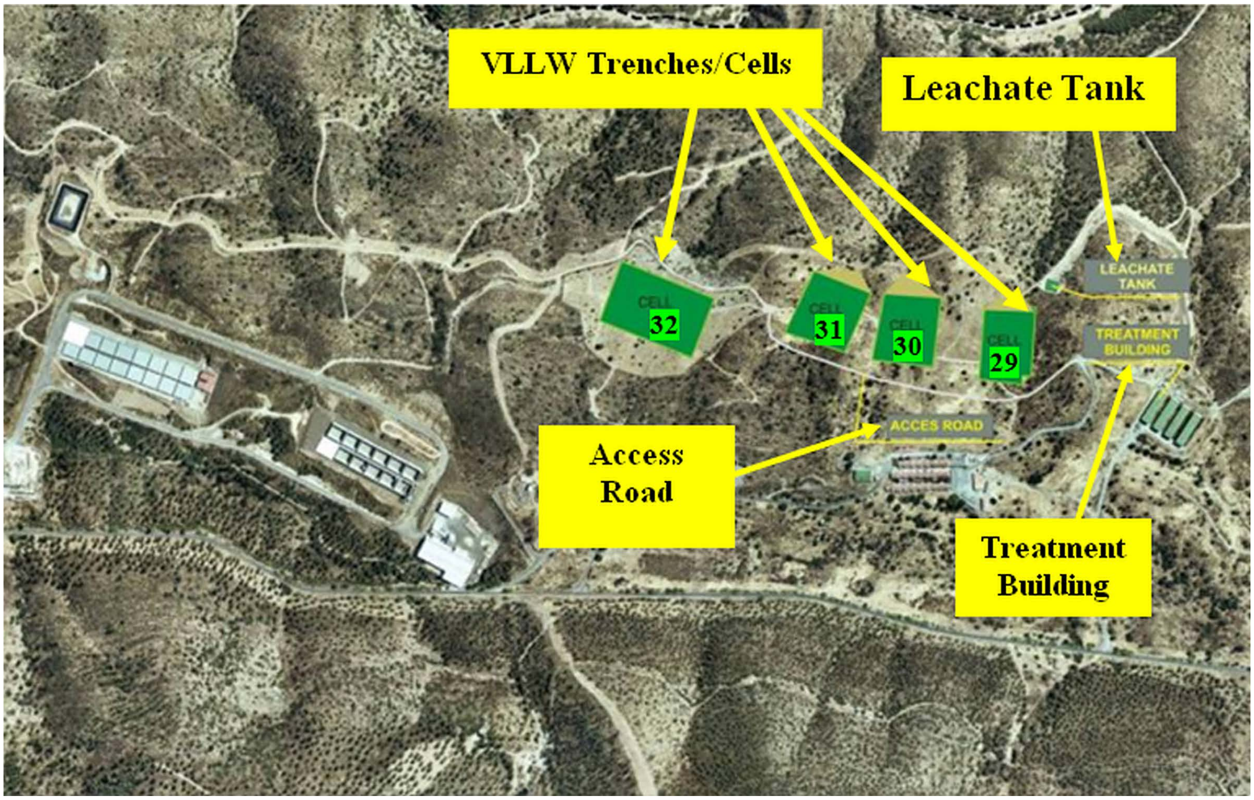

Figure 5-3
El Cabril VLLW Disposal Facility [3]

El Cabril's trench design for Very Low Level Waste is based on Spain's hazardous waste regulations. However, all other aspects are regulated as a radioactive waste. For example, the post closure site monitoring is required for 60 years, vs. 30 years for closure of a hazardous waste site.

The site has its own treatment building, located near the VLLW trenches. The treatment building functions as a reception area and unloading area for trucks transporting VLLW. It is used in the identification and control of different wastes, temporary storage, and classification of the waste for treatment and/or final disposal. It is also used for stabilizing waste, either by the addition of a hydraulic mortar or backfilling container gaps with metallic waste. [11, 12]

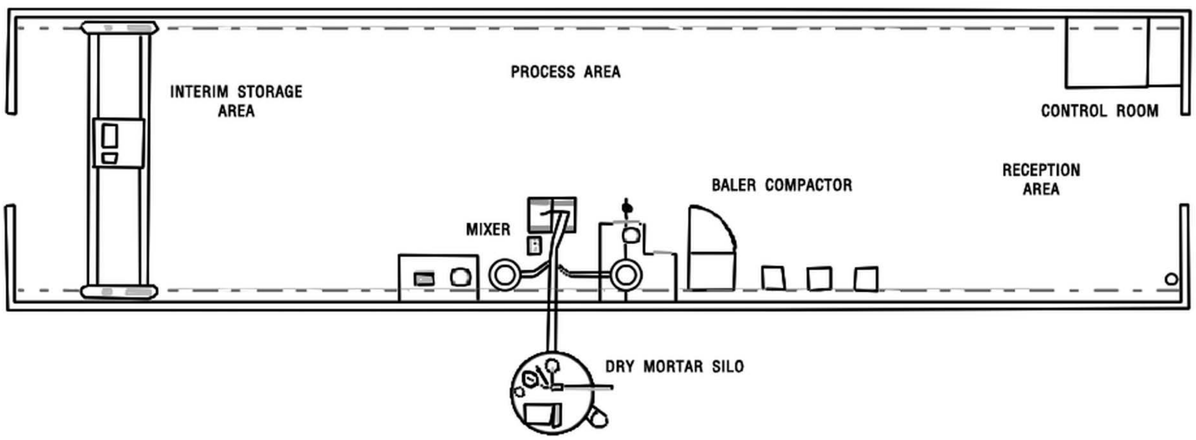

Figure 5-4
VLLW Treatment Building Schematic [12]

Phases of Construction and Use of Individual Trenches/Cells

Figure 5.5 shows, how a disposal trench is developed and operated. Construction and use of the trench occurs in two phases. First the ground is cleared and conditioned. Next the barrier system at the bottom of the trench is put in place. When waste emplacement reaches the upper level of the Riprap dike, an intermediate protective layer is put in place. Over this layer a new soil dyke is constructed and set in from the Riprap dyke. Once sufficient barrier layers are emplaced, this will become the basin for the second phase of waste emplacement.

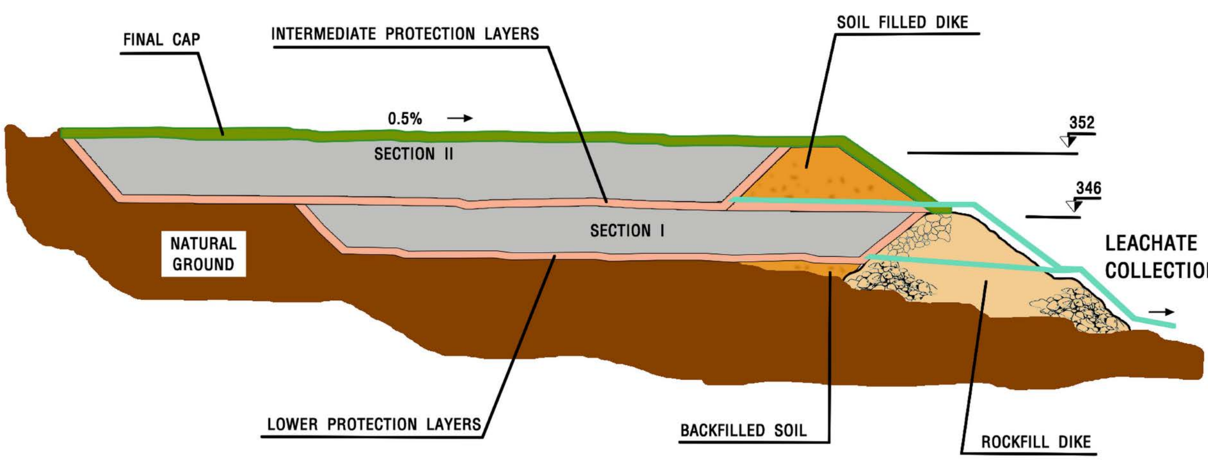

Figure 5-5
Longitudinal View of Trench 29 [11]

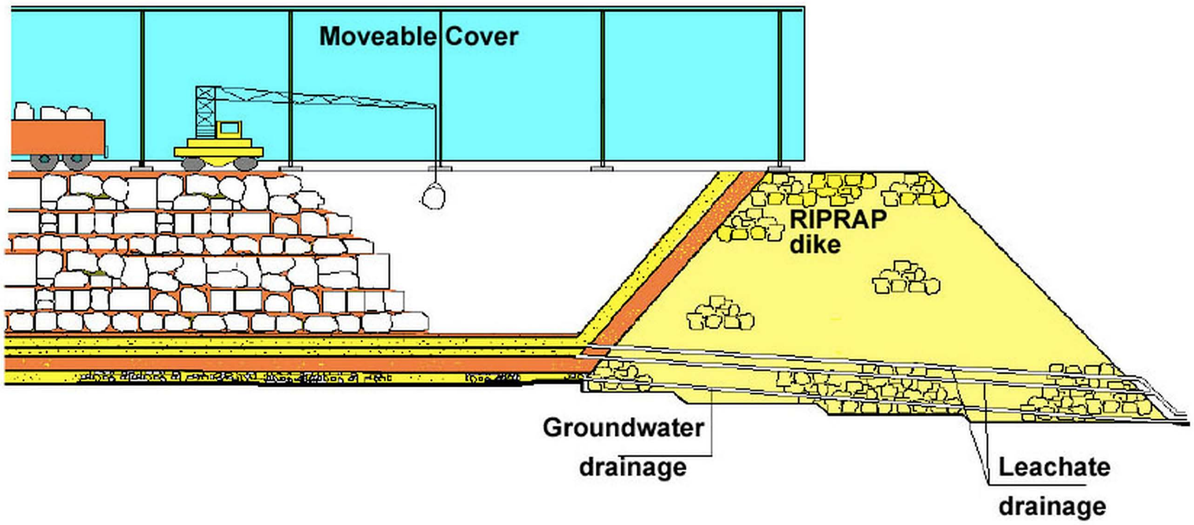

Figure 5-6
Side view of a Section 1 of a VLLW Disposal Trench at El Cabril [10]

Figure 5.6's "Movable Cover" is a tent that is placed over the active portion of the trench where waste is being placed. This eliminates rainwater from entering the active portion of the trench.

The waste is placed in layers and those layers are backfilled allowing the crane or truck to successfully place the waste across the entire trench sub section.

Figure 5-7
Waste and Backfill [11]

Figure 5-8
Protective Tent Over Active Section [11]

Figure 5-9
Interior View of Tent [11]

Which Operating Nuclear Power Plant and Decommissioning Waste Streams Qualify as VLLW?

In general, inert waste (soil and debris) and non-hazardous waste (scrap ferrous materials such as pipe, equipment, textiles, and plastics) from operating and decommissioning facilities qualify as VLLW. In addition, at operating plants dry active waste and contaminated soils also qualify. [9]

In Spain's Sixth General Radioactive Waste Plan, its latest plan for managing radioactive waste, published in June 2006, they estimate that 16% of operating plant waste is VLLW and 73% of decommissioning power plant waste is VLLW. [13]

What is the Waste Acceptance Criteria (WAC) for VLLW?

The Acceptance Criteria for VLLW apply to the El Cabril disposal trenches 29 – 32. [3]

Background

The Ministry of Industry, Tourism and Trade (MITYC) and the Nuclear Safety Council (CSN) have approved the license document (031-ES-IN-0002), containing the **Acceptance Criteria for Disposal Units of LILW and VLLW** at the El Cabril Disposal Facility. [3, 10]

Waste Acceptance Criteria for Disposal Units

Briefly stated, the Waste Acceptance Criteria for Disposal Units are based on:

1. **The total waste to be disposed**, including volume optimization requirements and volume dilution constraints,
2. **Total activity to be disposed,** where activity is determined as a function of waste classification, i.e. VLLW is 1≤ 100Bq/g),
3. **Scenarios considered and engineered barriers,** operational and closure scenarios exposure limits and trench design, water collection/diversion, capping, respectively,
4. **Waste treatment and conditioning**,
5. **Transportation to the disposal facility**, Enresa fulfills the transportation requirements contained in European Agreement concerning the International Carriage of Dangerous Goods by Road (A.D.R.), and
6. **Waste handling** by the generator/producer and the disposal facility, i.e., maximum package size, mass and dose rate. [14]

Three Specific Legislated Requirements

There are three specific legislated requirements, which generators/producers must follow, to successfully meet the Acceptance Criteria for VLLW-Disposal Units (VLLW-DUs). These are:

1. Acceptance Criteria for Packages,
2. Acceptance Method for Packages, and
3. Acceptable Methodologies for Activity Determination.

These acceptance requirements: 1. Identify the responsibilities of Enresa and the generator/producer, throughout the waste acceptance process; 2. Describe the requirements for waste acceptance and documentation; and 3. Delineate the process the generator/producer and Enresa will follow for VLLW-DU acceptance and disposal. [3, 14]

These acceptance requirements must be met at the site of generation before Enresa will accept the waste packages for transportation and disposal.

Packages and VLLW-Disposal Units

In the Spanish system, to accept VLLW packages ENRESA should accept them in batches. In the VLLW case, an accepted package can be a disposal unit, when no additional treatment is required at El Cabril. A Disposal Unit is not a disposal trench, or a segment of a disposal trench. Once the accepted package enters at El Cabril, it becomes a Disposal Unit, so this means that it can be disposed of directly at cell 29.

Prior to these groups of packages being defined as a Disposal Unit, the producer may have to perform one or both of the following processes:

- Stabilization, if they contain hazardous waste, or
- Gap filling

However, compaction, if require is always done at El Cabril where the equipment is located.

Because Enresa does not accept individual VLLW packages, the packages must be defined as part of a VLLW- batch.

These acceptance requirements must be met at the site of generation before Enresa will accept the waste packages for transportation and disposal.

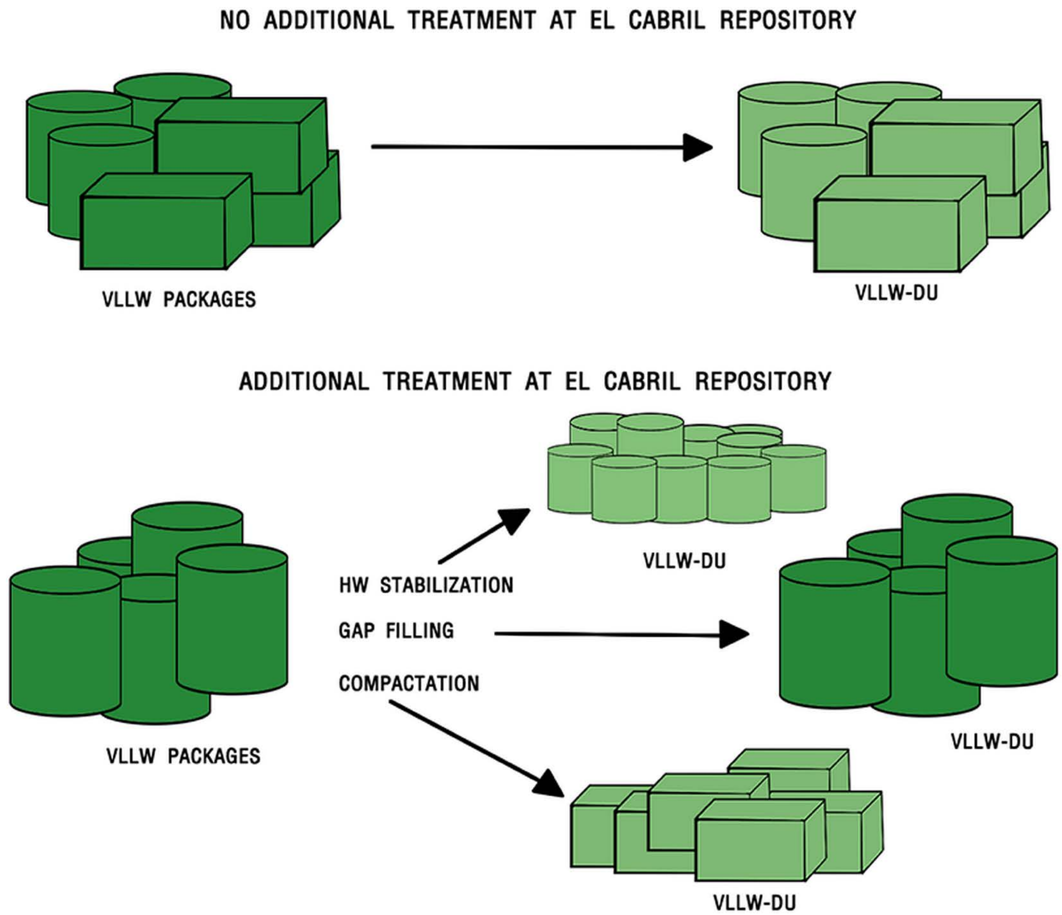

Figure 5-10
The Relationship Between VLLW Packages (a batch) and VLLW-Disposal Units (VLLW-DU) [9]

For both LILW and VLLW the generator first defines packages. Next the packages are grouped together to make up a VLLW- batch (either of packages or DU) or an LILW-DU. The Disposal Unit is a key element in the disposal approach used in Spain. An important difference between LILW-DU and VLLW-DU is the placement of an LILW-DU in a concrete overpack before being placed in a disposal vault. A VLLW-DU is not placed in an overpack. VLLW packages must be configured into VLLW-Batches, by Enresa prior to being accepted for disposal. Only VLLW-DUs can be disposed of in trenches 29-32 at El Cabril.

Enresa's LILW and VLLW Contract with Generators/Producers

Generators/producers, which generate radioactive waste, must put a contract in place with Enresa. The Enresa contract commits Enresa to accepting LILW-DUs and VLLW-DUs for disposal after Enresa has conducted a thorough waste verification process.

The objective of the Spanish regulations is to have the technical specifications for each waste stream attached to the Enresa contract before the waste is generated. This is not always possible and is addressed in the sub section Waste Acceptance Method for Packages.

VLLWs are not considered a new category of radioactive waste. VLLW is considered a sub category of the current definition of LILW. There are some packages/waste streams, originally identified under the LILW acceptance criteria, which are VLLW. These waste streams are contained in the Level 1 sub category of LILW (see Introduction, above). This is important because generators with Package Descriptive Documents addressing the VLLWs in LILW do not need to revise their documentation or existing contract to include those VLLW packages.

Acceptance Criteria for VLLW-Disposal Units (VLLW-DUs):

The general acceptance criteria for VLLW-DUs, consists of the following:

1. VLLW-DUs will be accepted in “Batches”.
2. The container is used to prevent material dispersion.
3. Hazardous waste requires stabilization by the generator/producer.
4. Inert and non hazardous wastes can be conditioned by using gap filling.
5. Fine grain inert waste, but not dust, can be used for volume optimization of other packaged VLLW.
6. VLLW can not contain explosive materials, oxidizing agents, flammable material, pyrophoric material, or gas. It cannot contain fermentable material or material that can rot in some conditions and with some limits. Finally the material cannot reach a temperature greater than 60^{o} C (140^{o} F).
7. Encapsulated sources, liquid aqueous or organic waste without solidification are not acceptable. [9]
8. The specific activity of the waste cannot be diluted to meet VLLW criteria and any hazardous wastes require stabilization. [9, 10]

The three specific legislated requirements, generators/producers must follow to successfully meet the Acceptance Criteria for VLLW-Disposal Units (VLLW-DUs):

1. Acceptance Criteria for Packages,

2. Acceptance Methods for Packages, and

3. Acceptable Methodologies for Activity Determination.

Acceptance Criteria for Packages

Waste Acceptance Criteria

The acceptance criteria include the following General Criteria, Radiological Criteria and certain Restrictions.

General Criteria

- Package identification and traceability: The waste package can be traced back to where it was generated and does not contain free liquids.
- Hazardous waste: These packages require stabilization.

Radiological Criteria

- Index of Acceptance (***IA***): Each package must meet the Index of Acceptance. The calculation is performed by the generator for each waste package being prepared for shipment to the El Cabril disposal site, using the following formula.

Equation 5-1
Enresa Index of Acceptance

$$IA_{DU} = \sum_i \frac{A_{Mi}}{A_{M\,\max i}}$$

IA	is the Acceptance Index
DU	is the Disposal Unit
A_{Mi}	is the specific activity of radionuclide *i* (Bq/g) in the mass waste
$A_{M\,max\,i}$	is the specific activity limit for radionuclide *i* (Bq/g) in a waste assuming radionuclide *i* is the only radionuclide in the waste
i	is an individual radionuclide [3, 9, 10, 11]

- Specific activity limits: These limits are provided for specific radionuclides. The following table lists the activity limits for some of the radionuclides of interest. Spain has identified over 100 radionuclides, as potentially being present in its VLLW.

Table 5-2
Maximum Activity per VLLW Package ($A_{i\,max}$) in Bq/g [6,15]

H^3	1,000
C^{14}	1,000
Co^{60}	10
Ni^{63}	1,000
Sr^{90}	1,000
Cs^{137}	30
Pu^{241}	1,000
Pu^{239},Pu^{240}	10
Am^{241}	10
U^{234} to U^{238}	100

Co^{60} and Cs^{137} are the nuclides of interest in VLLW and LILW disposal. [4, 9]

Restrictions [3]

- Non radiological: The waste will not contain substances that promote leaching, could explode, are corrosive, flammable, etc.
- Physical-chemical characteristics: The waste will not include spent sealed sources, aqueous liquids or organic waste, unless they are solidified.

Physical and Chemical Characteristics Requirements [3, 9]

- Inert Waste: A physical and chemical description of the waste, including its origin and the processes that generated it (non-hazardous soil, debris, etc., that show no significant change with time) are required.
- Non-Hazardous Waste: For non-metallic waste, information on its cellulose components; for metallic waste the ratios of the various metals (in the case of steel, the ratio of stainless steel to carbon steel) are required.
- Hazardous Waste: Report on the following characteristics to determine whether the waste is hazardous - pH >4 and < 13; soluble ratio < 10% of the dry waste mass; if leaching criteria cannot be verified then the waste should be stabilized; identify any complex organic or inorganic materials; identify whether asbestos is present.

Package Dose Rates and Allowable Contamination

As part of the WAC, and for verification purposes, prior to accepting the waste on site, the:

- Dose rate at any point of the package will not exceed 2mSv/hr.
- Removable surface contamination from packaging will not exceed 4 Bq/cm^2 for ß-γ emitters and 0.4 Bq/cm^2 for α emitters.
- Accessible surface material transported unpackaged (soil/rubble in an open truck), with Enresa agreement, will not exceed 4Bq/cm^2 for ß-γ emitters and 0.4 Bq/cm^2 for α emitters. [10]

How Does the Disposal Site Verify the Waste Packages' Radioactivity and Chemical Content?

The three specific legislated requirements, generators/producers must follow to successfully meet the **Acceptance Criteria for VLLW-Disposal Units (VLLW-DUs):**

1. Acceptance Criteria for Packages,
2. **Acceptance Methods for Packages**, and
3. Acceptable Methodologies for Activity Determination.

Waste Acceptance Method for Packages

The second legislated requirement, Acceptance Method for Packages, outlines the progression of events that result in Enresa's verifying that the Acceptance Criteria for VLLW-Disposal Units have been met.

Enresa provides generators/producers with The Acceptance Methodology for Packages. The Acceptance Methodology process identifies the required documentation to be provided by the generator/producer and the responsibilities of both Enresa and the generators/producers, for each step of the acceptance process. This includes aspects of the waste to be documented, required calculations, necessary verification of samples, etc.

The Acceptance Method for VLLW Packages and LILW packages is parallel. When the Acceptance Method is successfully followed and completed, VLLW-DUs are verified and Enresa accepts the management responsibility, i.e., transportation and disposal, of the Disposal Units.

Generally speaking, there are three options a VLLW package can take to verification, acceptance and disposal by Enresa. The three paths are: 1. Packages are Generated Specifically as VLLW, 2. Packages are Generated as LILW, and 3. Historical Packages were generated prior to putting an LILW contract in place with Enresa. (Until the packages are properly characterized they cannot be part of a VLLW batch and are not acceptable for disposal.)

Reviewing the VLLW batch for compliance with the Waste Acceptance Criteria (for packages) depends on a report that describes the waste, a report that documents WAC compliance and the development of a unique Acceptance File. The following identifies each process, the required reports and who writes those reports [3, 9, 10].

1. Packages Generated Specifically as VLLW

a. VLLW Package Descriptive Document - developed by generator.

b. Characterization Study[1] - developed by Enresa.

c. Acceptance File references reports a & b - compiled by Enresa.

2. Packages Generated as LILW

a. Package Descriptive Document - developed by generator.

b. Process Book - developed by Enresa.

c. Acceptance File references reports a & b - compiled by Enresa.

3. Historical Packages Without an LILW Acceptance Document

a. Generator/producer data and samples - requested by Enresa.

b. Characterization Dossier[1] - developed by Enresa.

c. Acceptance File references reports a & b - compiled by Enresa.

[1] Developed with data and complementary tests from the generator/producer.

The Acceptance File is the Acceptance Document for a Batch of Packages.

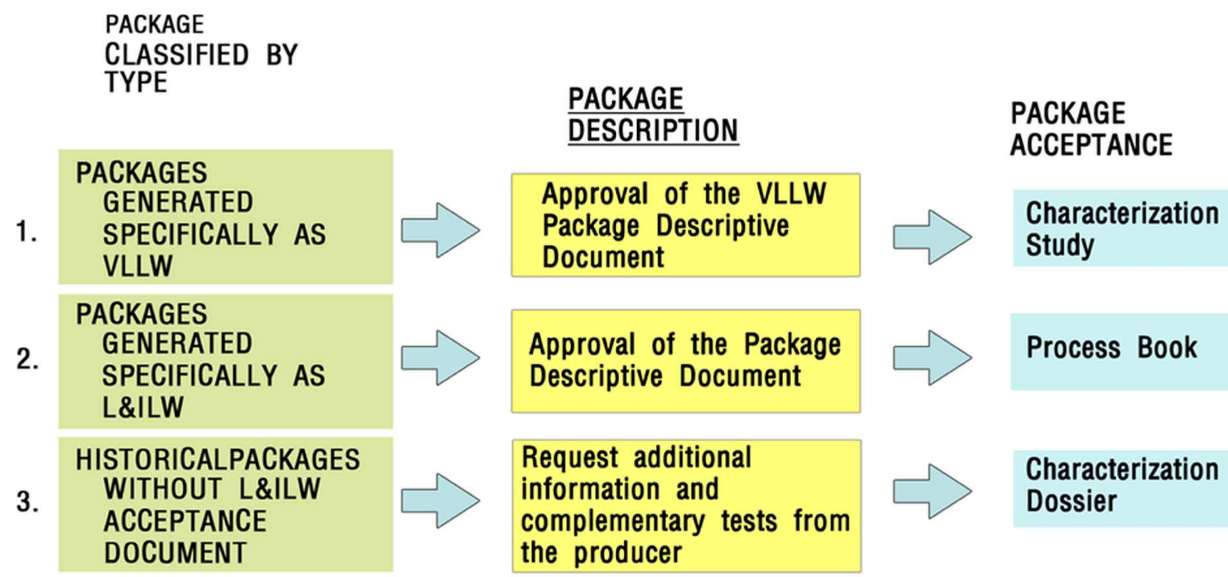

Figure 5-11
The Three LLW-DUs Acceptance Methods [3]

Proceeding to the "Package Description" outlined Figure 5.11 above, Enresa reviews the first set of documents for specific information.

Package Description

1. Packages Generated Specifically as VLLW

These technical specifications attached to the contract are the basis for the VLLW Package Descriptive Document written by the generator/producer. The VLLW Package Descriptive Document describes the waste in terms of where and how it was generated, its classification (activity content of Co-60 and Cs-137, whether it is considered inert, non hazardous or hazardous and provides details on the Quality Assurance program used by the generator. Because this document is based on technical specifications developed before the waste is generated, the document is generally accepted as representing the group of VLLW packages.

2. Packages Generated as LILW

The technical specifications are attached to the Enresa contract and form the basis for the Package Descriptive Document written by the generator/producer. It describes the waste in terms of where and how it was generated, its classification (activity content of Co-60 and Cs-137, whether it is considered inert, non hazardous or hazardous and details on the Quality Assurance program used by the generator. Because this document is based on technical specifications developed before the waste was generated, it is generally accepted as representing the group of qualifying LILW packages that contain VLLW.

3. Historical Packages Without an LILW Acceptance Document

Packages generated prior to putting the LILW contract in place with Enresa, will not have the technical specifications required to assemble a Package Descriptive Document. Because this is a historic waste, with limited information available, Enresa will request additional information and complementary testing of the waste. To the extent possible, Enresa will describe the packages in terms of where and how it was generated, its classification (activity content of Co-60 and Cs-137, whether it is considered inert, non hazardous or hazardous and detail the Quality Assurance program used by the generator. [3, 9]

In all three processes a second report is developed or reviewed, because this is the step where Enresa documents its hands on review to ensure that the Waste Acceptance Criteria are met.

Based on documentation and data sheets supplied by the generator/producer, sufficiently detailed information is known about the packages/waste streams that Enresa can develop a characterization document.

Proceeding to the “Package Acceptance” stop outlined in Figure 5.11 Enresa develops reports with the following specific information.

Package Acceptance

For Packages Generated Specifically as VLLW

Enresa develops a Characterization Study. Enresa’s Characterization Study [3, 9] specifically,

1. Reviews descriptions of the activity methodology used, to ensure that it complies with Enresa’s Acceptable Methodologies for Activity Determination (Enresa provides generators/producers with Acceptable Methodologies for Activity Determination. The document contains methods for calculating scaling factors, the specific activity of a radionuclide (the activity per unit mass of that nuclide, etc.).

2. Analyses the suitability of stabilization, gap filling and volume optimization techniques used on packages, and
3. Reviews how leaching tests were conducted on the hazardous wastes and the test results.

For Packages Generated as LILW

Enresa must compile a Process Book. The Process Book includes items 1-3 above.

For Historical Packages Without an LILW Acceptance Document

Enresa develops a Characterization Dossier covering the three items listed above. This dossier is put together from datasheets provided by the generator/producer as well as additional information Enresa has requested.

In addition to the three items listed above in each of the three categories, the (VLLW) Characterization Study, the (LILW) Process Book, and the (Historic Package) Characterization Dossiers, address the origin and production process for the package/waste stream. This includes a chemical description of the waste and generation process. The documents address whether a waste is hazardous, the cellulose components of non-metallic wastes, and the ratio of the different metals including the ratio of stainless steel to carbon steel in the case of iron wastes. They address the pH, solubility ratio, leaching limits, organic and inorganic complexing agents and identify the presence of asbestos. The index of acceptance (IA) calculations done for each package, are presented, in terms of specific activity limits for nuclides present. All these items are listed in the Waste Acceptance Criteria.

Document Reviews are Conducted in Three Stages. [3]

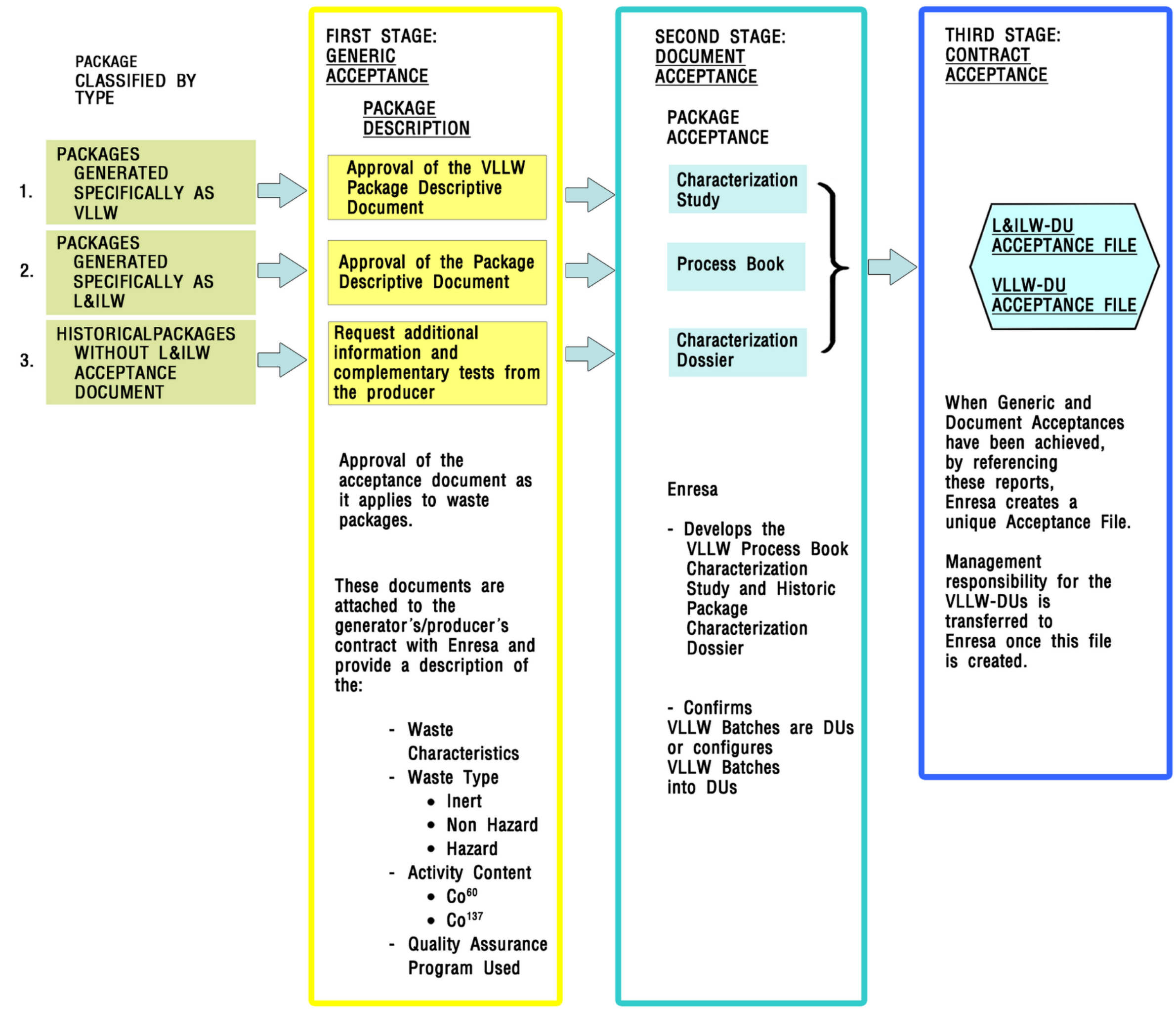

Figure 5-12
General Acceptance/Verification Methodology for VLLW [3]

See Figure 5.12 above where the first stage is Generic Acceptance. The second stage is Document Acceptance, and the third stage is Contractual Acceptance.

First Stage: Generic Acceptance

1. For Packages Generated Specifically as VLLW.

Enresa makes sure the VLLW Package Descriptive Document is still a reasonable and accurate description of the waste, in terms of where and how it was generated, its classification and whether it is considered non hazardous or hazardous.

2. For Packages Generated as LILW.

Enresa, reviews the Package Descriptive Document to be sure it accurately describes where and how the packages/waste was generated, it's classification and whether it is considered non hazardous or hazardous.

3. For Historic Packages Generated Without an LILW Acceptance Document.

Enresa request additional information and complementary testing of the waste, so a document can be put together so it will, to the extent possible, include where and how the waste was generated, its classification and whether it is considered non hazardous or hazardous. Again, ensuring that on a generic level the waste being verified is a VLLW.

Second Stage: Document Acceptance

1. For Packages Generated Specifically as VLLW.

Enresa conducts detailed reviews of the generator's/producer's datasheets. This review focuses on the chemical description of the waste and waste generation process, including the percent of cellulose in non hazardous wastes and the ratio of the different metals including the ratio of stainless steel to carbon steel. Hazardous wastes are reviewed to make sure the pH is between 4 and 13, the solubility ratio is less than 10% of the dry waste mass. The waste is stabilized if the leaching limits cannot be verified. Furthermore the report contains information on any organic or inorganic complexing substances, and identifies the presence of asbestos. Enresa reviews whether stabilization processes, gap filling and volume optimization techniques have been employed properly. Leachate test data from hazardous waste are then reviewed. Also included is the index of acceptance (IA) calculation(s), for each package. Enresa ensures that these calculations meet specific activity limits, (see Maximum Activity per VLLW Package on page 14). Enresa makes sure these calculations were done correctly for nuclides present in the packages/waste streams.

Based on the generator's/producer's report and datasheets, Enresa develops a Characterization Study for the waste packages. At this point Enresa is able to determine whether or not additional treatment will be required, at El Cabril, for the VLLW packages and proposed VLLW-Batches.

Finally the package contact dose rate data sheets are checked to see that they meet the following criteria [10]:

- Dose rate at any point of the package will not exceed 2mSv/hr
- Removable surface contamination from packaging will not exceed 4 Bq/cm^2 for ß-γ emitters and 0.4 Bq/cm^2 for α emitters and
- Accessible surface material transported unpackaged (soil/rubble in an open truck), with Enresa agreement, will not exceed 4Bq/cm^2 for ß-γ emitters and 0.4 Bq/cm^2 for α emitters.

2. For Packages Generated as LILW.

Enresa, conducts the same reviews to develop their Process Book.

3. For Historic Packages Generated Without an LILW acceptance Document.

Enresa conducts the same reviews on the generator's/producer's data sheets and ultimately develops the Characterization Dossier of the VLLW-DUs.

The three specific legislated requirements, generators/producers must follow to successfully meet the **Acceptance Criteria for VLLW-Disposal Units (VLLW-DUs):**

1. Acceptance Criteria for Packages,
2. Acceptance Methods for Packages, and
3. **Acceptable Methodologies for Activity Determination.**

Third Stage: Contract Acceptance

When the First Stage: Generic Acceptance and Second Stage: Document Acceptance, have been successfully achieved, the generator's/producer's contractual obligations to Enresa have been met. The result is Contractual Acceptance. Enresa references the Descriptive Document, the characterization document, the waste samples analysis and data sheets and puts together a unique VLLW-DU Acceptance File for DUs following the VLLW process. For LILW packages, and Historic Packages, following the LILW process, Enresa references the Descriptive Document (if there is one), the characterization document, the analyzed waste samples and data sheets and puts together a unique VLLW-DU Acceptance File for the corresponding batch.

This verification process has occurred while the waste was at the site of generation. Now Enresa is ready to accept management responsibility for the waste. Waste management responsibilities will be transferred to Enresa when the wastes are loaded on to Enresa's vehicle or the vehicle of an Enresa contractor, for transport to El Cabril.

Assignment of Responsibilities [10]

Up to this point the generator/producer has been responsible for:

1. Preparing the Package Descriptive Document,
2. Preparing packages according to the VLLW/ LILW Waste Acceptance Criteria,

3. Generating the waste packages according to the Package Descriptive Document approved by Enresa, and
4. Providing activity data and samples to Enresa, for package/waste stream verification.

Enresa's responsibilities have been:

1. Providing the generator/producer with the specifications for waste stabilization, activity assignment and scaling factors.
2. Verifying that the Waste Acceptance Criteria have been fulfilled.
3. Creating an Acceptance File for every batch formed by packages generated according to VLLW, LILW or historic documents. and
4. Accepting management responsibility for the waste and transporting it to El Cabril.

Prior to the management responsibility for the waste being transferred to Enresa, Enresa may reconfigure a batch or create a new one with the VLLW packages communicated by the generator/producer, or accept the VLLW-Batches as proposed by the generator. Generators can propose VLLW-Batches. However, Enresa has the final say on whether they will accept the Batches as proposed, reconfigure them with other VLLW-DUs or provide additional treatment to the proposed VLLW-Batches. Enresa makes these decisions prior to accepting management responsibility for the waste. If Enresa decides additional stabilization is necessary for a VLLW-Batch, it is done at the treatment building near the VLLW trenches at El Cabril. All of these changes are contained in the Acceptance File for the VLLW-Batch prior to transferring management responsibilities to Enresa. If Enresa chooses to change the configuration of a VLLW-Batch suggested by the generator/producer, it will notify the generator/producer of the change to the Batch configuration.

VLLW-Batch Specifics [3, 9]

Spain's LILW waste acceptance processes have always resulted in the generator/producer creating LILW-DUs for disposal. The VLLW acceptance process, being parallel to the LILW acceptance process also requires the formation of VLLW-DUs. The significant change in how VLLW is accepted versus the LILW is processed, is that a VLLW package must be included in a batch of VLLW packages. This concept was taken from the French VLLW disposal approach.

A VLLW-Batch has a particular composition and origin. However, there can be additional options available that influence its composition:

Composition of Batches:

A VLLW-Batch cannot contain both hazardous and non hazardous waste packages, because hazardous wastes require stabilization and non-hazardous wastes do not. For this reason, a VLLW-Batch cannot contain a group of packages, where some are stabilized and others are not.

Typical VLLW-Batch origin:

A typical Batch has the same producer/generator, type of waste, isotopic composition, stabilization process and type of container. After some years of experience, the batches are now more complex.

Alternative VLLW-Batch origin:

Enresa has the option of developing a VLLW-Batch from different types of containers, several different types of VLLW and waste that was produced by different generators/ producers.

If Enresa reconfigures a generator's/producer's VLLW Batch, they will notify the generator and communicate what changes were made.

Radiological Criteria for Disposal Units and Batches [3, 9, 10, 12]

As discussed earlier in this section, the Index of Acceptance for all VLLW-DUs is calculated by the generator/producer using the following equation provided by Enresa.

Note on a per package basis, the Index of Acceptance cannot be greater than ten.

Equation 5-2
Enresa Index of Acceptance

$$\mathrm{IA}_{\mathrm{DU}} \le 10 \quad \longrightarrow \quad IA_{DU} = \sum_{i} \frac{A_{Mi}}{A_{M\,\max i}}$$

IA	is the Acceptance Index
DU	is the Disposal Unit
A_{Mi}	is the specific activity of radionuclide *i* (Bq/g) in the mass waste
$A_{M\,max\,i}$	is the specific activity limit for radionuclide *i* (Bq/g) in a waste assuming radionuclide *i* is the only radionuclide in the waste

When a generator/producer proposes a VLLW-Batch to Enresa they are required to use the following Index of Acceptance formula. However, Enresa makes the final determination of what a VLLW-Batch will contain.

A VLLW- Batch Should Comply with the Following Radiological Criteria [3,6]:

Equation 5-3
Enresa Radiological Criteria

$$\mathrm{IA}_{\mathrm{Batch}} \le 1 \quad \longrightarrow \quad IA_{Batch} = \frac{\sum_{l} M_{l}\, IA_{DU\,l}}{\sum_{l} M_{l}}$$

IA	is the Acceptance Index
Batch	a group of Disposal Units as defined in **VLLW-Batch Specifics** sited above
l	is the number of packages/DUs in a batch
IA_{DUI}	is the Acceptance Index for each package/DU
M_i	is the mass of each package/DU

Final Verification

When VLLW-DUs arrive at the El Cabril site, final verification of the waste occurs with Enresa conducting the following tests at the El Cabril Laboratory:

1. Gamma Spectrometry,
2. Radiochemical analysis, and
3. Leaching tests.

Verifying Generator/Producer Documentation Through Periodic Monitoring [9]

Enresa's auditing of the generator's/producer's waste packages and waste acceptance documentation is the same for VLLW and ILLW.

Four Verification Types

There are four types of verification that can be carried out by Enresa:

Resource Control (Type I)

for approving a generator's VLLW Package Descriptive Document (developed by the generator) or the LILW Descriptive Document (see Figure 5.11)

Activity Control (Type II)

in cases of discrepancies between the declared activity and the activity determined by Enresa,

Audit Control (Type III)

Documental verification of the overall waste package production process. This occurs at least every 18 months for producers of LILW and VLLW or 500m3 of VLLW for VLLW producers.

Process Control (Type IV)

To check the correctness, according to the Package Descriptive Document and the LILW Descriptive Document, of the conditioning process for the type of waste package.

What are the Packaging and Transportation Requirements for VLLW?

Packaging Requirements

VLLW packaging is more for containing the waste than for hazard protection. Waste is packaged in plastic bags, liter drums, and metal boxes. [4]

Figure 5-13
"Big Bags" (plastic), 480- and 220-liter Drums and Metal Boxes [3, 10]

As identified in the "Acceptance Criteria for Packages" the generator's/producer's packages must meet the following dose rate and contamination levels prior to leaving the site of generation and are verified at El Cabril.

- Dose rate at any point of the package will not exceed 2mSv/hr
- Removable surface contamination from packaging will not exceed 4 Bq/cm^2 for ß-γ emitters and 0.4 Bq/cm^2 for α emitter.
- Accessible surface material transported unpackaged (soil/rubble in an open truck), with Enresa agreement, will not exceed 4Bq/cm^2 for ß-γ emitters and 0.4 Bq/cm^2 for α emitters.

In addition, each package is required to meet the criteria of the European Union (EU), 2007 International Carriage of Dangerous Goods by Road Act, which includes control of explosive materials, corrosive materials, oxidizing agents, flammable materials, pyrophoric materials (reactive metals), gases, biodegradable materials, free liquids, and a temperature >60°C (140^{o}F) [9].

Transportation Requirements

- Enresa uses the same shipping requirements for VLLW as those used for ILLW.
- Enresa is responsible for the shipment of radioactive waste from a generator's/ producer's site. As such, generators/producers are required to contact Enresa so Enresa can collect LILW and transport them to the El Cabil disposal facility.

- Prior to each shipment, Enresa inspects and checks all technical and administrative aspects of the waste, i.e., VLLW and LILW Package Descriptive Documents, and the vehicle involved to ensure safe transport of the wastes. For larger waste shipments Enresa's transportation contractors provide this function. The transport vehicles are specifically designed, with special shielding, automatic locking devices, etc. to further ensure safe transport of the wastes.
- The drivers of these trucks are knowledgeable, trained and able to respond appropriately in the event of an accident.

Regulations:

- Legislation on the transport of radioactive substances in Spain includes the European Agreement on the Road Transport of Hazardous Goods (ADR). Spain's radioactive waste shipments are carried out in accordance with the International Atomic Energy agency Regulations for the Safe Transport of Radioactive Materials [16].
- Enresa must give advance notice of its shipments to the Nuclear Safety Council (NSC, their nuclear regulator), the police, appropriate ministries, Civil Defense and the affected town council(s). The NSC inspects, on average, 100 LILW shipments per year. In the event of an incident during the transport of waste, a special contingency plan has been developed in accordance with the Civil Protection Board requirements [16].

What are the Disposal Site's Design Requirements?

Figures 5.2 to 5.7 provided general photos and drawings of the disposal trenches and how they are operated. The design of the El Cabril trenches, is based on hazardous waste disposal and not radioactive waste disposal. Each trench has a one-meter artificial geological barrier of compacted clay, complemented with geo-bentonite to provide a five-meter equivalent clay barrier. The isolation barrier also has a high-density polyethylene (HDPE) film with a leachate collection system located above it [12, 15, 17].

The following is a detailed listing of the layers that comprise a disposal trench liner and trench cap for the El Cabril VLLW trenches.

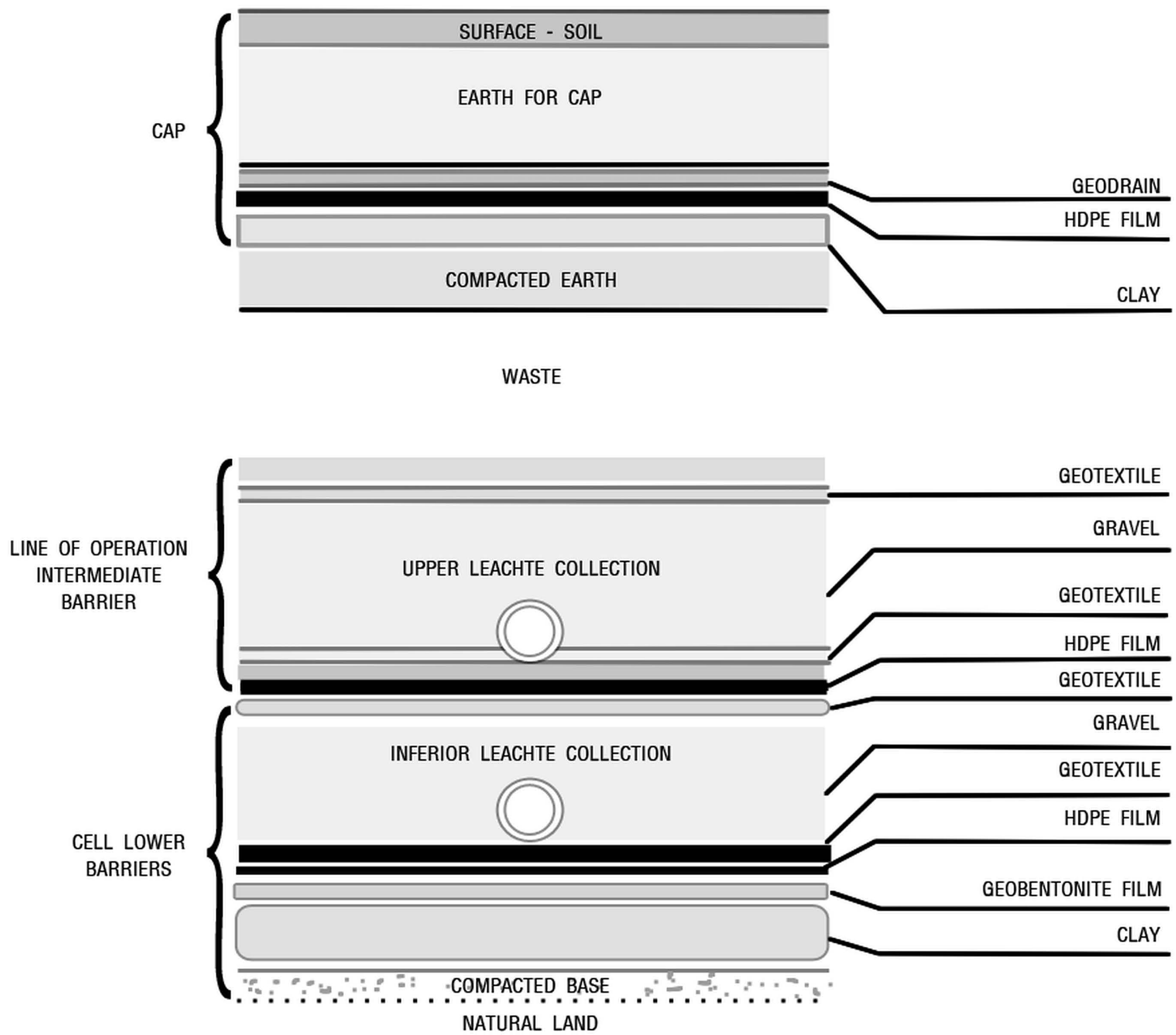

Figure 5-14
The Design of the El Cabril VLLW Trenches/Cells [12, 17]

The lower, "inferior leachate collection system", in the liner is capable of collecting ground water. The upper leachate collection systems have the potential to collect and identify which sections of the trench/cell are experiencing water intrusion. If there were a problem with water intrusion, Enresa would know which sub section of the trench/cell requires remediation.

The site's natural geologic barrier in combination with the complementary materials has a permeability equivalent to 5 meters of clay with K=1.00E-09 m/s [11].

What Scenarios are Used to Analyze the Potential for Radiation Exposure?

The Waste Acceptance Criteria for each of the disposal facilities at El Cabril (LILW and VLLW were based on, the radionuclide inventory after 300 and 60 years respectively [14]. The total activity level for the EL Cabril VLLW disposal facility (trenches 29-32) cannot exceed 1% of the total radioactivity planned for disposal in the LILW trenches (1-28) [14], below is a comparison of those concentrations.

Table 5-3
Maximum activity level for both El Cabril facilities [11,12]

Radionuclide	El Cabril Facilities Activity (TBq)	VLLW Facility Activity (TBq)
H-3	2.00E+02	2.00E+00
C-14	2.00E+01	2.00E-01
Ni-59	2.00E+02	2.00E+00
Ni-63	2.00E+03	2.00E+01
Co-60	2.00E+04	2.00E+02
Sr-90	2.00E+03	2.00E+01
Nb-94	1.00E+01	1.00E-02
Tc-99	3.20E+00	3.20E-02
I-129	1.50E-01	1.50E-03
Cs-137	3.70E+03	7.40E+00
Pu-241	1.15E+02	1.15E+00
Total alpha at:	300 years	60 years
	2.70E+01	2.70E-01

The regulatory dose limits are the design basis of the El Cabril facility [4,5]. The exposure scenarios were analyzed and resultant dose limits calculated, for the El Cabril facility using the RESRAD code. The above activity concentrations were derived from meeting the following exposure scenarios.

Table 5-4
Exposure Scenarios El Cabril LILW VLLW

		El Cabril LILW Facility [18]	El Cabril VLLW Facility [15]
Water			
	Normal ground water migration pathway	$\leq 10^{-1}$ mSv/y	
	Ground water pathway with cap failure	$\leq 10^{-1}$ mSv/y	
	Rise of water table	$\leq 10^{-1}$ mSv/y	
Accident			
	Aircraft crash (air exposure)	$\leq$ 5 mSv	
Human Exposure			
	Construction accident (air & external exposure)	**$\leq$ 1 mSv/y**	**0.1mSv/y**
	Air External exposure/Resident Scenario Resident (air & external exposure)	**$\leq$ 1 mSv/y**	**0.1mSv/y**
	Soil contamination (air & external exposure)	**$\leq$ 1 mSv/y**	**0.1mSv/y**
	Disposal Site Worker Exposure		**5 mSv/y**

How is the Disposal Site Monitored?

Enresa's VLLW license for the El Cabril VLLW site requires they collect and monitor any leachate from the trenches and check for cap subsidence for 60 years after site closure. [11, 12]

Although separated, the VLLW trenches are on the same property as the LILW trenches, which will be monitored for 300 years. [4]

Key Conclusions

Spain's VLLW Program

1. Dispose of VLLW in a hazardous waste disposal facility, owned and operated by the public company (Enresa) that operates their LILW facility.
2. Locate their VLLW facility on same site as operating LILW facility.
3. Accept only VLLW at the facility.
4. Construct and operate disposal trenches under a movable tent.
5. Require VLLW be submitted for disposal in batches.
   5.1. Individual package can have an Index of Acceptance (*IA*) < 10
   5.2. Batch of packages can have an Index of Acceptance (*IA*): < 1.
6. Enresa is heavily involved in the development of the Waste Acceptance File which is a key element of the waste characterization and ultimate acceptance of the waste.
7. Once waste is accepted, it is put on Enresa trucks or contracted trucks, all waste processing is done by Enresa and Enresa makes the final decision on the composition of all waste batches.

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6

THE USE OF RCRA HAZARDOUS WASTE DISPOSAL SITES FOR LOW ACTIVITY WASTE DISPOSAL

Introduction

The U.S. EPA's Resource Conservation and Recovery Act (RCRA) was enacted in 1976 [1]. RCRA provides a cradle to grave approach to regulating hazardous wastes [2]. EPA has divided hazardous wastes into four types [3], 1. Listed Wastes which EPA has determined is hazardous and has broken down into lists (F-list, K-list, etc.) and represent wastes from common manufacturing and industrial processes, and wastes that are more specific to certain industries. 2. Characteristic Wastes which are not in any of the lists above but are ignitable, corrosive, reactive or toxic. 3. Universal Wastes such as batteries, pesticides, items that contain mercury, like thermostats, and fluorescent bulbs. And 4. Mixed Waste which are wastes that contain both radioactivity and hazardous components [4].

When Congress enacted RCRA its intent was that the States have primary responsibility for implementing RCRA with EPA overview. All 50 states are authorized to implement RCRA. These state programs are at lease equivalent to and consistent with the federal law. [1]

Low Activity Waste (LAW) is referred to, by EPA, as Low-Activity Radioactive Waste (LARW) and as such is defined as waste with a low concentration of radioactive material [5] but there is no legal definition for LAW in the United States. However Low Activity wastes from a variety of sources have been disposed of at RCRA sites over the years. For a history of LAW in the United States the Nuclear and Radiation Studies Board's (NRSB's), Improving the Regulation and Management of Low-Activity Radioactive Wastes. It was published by The National Academies Press, Washington, D.C.in 2006 and contains an in depth review of this topic [6].

There are 21 operating Resource Conservation and Recovery Act (RCRA) Subtitle C hazardous waste landfills in the United States. Seventeen states have RCRA landfills and of those only two states, Texas and California have three RCRA facilities each.

Table 1 lists the city and state of each RCRA site, the facility operator and whether or not the site is known to have a Toxic Substances Control Act (TSCA) Permit. The availability of the TSCA permit listed here because these permits allow the RCRA facility to receive and dispose of Polychlorinated biphenyls (PCBs) [7].

Table 6-1
RCRA Subtitle C Hazardous Waste Facilities Operating in the United States

State	City	Facility Operator	TSCA Permit
Alabama	Emelle	Chemical Waste Management	Yes
California	Kettleman City	Laidlaw	
	Buttonwillow	Laidlaw	No
	Westmorland	Laidlaw	
Colorado	Deer Trail	Laidlaw	
Idaho	Grandview	US Ecology	Yes
Illinois	Peoria	Peoria Disposa	No
Indiana	Fort Wayne	Chemical Waste Management	
Louisiana	Carlyss	Chemical Waste Management	No
Michigan	Belleville	Wayne Disposal	Yes
Nevada	Beatty	US Ecology	Yes
New York	Model City	Chemical Waste Management	Yes
Ohio	Oregon	US Ecology	
Oklahoma	Waynoka	Laidlaw	
Oregon	Arlington	Chemical Waste Management	Yes
Pennsylvania	Pittsburgh	MAX Environmental	No
S. Carolina	Pinewood	Laidlaw	No
Texas	Deer Park	Laidlaw	Yes
	Robstown	US Ecology	No
	Andrews	Waste Control Specialists	Yes
Utah	Lake Point	Laidlaw	

There are eight RCRA sites (Grand View, ID, Cattleman City and Buttonwillow, CA, Deer Trail, CO, Andrews and Robstown, TX, Carlyse, LA and Belleville, MI) that have accepted Low Activity Wastes. Texas refers to this waste as Exempt Radioactive Waste (ERW), Two examples, as of 2006, The Deer Trail, Colorado site, had an authorization to receive wastes up to 74 Bq/g (2000 pCi/g) and the Buttonwillow, California site had an authorization to receive wastes up to 66.6 Bq/g (1800 pCi/g) for Naturally Occurring Radioactive Material (Norm) and Technologically Enhanced Naturally Occurring Radioactive Material (TENORM) [8].

The Grand View, ID, Robstown, TX and Andrews, TX facilities are unique in that they are permitted to accept LAW/ERW by their state regulators. Both Idaho and Texas are Agreement States and each state has its own approach to regulating LAW/ERW.

Agreement States

In 1959 Congress established the Nuclear Regulatory Commission (NRC) Agreement State Program. In 1962 the first states entered into individual agreements with the NRC, which allowed them to assume responsibility for licensing radioactive material that is under NRC's jurisdiction. Agreement States carry out the NRC regulations as they stand but have the option to impose stricter requirements. It is within NRC's purview to periodically review each Agreement State's regulatory program [9]. Today there are 37 Agreement States [10].

Andrews, TX Subtitle C Landfill

Briefly, in Texas radioactive material is regulated by both the Department of State Health Services (DSHS) and the Texas Commission on Environmental Quality (TCEQ). Under Texas regulations low activity wastes are exempt. The DSHS and TCEQ have a Memorandum of Understanding in place stating that material exempted by the DSHS rules can be disposed without regard. The details on the wastes that Texas RCRA facilities can accept are contained in their waste acceptance criteria [11].

As an Agreement State with ERW regulations, Texas has chosen not to involve the NRC (via the 10 CFR 20.2002 application process [12]) in the process for accepting ERW at Texas RCRA facilities.

Waste Control Specialists' (WCSs') Andrews, *TX* RCRA facility accept*s* ERW and will continue to *do so* when the *co-located,* Texas Compact *Low Level Waste disposal facility* begins operation this year [13]. *LLW is regulated by the NRC and EWR are those wastes not regulated under the Atomic Energy Act, as amended.*

Grand View, Idaho Subtitle C Landfill

US Ecology's Robstown, TX facility often refers potential ERW/LAW clients, to the Grand View, Idaho RCRA facility [14].

The State of Idaho has "Rules Regulating the Disposal of Radioactive Material Not Regulated Under the Atomic Energy Act of 1954, As Amended" which allows for the disposal of a variety of low activity wastes [15, 16]. In 2005 the Idaho Department of Environmental Quality (IDEQ) had exempted fission and activation products to Grand View's existing permit [17].

The Grand View, Idaho site permit, from the State of Idaho's Department of Environmental Quality, defines the activity limits the site can accept for LAW in Table C.4b of its Waste Acceptance Criteria [18]. The activity limit for byproduct materials is set at 111 Bq/g (3,000 pCi/g). This number takes into account daughter products of specific nuclides and in those instances results in lower Bq/g limits.

However every NRC licensee, interested in sending Low Activity Waste to the Grand View RCRA facility must file a 10CFR20.2002 request for alternate disposal with the NRC. Once the NRC approval of alternate disposal procedures is granted, the RCRA site owner US Ecology, submits the Safety Assessment to the Idaho Department of Environmental Quality (IDEQ). Additional reviews, which may include requests for additional information, are performed by the IDEQ to ensure that the material proposed for disposal will meet Grand View's Waste Acceptance Criteria for LAW [17, 19]. When approved by the IDEQ the waste can begin the process of preparation and disposal at the RCRA facility.

How is LAW Defined in Terms of Radioactivity and Chemical Content?

Radioactivity

There is no regulatory definition of Low Activity Waste. Though there have been a number of NRC alternate disposal procedure approvals, these have been evaluated based on a "dose standard" of "less than a few millirem per year" [20]. Therefore, it is difficult to define activity concentrations associated with LAW based on these approvals alone.

Chemical Content

Because there is no regulatory definition of LAW there is currently no chemical definition for this waste stream. However, as with any waste stream for disposal, the regulators are interested in the physical and chemical makeup of the nuclides present in the waste.

Since the waste will be disposed of in a RCRA Subtitle C facility, it is reviewed in terms of RCRA disposal requirements to ensure that the waste will meet those requirements. These are discussed in the "What are the Waste Acceptance Criteria (WAC) for LAW at a RCRA site?".

Does the U.S. Have an Operating VLLW or LAW Disposal Facility?

There are no operating VLLW disposal facilities per se in the U.S. However, Low Activity Waste can be sent to RCRA disposal facilities. In most states with a RCRA site, that do not have regulations that preclude the disposal of radioactive material in their RCRA facilities, this type of disposal occurs when an NRC Title 20.2002 request for alternate disposal has been filed with, and approved by the NRC. Currently, NRC's policy is to issue an exemption, under 10.CFR 30.11 concurrently with its 20.2002 approval [19].

Over the years NRC has approved a number of 20.2002 alternate disposal requests. These alternate disposal requests have included a variety of disposal alternatives. This report focuses on utility request for RCRA disposal alternatives. There have been two instances in the past, when the NRC approved of an alternative disposal request and the waste was approved for disposal in a non hazardous municipal RCRA Subtitle D landfill. See Consumer Power 2004 and 2001 in Table 6-2. However there have been a number of alternative disposal requests for disposing of decommissioning wastes in RCRA hazardous waste Subtitle C facilities. The most recent example was in November 2010, when the NRC approved the disposal of ~5,663 m3 (~200,000 ft3) from PG&E's Humboldt Bay NPP decommissioning waste. These materials contained 0.19Bq/g (5pCi/g) of Cs-137, 0.19Bq/g (5pCi/g) o Co-60 and 0.04 Bq/g (1pCi/g) of C-14 [19].

Which Nuclear Power Plant, Operating and Decommissioning, Waste Streams Qualify as LAW?

The following information is taken from the NRC's Enclosure 4 to SECY-06-0056, Improving Transparency in the 10CFR20.2002 Process, March 9, 2006 [21]. It is a list of only the nuclear power plant requests for alternate disposal that NRC approved. The table includes information on the nuclides of interest and a brief description on the wastes, which in each instance is demolition debris from decommissioning nuclear power plants.

Table 6-2
Utility 20.2002 Request Received by the NRC Between January 2000 and March 2006

Licensee	Submitted	RCRA Disposal	Materials	Exposure Limit 0.01 mSv - <0.05mSv
Connecticut Yankee (Note 1)	1/2005	√	28,317 m3 Demolition debris containing byproduct materials.	√
Yankee Atomic (Note 1)	12/0204	√	16,990 m3 (21,215 metric tons) Demolition debris Co-60 - Up to 0.74 Bq/g (20pCi/g) Cs-137 - 3.7 Bq/g (100 pCi/g) H-3 - 7.33 Bq/g (198 pCi/g)	√
Connecticut Yankee	09/2004	√ Note2	28,317 m3 Demolition debris. Cs-137, Co-60, C-14 and H-3 Concentrations very small. Other radionuclides present.	√

Table 6-2 (continued)
Utility 20.2002 Request Received by the NRC Between January 2000 and March 2006

Licensee	Submitted	RCRA Disposal	Materials	Exposure Limit 0.01 mSv - <0.05mSv
Consumers Energy Note 3 Big Rock Point	09/2004	Note 4	41,158 m3 Demolition debris. Cs-137, Co-60, and H-3 at low concentrations	√
Consumers Energy	12/2000	Note 4	9,910 m3 Demolition debris. Cs-137 – 0.006 Bq/g (0.17 pCi/g) Co-60 – 0.31 Bq/g (0.83 pCi/g) H-3 – 0.29 Bq/g (7.86 pCi/g)	√

Note:
1. Additional information is provided in Appendix B
2. Site owner decided not to pursue the disposal.
3. Amendment to a 2001 request to use Michigan municipal landfill.
4. Municipal landfills in Michigan (different than 2001 landfill).
5. Municipal landfill in Michigan.

What is the Waste Acceptance Criteria (WAC) for LAW at a RCRA Site?

This section looks at the RCRA Waste Acceptance Criteria (WAC) for a RCRA site [21]. And to the degree possible the WAC for a RCRA site accepting, Nuclear Power Plant decommissioning Low Activity Waste.

RCRA Waste Acceptance Criteria [21]

Determine waste acceptability for disposal at the RCRA site

A RCRA site's Part B Permit does not allow the disposal of: 1. Highly water reactive waste, 2. Explosive, pyrophoric, or sensitive to shock wastes, 3. Medical or biological wastes, or 4. Compressed gasses.

Pre-acceptance Protocol

To determine whether the site can properly treat, store and dispose of the waste, the generator must complete the Waste Product Questionnaire (WPQ). This document details on the process that generates the waste (process knowledge). Once the form is completed, the generator certifies that, 1. A representative sample was characterized in accordance with RCRA requirements, 2. All waste constituents have been identified, 3. The waste meets all applicable Land Disposal Restriction treatment standards, requires treatment or is subject to a variance, and 4. All information submitted in the WPQ correct.

The completed form is thoroughly reviewed by the site. Samples may be requested if the waste is being proposed for stabilization.

Waste Receipt Summary

When all reviews of the WPQ are complete a summary sheet of the WPQ information is produced. The Summary sheet establishes laboratory testing parameters for accepting the waste, identifies necessary personal protective equipment needed to handle the waste onsite, and any process testing that may be required to ensure that the stabilization process to be employed will be done satisfactorily.

Terms, Conditions, and Contract

Based on a successful WPQ review a contract is drawn up including price and terms and conditions. The Contract has to be on file with the site owner/operator prior to approval of a waste stream.

Approval Notification

When the Approval Notification is received this verifies that the owner/operator of the site has all necessary required permits and can receive the waste stream approved.

Waste Receipt Process

This process verifies that the waste received matches the acceptance criteria. In addition the generator can now schedule the waste shipment and fill out the Uniform Hazardous Waste Manifest. The manifest includes the waste stream identification number, shipment mode, number of containers and the transporter. The site transportation coordinator will verify the waste approval and date the waste can be accepted at the disposal facility.

When the waste arrives, the Receiving Department verifies the manifest, shipping papers and the Land Disposal Requirements Certificate are accurate. A work order is generated to track the waste acceptance, processing that is scheduled to occur on site and final disposal.

Depending on the requirements of the waste it may either go to storage, treatment or disposal. Samples of the waste may be taken, i.e., each bulk load and all containers are uncovered/opened, respectively, to be inspected and samples are taken from 10% of the containers. (TSCA (PBC) wastes are rarely sampled.)

These samples are taken to the onsite laboratory. This is done through a chain of custody. Depending on the waste there are a variety of tests that can be conducted. Upon completion of the laboratory analysis, if the sample passes the receipt parameters the waste is accepted

If there are any discrepancies found during the sample analysis and they cannot be resolved, the waste is rejected. This is usually done before the transporter is allowed to leave the site.

Once the waste has been verified it will either be treated, go into temporary storage or be disposed. As the waste is handled it is tracked using the work order generated when the waste was first received.

Testing Requirements after Treatment of the Waste

Any waste that has been treated has a specific sampling and testing protocol depending on the treatment process used. These testing requirements are related to the Land Disposal Restriction requirements.

Final Documentation Package

Depending on the number of processes the waste goes through there can be up to 11 documents that make up this documentation. This paperwork is maintained on site for three years.

For generators who have PCB waste there are additional procedures that need to be followed. Sites with a TSCA permit can receive PCB wastes.

LAW Waste Acceptance Criteria [18]

In this process, similar to the RCRA WAC, the waste goes through a pre-acceptance review to make sure the site can meet the treatment, storage and disposal requirements of the waste. This is done in a two step process. First the generator provides a chemical, radiological and physical characterization of the waste stream. Second is the pre-acceptance evaluation conducted by the RCRA site owner/operator determines whether the waste meets the requirements for acceptance in light of the facility's current permit(s). If the waste is acceptable the next step is for the generator (a radioactive waste licensee) to proceed with an NRC 20.2002 alternate disposal request.

Note: Because a RCRA facility is not licensed by the NRC, once the 20.2002 request is approved the current NRC policy is to exempt the low activity material from any further AEA and NRC licensing requirements. [18].

Steps:

1. Waste characterization for NRC Part 10 CFR20.2002 approval.

   a) If the RCRA site has a pre approved activity concentration the generator/licensee should submit sufficient information to the site to show that its waste meets the facility's approved limits and RCRA WAC. Assuming the waste does, the generator/licensee begins the process necessary to attain NRC approval for alternate disposal.

b) b. The licensee is responsible for preparing the radiological assessment. The RCRA site owner/operator is responsible for providing the generator/licensee site-specific data for the radiological assessment/RESRAD Safety Analysis for the 20.2002 submission. The licensee may perform the radiological assessment in consultation with the RCRA site owner/operator.

c) c. Once the NRC approves waste stream, in Idaho, the site owner presents the safety assessment to the state RCRA site regulator. This process may include additional requests for information from the regulator. When state approval is secured the generator can start shipping the waste following the RCRA site's Waste Acceptance Criteria (WAC).

Approval to Dispose of LAW in a RCRA Disposal Facility

2. Generator completes a Hazardous Waste Manifest for each shipment of material – this is a one page Environmental Protection Agency (EPA) developed form [23].
3. Generator also completes a Land Disposal Restriction Notification Requirements Checklist that RCRA site owner/operator can turn into a form. These requirements are part of the general notification and reporting requirements that are the basis of the cradle to grave (generation to disposal) philosophy of RCRA. These reporting requirements are in 40 CFR 268.7 (a) which contains the generator requirements and 40 CFR 268.7 (b) which contains the waste processor/treater requirements (40 CFR 268.7 (c) contains the disposal facility requirements) [24].

Container Requirements

Containers must:

1. Display 40 CFR Part 262.31 and 262.32(a)(b), markings and/or labels, if applicable,
2. Be compatible with the wastes,
3. Are required to be in good condition, and
4. Be approved by site owner/operator, prior to receipt if weighing more than 363 kg (800 pounds).

How Does the Disposal Site Verify the LAW Packages' Radioactivity and Chemical Content?

Since the Grand View, Idaho disposal facility has a history of accepting LAW; the site will be used as an example of how a RCRA facility can and does verify LAW packages as they arrive on site.

The steps completed prior to a LAW shipment being received at the RCRA disposal facility are an important piece of the waste verification process. The waste characterization required and conducted by the RCRA facility is the first and most important step in verifying the waste is acceptable.

Prior to Approval to Dispose of LAW in a RCRA Disposal Facility.

Steps:

1. Waste characterization for NRC Part 10 CFR20.2002 request for alternative disposal.

    a) The RCRA site owner/operator may be involved in developing or reviewing the generator's waste analysis

    b) RCRA site owner/operator is responsible for providing site-specific data for RESRAD Safety Analysis and. may be involved in developing the safety analysis submitted to the NRC.

    c) Once the NRC approves the proposed disposal procedures, The RCRA site owner/operator submits the safety analysis to the Idaho Department of Environmental Quality (IDEQ) for their review and approval.

    d) Once IDEQ approval has been secured the generator can start shipping the waste following the RCRA site's Waste Acceptance Criteria (WAC).

Post Approval to Dispose of LAW in a RCRA Disposal Facility

2. Generator completes a Hazardous Waste Manifest for each shipment of material – this is a one page EPA developed form [23].
3. Generator also completes a Land Disposal Restriction Notification Requirements Checklist that RCRA site owner/operator can turn into a form. These requirements are part of the general notification and reporting requirements that are the basis of the cradle to grave (generation to disposal) philosophy of RCRA. These reporting requirements are in 40 CFR 268.7 (a) which contains the generator requirements and 40 CFR 268.7 (b) which contains the waste processor/ treatment requirements (40 CFR 268.7 (c) contains the disposal facility requirements) [24].

Arrival of Waste Shipment – On Site Verification

The complete details of how the paperwork is handled are presented earlier in this section and apply equally to LAW as they do to RCRA hazardous wastes. The following steps are listed to include the specifics of a LAW shipment.

4. Guard shack notes incoming vehicle.
5. Truck proceeds to scale to confirm shipment weight.
6. Shipment paperwork, manifest, etc. are checked to see that signatures are in order, shipment weights are correct, and the wastes match the paperwork, etc.
7. The truck is survey – sides, bottom and top using gamma spectroscopy.

8. If the shipment is a DOT Class 7, a surface contamination survey is conducted on the wastes. This is to ensure they do not exceed DOT's surface contamination levels.

Upon successful completion of these verification steps the truck is directed to the operating cell for disposal of the waste [25].

What are the Packaging and Transportation Requirements for LAW?

Packaging Requirements

RCRA packaging requirements appear to be waste specific and most influenced by the degree of hazard, i.e., potential emissions from the waste. Most, RCRA facilities accept LAW waste in bulk shipments by trucks, intermodal boxes and roll-off boxes. They also accept a variety of truck and boxcar loads in drums, plastic bags and boxes (metal and non metal). For these three smaller containers there can be weight and strength related requirements for handling purposes.

Transportation Requirements

RCRA's transportation of hazardous waste requirements are found in 40 *CFR* 263. EPA's approach to addressing the transportation requirements was to adopted by reference most of DOT's Hazardous Materials Transportation Act (HMTA) (49 *CFR* 171 through 179). Because EPA references rather than incorporates the DOT's HMTA regulations, transporter companies transporting RCRA wastes must comply with both EPA's RCRA and DOT's HMTA regulations. [26]

When the material being shipped is considered radioactive by DOT standards, (§ 172.310) the transport vehicle will need to be shipped as Class 7 (radioactive) materials.

What are the Disposal Site's Design Requirements?

Table 6-3 and 6-4 lists the design elements of the cap and trench liner system for the Grand View, Idaho site and general RCRA requirements to identify similarities and differences. All RCRA disposal facilities have the RCRA required design features which may include particular types of barriers or specific permeability coefficients.

Table 6-3
RCRA Generic Cap Design and Grand View, Idaho RCRA Site Cap Design

US RCRA /Grand View, ID [16]		**US RCRA [28]**		
Radon Barrier 3.6m		General Trench Cap Design		
Trench Cap Design				
Top of Cap	76cm soil	Top of Cap	top soil with vegetation	
	Geotextile filter		60cm soil	
	Geonet drainage		Granular or Geotextile filter	
	Synthetic high density polyethylene liner		30 cm drainage layer	
	Granulated bentonite clay			
				Hydraulic Conductivity:
			- Geomembrane with overlying protective geotextile	- Impermeable
			- 0.6 m compacted clay[1]	- no greater than 1 x E-06^{-6} cm/sec
Bottom of Cap[2]	3m non-radioactive buffer material	Bottom of Cap	Geotextile gas collection layer	

Note: 1. Not recommended for arid environments due to potential cracking of clay layer should it dry out.
2. Below the bottom of the cap is the waste material.

Table 6-4
Trench Design

US RCRA /Grand View, ID [17,19]			US RCRA [22]		
Sidewall System Design			Sidewall System Design		
Surface of Sidewall			Storm water run-on and run-off design to meet to meet 25year storm		
	Filter fabric				
	Geonet for drainage				
	60mil HDPE Synthetic liner				
	Geonet for drainage				
	60mil HDPE Synthetic liner				
	Compacted Clay – 91 cm minimum, permeability < 1 X 10^{-7} cm/sec.				
Surface of Trench Liner			Surface of Trench Liner		
	76 cm Protective soil layer			- Drainage Layer	
	Protective Fabric				
	HDPE Geonet for drainage				
	60 mil HDPE Synthetic Liner				
	HDPE Geonet for drainage				
Bottom of Trench 1.75% slope			Bottom of Trench		
	60 mil HDPE Synthetic Liner				
	Perforated Leachate Collection Lateral, length of trench			- Primary Leachate Collection and Removal System -Top Liner	
	Two Layer Sump			- Drainage layer	
		Riser pipes – Leachate collection system		Leak detection System Secondary leachate Collection and Removal system Bottom Composite Liner	
Below bottom of Trench		Riser pipes – Secondary (Leachate) Collection		Low Permeability Soil Native Soil Foundation	

What Scenarios are Used to Analyze the Potential for Radiation Exposure and Dose Limits for LAW in a RCRA site?

Grand View, Idaho RCRA Facility

U.S. Ecology's Grand View, Idaho facility has accepted Low Activity Waste since 2000. The percent of LAW versus Hazardous/Non-Hazardous waste the facility accepts is increasing.

Table 6-5
Grand View, Idaho RCRA facility, Percent of Waste Accepted in Tons

Years	Percent By Weight	
	LAW	Haz/Non- Haz
2011 [25]	~50	~50
2000-2008 [27]	54	46
2000-2007 [17]	45	55

Table 6-6 lists the exposure scenarios that are analyzed using the RESRAD code by the licensee. The table also includes the dose limits that the licensee must meet. Besides the 15 mrem/yr criteria, the licensee must ensure that the onsite worker, the truck driver, transporting the waste to the disposal facility, maintain As Low As Reasonably Achievable (ALARA) doses.

Table 6-6
Exposure Scenarios Evaluated and Their Exposure Limits [18,30]

Exposure Scenarios	Dose Limits	
RESRAD Code During facility operation Member of the Public	1Sv/y, (100 mrem/y)	
Post Closure, Resident Farmer	0.15Sv/y, (15 mrem/y)	1,000 year period
Inadvertent Intruder	0.15Sv/y, (15 mrem/y)	
MicroShield Code Potential driver and worker doses during: Transportation, Transfer, and Disposal Operations	ALARA	
Disposal site worker, on site 25% of the year.	Not to exceed 400mrem	

The Grand View site has a 10-year permit. Every five years the site must conduct a review of its permit and submit to the Idaho Department of Solid Waste Management any changes that may need to be made or typos that were found in the original document [25]. Every 10 years they renew their license. The renewal process benefits from the five-year reviews because the permit has already been reviewed in detail by US Ecology.

The Idaho Department of Environmental Quality is the State agency responsible for regulating RCRA sites in Idaho. The state of Idaho passed a regulation that says the state will comply with EPA regulations and not impose regulations that are more stringent.

How is the Disposal Site Monitored Including Duration of Post Closure Monitoring [31]?

From EPA's takes a three prong approach to hazardous waste management. First the land disposal restrictions ensure that toxic constituents in hazardous waste are treated prior to land disposal. Next, the unit specific disposal requirements, where EPA's established design and operating requirements for disposal units, reduce toxic waste and prevent the release of hazardous substances into the environment. And finally the third element of their protection approach is ground water monitoring and is considered the last line of protection. Should land disposal restrictions and unit specific disposal fail, ground water monitoring will detect any releases so the cause of the release can be corrected.

Generic RCRA Requirements

EPA does not have a prescriptive ground water monitoring program. The regulations 40 CFR 264 Sub part F lists general requirements and what a successful ground water monitoring program must accomplish. The ground water protection system consists of three stases at a RCRA site are; Stage I. Detection Monitoring Program used during operation of the site to detect and characterize any releases of hazardous constituents from a disposal unit, Stage II. Compliance Monitoring Program is used once a leak is detected. The goal of this monitoring this program is to ensure the releases don't exceed compliance levels, and Stage III. Corrective Action Program used to bring a disposal unit or units back into compliance with the Ground water Protection Standard.

Closure and Post Closure Monitoring

The operator's closure plan is submitted at the time of applying for a site permit. The closure plan includes a description of how each hazardous waste management unit will be closed, how and when final closure will be achieved, an estimate of the maximum quantity of hazardous material will be on site for the life of the facility, how the site will be decontaminated and how groundwater and leachate monitoring, depending on disposal unit type, will be monitored.

RCRA hazardous waste landfills are required to be monitored for 30 years post closure. This period can be increased or decreased by the EPA Regional Administrator. The site owner or operator designs the post-closure plan for the facility. Post closure plans include required monitoring and maintenance of the site.

The plan describes the groundwater monitoring activities and maintenance activities. The focus of the post-closure care program is to maintaining the waste containment system and includes the following, 1. Maintaining the final cover, the leak detection system and the ground water monitoring system to detect any infiltration of water. 2. Protecting the disposal cell from the infiltration of water by promoting surface drainage and accommodating the settlement of waste, and 3. Making sure all disposal cell containment systems (final cover, liners, etc.) are not disturbed, and 4. Using ground water monitoring to detect any release of hazardous constituents from the site.

Key Conclusions

LAW Disposal in the U.S.

- In NRC's request for alternate disposal process has worked relatively successfully.
  - Delays in processing requests have been reduced.
  - NRC is investigating options for transparency to the 10 CFR 20.2002 process (for the public) [16].
- RESRAD analysis, safety assessment based on NRC's "a few hundredths of a mSv ($\leq$ 0.05 mSv) for maximally exposed individual.
- Grand View site located in arid environment.
- Shipping manifest more requires more information than LLW manifest but not very difficult.
- Pre shipping testing, etc. to determine hazardous waste or LAW acceptance for disposal is a rigorous process due to hazardous waste requirements.
- Packaging and Transportation requirements are familiar to operating and decommissioning power plant personnel.

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7 AGREEMENT STATE, TENNESSEE BULK SURVEY FOR RELEASE (VERY LOW ACTIVITY WASTE) PROGRAM

Introduction

Agreement State

In 1959 Congress established the Nuclear Regulatory Commission (NRC) Agreement State Program. In 1962 the first states entered into individual agreements with the NRC, which allowed them to assume responsibility for licensing radioactive material that is under NRC's jurisdiction [1]. Agreement States carry out the NRC regulations as they stand but have the option to impose stricter requirements. It is within NRC's purview to periodically review each Agreement State's regulatory program [1]. Today there are 37 Agreement States [2] including Tennessee [3].

Tennessee

The state of Tennessee has had and continues to have a number of companies that process radioactive waste from the Department of Energy, nuclear power plants, industry, medical facilities, etc. Over the years Tennessee has put in place a program that allows waste processors to verify that some wastes, post processing, are suitable for disposal in specified local landfills. The following is a brief discussion of the evolution of Tennessee's Bulk Survey for Release (BSFR) Program, its responsibilities, the Waste Acceptance Criteria, processing site verification program and landfill disposal site verification program.

Tennessee's Bulk Survey for Release (BSFR) Program

Under the umbrella of the Tennessee Department of Environment and Conservation, the Division of (Hazardous and) Solid Waste Management (DSWM) and the Division of Radiological Health (DRH) jointly implemented the state's Bulk Survey for Release (BSFR) Program.

The Division of Solid Waste Management (DSWM) is responsible for regulating all solid waste landfills in Tennessee. For the BSFR Program they regulate the receipt and disposal of BSFR in Class 1 (municipal solid waste) landfills. This is done through a Special Waste Approval process [4]. Using the Special Waste Approval process DSWM reviews an applicant's data on the BSFR waste it proposes to dispose of and determines whether or not it is acceptable solid waste for a TN Class 1 landfill (equivalent to a RCRA Subtitle D facility).

The Division of Radiological Health (DRH) is responsible for regulating the four waste processing companies that send BSFR wastes to approved municipal Class 1 landfills. This is done through its licensing and inspection process. A brief description of this process is presented later in this Section.

Tennessee Department of Environment and Conservation (TDEC)

In the early 1980s the TEDC's Department of Radiological Health began receiving requests from processors to evaluate and approve the disposal of very low activity waste in municipal landfills. Eventually the DHR decided to fashion their review process after the NRC's Title 10 CFR20.2002 request for alternate disposal. Over time the DRH has moved away from approving each request individually and in 1997 put a BSFR Program in place. The BSFR Program now refers to a "license process", where the DRH approves a radioactive materials processor's request to allow "extremely low levels" of radioactive material to be disposed of in specific Class 1 municipal landfills [5]

As an Agreement State, Tennessee's DRH evaluates a request (by a Tennessee radioactive waste processor) for disposal of radioactive material in a facility other than a Low Level Waste Disposal Facility, i.e., the use of "alternative disposal".

Because the state of Tennessee has chosen to design their process on the NRC's Title 10 CFR 20.2002, the organization or person requesting the alternative disposal option, must first describe the waste. This description includes radiological, physical and chemical properties. These characteristics determine how the waste is to be disposed. The description will also include information on the environment of the disposal facility and the conditions of waste disposal.

The applicant then provides an analysis and evaluation that characterizes the environment of the disposal facility or area where the waste will be disposed. The applicant then outlines the procedures that will be followed to ensure potential doses are As Low As Reasonably Achievable (ALARA)[5]. The state of Tennessee makes reference to NRC's occupational dose limits for adults (annual total effective dose of 0.05 Sv [5 rem]) and limits for individual members of the public (doses not expected to be greater than 1 mSv/y [100 mrem/y]) [5], etc. Tennessee's dose limits are outlined in the Department of Radiological Health's CHAPTER 1200-2-5, Standards for Protection Against Radiation [6]. However the actual exposure limits used by the BSFR Program are significantly lower, and are shown in Table 7-3.

The following radioactive waste processors in Tennessee have licenses to dispose of BSFR wastes, in specified local landfills [7, 8]:

- IMPACT, Oak Ridge
- Studsvik, Memphis
- TOXCO, Oak Ridge
- EnergySolutions, Oak Ridge

How Does the State of Tennessee Define BSFR in Terms of Radioactivity and chemical Content?

Radioactivity

Division of Radiological Health

The IAEA Section discusses the difficulty in providing generic criteria for VLLW because each disposal facility requires an independent safety assessment. DRH is responsible for determining the activity limits allowable for the BSFR license. Therefore a safety assessment of each disposal facility is needed. These assessments consist of using the RESRAD code to determine allowable activity concentrations, based on the state regulator's dose limits. The radioactive material processor requesting the license provides the regulator with the RESRAD safety assessments.

Table 7-1 lists the limits for a subset of activity concentrations that can be disposed of at the Carters Valley landfill in Hawkins County, the North and South Shelby landfills in Shelby County and the Chestnut Ridge landfill in Anderson County. These values were derived by each of the processors applying to dispose of BSFR wastes at these disposal facilities. As Table 7-1 shows, there are differences between landfills when running the RESRAD code with the same exposure limits.

Table 7-1
Activity Limits for Specific Landfills in Three Tennessee Counties (0.01mSv/y) [9]

Nuclides	Hawkins County	Shelby County North	Shelby County South	Anderson County
Nuclide limits in Bq/g				
H-3	37	5	0.074	208
C-14	1	0.02	0.03	0.3
Co-60	1	0.05	0.05	3
Ni-63	37	6	6	136
Sr-90	0.2	0.01	0.01	0.5
Cs-137	0.01	0.2	0.2	1
Pu-241	11	3	3	134
Pu-239,	0.6	0.1	0.1	4
Pu-240	0.6	0.1	0.1	4
Am-241	0.6	0.1	0.1	4
U-234	2	0.4	0.4	10
U-236	3.7	0.4	0.4	21
U-238	3.7	0.2	0.2	11

Chemical Content

Division of Radiological Health (DRH)

The DRH requires that the physical and chemical aspects of the radionuclides in the waste be included in the license application.

Division of Solid Waste Management (DSWM)

Tennessee's Class 1 solid waste landfills accept a variety of wastes including, municipal solid waste, household waste, waste tires, landscaping and land clearing wastes, construction and demolition wastes and farming wastes [10].

BSFR waste requires an approval under the DSWM "Special Waste" program. These applications are separately reviewed and approved by the DSWM and include the chemical content of the waste.

There are no RCRA Hazardous Waste (Subtitle C) Landfills in Tennessee [11]. Class 1 landfills may have a stabilization pit where liquid wastes are stabilized prior to disposal. Waste materials that are a) Flammable (b) Corrosive; (c) Reactive; (d) Pyrophoric; or (e) Promote rapid oxidation are not accepted in a Class 1 landfill.

Does the State of Tennessee Have Landfill Disposal Facilities Accepting BSFR Waste?

The State of Tennessee has four landfills that accept Bulk Survey for Release (very low activity) wastes from four Tennessee radioactive materials processors with BSFR licenses. A fifth landfill the Middle Point landfill in Rutherford County, accepted BSFR wastes until the summer of 2007. At that time the operator stopped accepting BSFR due to the local community's concerns. The following table provides the names, locations, and classifications for these landfills.

Table 7-2
Landfills Accepting BSFR Licensed by the TDEC's Solid Waste Division [7, 9]

Name	Location	Owner/Operator	Processor with BSFR Permit	Tennessee Classification	Federal RCRA Classification
North Shelby	Shelby County	BFI / Allied Waste	Studsvick	Class 1 landfill	Subtitle D landfill
South Shelby	Shelby County	BFI / Allied Waste	Studsvick	Class 1 landfill	Subtitle D landfill
Chestnut Ridge	Anderson County	Waste Management	IMPACTS and TOXCO	Class 1 landfill	Subtitle D landfill
Carters Valley	Hawkins County	BFI /Allied Waste	EnergySolutions	Class 1 landfill	Subtitle D landfill

Which Nuclear Power Plant (NPP), Operating or Decommissioning, Waste Streams Qualify as BSFR?

There are a variety of BSFR wastes from operating and decommissioning nuclear power plants that have been disposed of at these Tennessee landfills. The following is a representative list:

- Contaminated soils,
- Ion exchange resins, such as blow down resins,
- Contaminated concrete in the form of rubble,
- Construction/demolition debris,
- Wood,
- Asphalt,
- Dry Active Waste (DAW), Paper, Plastic, Clothing [5, 7].

Though the regulations have changed, see list of nuclide concentrations for 2012 in Table 7-1, it is safe to assume that the majority of wastes listed above will continue to qualify for disposal as BSFR.

What is the Waste Acceptance Criteria (WAC) for BSFR?

The TDEC requires that the processor seeking a license to dispose of BSFR conduct an environmental assessment of the landfill where they intend to dispose of the wastes. As such, the North Shelby, South Shelby, Chestnut Ridge and Carters Valley landfills have undergone separate modeling analyses of their site characteristics. Based on the modeling analyses and the projected volume of waste to be disposed of, the activity limits for each specified landfill are calculated and set by the regulator. These activity limits form the basis of the Waste Acceptance Criteria [12].

DRH, Environmental Assessment/Pathway Analysis Modeling

Tennessee's Department of Radiation Health requires the use of the RESRAD code for assessing the suitability of a potential landfill facility for accepting BSFR wastes. The processes modeled and analyzed are:

1. Water infiltration through engineered barriers,
2. Radionuclide leaching, from a variety of waste types,
3. Projected degree of transport of radionuclides through the environment,
4. Analysis of a variety of human exposure pathways, and
5. Estimating the dose to humans [5].

The potentially affected Resident Farmer, Inadvertent Intruder and Site Worker (on site 25% of the year) are analyzed using the RESRAD code. The Truck Driver's exposures (transporting BSFR waste from the processor site to the landfill) are calculated using the ALARA principals. In order to accomplish this, the code analyzes a variety of site specific characteristics (composition of the soil, dimensions of compacted soil under the disposal cell, distance to the groundwater, annual rainfall, etc.) and the proposed or existing engineered barriers, i.e., trench liners and trench cap. This is standard procedure in environmental assessment/exposure pathway modeling, as outlined by the IAEA, and accepted by the NRC, the DOE, and others.

DRH requires analysis of The Resident Farmer Scenario [5] because it takes into account the most exposure pathways of any scenario analyzed by the model and results in the highest predicted lifetime doses [5, 13]. DRH places the following requirements on how parameters are handled in the code analyses [5].

- 0.01mSv/y (1 mrem/year) dose to the maximally exposed individual.
- The BSFR Material is uniformly distributed throughout the landfill and results in a 1 mrem/y Total Effective Dose Equivalent (TEDE).
- Model assumptions do not include clean cover soil between the Resident Farmer and the waste.
- In the Unsaturated Zone, no credit is taken for the landfill's plastic/synthetic liner.
- In the Unsaturated Zone, below the disposal cell, only the landfill's clay liner and soil buffer are included in the analysis.
- No credit is given for the cover over the landfill (which can be included in a RESRAD analysis and State and Federal Solid Waste Management programs require a cover be constructed prior to closing a landfill facility).
- Of the total annual waste quantity received by the Class 1 landfill disposal facility, no more than 5% can be BSFR material.
- When the site is closed it is assumed that the BSFR waste disposed to date has not decayed. Therefore over estimating the amount of radioactivity present. [5]

Should the DRH find any of the numbers used in the applicant's pathway analysis are not sufficiently documented and justified, DRH will use the RESRAD probability and sensitivity analysis to develop its own activity concentration values. These values will be more conservative.

The characteristics of a disposal facility can vary greatly. Because of this, the DRH requires that either the most restrictive parameter value is used in the codes, or a new analysis be submitted each time a new cell is developed at the site [12]. However, each approval is for a specific area of the disposal facility and each new area of the disposal facility proposed for BSFR disposal will require the latest modeling and updated parameter values. Annually, the license holders are required to analyze the disposal facility's Resident Farmer scenario, using the cumulative activity the site has received to ensure that at the time of closure the Resident Farmer will receive no more than a 0.01 Sv/yr TEDE. This analysis is run for a 20 year post closure period and a 1000 year post closure period [14].

Mixtures of Nuclides - Sum of the Fractions Rule

The DRH provides processors with the following equation to evaluate mixtures of radionuclides for disposal and provides the following basic formula to determine the sum of the fractions which must not exceed 1. [14]

Equation 7-1
DRH Evaluation Mixture Formula

$$\frac{concA}{\lim A}+\frac{concB}{\lim B}+\frac{concC}{\lim C}=\leq 1$$

The concentrations, in a package, of each radionuclide *A*, *B* and *C* are listed as *concA*, *concB* and *concC*. The regulatory limits, allowed in a package, for each nuclide are *limA*, *limB* and *limC*.

Waste Packages [14]

DRH requires that surface radiation levels of a BSFR container not exceed 0.5 µSv or 50 µrem/hr at one (1) meter from each surface including the top and the bottom of the container.

Mixing Wastes

Wastes above the license limit cannot be mixed with clean or lower activity BSFR wastes.

Nuclide Concentration

Nuclide concentration is based on the waste itself and does not include any stabilization material that may be added.

Radionuclides that are greater than 0.1% (by activity) of the waste mixture must be quantified. These results shall be used in deriving pass/fail limits for the BSFR waste being analyzed.

Material Evaluation

Material for BSFR disposal is evaluated based on the size of the container received in a shipment of waste. It cannot be analyzed in a larger container or a container that has a smaller container of the waste placed in it.

Annual Mass Allowed for Disposal

The mass limit is 5% of the total waste received at the disposal facility during the three (3) years preceding the last full year of disposal at the site. An adjusted mass total can be calculated by multiplying the actual mass of each waste package by the dose fraction (df) for that package.

Scaling Factors

A detailed isotopic and concentration analysis of each shipment must be provided. Process knowledge and scaling factors cannot be the only methods used in these calculations. Scaling factors have to be established annually for each specific waste stream and must be based on appropriate laboratory methodology.

The processor is tasked with demonstrating that the waste is homogeneous and the scaling factors are consistent throughout the waste.

Quality Control Program

Representative, Quality Control samples are required for each shipment. These samples will be analyzed to determine the Lower Level of Detection (LLD, equivalent to10% of the disposal limit) for each radionuclide used as a basis for the waste stream's scaling factors. If the nuclide's LLD is not determined to be 10% or less of its limit then the processor is to assume the nuclide is present at its LLD value.

Wastes without a gamma component will have an assay measurement for each 400 cubic feet of waste again determining LLDs equivalent to 10% or less. If these LLDs cannot be determined the processor will assume the nuclide is present in the LLD quantities [14].

License Requirements

The DRH license requirements are the basis for the Waste Acceptance Criteria outlined above. For additional license details, refer to Tennessee's Chapter 1200-02-11, Requirements for Land disposal of Radioactive Waste, Nashville, revised in March of 2010 [15].

Division of Solid Waste Management (DSWM)

Special Waste Re-Certification

Generators are required to annually re-certify that there has been no change in the waste stream since the original special waste approval was granted by the DSWM. For further details on the Special Waste approval process see Tennessee Rule 1200-1-7 [15, 16].

How Does the Disposal Site Verify the Waste Packages' Radioactivity and Chemical Content?

Because the radioactive material processor is licensed to send the BSFR to a specific disposal facility, verification of the waste is predominantly its responsibility. The disposal facility operator's verification is limited to a preset gate monitor.

The following describes the waste verification process from the generator to the gates of the landfill facility.

Waste Generator Defines the Waste

Prior to shipping the waste to a licensed radioactive material processor, the generator analyzes the waste on site. From this analysis the generator provides the processor with a description of the waste. This is done with a pre shipment summary form. The form requests general information such as, shipment weight, total activity in pCi and the highest contact dose rate. It requests more detailed information on the material properties of the waste such as physical state, flash point, reactivity, pH, etc. It addresses chemical properties such as water reactivity, alkaline reactivity, presence of organic compounds, volatile organics, oxidizing agents, etc. It also requests information on potential hazardous characteristics. For example, are specific RCRA listed wastes present, and weather the waste is defined as hazardous by local or state regulations. The actual forms are more detailed and are available from each of the four processors listed in Table 7-2.

The waste is then shipped to the processor. Shipments must be conducted according to U.S. Department of Transportation (USDOT) regulations [5].

Radioactive Materials Processor

The processor reviews the information on the shipment to make sure it can receive the waste. An external gamma detector measures the activity levels at various locations on the container to confirm the actual activity levels agree with the generator's activity analysis. Shipments will either be BSFR upon receipt or are processed with the resulting material meeting the BSFR criteria for disposal. In either case the processor determines that the activity levels meet the predetermined levels authorized by their Radioactive Material License for BSFR Disposal.

Once confirmed that the authorized levels have been met, the processor ensures that the container surface dose rate does not exceed the license limit of 0.5 µSv/hr (50 µrem/hr). Finally if the waste qualifies as radioactive material (USDOT's definition) it cannot be shipped to the landfill facility [5].

As addressed earlier in the approval process, the Division of Solid Waste Management (DSWM) is responsible for the materials that are sent to the landfill. When a processor makes a "special waste" request, they provide the DSWM with a description of the BSFR material for approval. After the approval is received, it submits to DSWM a "profile" of each waste shipment. The profile contains information on the maximum radiation distribution in the waste, the composition of the waste, how it was generated, waste generator information, how the waste was packaged for final disposal and to which disposal facility the waste will be sent. The fee for reviewing each profile is $300. The waste shipment cannot include anything that was not in the originally submitted BSFR profile.

Preauthorized Landfill

When the waste shipment arrives at the landfill facility, the facility's gamma spectrometer is set to alarm if the shipment exceeds activity level approved for BSFR shipments. If a shipment were to trip the alarm the truck would immediately return to the processor.

Processor Reporting to DRH

Processors have to report quarterly on the shipments of BSFR waste. These quarterly reports include the total activity and total volume shipped to the approved landfill. The report includes any rejected shipments to the processor's site and any shipments that may set off the landfill's radiation gate alarm.

What are the Packaging and Transportation Requirements for BSFR?

Packaging Requirements

There are no packaging requirements placed on generators or processors for BSFR wastes. In DRH's Licensing Requirements document [14], DRH does refer to large containers such as intermodals. Package types that are received by the processor often include drummed and bagged wastes (plastic bags).

Transportation Requirements

BSFR material shipped to approved landfills cannot exceed the USDOT activity concentration limits for exempt material given in 49 CFR 173.436 or determined according to procedures in 49 CFR 173.433 [14].

What are the Disposal Site's Design Requirements?

Division of Solid Waste Management Class 1 Landfill Requirements [10]

The Division of Solid Waste Management implemented new Liner requirements in 1993 and updates its Class 1 landfill requirements periodically. A very brief description of Tennessee's May 2001 liner, leachate and gas collection requirements follows. For a detailed discussion see RULES OF TENNESSEE DEPARTMENT OF ENVIRONMENT AND CONSERVATION

DIVISION OF SOLID WASTE MANAGEMENT, CHAPTER 1200-1-7, SOLID WASTE PROCESSING AND DISPOSAL, May, 2010 (Revised).

Liner requirements

The liner system must function for the life of the site and the post closure care period (30 years).

- A composite liner system consisting of:
  - Upper component – Minimum 30-mil flexible membrane liner,
  - Lower component - A 0.6 meter (two foot) layer of compacted soil with a hydraulic conductivity of no more than 1 x 10^{-7} cm/sec, overlaid with
  - A flexible membrane liner,
    - At least 60 mil High Density Polyethylene,
    - In direct contact with the soil (lower component, and
    - Welded and tested.

    Underlying the liners – a geologic buffer with either 10E-05 or 10E-06 hydraulic conductivity.

Leachate migration control

The leachate collection system will be installed immediately above the liner.

- A leachate collection system will consist of the following:
  - Chemically resistant material suitable to waste in facility and strong enough to prevent collapse under pressure of overlying waste, etc.
    - Leachate collected, will be treated either at a wastewater treatment facility permitted to receive such waste water or other methods approved by the Commissioner of the Tennessee Department of Environment and Conservation.

Gas collection system

A gas extraction/collection system will be installed during operation of the landfill cells.

- Gas will be collected and vented, recovered or otherwise managed to preclude pressure buildup, concentration of explosive gases.

Groundwater monitoring

The groundwater monitoring program will be conducted during the facility's operating life and during the post closure care period.

- Groundwater monitoring programs will include the downstream sampling of:
  – Soils above bedrock to communicate with shallow water in soils, and
  – Bedrock fractures, to communicate with deep groundwater [8].

What scenarios are used to analyze the potential for radiation exposure and what are the BSFR dose limits?

Table 7-3 lists the exposure scenarios that are analyzed by the licensee. The table also includes the dose limits that the licensee must meet. Besides the 1 mrem/y criteria, the licensee must ensure that the worker, who works at the processor's site processing the waste and the truck driver, bringing the waste from the processor's site to the disposal facility, maintain As Low As Reasonably Achievable (ALARA) doses.

Table 7-3
Exposure scenarios evaluated and their exposure limits [5, 10]

Exposure Scenarios	Dose Limits
Resident Farmer	0.01Sv/yr, (1 mrem/y)
Inadvertent Intruder	0.01Sv/yr, (1 mrem/y)
Truck Driver bringing waste from processor site to disposal facility	ALARA
Disposal Site worker, on site 25% of the year.	ALARA
Post Closure (Note 1)	0.01Sv/y, (1 mrem/y)

Table 7-3's exposure scenarios include the Resident Farmer, the Inadvertent Intruder and Post Closure scenarios. The Resident Farmer and Inadvertent Intruder are analyzed for when the site is closed and the land is released. Post Closure is identified, because when the site is actually closed and the cap, etc. is put in place the regulator will compare the site's original closure plan with how the closure was actually conducted. If anything is different the RESRAD analyses will be rerun with the changes to verify that the Resident Farmer and Inadvertent Intruder scenarios continue to meet the 0.01Sv/yr, (1 mrem/yr) or less exposure limit set by the TDEC. Otherwise, the only time the disposal facility would need to update their RESRAD analysis would occur if the landfill were changed in a way that could affect the outcome of the original RESRAD analysis, i.e., a new/different type of leachate system were installed.

How is the Disposal Site Monitored Including Duration of Monitoring?

The landfill operator's involvement prior to receiving BSFR waste is to cooperate with the processor's pathway analysis, providing available site-specific disposal facility data. This includes items such as site meteorology data, geology data and design specifications of engineered barriers used at the site. Note that the synthetic liner and the final cover are not used in the calculations.

Upon obtaining a license to dispose of its BSFR material at the specified landfill, the processor begins sending BSFR to the landfill. It is the responsibility of the landfill operator to monitor the disposal facility.

For the most part, the monitoring requirements are the same as they were before BSFR waste was licensed for disposal at the landfill. For this reason the general monitoring requirements are listed but only elaborated upon when additional monitoring is required due to the acceptance of BSFR. The latest rules for solid waste processing and disposal [10] do not contain explicit time frames for sampling. The August 2006 regulations [17] include sampling time frames and are included as a framework for the discussion.

Monitoring requirements during operation include:

- Conducting routine methane monitoring, via on-site methane monitoring wells quarterly [10]
- Collecting groundwater samples,
- Collecting leachate for sample analysis (and treatment)
- Keeping records that show compliance [11].

Monitoring will be conducted at least semi-annually during the active life of the BSFR solid waste disposal facility. Monitoring will continue during closure activities and the post-closure period. What will be monitored and frequency will be finalized at the time of site closure.

Closure Requirements

A compacted final cover material, such as soil, will be placed on the disposal facility within at least 90 days of the site's initiating closure. This final cover system will consist of at least 36 inches of soil a minimum of 12 inches shall be for the support of vegetative cover. The final cover system's infiltration rate can be no greater than 1 x 10-7 cm/sec and the regulations offer alternative options that could result in lower rates [15].

Post-Closure Care Period

After the date of final closure, post-closure care will continue for 30 years unless a shorter period was approved in the closure/post closure care plan.

Post-closure will include maintaining:

1. Final contours and drainage system,
2. An established healthy vegetative cover,
3. Drainage facilities, sediment ponds, and other erosion/sedimentation control systems until the vegetative cover is sufficiently established,
4. A leachate collection, removal, and treatment system, if present,
5. A gas collection, monitoring and control system, and
6. A ground and surface water monitoring system will remain in place. Semiannual monitoring will be continued during the post-closure care period and reports will be sent to DSWM within 30 days of sample analysis.

Finally, upon completion of the post closure care period for the solid waste landfill facility, the owner or operator must file a certification of closure with the DSWM (2006 regulations are more specific in listing requirements than 2010) [17].

It is widely believed among landfill operators that post closure monitoring could continue past 30 years.

Key Conclusions

BSFR Disposal in Tennessee

- BSFR levels are equivalent to Free Release.
  - TDRH will make the BSFR program transparent in 2012 by making the individual radionuclide activity concentrations available to the public.
- RESRAD analysis, safety assessment is based on a 0.01mSv dose to the maximally exposed individual.
- Each of the four landfills that are allowed to accept BSFR are RCRA subtitle D municipal landfills as opposed to RCRA subtitle C hazardous waste disposal facilities.
- There is nothing unusual about the shipping and handling of these wastes, either on their way to the processors or to the municipal landfill.

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8
REVIEW OF EXPERIENCES WITH VERY LOW LEVEL AND LOW ACTIVITY WASTE DISPOSAL

Focus of Report

The focus of this report can be broken down into three specific areas,

1. The development of the VLLW category by the International community and its use in Europe,
2. A review of how and where Low Activity Waste is disposed of in the U.S. and
3. A comparison of European and U.S. experiences to determine whether using a VLLW category is a viable option for U.S, utilities.

Development of the VLLW Category

IAEA published a new waste categorization system in 2009 to address the need for a more comprehensive radioactive waste classification system. In addition the report linked each waste category to the appropriate form of disposal based on the hazard of the waste and ultimately the long term safety of the public. When linking wastes with specific disposal types the IAEA was careful to point out that actual disposal would depend on a site specific safety analyses to ensure that it would deliver doses less than or equal to the doses mandated by each countries regulator(s).

Supporting Factors for a VLLW Category

Ultimately the IAEA provided the foundation for implementation of a VLLW category when it defined Exemptions in 2004 in terms of an acceptable exposure limit to the public of 1 μSv/y. With this number they were able to calculate the specific activity limits for each radionuclide. With this in place the next steps were to:

4. Define VLLW. VLLW's activity level is equal to or greater than the activity level for Exempt Waste (EW) but according to the IAEA VLLW does not require a high level of containment or isolation. Furthermore VLLW generally has very limited concentrations of longer lived radionuclides.
5. Target definition of limits for concentrations. IAEA suggest that the VLLW activity concentrations could be one or two orders of magnitude greater than the EW concentrations.

Besides Exempt Waste, VLLW is the only other waste the IAEA attempts to numerically define.

6. Provide a starting point for defining the exposure limit for Exempt Waste. The EW exposure limit is no more than 1 μSv/y (0.1 mrem/y). This does not suggest that IAEA intends for this to be the limit for VLLW. Having a number for one category helps frame the discussion when evaluating the next category. Hence it gives regulators a starting point for determining an acceptable exposure limit for VLLW waste disposal.
7. Link disposal options with waste hazard. The latest IAEA waste categorization system not only provides more categories of waste but it links each category with a disposal option(s) commensurate to the hazard of the waste being disposed of. In the case of VLLW, engineered “near surface” landfills, having limited regulatory control, may also contain other hazardous waste.
8. Point out the necessity of a site specific safety assessment. From a site specific safety assessment the specific activity concentrations for the waste can be derived along with relevant Waste Acceptance Criteria.
9. Note that radiation protection will be greater than the requirements for exemption from regulatory control, but the extent of protection is limited compared to higher classes of radioactive waste.

France and Spain Adopt the VLLW Category

Prior to 2000, France and Spain, recognize the increasing volumes of VLLW that were being, or about to be generated in their respective countries. Their solution was to add VLLW to their radioactive waste categories and develop VLLW disposal facilities. These disposal facilities are non nuclear in nature which would reduce the cost of disposal while preserving the dwindling LILW disposal capacity. Both countries use “public companies”, ANDRA in France, and Enresa in Spain, to manage their radioactive waste, from treating, conditioning, storing, and disposing of the material.

VLLW Management

The similarities in both countries approach to VLLW management are:

1. Dispose of VLLW in a hazardous waste disposal facility, owned and operated by the public company that operates their LILW facility.
2. Locate their VLLW facilities on the same site as or within two kilometers of their operating LILW facility.
3. Accept only VLLW at the facilities.
4. Construct and operate disposal cells/trenches under a movable tent.
5. Require VLLW be disposed of in batches.
6. Characterize the waste up front and rely heavily on process knowledge.

BSFR and LAW disposal in the U.S.

- In the U.S. both programs have worked relatively successfully.
- Quicker review times of 10 CFR 20.2001requests for alternate disposal, by the NRC are being pursued.
- Tennessee's BSFR Program is in place and case-by-case reviews are no longer required.

A Comparison of the Three Programs Reviewed

The following table highlights the differences between VLLW, BSFR and LAW.

Table 8-1 shows the activity limits of a number of radionuclides associated with each program. The defined values for VLLW in France and Spain are higher than the BSFR and NRC 20.2002 process. Due to the variability in how these numbers are derived it is not surprising that the VLLW and LAW do not have exactly the same values.

Table 8-1
Summary of Limits on the Transportation and Disposal of VLLW

		VLLW Disposal Programs		BSFR Program	NRC 20.2002 permit	
Radio-nuclide	IAEA Bq/g	France Bq/g	Spain Bq/g	Tennessee BSFR Processors Bq/g	Yankee Rowe 20.2002 Exemption for WCS Bq/g	Connecticut Yankee 20.2002 Exemption for US Ecology Idaho Bq/g
H-3	10,000	1,000	1,000	0.074 to 208	N/A	48,000
C-14	100	1,000	1,000	0.02 to 1	N/A	16.2
Co-60	10	10	10	0.05 to 3	20	1
Ni-63	10,000	1,000	1,000	6 to 136	N/A	Note 2
Sr-90	100	1,000	1,000	0.01 to 0.5	N/A	Note 2
Cs-137	10	10	30	0.01 to 1	100	3.4
Eu-152	10	10	N/A	0.3 to 1.8	N/A	Note 2
Pu-241	1,000	1,000	1,000	3 to 134	N/A	Note 2
Pu-239,	10	10	10	0.1 to 4	N/A	Note 2
Pu-240	10	10	10	0.1 to 4	N/A	Note 2
Am-241	10	10	10	0.1 to 4	N/A	Note 2

Notes: 1. N/A - Not Available
2. These radionuclides have very low dose consequences but due to their ratio to Co-60 and/or Cs-137 would be limited by the concentrations of those radionuclides present.

Determining VLLW Impact

To determine the impact of a VLLW category in the U.S. specific activity limits were selected that would make sense in the U.S. The final column in Table 8-2 defines the limits for a new possible U.S VLLW classification proposed in this report. These limits have been determined so as to meet all of the following criteria:

- Are no higher (lower for some of the radionuclides) than the French and Spanish VLLW limits and the IAEA guidance on VLLW limits.
- Are no higher (lower for some of the radionuclides) than the U.S. DOT exempt limits for transport of radioactive material.

Table 8-2
Summary of Limits on the Transportation and Disposal of VLLW

Radio-nuclide	IAEA Bq/g	France Bq/g	Spain Bq/g	Tennessee Processors Bq/g	Yankee Rowe 20.2002 Exemption for WCS Bq/g	Connecticut Yankee 20.2002 Exemption for US Ecology Idaho Bq/g	U.S.DOT Exempt. Limits Bq/g pCi/g	Possible U.S. VLLW Limits Bq/g pCi/g
H-3	10,000	1,000	1,000	0.074 to 208	N/A	48,000	1,000,000 27,000,000	1,000 27,000
C-14	100	1,000	1,000	0.02 to 1	N/A	16.2	10,000 270,000	100 2,000
Co-60	10	10	10	0.05 to 3	20	1	10 270	10 270
Ni-63	10,000	1,000	1,000	6 to 136	N/A	Note 2	100,000 2,700,000	1,000 27,000
Sr-90	100	1,000	1,000	0.01 to 0.5	N/A	Note 2	100 2,700	100 2,700
Cs-137	10	10	30	0.01 to 1	100	3.4	10 270	10 270
Eu-152	10	10	N/A	0.3 to 1.8	N/A	Note 2	10 270	10 270
Pu-241	1,000	1,000	1,000	3 to 134	N/A	Note 2	100 2,700	100 2,700
Pu-239,	10	10	10	0.1 to 4	N/A	Note 2	1 27	1 27
Pu-240	10	10	10	0.1 to 4	N/A	Note 2	1 27	1 27
Am-241	10	10	10	0.1 to 4	N/A	Note 2	1 27	1 27

Notes: 1. N/A - Not Available
2. These radionuclides have very low dose consequences but due to their ratio to Co-60 and/or Cs-137 would be limited by the concentrations of those radionuclides present

Table 8-3 shows the waste volumes and cost benefits from the resulting reduction in waste volumes.

Change in Volume of Decommissioning LLW from the Introduction of the VLLW Waste Class

Table 8-3 shows that it is projected that a total of 70.5 Million ft^3 of Class A waste could be reclassified to Very Low Level Waste with the institution on this new waste classification. As previously noted this total is likely below what the actual reduction would be due to the conservatisms used in determining the estimate.

Table 8-3
Projected Waste Volumes for Operating and Decommissioning of All Plants

Waste Classification	Total Operational and Decommissioning Waste from 2011 to 2059 (All Plants) ft3/yr	Percentage of Total LLW Volume from All Sources
Current Classification Situation		
Class A Waste from Decommissioning	84.4 Million	73.5 %
Class B/C Waste from Decommissioning	166,000	0.2 %
Total Decommissioning LLW	84.6 Million	73.7 %
Class A Waste from Operating Plants	29.8 Million	26.0 %
Class B/C Waste from Operating Plants	344,000	0.3 %
Total Operational LLW	30.2 Million	26.3 %
Total LLW - All Sources	114.8 Million	100 %
Changes to Waste Classification With Establishment of VLLW Classification (B/C Volume Unchanged)		
Class A Waste from Decommissioning	28.7 Million	25.0 %
Very Low Level Waste from Decommissioning	55.7 Million	48.5 %
Class A Waste from Operating Plants	15.0 Million	13.1 %
Very Low Level Waste from Operating Plants	14.8 Million	12.9 %
Total Class A Waste Reclassified to VLLW	70.5 Million	61.4 %

From these potential volumes, that can be disposed at a lower cost, we can calculate the expected Disposal cost saving.

$6 Billion in Cost Savings Potential

Figure 1-2 shows a projected total savings of $6.2 Billion (expressed in 2011 dollars without escalation) should a VLLW classification be made available as described in this report. As has been previously discussed, this is likely an underestimate as the assumptions used to determine the waste volume and cost savings estimates are conservative.

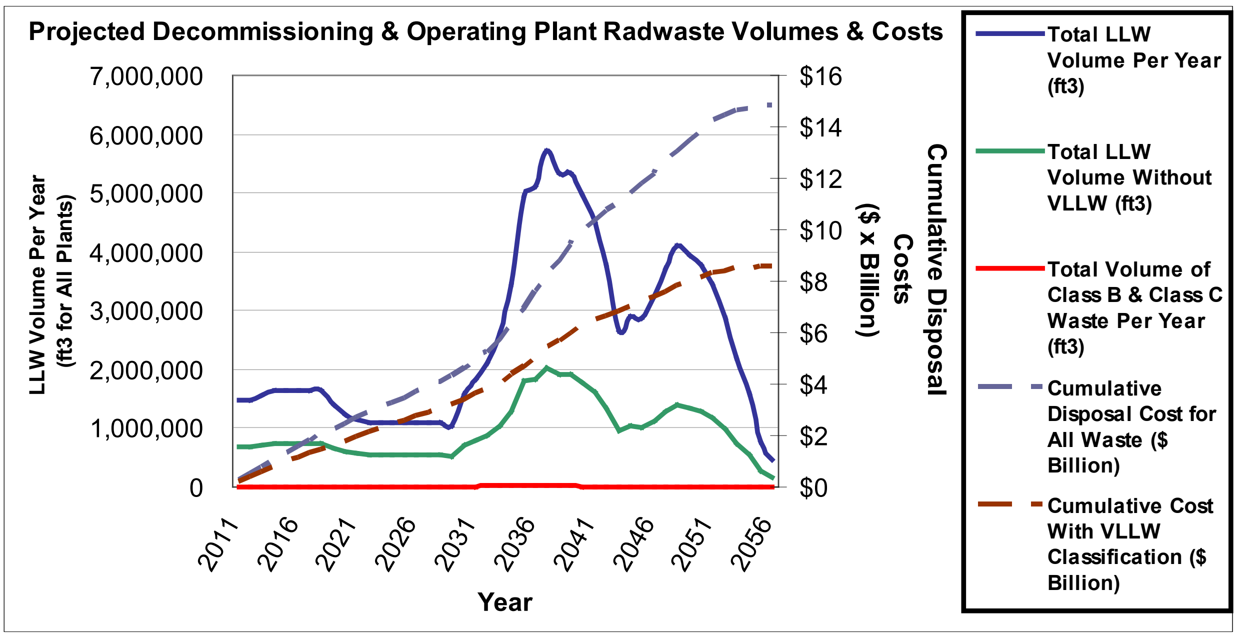

Figure 8-1
Estimated Disposal Cost Savings from a VLLW Classification

Will VLLW be Difficult to Implement?

The next consideration is how difficult it will be to implement a VLLW classification on an operational basis.

Table 8-4 shows that except for France the 0.01mSv exposure limit is used in most cases, and as explained in the note after Table 8-4, the actual results are an order of magnitude less than reported in the literature. Note the range of numbers for the BSFR activity limits were derived at four sites using the same exposure limit.

Looking specifically at the BSFR Activity Limits we can see that the numbers reported for BSFR vary rather significantly. The numbers happen to be for tritium from two of four municipal landfills analyzed in Tennessee. The numbers show that the testing from actual site analyses can result in significantly different values.

Table 8-4
Landfill Parameter for VLLW, BSFR and LAW

Waste type	Disposal Facility	Activity Concentration Bq/g	Exposure Limit mSv/y
VLLW			
IAEA Reports: Safety Standards No. GSG-1 Safety Standards Series RS-G-1.7	Municipal Landfill or Landfills with other chemicals	10 – 10,000	No VLLW Exposure Limit Exempt Waste Exposure Limit of 1 µSv/y
France	Hazardous Waste	10 – 1,000	Public Exposure << 0.25mSv/y
			Disposal Site Worker 5mSv/y
Spain	Hazardous Waste	10 – 1,000	Maximally Exposed Individual 0.01mSv/y
			Disposal Site Worker 5mSv/y
LAW			
U.S.NRC10 CFR 20.2002 Request for Alternative Disposal	RCRA Subtitle C Hazardous Waste Landfill	1- 48,000	Maximally Exposed Individual a few mSv/y - < 0.05 mSv/y (“a few mrem/y”, < 5mrem/y)
	RCRA Subtitle D Municipal Waste Landfill		
BSFR			
Tennessee	RCRA Subtitle D Municipal Waste Landfill	.01 - 208	Maximally Exposed Individual 0.01mSv/y (1mrem/y)

Note: 1. France uses 0.25 mSv/y for the disposal sites exposure limit the actual number is at least an order of magnitude lower (see Section 4 “Use of VLLW in France”).

VLLW is Simple to Implement for Packaging and Transportation Requirements.

From a packaging and transportation requirement, there is nothing out of the ordinary for using the VLLW category here in the U.S.

Table 8-5
Packaging and Transportation Operational Requirements

WASTE CATEGORIES	PACKAGING REQUIREMENTS	PACKAGES USED	TRANSPORTATION REQUIREMENTS
VLLW			
IAEA	Waste does not need a high level of containment	NA	ADR Low specific activity waste
FRANCE	Contain wastes and Ease of handling package	Big Bag 480 & 220 liter Drums Boxes	ADR Generator hires transportation company
SPAIN	Contain wastes and Ease of handling package	Big Bag 480 & 220 liter Drums Metal Boxes	ADR Enresa transports small quantities Contracts transports for larger quantities of waste
LAW			
NRC 10 CFR20.2002	NA	Truck Roll-off Intermodal boxes	U.S. DOT Exempt Limits or Class 7 – Radioactive Material
BSFR			
TENNESSEE Agreement State 10 CFR20.2002	Rolloff Drums Bags	Roll-offs Drums Bags	DOT Exempt Limits

Comparing the French and the Spanish disposal site verification programs there appear to be differences. Enresa appears to be more inclined to request samples from the generator/producer and analyzing them.

A preliminary review of the waste manifests indicates that there can be more forms to complete, requesting more information than typically required by a LLW waste manifest. However the difference in paperwork should not be an obstacle.

A thorough evaluation of how France and Spain run their VLLW programs provides the data needed to determine whether or not there will be generator/producer specific problems in using the VLLW option.

Summary

There are European countries currently using the Very Low Level Waste category. Their approaches are based on sound practices that specifically address the continuing safety of the public. If the U.S. were to implement a VLLW category where the wastes were disposed of in RCRA Subtitle C Disposal Facilities, the nuclear industry and the public would benefit from lower costs and the public would continue to be adequately protected.

9
PROJECTED VOLUME OF VLLW IN THE U.S.

Estimated U.S. LLW Volumes that Qualify for a VLLW Classification

As is discussed in previous Sections of this report, a technical basis exists for the establishment of a Very Low Level Waste (VLLW) disposal classification. This Section provides an estimate of the volumes of LLW that would potentially qualify for the various waste classifications. Estimates of the total volume of Class A, B and C LLW and VLLW have been prepared for the remaining operating life and/or decommissioning period for:

- Plants currently operating *and,*
- Plants that are Permanently Shutdown

Operational LLW

Table 9-1 shows the references and assumptions used to prepare the operational waste volume estimates. EPRI Report # 1013506 (Title: Technical Development of New Low Level Waste Disposal Options: Industry Strategic Database) stated a value of 49% of the total volume of LLW had a contact dose rate that was less that 1 mr/hr and that this waste would likely be within the disposal limits for Green is Clean (GIC) waste. Table 2-1 gives limits for a number of processors that operate GIC or Bulk Survey for Release (BSFR) processes. It can be seen in Table 2-1 that all of the GIC and BSFR limits are well below the Projected VLLW limits shown in Table 2-1. Therefore, using percentage of LLW estimated to meet GIC limits in EPRI Report #1013506 to estimated the potential volume of VLLW that would come from operating plants is conservative.

Table 9-1
Estimated Operational LLW Volume

Waste Classification	ft3/yr	Basis for Volume Used
PWR Operational Waste		
Total A, B and C Waste (for each PWR)	10,311	EPRI Radbench Summary Data- Industry Average for Total Low Level Radwaste for 2009
Total B/C Waste (for each PWR)	146	EPRI Radbench Summary Data- Industry Average for B/C Low Level Radwaste for 2009
Volume Estimated to Meet VLLW Limit	5,124	49% of total volume of LLW per EPRI Report # 1013506
BWR Operational Waste		
Total A, B and C Waste (for each BWR)	10,314	EPRI Radbench summary data- Yearly Average for Total Dry Solid waste for 2007 to 2009
Total B/C Waste (for each BWR)	66	EPRI Radbench Summary Data- Industry Average for Total Low Level Radwaste for 2009 (See Detail Below)
Volume Estimated to Meet VLLW Limit	5,082	49% of total volume of LLW per EPRI Report # 1013506

LLW Volumes During Decommissioning

EPRI has published a number of experience reports for the nuclear power plant decommissionings that have been conducted in the United States. Low Level Waste volume data was taken from the following reports to complete Table 9-2 for the decommissioning of PWR plants:

- EPRI Report # 1011734, Maine Yankee Decommissioning – Experience Report (Reference 9-1)
- EPRI Report # 1013511, Connecticut Yankee Decommissioning – Experience Report (Reference 9-2)
- EPRI Report #1015121, Rancho Seco Nuclear Generating Station Decommissioning Experience Report (Reference 9-3)

Table 9-2
Decommissioning Waste Volume Estimates

Waste Classification	**Total Waste Volume ft3/plant**	**Basis for Volume or Fraction Used**
PWR Decommissioning Waste		
Total A, B and C Decommissioning Waste Volume	612,200	Conservatively based on total volume of A, B and C waste reported for Rancho Seco. Volumes reported for Connecticut Yankee and Maine Yankee were on the average 5.7 times higher than Rancho Seco.
Fraction of Total that is Very Low Level	0.66	Conservatively based on "Green is Clean" waste percentage (66%) of the total reported for Rancho Seco. Percentage at CY was > 98%, Percentage at Maine Yankee was at least 88%
Total B/C Decommissioning Waste Volume	15,090	Used the Average of the Volume for Following Three Plants: **Rancho Seco** - 3,270 ft3 (Only plant with detailed breakdown), **Connecticut Yankee** - 14,500 ft3 (Conservatively Assumes 50% of Primary Components and High Activity LLW was Class B/C) **Maine Yankee** - 27,500 ft3 (Conservatively Assumes 50% of Primary Components and High Activity LLW was Class B/C)
BWR Decommissioning Waste		
Total A, B and C Decommissioning Waste Volume	1,000,000	Based on Volume of Waste Estimated for the Humboldt Bay Decommissioning (658,000 ft^3) and the average decommissioning waste volumes published in the reference. As the Humboldt Bay plant is relative small (65 MWe), It is expected that a more typically sized BWR will have higher waste volumes. The volumes estimated in this report for BWRs is approximately half of the average decommissioning waste volume for the plants listed in the reference. (Reference: EPRI Report # 1023025, decommisioning EWxperiences and Lessons Learned: Decommisioning Costs, November 2011)
Fraction of Total that is Very Low Level	0.66	Conservatively Use Same Basis as PWRs Above. Humboldt Bay estimates indicate that more than 75 % of the total waste volume would qualify as VLLW.
Total B/C Decommissioning Waste Volume	15,090	Conservatively Use Same Basis as PWRs Above. Humboldt Bay is projecting only 3,649 ft3 of B/C waste

Additionally, Appendix A contains a summary of radionuclide characterization data for soil and building demolition concrete LLW from the decommissioning of various power plants in the US. The conclusion of that appendix is that a very high percentage of this decommissioning waste has activity levels that would be below the VLLW limits proposed in this report. This data and conclusion support the assumptions used to determine the volumes in Table 9-2.

The volumes that have been estimated for decommissioning waste have used conservatisms that subsequently reduce the estimated cost savings resulting from instituting a VLLW classification by:

- Underestimating the total volume of Decommissioning LLW
- Underestimating the percentage of LLW that would potentially qualify as VLLW
- Overestimating the volume of Class B/C waste such that the saving from VLLW is a smaller fraction of the total radioactive waste cost.

Determination of Yearly LLW Volumes

To determine the yearly volumes of waste projected for each waste classification a spreadsheet was created such that the yearly volumes for each plant could be entered and the sum for each year determined.

For the 11 plants that have been permanently shutdown:

- It is assumed that the decommissioning will take 8 years and that the waste volume for each year will be 1/8th of the total.
- The Humboldt Bay (Decommissioning complete in 2015) and the Zion (Decommissioning complete in 2020) plants are in active decommissioning. Their LLW volume was evenly spread over the decommissioning period.
- For the remaining 9 plants that have been permanently shutdown, the schedules shown on the "Decommissioning" page of the NRC Website were used to determine the start of their decommissionings.

For the 104 operating nuclear power plants:

- Each plant was assumed to operate to the end of its current NRC operating license period except for the plants that have not yet gained a license extension. In the case plants that have not yet had their licenses extended, it was assumed that a 20 year extension was granted and operational waste generation rates were used for those additional 20 years.
- Once an operating plant had been permanently shutdown, it was assumed that the decommissioning would start immediately and would take 8 years to complete. As with the currently shutdown plants, the total decommissioning waste volume was assumed to be spread evenly over the 8 years of the decommissioning.

Total Yearly Waste Volumes:

To calculate the estimated total yearly waste volumes, the values in the spreadsheet column for each year were totaled. To determine the volume of VLLW and Class B/C waste for each year, the fractions of the total volume stated in Tables 9-1 and 9-2 were applied to the applicable total volume. By subtracting the volumes of VLLW and Class B/C waste from the total of LLW, the volume of waste that would be Class A with a VLLW classification in place was determined.

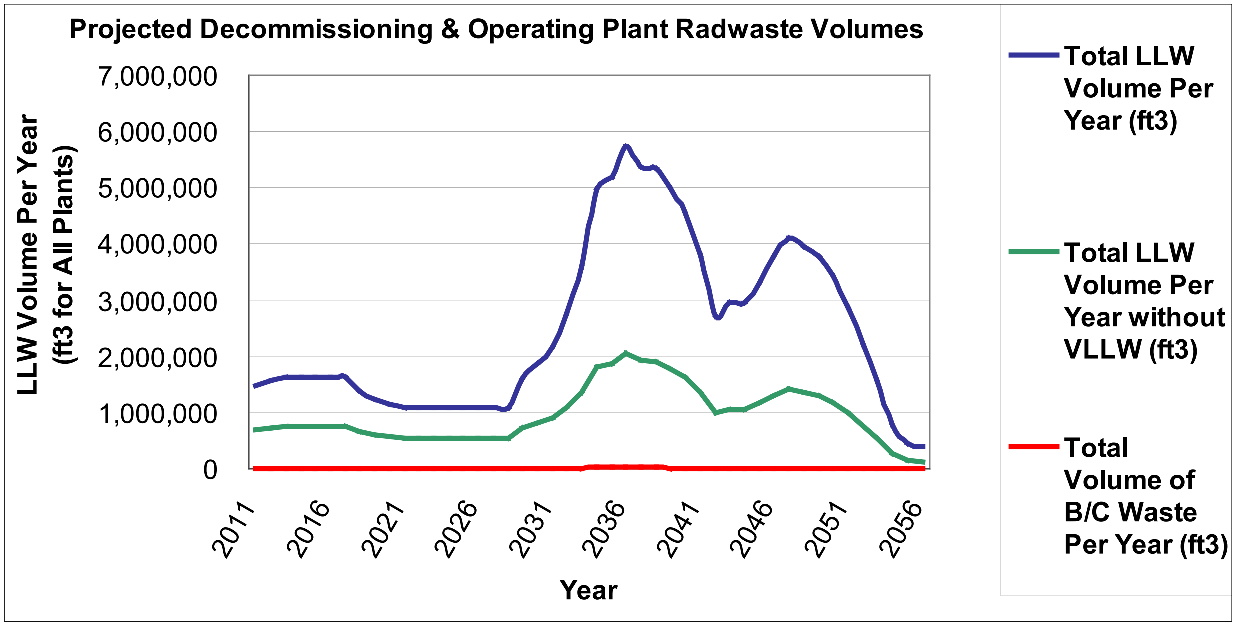

Figure 9-1
Volumes of Radioactive Low Level Waste from the Plant Operations and Decommissioning

Figure 9-1 shows the summary of the above analysis of projected future radioactive waste volumes from the operation and decommissioning of the nuclear power plants in the US. The following can be observed from Figure 9-1:

- Except for some additional volume due to the decommissioning through 2020, there is a fairly even projected volume of LLW of approximately 1 Million ft^3 per year through 2029 when the first of the current operating plants reach the end of their NRC operating license period.
- While the current operating plants are being decommissioned from 2029 thru 2056 there is a significant increase in yearly LLW volumes. The highest projected volume is 5.7 Million ft^3 in 2037 with a second smaller peak volume of 4.1 Million ft3 in 2048.
- The volume of Class B/C waste is very small compared to the total volume of Class A waste (Less than a half of a percent)

Utilizing the yearly estimates determined above, Table 9-3 shows the total volume estimates for the various waste classifications from 2011 to 2056 calculated with or without the institution of a VLLW classification.

Table 9-3
Projected Waste Volumes for Operating and Decommissioning of All Plants

Waste Classification	Total Operational and Decommissioning Waste from 2011 to 2059 (All Plants) ft3/yr	Percentage of Total LLW Volume from All Sources
Current Classification Situation		
Class A Waste from Decommissioning	84.4 Million	73.5 %
Class B/C Waste from Decommissioning	166,000	0.2 %
Total Decommissioning LLW	84.6 Million	73.7 %
Class A Waste from Operating Plants	29.8 Million	26.0 %
Class B/C Waste from Operating Plants	344,000	0.3 %
Total Operational LLW	30.2 Million	26.3 %
Total LLW - All Sources	114.8 Million	100 %
Changes to Waste Classification With Establishment of VLLW Classification (B/C Volume Unchanged)		
Class A Waste from Decommissioning	28.7 Million	25.0 %
Very Low Level Waste from Decommissioning	55.7 Million	48.5 %
Class A Waste from Operating Plants	15.0 Million	13.1 %
Very Low Level Waste from Operating Plants	14.8 Million	12.9 %
Total Class A Waste Reclassified to VLLW	70.5 Million	61.4 %

Table 9-3 shows that it is projected a total of 70.5 Million ft^3 of Class A waste could be reclassified to Very Low Level Waste with the institution on this new waste classification. As previously noted this total is likely below what the actual reduction would be due to the conservatisms used in determining the estimate.

The above estimated volumes will used in the next Section to determine an estimate cost saving from the institution of a VLLW Classification.

10
POTENTIAL COST SAVINGS WITH VLLW CLASSIFICATION

The last Section determined estimates of the volume of LLW waste that would potentially qualify as VLLW with the institution of such a new waste classification category. This Section will determine the estimated waste disposal cost savings should a VLLW classification become available.

Unit Costs of Waste Disposal

In order to calculate waste disposal costs based on the volume estimated above, unit disposal costs were applied based on the values in Table 10-1

Table 10-1
Unit Waste Disposal Cost Assumptions

Disposal Cost Assumptions	$/ft3	Basis
Cost of Class A Disposal at NRC Licensed Part 61 Site	100	Based on Industry Experience
Disposal of VLLW at RCRA facility	10	Conservatively based on 10:1 Ratio of LLW to RCRA Disposal Cost. It is believed that actual disposal of this category of waste at a RCRA facility would be lower.
Disposal Costs for Class A Resins and Filters	500	Based on Industry Experience
Waste Disposal for B/C Waste at a Part 61 Site	4,500	Estimate based on expected cost of Class B/C waste disposal

Cost Savings Calculation

The above unit pricing was applied in the volumes estimated for the various waste classifications in the spreadsheet discussed in the last Section for two cases as follows:

- Case 1: No VLLW Classification (current situation)
- Case 2: VLLW classification is established

Once the various cost segments were determined in the spreadsheet, they were summed for Case 1 and Case 2 for each year. Next the total for Case 2 was subtracted from the total for Case 1 for each year to determine the yearly cost saving projected with a VLLW classification available. Finally for each year the cumulative savings from 2011 to that year was determined. It should be noted that unit costs were not escalated for the years after 2011 such that all costs and savings are in 2011 U.S. Dollars. This likely underestimates the disposal cost savings as waste disposal cost have historically escalated at a rate which is approximately 5% higher than the escalation of other costs.

The results of these calculations are shown in figure Y-1 along with the waste disposal volume estimates determined in Section 2.

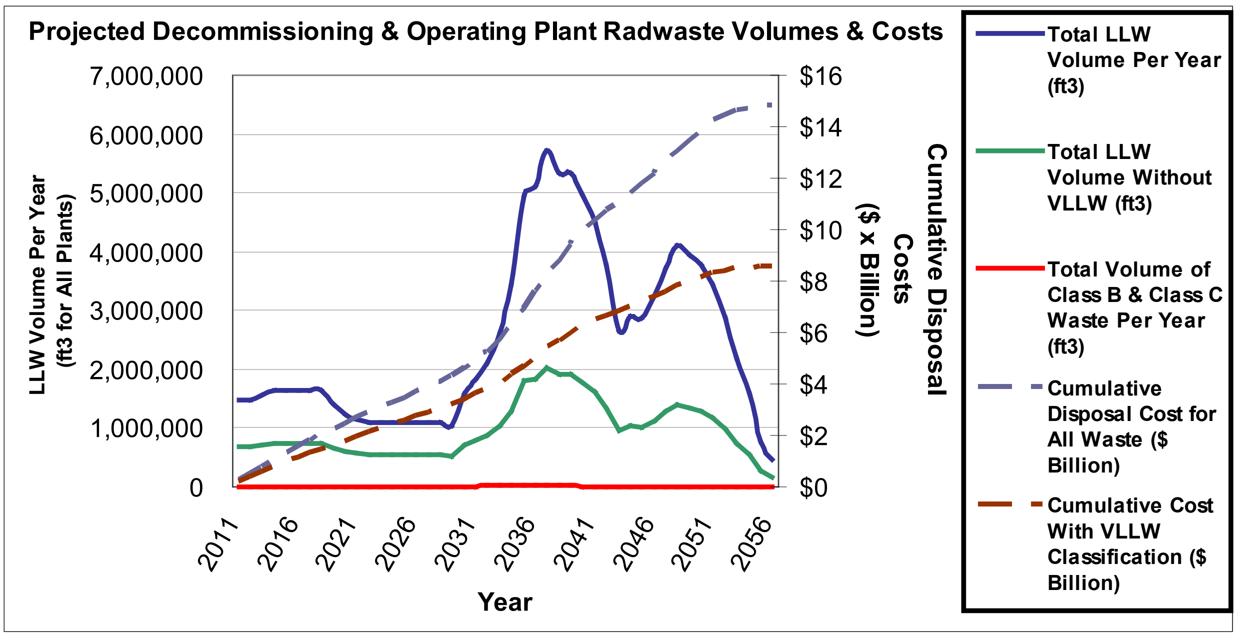

Figure 10-1
Estimated Disposal Cost Savings from a VLLW Classification

Estimated Total Cost Savings

Figure 10-1 shows a projected total savings of $6.2 Billion (expressed in 2011 dollars without escalation) should a VLLW classification be made available as described in this report. As has been previously discussed, this is likely an underestimate as the assumptions used to determine the waste volume and cost savings estimates are conservative.

A
DECOMMISSIONING WASTE CHARACTERIZATION DATA

EPRI has published two reports (References A-1 and A-2) that provide detailed information on the volumes and concentrations of soil and building demolition debris that has been disposed of as radioactive waste during power plant decommissioning. A large percentage of the radioactive waste resulting from power plant decommissioning has been of these two types of material. The following is a summary of the characterization information contained in these references for this material.

A.1 Concrete Radionuclide Characterization Data Results

Most of the information contained in this section was taken from EPRI Report # 1015502, Concrete Characterization and Dose Modeling During Nuclear Power Plant Decommissioning, (Reference A-1).

This section provides a summary of the concrete characterization results determined for a number of power plant sites that have gone through the decommissioning process. The characterization process for a nuclear power plant has been shown to result in a very large amount of data on radionuclide concentration and contamination levels. As reporting of all of that information is beyond the scope of this report, a summary of the results will be provided. For each plant, the following will be described:

- The characterization methodologies used
- A summary of the results obtained

A.1.1 Yankee Rowe Characterization Results

The Yankee Nuclear Power Station (YNPS also known as Yankee Rowe) was owned and operated by Yankee Atomic Electric Company and was a 4 loop PWR of Westinghouse design. Its final output was 185 MWe. YNPS was permanently shut down in October 1991 after about 31 years of commercial operation.

Scoping and initial characterization surveys of concrete began in 1993. Where possible, concrete surfaces were assessed by direct radiation measurements and removable contamination smears. In high background areas, a shallow 100 cm^2 area of paint and concrete was removed and measured with a GM detector. Selected areas were measured by core sampling to assess the depth profile of contaminants. To allow easy comparison to the VLLW concentration limits proposed in this report, only the core sample results will be reported here. Figure A-1 shows a cross-section of the Yankee Rowe Containment Building.

Appendixes present material too detailed to be included in the main report text–for example, computer printouts or lengthy comparative data.

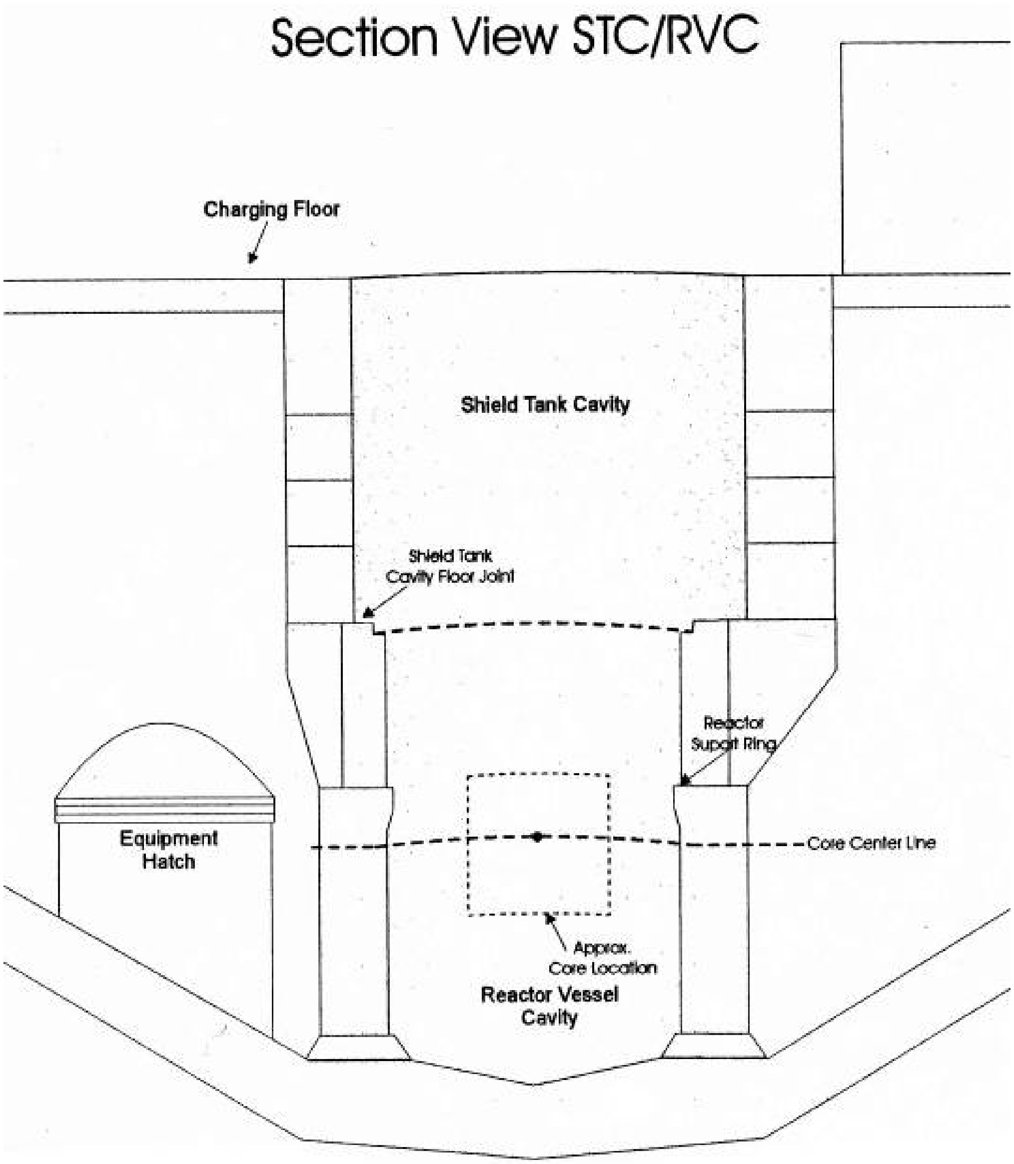

Figure A-1
Cross Section of Yankee Rowe Containment Building

Before the removal of highly radioactive primary system components from containment, neutron-activated concrete surfaces were not accessible for measuring, sampling, or coring. Following the removal of the reactor vessel and related components, exposure rate surveys were conducted to determine the extent of contamination and/or activation in containment concrete. A concrete core sampling plan was developed in 1997 to take cores inside the Reactor Vessel

Cavity (RVC) and Shield Tank Cavity (STC) at both systematically-spaced locations, and at locations where previous radiation surveys had indicated the greatest exposure rates, and where staining suggested STC leakage. After initial core analyses were completed, several more cores were collected in areas surrounding the original core locations with the highest radionuclide concentrations. A total of 28 cores were collected under this plan, each core measuring 3 inches in diameter and approximately 12 inches in depth. Due to the tight mesh of rebar in this area, a diamond-tipped drill bit was used. Each core was sliced into 0.6 inch (15 mm) segments and analyzed by gamma spectroscopy. The results of the analyses are found below in Tables A-1 and A-2.

Table A-1
Summary of Radionuclide Concentrations in 16 Concrete Cores from the YNPS Reactor Vessel Cavity

Core Depth inch (mm)	Concentration, in pCi/g							
	Co-60		Cs-134		Cs-137		Eu-152	
	Mean	Max	Mean	Max	Mean	Max	Mean	Max
0 (0)	304	674	104	426	2620	8400	21	68
0.2 (15)	10	38	0.4	3.2	5.1	44	29	101
1.2 (30)	8.8	48	0.3	2.1	2.0	11	35	123
1.8 (45)	7.9	37	0.1	1.5	1.4	9.0	38	130
2.4 (60)	6.7	33	0.2	1.8	1.0	7.8	34	121
3.0 (75)	10	79	0.2	2.1	1.1	9.2	31	99
3.5 (90)	5.0	25	0.2	1.5	6.6	87	28	96
4.1 (105)	4.4	20	0.1	0.7	2.7	18	27	77
4.7 (120)	5.0	17	0.2	1.2	1.6	12	25	89
5.3 (135)	4.1	14	0.0	0.0	3.6	35	19	52
5.9 (150)	6.2	14	4.1	29	2.7	9.3	24	38
6.5 (165)	5.2	10	0.0	0.0	2.1	4.8	32	106
7.1 (180)	5.2	19	0.4	2.6	6.1	31	14	20
7.7 (195)	2.9	6.6	0.0	0.0	4.2	16	12	22
8.3 (210)	2.6	7.5	0.0	0.0	2.7	13	6.0	12
8.9 (225)	2.4	5.9	0.0	0.0	2.3	11	8.1	13
9.5 (240)	2.9	8.5	0.0	0.0	4.2	15	5.6	8.7
10.0 (255)	2.5	7.8	0.0	0.0	3.8	15	4.4	7.4
10.6 (270)	1.4	4.3	0.0	0.0	2.6	10	3.7	6.1

Table A-2
Summary of Radionuclide Concentrations in 11 Concrete Cores from the YNPS Shield Tank Cavity

Core Depth inch (mm)	Concentration, in pCi/g							
	Co-60		Cs-134		Cs-137		Eu-152	
	Mean	Max	Mean	Max	Mean	Max	Mean	Max
0 (0)	148	732	7.9	41	249	928	0.0	0.0
0.2 (15)	1.2	6.7	0.0	0.0	0.2	1.4	0.1	1.2
1.2 (30)	1.2	5.2	0.0	0.0	0.1	1.3	0.2	2.4
1.8 (45)	1.0	4.6	0.0	0.0	0.0	0.0	0.4	2.0
2.4 (60)	0.7	3.4	0.0	0.0	0.1	0.9	0.2	2.1
3.0 (75)	1.0	6.0	0.0	0.0	0.1	0.7	0.7	4.0
3.5 (90)	0.7	3.1	0.0	0.0	0.0	0.0	0.0	0.0
4.1 (105)	0.4	2.7	0.0	0.0	0.0	0.0	0.0	0.0
4.7 (120)	1.8	15	0.0	0.0	0.1	1.4	0.0	0.0
5.3 (135)	24	122	0.0	0.0	0.6	3.1	0.0	0.0
5.9 (150)	33	98	0.0	0.0	0.0	0.0	0.0	0.0
6.5 (165)	0.7	1.4	0.0	0.0	0.6	1.1	0.0	0.0
7.1 (180)	2.2	2.2	0.0	0.0	1.3	1.3	0.0	0.0

Note: Low concentrations of silver-108m were also detected in the surface segment from two cores.

A review of Tables A-1 and A-2 shows that only the concrete in the first 0.2 inches (15mm) of the surfaces of the RCV and STC had levels of contamination that were above VLLW concentrations proposed in this report. As remediation was needed to depths of approximately 10 inches in the RVC and 6 inches in the STC, the remediation process would be expected to result in average waste concentrations well below the VLLW concentrations proposed in this report.

The plant eventually disposed of all of the concrete from these structures as radioactive waste due to contamination detected in construction joints, at points where the STC liner had been welded, and at points where anchors held the liner to the concrete wall. One possible source of such contamination was the significant amounts of water that had leaked out of the STC during refueling operations, and then presumably seeped into these joints and penetrations. The average radionuclide concentrations of the actual waste shipped for disposal would be much lower then that in the first few inches and therefore all of the concrete waste would be expected to qualify as VLLW.

A.1.2 Maine Yankee Characterization Results

The Maine Yankee plant owned and operated by the Maine Yankee Atomic Power Company (MYAPC) was an 864 MWe 3 loop PWR designed by Combustion Engineering. It commenced power operation in 1972 and was permanently shutdown in 1997.

For decommissioning purposes, the initial effort to characterize the radiological status for all concrete structures at the Maine Yankee site began in the fall of 1997. Each structure was placed into one of two categories based on the likelihood of contamination. Affected areas included structures located inside the radiation restricted area (RA), such as the Containment Building and the Primary Auxiliary Building (PAB). Unaffected areas included structures located outside of the RA, such as the Turbine Building.

The initial characterization effort included approximately 6,400 measurements from structures in the Affected Area, and approximately 7,900 measurements from structures in the Unaffected Area. Maine Yankee collected more characterization measurements from unaffected structures to ensure that all locations were surveyed and to verify that plant-related contamination was not present in these areas. In addition to the survey measurements, 18 concrete core samples were collected from affected buildings during the initial characterization effort. Because concrete basement surfaces represented the key remaining structures upon the termination of Maine Yankee’s license, an additional 51 concrete core samples were obtained in subsequent characterization efforts.

The concrete characterization results provided the information needed to establish radionuclide profiles, estimate radioactive waste volumes, and target those structures within the RA that required remediation. Characterization data from several buildings are presented below as examples of the type of data acquired during the characterization efforts at the Maine Yankee site.

Maine Yankee conducted several concrete sampling campaigns designed to further characterize the radiological nature of concrete structures. Concrete cores were collected from the Containment Building, PAB, Fuel Building, RCA Building, and Spray Building. These data were used to develop radionuclide profiles to support radiological assessment for concrete structures as well as to support decommissioning activities. The concrete core campaigns conducted for the Containment Building, PAB, and Fuel Building are discussed below.

Not unexpectedly, the surface contamination was significantly lower in the unaffected structures, such as the Turbine Building, Service Building, and Warehouse. Measurements for removable beta-emitting surface contamination had a maximum value of approximately 200 dpm/100 cm^2, which was observed in the Turbine Building. Concrete core samples were not needed for the unaffected concrete structures.

A.1.2.1 Containment Building

Samples from Areas not Subject to Neutron Flux

Concrete cores were collected from various locations within the Containment Building, including the loop areas, the 20 ft, 32 ft, and 46 ft elevations, and the Containment Outer Annulus Trench. Most of the concrete cores were collected from the general area floor surfaces. The concrete cores from the 32 ft elevation represented activated concrete, and the cores from the Containment Outer Annulus Trench represented Maine Yankee's effort to refine the contamination profile because the trench had a different history of radionuclide contact than the general area floor surfaces.

Analyses of the core samples revealed Co-60, Cs-134, and Cs-137 to be the primary gamma-emitting radionuclide-of-concern in concrete that was not subject to neutron activation. In core samples from concrete that was exposed to neutrons, Eu-152 and Eu-154 were also identified. The gamma analysis results for these relatively shallow concrete core samples are summarized in Table A-3.

Table A-3
Radionuclide Concentrations in Concrete Core Samples from the Containment Building

Elevation (# of cores)	Concentration Range (pCi/g) (Note 1):				
	Co-60	Cs-134	Cs-137	Eu-152	Eu-154
-2 ft (7)	2545 – 50	125 - 9	11914 - 307	ND	ND
20 ft (2)	6 – 3	1 – 0.4	16 - 11	ND	ND
32 ft (3) (Note 2)	307 - 157	39 - 28	359 - 36	290 - 280	35 - ND
46 ft (5)	8 - 1	6 - ND	388 - 14	ND	ND
Outer Annulus Trench (2)	935 - 931	9	636 - 535	ND	ND

Notes: 1. ND = Not Detected
2. Activated Samples.

The radioactivity found in the cores was attributed to penetration of surface contamination to a relatively shallow depth of the concrete matrix. Several of the concrete cores were analyzed for HTDNs, such as low energy beta-emitters and TRUs. H-3, Ni-63, and Sr-90 were prevalent in the core samples. However, TRUs (Pu-238, Pu-239, Pu-240, Pu-241, Am-241, Cm-242, Cm-243, and Cm-244) were positively identified only in the core samples from the Containment Outer Annulus Trench. The nuclide fractions from analytical results for HTDNs in the concrete core samples are summarized in Table A-4. It can be seen in Table A-4 that Cs-137 and to a lesser extent Co-60 have the highest nuclide fractions.

Table A-4
HTDN Concentrations in Concrete Core Sample from the Containment Building

Nuclide	Nuclide Fraction Range (ND - Not Detected):				
	Containment Loop 1 (2 cores)	Containment Loop 1 (2 cores)	Containment Loop 1 (2 cores)	Outer Annulus Trench (2 cores)	Activated Concrete (Average of Several Cores)
H-3	0.12 – 0.036	0.0063 – 0.0043	0.0021	ND	0.65
C-14	ND	ND	ND	ND	0.058
Fe-55	0.016 – 0.0088	0.036 - ND	0.0012	0.0026 – 0.0010	0.124
Mn-54	0.000029 - ND	0.000013 - ND	ND	0.00036 – 0.000042	ND
Co-57	<0.0014 - <0.00005	<0.0001 - <0.00003	<0.0002	<0.0005 - <0.0004	ND
Co-60	0.10 – 0.046	0.085 – 0.006	0.041	0.55 – 0.52	0.04
Ni-63	0.26 – 0.11	0.13 – 0.015	0.20	0.057 – 0.042	0.007
Sr-90	0.0013 – ND	0.00098 – ND	0.0011	0.0036 – 0.0031	ND
Sb-125	ND	0.00023 - ND	ND	0.0028 - ND	ND
Cs-134	0.0022 – ND	0.0017 – 0.0015	0.0012	0.0020 – 0.0014	0.0084
Cs-137	0.79 – 0.26	0.93 – 0.73	0.75	0.37 – 0.33	ND
Eu-152	ND	ND	ND	ND	0.11
Eu-154	ND	ND	ND	ND	0.009
Pu-238	ND	ND	ND	0.000038 – 0.000036	ND
Pu-239	ND	ND	ND	0.000026 – 0.000014	ND
Pu-240	ND	ND	ND	0.000026 – 0.000014	ND
Pu-241	ND	ND	ND	0.0015	ND
Am-241	ND	ND	ND	0.000017 – ND	ND
Cm-243	ND	ND	ND	0.0000012 – ND	ND
Cm-244	ND	ND	ND	0.0000012 - ND	ND

Although many of the sample result in Table A-3 for shallow cores exceed the VLLW concentrations proposed in this report, Maine Yankee, as did most of the plant that have been decommissioned, disposed of all of the concrete inside of the containment liner as radioactive waste. The process of removing all the concrete inside the containment liner resulted in waste that had an average concentration that was below the VLLW concentrations proposed in this report.

Samples from Areas Subject to Neutron Flux

The radioactivity found in cores collected in concrete exposed to neutrons was attributed to neutron activation. Maine Yankee collected a core sample from concrete in these areas and determined the depth of activation by slicing the core and analyzing each slice separately. At Maine Yankee, activated concrete comprised approximately 5 percent of the concrete in containment. Table A-5 shows the measured radioactivity with depth into a 22-inch core subject to neutron activation.

Table A-5
Radionuclide Concentrations in Neutron-Activated Containment Building Concrete Core

Depth (inch)	Total Activity (pCi/g)*	Depth (inch)	Total Activity (pCi/g)*	Depth (inch)	Total Activity (pCi/g)*
0 – 0.5	677	7.75 – 8.5	233	14.5 – 15.25	14
0.5 – 1.0	828	8.5 – 9.25	206	15.25 – 16.0	11
1.0 – 1.5	845	9.25 – 10.0	182	16.0 – 16.75	7
1.5 – 4.0	824	10.0 – 10.75	103	16.75 – 17.5	6
4.0 – 4.75	771	10.75 – 11.5	87	17.5 – 18.25	6
4.75 – 5.5	329	11.5 – 12.25	23	18.25 – 19.0	1
5.5 – 6.25	534	12.25 – 13.0	23	19.0 – 20.0	1
6.25 – 7.0	365	13.0 – 13.75	17		
7.0 – 7.75	290	13.75 – 14.5	14		

*The radionuclide profile for activated concrete is presented in Table A-4.

A review of Table 10-6 shows that for the Maine Yankee neutron activated concrete the primary radionuclides are:

- H-3 - Average of samples is 66 %

- Primary Gamma Emitters Present (Co-60 and Eu-152) - Average of Samples is 15%

Contamination in the concrete exceeded the site release limits to a depth of 12 inches. Maine Yankee decided it was more cost effective to dispose of all of the activated concrete as radioactive waste rather than remediate to the site release limits. When these nuclide fractions are applied to the values in Table A-5, all of the sample results are below the VLLW concentrations proposed in this report.

A.1.2.2 Primary Auxiliary Building (PAB)

Concrete cores were collected from the PAB pipe trench, the PAB pipe tunnel, and the PAB evaporator cubicle. Normal system leakage was responsible for the contamination levels found within the PAB. As with the Containment core samples, the penetration of surface contamination was limited to the first 0.25 inches. Analyses of the core samples revealed Co-60 and Cs-137 to be the primary gamma-emitting radionuclides of concern, although they were not found in all of the PAB core samples. Co-60 was identified in approximately 45% of the core samples and Cs-137 was identified in approximately 72% of the samples. Cs-134 was the only other gamma-emitting radionuclide identified in the samples, but at a much lower frequency (< 30% of the core samples) and at a much lower concentrations. The gamma analysis results for the concrete core samples are summarized in Table A-6.

Table A-6
Radionuclide Concentrations in Concrete Core from Primary Auxiliary Building (PAB)

Elevation (# cores)	Concentration Range (pCi/g) (ND - Not Detected):		
	Co-60	Cs-134	Cs-137
11 ft (9)	208 – ND	10 - ND	1030 - ND
21 ft (2)	1 – ND	ND	7 - ND

Several of the PAB concrete cores were analyzed for HTDNs. H-3, Ni-63, and Sr-90 were prevalent in the core samples. TRUs (Pu-238, Pu-239, Pu-240, Pu-241, and Am-241) were positively identified only in one of the core samples from the PAB pipe tunnel and at nuclide fractions less than 1.5%. Full results of the HTDN fractions are shown on Table A-7.

Table A-7
HTDN Fractions in Concrete Core Samples from Primary Auxiliary Building (PAB)

Nuclide	Nuclide Fraction Range (ND - Not Detected):		
	Pipe Trench (2 cores)	Evaporator Cubicle (1 core)	Pipe Tunnel (2 cores)
H-3	0.0036 – 0.0004	0.072	ND
Fe-55	0.0029 – 0.0024	0.0085	0.056 – 0.024
Co-57	<0.000065 - <0.000028	<0.0008	<0.002 - <0.00032
Co-60	0.15 – 0.05	0.082	0.21 – 0.097
Ni-63	0.74 – 0.17	0.12	0.62 – 0.43
Sr-90	0.00044 – 0.00013	ND	0.017 – ND
Cs-134	0.0017 – 0.0016	ND	0.0043 - ND
Cs-137	0.74 – 0.08	0.65	0.22 – 0.17

When the values of Table A-6 and A-7 are considered together some of the samples taken to a 1/4 inch depth exceed the VLLW limits proposed in this report only for Cs-137. When it is considered that remediation to a depth of a few inches was needed to meet the site release limits in these areas of higher contamination, the average concentration in the remediation waste was below the proposed VLLW limits.

A.1.2.3 Fuel Building

Concrete cores were collected from both the 21 ft and 31 ft elevations of the Fuel Building, as shown in Table A-8. Maine Yankee attributed normal system leakage as the source for the contamination levels found within the Fuel Building. Surface contamination penetration into concrete was limited to the first 0.25 inches. Cs-137 was identified as the primary gamma-emitting radionuclide of concern, although Co-60 was also identified.

Table A-8
Radionuclide Concentrations in Concrete Cores from the Fuel Building

Elevation (# cores)	Concentration Range (pCi/g) (ND - Not Detected):		
	Co-60	Cs-134	Cs-137
21 ft (4)	156 - ND	ND	1186 - 4
31 ft (2)	ND	ND	64 - 20

The analytical results for HTDNs are summarized below in Table A-9. The source of the information was a single core sample collected from the Decontamination Room in the Fuel Building. No TRU radioactivity was identified.

Table A-9
Hard-to-Detect Radionuclide Fractions in Concrete Cores from the Fuel Building

Nuclide	Fraction	Nuclide	Fraction
H-3	0.00062	Ni-63	0.21
Fe-55	0.00046	Sr-90	Not Detected
Co-57	<0.00009	Cs-134	Not Detected
Co-60	0.0041	Cs-137	0.75

As was the case with the Maine Yankee Fuel Building, when the values of Table A-6 and A-7 are considered together some of the samples taken to a 1/4 inch depth exceed the VLLW limits proposed in this report only for Cs-137. When it is considered that remediation to a depth of a few inches was needed to meet the site release limits in these areas of higher contamination, the average concentration in the remediation waste was below the proposed VLLW limits.

A.1.3 Connecticut Yankee Characterization Results

The Connecticut Yankee (CY) Haddam Neck Plant (HNP) was operated by the Connecticut Yankee Atomic Power Company. Connecticut Yankee, a single unit 4-loop PWR. On December 4, 1996, HNP permanently shut down after approximately 28 years of operation.

Connecticut Yankee performed concrete characterization during various phases of the decommissioning in order to support project planning and waste characterization and to provide information needed for submittals to the NRC such as the License Termination Plan (LTP). CY utilized core boring exclusively as the method to assess contamination in concrete. One of the reasons for this choice was the presence of numerous HTDNs at CY. A significant amount of sample mass was needed at each core location.

For CY the required number of samples to support regulatory reviews is shown in Table A-10.

Table A-10
Volumetric Concrete Sample Requirements

Basement Area Sampled	Number Of Samples Needed for the Inventory Calculation	Basement Area Sampled	Number Of Samples Needed for the Inventory Calculation
Containment Base Slab	14	Cable Vault	13
Containment Walls	10	"B" Switchgear Building	8
In Core Sump	9	Discharge Tunnels	10
Spent Fuel Pool	12	Intake Structure	8

Characterization results for the structures listed in Table A-10 that contained significant radionuclide concentrations are listed in the following pages. CY also collected characterization samples from other structures such as the Service Building, the Waste Disposal and Yard Crane Support Footings that contained radionuclide concentrations. Tables A-11 to A-19 show a summary of the concrete characterization results. Basements of buildings such as the Auxiliary Building, the Ion Exchange Building and most of the Waste Disposal Building along with all concrete inside the Containment Building Inner Liner were determined to be too contaminated to remain at the time of site release. There structures were totally removed and disposed of as radioactive waste. Characterization results for these structures are provided later in this section and were helpful in providing the radionuclide ratios needed for waste disposal.

A.1.3.1 Containment Building Non-Activated Areas

Table A-11 presents results for the areas inside of the containment building not subject to a significant neutron flux (considered non-activated areas).

Table A-11
Connecticut Yankee Concrete Characterization Results for Areas Inside the Containment Liner (Not Subject to Neutron Flux)

Radio-nuclide	Containment Walls and Floors above Basement Level: Surface Wafer (pCi/g)	Containment Sump: Surface Wafer (pCi/g) (Note 1)	Containment Sump: Deeper then Surface Wafer (pCi/g)	Containment Basement Floor: Surface 0.5 inch Wafer (pCi/g) (See Note 2)
H-3	350 to 1780	1170 to 1400	Not Analyzed	Not Analyzed
C-14	<0.1 to 720	25.4 to 70	<0.57	Not Analyzed
Mn-54	<0.1	<0.38	<0.07	< MDA
Fe-55	Not Analyzed	10.2 to 74	Not Analyzed	Not Analyzed
Co-60	<0.02 to 23.1	70.9 to 240	<0.02 to 0.38	23.4
Ni-63	Not Analyzed	415 to 1620	Not Analyzed	Not Analyzed
Sr-90	Not Analyzed	0.95 to 20.1	Not Analyzed	Not Analyzed
Nb-94	<0.07	<0.29	<0.07	< MDA
Tc-99	<0.68	<0.91 to 2.84	<0.77	Not Analyzed
Ag-108m	<0.11	<0.57	<0.06	Not Analyzed
Cs-134	<0.11 to 0.27	1.25 to 25.5	<0.08	2.76
Cs-137	<0.01 to 34.9	583 to 1270	<0.01 to 6.02	279
Eu-152	<0.25	<1.3	<0.06 to 0.15	< MDA
Eu-154	<0.25	<0.23 to 1.86	<0.87	< MDA
Eu-155	<0.18	<0.8	<0.13	< MDA
Pu-238	Not Analyzed	0.63 to 5.08	Not Analyzed	Not Analyzed
Pu-239	Not Analyzed	0.24 to 1.92	Not Analyzed	Not Analyzed
Pu-241	Not Analyzed	9.86 to 54.8	Not Analyzed	Not Analyzed
Am-241	<0.27	0.74 to 7.06	<0.22	Not Analyzed
Cm-243	Not Analyzed	0.11 to 1.7	Not Analyzed	Not Analyzed

Notes: 1. Unless otherwise noted wafers vary between 1 to 2 inches in thickness.
2. Results for all deeper floor wafers were less than the Minimum Detectable Activity (MDA)

Considering the levels of contamination inside of the containment building, Connecticut Yankee decided that it was more cost beneficial to dispose of all the concrete inside of containment as radioactive waste rather then decontaminate to site release levels. The worst case concentrations were in the 1 to 2 inch surface wafers at a couple of locations. These elevated samples, exceed the VLLW limits proposed in this report only for Cs-137. When it is considered that remediation to a depth of a few inches was needed to meet the site release limits in these areas of higher contamination, the average concentration in the remediation waste was below the proposed VLLW limits.

Table A-12
Connecticut Yankee Concrete Characterization Results for Areas Outside the Containment Liner (Not Subject to Neutron Flux)

Radionuclide	Containment Mat: Surface Wafer (pCi/g)	Containment Mat: Deeper than Surface Wafer (pCi/g)	Containment Below Grade Walls: Surface Wafer (pCi/g)	Containment Below Grade Walls: Deeper Wafers (pCi/g)
H-3	<1.01 to 27.8	<MDA to 31.5	<MDA to 4.72	<MDA to 13.2
C-14	<MDA	<MDA	<MDA	<MDA to 0.32
Mn-54	<MDA	<MDA	<MDA	<MDA to 0.02
Fe-55	Not Analyzed	<MDA	<MDA	<MDA
Co-60	<MDA to 0.5	<MDA to 0.18	<MDA to 0.29	<MDA to 0.83
Ni-63	<MDA	<MDA to 12.1	<MDA	<MDA
Sr-90	<MDA to 0.2	<MDA to 0.12	<MDA to 0.01	<MDA to 0.03
Nb-94	<0.02	<0.05	<MDA	< MDA
Tc-99	<MDA	<MDA	<MDA	< MDA
Ag-108m	<MDA	<MDA	<MDA	< MDA
Cs-134	<MDA	<MDA to 0.06	<MDA	<MDA to 0.03
Cs-137	<MDA to 0.1	<MDA to 0.58	<MDA to 0.2	<MDA to 0.23
Eu-152	<MDA	<MDA to 0.05	<MDA	< MDA
Eu-154	<MDA	<MDA	<MDA	< MDA
Eu-155	<MDA	<MDA to 0.05	<MDA	< MDA to 0.08
Pu-238	<MDA	<MDA	<MDA	< MDA
Pu-239	<MDA	<MDA	<MDA	< MDA
Pu-241	<MDA to 2.71	<MDA	<MDA	< MDA
Am-241	<MDA	<MDA to 0.03	<MDA	< MDA
Cm-243	<MDA	<MDA	<MDA	< MDA

By comparing the results in Tables A-11 and A-12 it can be seen that the carbon steel liner that covered the inside of the Containment Walls and Base Slab (Mat) was very effective in protecting the concrete outside the liner from system leakage and gaseous contamination mechanisms that occurred inside of the containment building. The low levels of contamination that are observed behind the liner appear to be from contaminated groundwater that passed through construction joints in the containment basement structure. Samples taken of the outside surface of the containment structure, i.e., below grade where contaminated groundwater was known to be present showed minor diffusion of contamination into the concrete. Samples of the popcorn concrete (part of the construction of the base mat of the Containment Building to allow pumping of groundwater to reduce the buoyant forces on Containment) showed only low levels of contamination from the contaminated groundwater passing through this structural component. Although the levels were below the site release, CY decided to dispose of all the above grade concrete outside of the containment liner as radioactive waste rather than incur the expense of a site release survey of the structure. All the sample results were well below the VLLW limits proposed in this report.

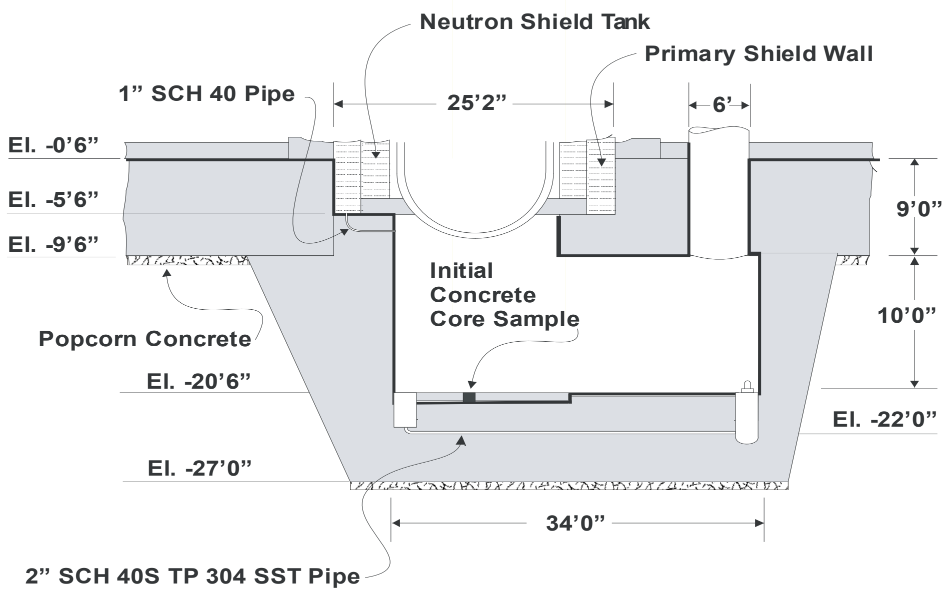

Figure A-2
CY Incore Instrumentation Sump

A.1.3.2 Containment Building Neutron Activated Areas

Concrete characterization data indicates that the Neutron Shield Tank that was located around the active fuel region of the CY Reactor Vessel (See Figure A-2) was effective in reducing activation of the concrete outside of the tank. In the area below the Neutron Shield Tank where the Containment In-Core Sump was located, radioactivity levels in the concrete were found to be significant both in terms of level and depth. Figure A-2 shows the location of the CY ICI Sump in relation to the Reactor Vessel and the Neutron Shield Tank. Sample results for the concrete and the protective inner liner located in the ICI Sump are summarized in Table A-13.

Table A-13
Connecticut Yankee Concrete Characterization Results for the In-Core Instrumentation (ICI) Sump

Radionuclide	ICI Sump: Upper Walls (pCi/g) (Note 1)	ICI Sump: Lower Walls (pCi/g) (Note 1)	ICI Sump: Floor (pCi/g) (Note 1)	Sample of Carbon Steel Liner (pCi/g)
H-3	6.12 to 6460	<MDA to 3440	<MDA to 3340	Note 2
C-14	2.42 to 2.84 (1st Wafer Only)	1.0 to 1.54 (1st 2 Wafers Only)	1.48 to 2.35 (1st 2 Wafers Only)	Note 2
Mn-54	<MDA	<MDA	<MDA to 2.54	<MDA
Fe-55	<MDA to 1950	<MDA to 533	<MDA to 403	<11.4 to 20100
Co-60	<MDA to 2140	<MDA to 688	<MDA to 301	<0.51 to 1980
Ni-63	<MDA to 9.02	<MDA to 3.37	<MDA to 2.62	Note 2
Sr-90	<MDA to 0.31	<MDA to 0.09	<MDA to 0.16	Note 2
Nb-94	<MDA to 0.43	<MDA to 0.43	<MDA	<MDA
Tc-99	<MDA to 0.68	<0.42 to 0.46	<MDA to 0.40	Note 2
Ag-108m	<MDA	<MDA	<MDA	<MDA
Cs-134	<MDA to 9.52	<MDA to 5.01	<MDA to 4.54	Note 2
Cs-137	<MDA to 1.24	<MDA to 0.47	<MDA to 0.45	Note 2
Eu-152	<MDA to 2740	<MDA to 799	<MDA to 515	Note 2
Eu-154	<MDA to 256	<MDA to 70.1	<MDA to 256	Note 2
Eu-155	<MDA to 2.72	<MDA to 1.26	<MDA to 50.9	Note 2
Pu-238	<MDA to 0.02	<MDA to 0.06	<MDA	Note 2
Pu-239	<MDA to 0.1	<MDA to 0.08	<MDA	Note 2
Pu-241	<MDA to 1.43	<MDA to 4.04	<MDA	Note 2
Am-241	<MDA to 0.06	<MDA to 0.49	<MDA to 0.06	Note 2
Cm-243	<MDA to 0.05	<MDA	<MDA	Note 2

Notes: 1. Highest Results were always the first 1.5 inch wafer. Total core depth 20 inches
2. Not reported as results are indicative of surface contamination and not activation.

Levels of contamination in CY the in-core sump concrete were as much as 10 times the VLLW limits proposed in this report. Although the average concentrations in any remediation waste would likely be lower, this remediation waste would not be expected to qualify as VLLW under the proposed limits of this report.

A.1.3.3 Primary Auxiliary Building (PAB) and Waste Disposal Buildings

- The PAB and Waste Disposal Building at CY housed systems that contained fluids with relatively high concentrations of radionuclides. Significant leakage of these systems caused high levels of surface and deep contamination of the concrete in these buildings. The floors in the basements exhibited the highest levels of contamination. Tables A-14 and A-15 respectively summarize the concrete characterization results for these buildings.

Table A-14
Connecticut Yankee Concrete Characterization Results for the Auxiliary Building Residual Heat Removal (RHR) Pit

Radio-nuclide	RHR Pit Floor: Surface Wafer (pCi/g)	RHR Pit Floor: Deeper than Surface Wafer (pCi/g)	RHR Pit Walls: Surface Wafer (pCi/g)	RHR Pit Walls: Deeper than Surface Wafer (pCi/g)
H-3	11.2 to 23.7	<1.29 to 16.4	6.5 to 13.8	1.04 to 25.9
C-14	<0.77	<0.77	<0.74	<0.74
Mn-54	<0.18	<0.4	<0.09	< 0.09
Fe-55	<3.9 to 49.9	<4.1	<4.02	<4.94
Co-60	6.93 to 67.7	<0.02 to 5.5	<0.17 to 0.91	<0.03 to 1.04
Ni-63	21.8 to 23.7	1.43 to 2.09	<1.52	<1.65
Sr-90	1.74 to 4.59	0.01 to 0.03	<0.01 to 0.67	<0.01 to 0.09
Nb-94	<0.13	<0.04	<0.08	<0.09
Tc-99	<0.86	<0.82	<0.89	<0.77
Ag-108m	<0.24	<0.03	<0.03	<0.07
Cs-134	0.15 to 0.32	<0.05	<0.03 to 0.28	<0.03 to 0.04
Cs-137	39.8 to 226	<0.03 to 1.55	<0.02 to 7.59	<0.02 to 4.08
Eu-152	<0.59	<0.1	<0.3	<0.23
Eu-154	<0.29 to 0.9	<0.12	<0.26	<0.29
Eu-155	<0.45	<0.11	<0.27	<0.76
Pu-238	0.76 to 0.92	<0.07	<0.01 to 0.06	<0.08
Pu-239	0.21 to 0.28	<0.05	<0.08	<0.06
Pu-241	7.94 to 11.9	<4.11	<2.74	<2.74
Am-241	0.9 to 0.97	<0.04 to 0.08	0.02	<0.2
Cm-243	0.11 to 0.24	<0.07	<0.05	<0.05

Notes: 1. All wafers were 0.5 inches thick.
2. Results for all other radionuclides were less than Minimum Detectable Activity (MDA).

Table A-15
Connecticut Yankee Concrete Characterization Results for Other Areas Outside the Containment

Radionuclide (Note 2)	Waste Disposal Building Basement Floor: Surface Wafer (pCi/g) (Note 1)	Waste Disposal Building Basement Floor: Deeper Wafers (pCi/g)	Auxiliary Building Pipe Chase Floor: Surface Wafer (pCi/g)	Auxiliary Building Pipe Chase Floor: Deeper Wafers (pCi/g)
Co-60	160.55	<0.65	34.1	<1.0
Nb-94	0.28	<0.43	<MDA	<MDA
Cs-134	1.3	0.18 to 0.54	5.18	<0.07 to 0.13
Cs-137	264	<0.72	74	0.28
Eu-154	4.05	<MDA	<MDA	<0.29
Eu-155	0.86	<MDA	<MDA	<0.76
Am-241	11.03	<MDA	<MDA	<MDA

Notes: 1. All wafers were 0.5 inches thick.
2. Results for all other radionuclides were less than Minimum Detectable Activity (MDA)

CY decided to completely remove the PAB and Waste Disposal Building due to the following:

- Removal of only the floors would have been very difficult due to limited accessibility.
- Soil contaminated by groundwater-conveyed radioactivity was present outside of the basement walls and required removal.

All of the sample results for the concrete in the PAB and Waste Disposal Building are below the VLLW limits proposed in this report. As the concentrations of the deeper samples are much lower than the surface samples, the average concentration for the final waste from the complete demolition of the buildings was much lower than the highest sample concentrations.

A.1.3.4 Spent Fuel Building

Samples of the water in the Spent Fuel Pool leak detection system collected during plant operation had shown that the stainless steel pool liner at CY had leaked slightly during its use. The concrete core results shown in Table A-16 indicate only modest penetration of the contamination into the walls outside the pool liner and only in certain areas.

Table A-16
Connecticut Yankee Concrete Characterization Results for Spent Fuel Pool: Samples from Pool Side

Radionuclide	Fuel Pool Floor: First 3 Wafers under the Liner (pCi/g) (Note 1)	Fuel Pool Floor: Deeper than First 3 Wafers (pCi/g)	Fuel Pool Walls: First 3 Wafers under the Liner (pCi/g) (Note 1)	Fuel Pool Walls: Remainder of Wall Thickness (pCi/g)
H-3	<MDA to 1150	<MDA to 560	<56.6 to 970	<MDA to 13.4
C-14	<MDA	<MDA to 0.9	<MDA to 1.63	Not Analyzed
Mn-54	<MDA	<MDA	<MDA	<MDA
Fe-55	<MDA	<MDA to 2.2	<MDA	<MDA
Co-60	<MDA	<MDA to 0.9	<MDA to 0.44	<MDA to 1.35
Ni-63	<MDA	<MDA	<MDA to 9.96	<MDA
Sr-90	<MDA to 0.37	<MDA to 0.71	<MDA to 0.2	<MDA to 0.1
Nb-94	<MDA	<MDA	<MDA	<MDA
Tc-99	<MDA	<MDA	<MDA	<MDA
Ag-108m	<MDA	<MDA	<MDA	<MDA
Cs-134	<MDA to 0.11	<MDA	<MDA	<MDA
Cs-137	<MDA to 20.2	<MDA to 0.54	<MDA to 29.1	<MDA to 0.05
Eu-152	<MDA	<MDA	<MDA	<MDA
Eu-154	<MDA	<MDA	<MDA	<MDA
Eu-155	<MDA	<MDA	<MDA	<MDA
Pu-238	Not Analyzed	Not Analyzed	Not Analyzed	Not Analyzed
Pu-239	Not Analyzed	Not Analyzed	Not Analyzed	Not Analyzed
Pu-241	Not Analyzed	Not Analyzed	Not Analyzed	Not Analyzed
Am-241	<MDA	<MDA to 0.04	<MDA to 0.02	<MDA to 0.24
Cm-243	Not Analyzed	Not Analyzed	Not Analyzed	Not Analyzed

Notes: 1. Total thickness of these results varies from 6 to 10 inches.

The first 12 inches (30 cm) of the floor below the liner was highly contaminated throughout (See Table A-16) and required removal as did 2 inches (5 cm) of the walls where they intersected the floors. Samples taken at the location of a horizontal construction joint 8.8 feet below the pool floor (see Table A-17) showed contamination from a pool leak that probably moved through the spaces in some of these joints and resulted in moderate contamination levels in the concrete.

A modest amount of remediation of the CY spent fuel pool concrete was needed to meet site release limits. All samples results of the remediated material were below the VLLW limits proposed in this report except for Cs-137 in one construction joint sample. When it is considered that remediation to a depth of a few inches was needed to meet the site release limits in these areas of higher contamination, the average concentration in the remediation waste was below the proposed VLLW limits.

Table A-17
Connecticut Yankee Concrete Characterization Results for Spent Fuel Pool: Construction Joint

Radionuclide	Construction Joint 8.8 ft Below Pool Floor (pCi/g) (Note 1)
H-3	105 to 282
C-14	Not Analyzed
Mn-54	<MDA
Fe-55	2.85 to 9.45
Co-60	0.1 to 0.35
Ni-63	<MDA to 5.84
Sr-90	0.71 to 7.05
Nb-94	<MDA
Tc-99	<MDA
Ag-108m	<MDA
Cs-134	<MDA to 0.16
Cs-137	108 to 311
Eu-152	<MDA
Eu-154	<MDA
Eu-155	<MDA
Pu-238, -239, -241	Not Analyzed
Am-241	<MDA
Cm-243	Not Analyzed

Note: 1. Results are for first 2 inch wafer each side of the joint

A.1.3.5 Areas Affected by Groundwater Contamination

Due to significant levels of radionuclide contamination in groundwater at Connecticut Yankee, concrete that could have been affected by only groundwater contamination were characterized. The following structures were characterized to evaluate contamination from groundwater:

- The Cable Vault contained no systems containing radioactive fluids but was subject to in leakage of contaminated groundwater
- The Service Building Footer Wall was subjected to contamination that had been conveyed by groundwater from a spill at another site location
- The Containment Mat Sump contained only groundwater that had passed through the popcorn concrete in the Containment Mat.

A review of concrete sample results showed that although the concrete in these structures had come in contact with groundwater contaminated with H-3 and Sr-90, the resulting concentrations in the concrete were relatively low. No concrete exceeded the CY site release limits and therefore was well below the VLLW limits proposed in this report.

A.1.3.6 Discharge Tunnels, Intake Structure

The Discharge Tunnels at CY were large structures located below the Turbine Building which were used to convey the circulating water which had passed through the plant Turbine Water Boxes (where it had cooled plant secondary side water) to the plant discharge canal. The plant liquid radioactive waste system discharge was also routed into these tunnels so as to be diluted by the high flow rate of the circulating water. The data show that the diffusion of the contamination into the concrete at these locations was relatively low. Based on samples taken at various depths, the contamination penetrated to only a depth of approximately 1 inch (2.5 cm) or less. No concrete exceeded the CY site release limits and therefore was well below the VLLW limits proposed in this report.

Portions of the Circulating Water Intake Structure at CY came in contact with a portion of the heated circulating water (which contained water from the liquid radwaste system discharges as described in the last paragraph) was piped into the Intake Structure to avoid freezing of the water in that area during the cold winter months. This caused only trace levels of contamination in the Intake Structure concrete. No concrete exceeded the CY site release limits and therefore was well below the VLLW limits proposed in this report.

A.1.4 SONGS-1 Characterization Results

The San Onofre Nuclear Generating Station Unit 1 (SONGS 1) was a 410 MWe Westinghouse PWR owned (80%) and operated by Southern California Edison Company. The remaining 20% of this plant was owned by San Diego Gas & Electric Company (SDG&E). The unit began commercial operation in 1968 and was permanently shut down in November 1992. Major decommissioning activities began in 1999 when it was determined that space on the site needed to be made available for an Independent Spent Fuel Storage Facility (ISFSI) for all three units.

The situation for the SONGS 1 site is somewhat different then that at the other sites previously discussed. The property on which the SONGS-1 site is located is owned by the United States Marine Corps. Under a lease agreement with the Marines, after the site has been closed and released from its license it will be returned with essentially all structures removed. As few or none of the concrete structures at SONGS-1 will remain after eventual license termination, characterization to support partial remediation has not been needed to this point. What have been needed are radioactivity concentrations that will support waste segregation activities. A primary example of this is the characterization of the bioshield around the Reactor Vessel. Due to neutron activation, some of this 6' 3" thick wall was highly activated while some of the less neutron activated areas would not contain high levels of radioactivity. Approximately 28 locations were sampled with the `TruPro`® drill-type sampling equipment. Starting on the outside of the bioshield, the system was used to drill and collected samples for each 1 foot thickness of wall (for areas not suspected of high activation) or for each 3 inch thickness for areas in the highly activated area.

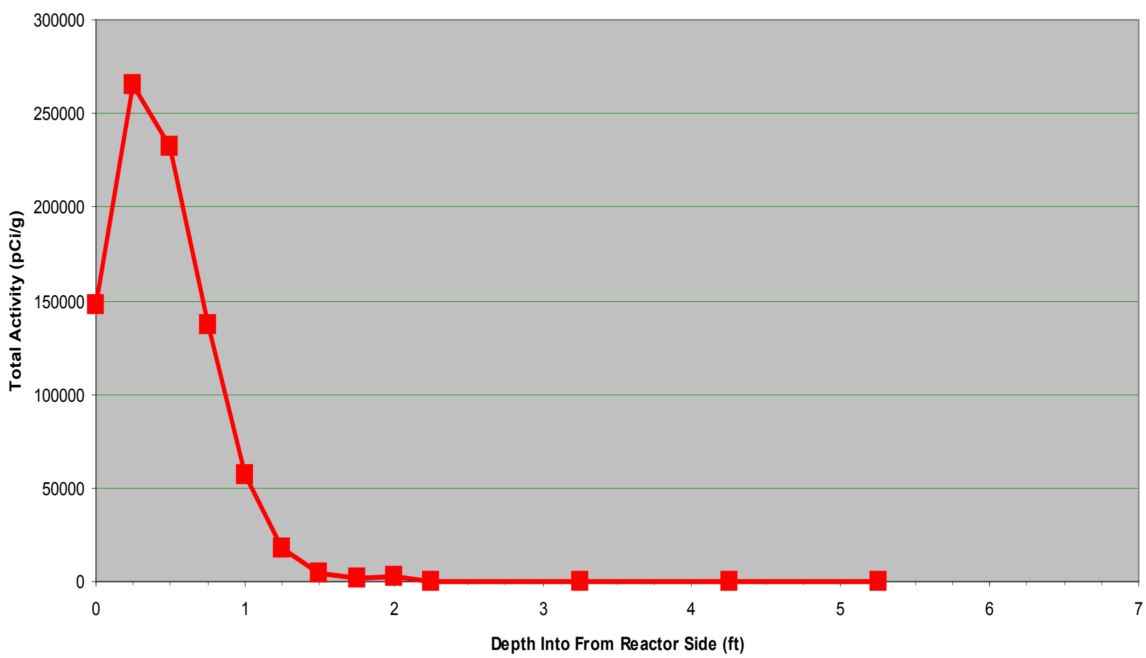

Figure A-3
Activation Profile for Songs-1 Bioshield Concrete

Figure A-3 shows a graph depicting the profile of total activity in the samples as a function of the depth of the sample from the outside of the bioshield. It can be seen that the highest activities occurred in the inner foot of concrete wall in the beltline region (8' 2" of wall height). Note that first thickness of concrete does not have the highest activity due to the thermalization of the neutrons that occurs as they pass through the first thickness of concrete.

Although detailed radionuclide concentration data is not available, it is likely that the first 18 inches of from the inside of the SONGS-1 bioshield would not qualify under the possible VLLW limits proposed in this report. Although a factor of approximately 4 concentration reduction would be achieved if the elevated waste were mixed with the lower level waste of the bioshield during the demolition process, this was would still be expected to exceed the proposed VLLW limits.

A.1.5 Saxton Characterization Results

The Saxton Plant was operated by the Saxton Nuclear Experimental Corporation, a wholly owned subsidiary of GPU Incorporated. Saxton was a single unit PWR facility with an electrical output of 8 MWe located near Saxton, PA. The site operated from 1962 to 1972 during which experiments with mixed oxide fuel were conducted where the fuel was intentionally “failed”. The station was placed in SAFSTOR for a period of 27 years after shutdown in 1972. Major decommissioning activities began in 1999 and the site was unconditionally released from its NRC license in the mid 2005.

A.1.5.1 Sampling of Concrete Inside the Saxton Containment Vessel

As with the other plants discussed above, Saxton found that the contamination levels in the concrete inside of the containment building were too high to allow survey and unconditional release at the time of license termination. This concrete was totally removed and disposed as radioactive waste.

Concrete Affected by Neutron Flux

Table A-18 summarizes the contamination levels in the high neutron flux region inside the Containment Vessel at the Saxton Plant.

Table A-18
Concrete Sample Results in the High Flux Region of the Containment Vessel (CV)

Location Sampled	H-3 (pCi/g)	Ni-59 (pCi/g)	Ni-63 (pCi/g)	Co-60 (pCi/g)	Cs-137 (pCi/g)	Eu-152 (pCi/g)	Eu-154 (pCi/g)
Composite from CV Outer Wall, N/E of Reactor Vessel	550.7	6.1	161.6	21.2	65.6	30.6	1.7

Notes: 1. Cs-137 contamination was contained in first core slice
2. The following radionuclides were not detected above their Minimum Detectable Activities: Ag-108m, Am-241, Ce-144, Cs-134, C-14, Eu-155, 1-129, Nb-94, Ru-106, Sb-125, and Tc-99

Although all of the concrete affected by neutron flux at Saxton needed to be disposed of as radioactive waste, all the sample concentrations shown in Table A-18 are below the VLLW limits proposed in this report.

Other Samples from Concrete Inside the Saxton Containment Vessel

In addition to analyzing the inside containment concrete for gamma-emitting radionuclides, Saxton also analyzed a subset of the cores for HTDNs. Table A-19 summarizes those results for some of the more noteworthy cores.

Table A-19
Concrete Sample Results (w/HTDNs) from Inside of the Containment Vessel (CV)

Location Sampled	H-3 (pCi/g)	C-14 (pCi/g)	Co-60 (pCi/g)	Cs-137 (pCi/g)	Pu-239 (pCi/g)	Am-241 (pCi/g)	Pu-241 (pCi/g)
Primary North Wall	371.6	74.5	4.0	275.0	<0.008	0.02	4.2
Reactor Cavity Wall: South	165.7	51.1	17.1	597.0	0.04	0.9	<0.8
Reactor Cavity Wall: N/W	259.3	85.5	52.2	2510	0.1	1.2	2.0
Basement Level	66.6	22.4	<0.09	5.1	<0.4	<1.3	46.8
Basement Level	190.6	114.7	<0.07	58.5	<0.6	0.6	31.0
Basement Level	518.6	98.4	<0.08	27.7	<0.6	<1.5	10.1
Basement Level	225.3	67.5	<0.07	11.1	<0.6	<0.5	23.5
Basement Level	270.9	90.3	<0.08	0.6	<0.8	1.08	39.7
Basement Level	40.9	33.6	0.04	1.4	<0.8	<1.2	275.0
Basement Level	376.2	80.4	<0.07	0.2	<0.4	<0.4	28.8

Notes: 1. The following radionuclides were not detected above their average Minimum Detectable Activities: Ni-63, Sr-90, Eu-152, and Pu238

All samples results of the remediated material were below the VLLW limits proposed in this report except for 3 wall samples which exceed the proposed limits by as much as 9 times. As the high levels of contamination appear in a relatively low number of areas, there is the potential that the final remediated waste from all areas taken together would be below the proposed VLLW limits.

A.1.5.2 Sampling of Concrete Outside the Saxton Containment Vessel

Saxton also performed characterization sampling on concrete structures outside of the Containment Vessel (CV). These results are summarized in Table A-20.

Table A-20
Concrete Sample Results from Outside of the Containment Vessel

Location Sampled	Core Depth (cm)	Co-60 (pCi/g)	Cs-137 (pCi/g)
Concrete slab below Personnel Airlock	11.9	<0.32	1.09
Concrete shield wall external to CV	10	<0.28	<0.30
Concrete ledge N/E side of CV	11.6	<0.23	2.00
Concrete pad - former ventilation system location	15	<0.91	156.1
Through concrete support wall in tunnel (west end)	23.8	<0.34	8.51
Through concrete support wall in tunnel (east end)	N/A	<0.36	<0.26
Through concrete ceiling of tunnel (end in contact with outside wall)	37.3	<0.33	12.56
Tunnel Wall at Southeast Hatch	15.9	0.31	1.35

As can be seen from Table 3-33, contamination levels in the concrete outside of the containment building were generally much lower than those inside of the containment vessel. Remediation was only anticipated at the higher activity locations.

All the sample results in Table A-20 are below the VLLW limits proposed in this report.

A.1.6 Summary of Concrete Waste Activity levels for Decommissioning Plants

The characterization results discussed in this section indicate that essentially all the concrete waste resulting from remediation during decommissioning for the plants reviewed would be below the possible VLLW limits proposed in this report.

A.2 Soil Radionuclide Characterization Data Results

Most of the information contained in this section was taken from EPRI Report # 1019228, Characterization and Dose Modeling of Soil, Sediment and Bedrock During Nuclear Power Plant Decommissioning, November 2009.

A.2.1 Yankee Rowe Characterization Results

A.2.1.1 Events with Potential Impact on Soil

Although there were other events that resulted in lower levels of contamination in areas of the Radiological Control Area at Yankee Rowe, the following events had significant impact on subsurface soil and/or groundwater:

- Leaks in the radioactive systems in the Ion Exchange (IX) Pit resulted in contamination of the water in the IX Pit. A defect in the construction of the IX Pit concrete allowed the contaminated water to leak, resulting in contamination of the subsurface soils, asphalt and concrete around the IX Pit and adjoining structures.
- In September of 1984 an excavated drainpipe from the storage building to the Waste Disposal building was found to be leaking. The area of maximum contamination was measured at 0.25-0.35 mSv/hr (25-35 mr/hr) (specific radionuclide data are not available), with a hot spot of 1,084 Bq/g (29,300 pCi/g) Co-60 in this same area. The pipe from the edge of the old Primary Containment Access (PCA) building to the edge of the Waste Disposal building and approximately 420 ft^3 ($12m^3$) of dirt and rock were removed as radioactive waste. The soil remaining at the bottom of the excavation contained Co-60 at an average concentration of 1.1 Bq/g (30 pCi/g), which was remediated to the site release limits of approximately 0.11 Bq/g (3 pCi/g). Although the soil from the initial remediation was well in excess of the VLLW limits proposed in this report, the soil in the bottom of the excavation that would need to be remediated to meet the site release limits would meet the possible proposed limits.

A.2.1.2 Yankee Rowe Soil Characterization Results

The investigation of the events during plant operations and other characterization sampling determined the soil concentrations listed in Table A-1.

Table A-21
Yankee Rowe Soil/Sediment Characterization Results

Location	Radio-nuclide	Number of Samples	Number of Detects	Concentration pCi/g (Bq/g) *		Fraction of DCGL
				Mean	Maximum[a]	
East Lower RCA Yard	Co-60	19	6	0.94 (0.035)	3.87 (0.14)	0.67
	Cs-137	48	47	10.31 (0.38)	160 (5.92)	3.43
			Sum of the Fractions			4.1
Northeast Upper RCA Yard	Ag-108m	49	2	0.295 (0.011)	0.536 (0.02)	0.118
	Co-60	53	21	0.362 (0.013)	1.74 (0.064)	0.259
	Cs-137	53	21	0.264 (0.01)	0.999 (0.037)	0.088
			Sum of the Fractions			0.465
Southeast Upper RCA Yard	Ag-108m	92	44	8.99 (0.33)	99.7 (3.69)	3.596
	Co-60	95	61	21.358 (0.79)	1008.8 (37.3)	15.256
	Cs-137	95	72	3.343 (0.12)	61.209 (2.64)	1.114
			Sum of the Fractions			19.966
Southwest Upper RCA Yard	Ag-108m	36	3	0.188 (0.007)	0.247 (0.009)	0.075
	Co-60	40	11	0.567 (0.021)	2.635 (0.097)	0.405
	Cs-137	40	16	0.322 (0.012)	1.717 (0.064)	0.107
			Sum of the Fractions			0.588
Northwest Upper RCA Yard	Co-60	57	10	0.11 (0.004)	0.383 (0.014)	0.079

Table A-21 (continued)
Yankee Rowe Soil/Sediment Characterization Results

Location	Radio-nuclide	Number of Samples	Number of Detects	Concentration pCi/g (Bq/g) *		Fraction of DCGL
Western Lower RCA Yard	Co-60	40	23	0.347 (0.013)	1.816 (0.067)	0.248
	Cs-137	40	23	0.176 (0.007)	0.535 (0.02)	0.059
			Sum of the Fractions			0.307
Sherman Pond Sediments	Co-60	19	9	0.177 (0.007)	0.764 (0.028)	0.126
	Sr-90	10	10	0.188 (0.007)	0.33 (0.012)	0.313
	Cs-137	48	47	0.922 (0.034)	3.03 (0.112)	0.307
			Sum of the Fractions			0.746

*Radionuclides Present at less than 5% of their DCGLs not included

Remediation was required for all areas where the sum of DCGL fractions exceeded 1, and some small area remediation where the sum of the fractions approached 1 in order to insure passing the site release limit statistical testing. Although the maximum concentrations of Co-60 in South East RCA Yard exceeded the VLLW limits proposed in this report, the mean concentrations in all areas requiring remediation to meet the site release limits were below the possible proposed limits.

A.2.1.3 Remediation of PCB Contaminated Soil at Yankee Rowe

The Yankee Rowe plant had utilized paint containing Poly Chlorinated Biphenyls (PCBs) to paint a number of outside structures at the site including the exterior of the containment sphere. Due to degradation of this paint, paint chips flaked off of these structures and mixed with soil and sediments on the site. Table A-22 lists the volumes estimated from characterization sampling at the site and the actual volumes of waste that was remediated.

Table A-22
Estimated and Actual Volumes of PCB Contaminated Soil and Sediment at Yankee Rowe

Medium	PCB Contamination Location	Estimated Initial Volume yd3 (m3)	Actual Volume yd3 (m3)
Sediment	East Storm Drain	520 (398)	305 (233)
	West Storm Drain Ditch	20 (15.3)	365 (279)
	Sediment subtotal	*540 (413)*	*670 (512)*
Soil	Industrial Area Soils	4,999 (3,822)	10,183 (7,785)
	SCFA[1]	3,100 (2,370)	13,050 (9,977)
	PCB Other Areas[2]	None	136 (104)
	Soil subtotal	*8,099 (6,192)*	*23,369 (17,867)*
	Total Sediment Plus Soil	8,639 (6,605)	24,039 (18,379)

[1]SCFA = Southeast Construction Fill Area
[2] Other areas = Mid-Lot Waste Debris Pile Area and painted blocks along the Deerfield River.

PCB Contaminated soil at the Yankee Rowe site was dispositioned depending on its radiological contamination status as follows:

- Radiologically clean soil with PCB contamination was processed onsite by heating to destroy the PCB. A truck monitor equipped with germanium detectors was used to determine the radiological status of PCB-contaminated soil and to verify acceptance of post-treated PCB soil as backfill.
- Soil containing both radiological contaminants and PCB contamination was disposed of as mixed waste. The impact of the PCB contamination was that, in addition to the radiological controls, measures to address the PCB component in the soil were required for shipping and disposal of the waste soil.
- The radiological concentrations in this soil were very low and well below the possible VLLW limits proposed in this report.

A.2.1.4 Other Hazardous Chemical Contaminated Soil at Yankee Rowe

Table A-23 list the volumes of soil contaminated with hazardous chemicals other than PCBs at Yankee Rowe.

Table A-23
Volume of Soil Contaminated with Hazardous Chemicals (other than PCB) at Yankee Rowe

Soil Cont-eminent	Location	Estimated Volume yd^3 (m^3)	Actual Volume yd^3 (m^3)	Rad Contaminated (Yes/No)
Dioxin	Dioxin area	278 (213)	300 (229)	N/A
Petroleum	Bulldozer Spill Area	0	9 (6.8)	N/A
	Drum in Woods	0	1 (0.76)	No
	Firewater Pump Drywell	0	25 (19)	Yes
	Firewater Tank (Tank 55)	0	275 (210)	Yes
	Fuel Oil Tank Area	150 (115)	242 (185)	N/A
	Fuel Spill 164 Area	0	2 (1.5)	No
	Furlon House Basement	0	40 (30.6)	Yes
	Radwaste Area	0	1 (0.76)	Yes
	Railroad Tie Area	0	1 (0.76)	Yes
	Turbine Building Office area	0	265 (203)	Yes
	Total volume	*150 (115)*	*861 (658)*	
Lead	Old Shooting Range	10 (7.6)	80 (61.2)	N
	Peninsula Sand Blast Area	0	430 (329)	N/A
	South Yard Sand Blast Area	0	180 (137.6)	Yes
	Total Volume	*10 (7.6)*	*690 (528)*	

N/A: Uncertain radiological status of soil

- As with the PCB contaminated waste discussed in the last section, the radiological concentrations in this soil was very low and well below the possible VLLW limits proposed in this report.

A.2.2 Maine Yankee Soil Characterization Results

Table A-24 shows the average nuclide fractions from the soil characterization results for Maine Yankee. The data in Table A-25 are from areas where soil concentrations were above or approaching the Maine Yankee soil site release limits and are therefore for areas where remediation was conducted and radioactive waste resulted. It should be noted that samples were taken in all areas of the site but are not shown here as they were generally below the soil site release limits. The contamination in the areas shown in Table A-25 is primarily from leaking components or spills in RCA yard areas.

Table A-24
Radionuclide Fractions in Soil at Maine Yankee

Radionuclide	Average Fraction of Total Activity
H-3	0.053
Ni-63	0.048
Co-60	0.009
Cs-137	0.890

Table A-25
Summary of Soil Characterization Results at Maine Yankee (Only Significantly Contaminated Areas)

Site Area	Number of Samples	Co-60 pCi/g (Bq/g)			Cs-137 pCi/g (Bq/g)		
		Positive Samples	Mean Co-60	Maximum Co-60	Positive Samples	Mean Cs-137	Maximum Cs-137
RCA: West Yard	58	23	0.62 (0.023)	3.29 (0.122)	55	10.99 (0.407)	156 (5.77)
RCA: East Yard	35	12	0.28 (0.010)	1.94 (0.072)	33	4.88 (0.181)	133 (4.92)
Roof Drains	7	4	4.09 (0.151)	11.2 (0.414)	6	0.33 (0.012)	0.53 (0.020)
Balance of Plant Areas	36	6	1.22 (0.045)	5.11 (0.189)	24	11.06 (0.409)	85.6 (3.17)
Plant RCA Areas	8	3	11,213 (415)	33,600 (1,240)	7	0.13 (0.005)	0.21 (0.008)
Waste Storage Building Yard	30	0	N/A	N/A	5	0.1 (0.004)	0.13 (0.005)

* The activity in roof drains is the result of concentration of airborne radionuclides from plant effluents which deposited on roofs. This activity was flushed into roof drains during rain events. The activity in roof drains is the result of concentration of airborne radionuclides from plant effluents which deposited on roofs. This mechanism was also seen at Connecticut Yankee.

A review of Table A-5 shows that of the areas showing significant radionuclide contamination at Maine Yankee:

- Only the "Plant RCA Areas" had levels that exceeded the VLLW limits proposed in this report.
- Most of the other areas of significant contamination had levels above the site release limits (i.e., would require remediation) but below the possible proposed VLLW limits.

A.2.3 Connecticut Yankee Characterization Approach and Results

A.2.3.1. Determining Areas of Potential Soil Contamination

The following is a discussion of the soil characterization work conducted at the Connecticut Yankee Plant. Figure A-4 shows the areas of the site with the potential for soil contamination based on a Historical Site Assessment of radiological events at the site.

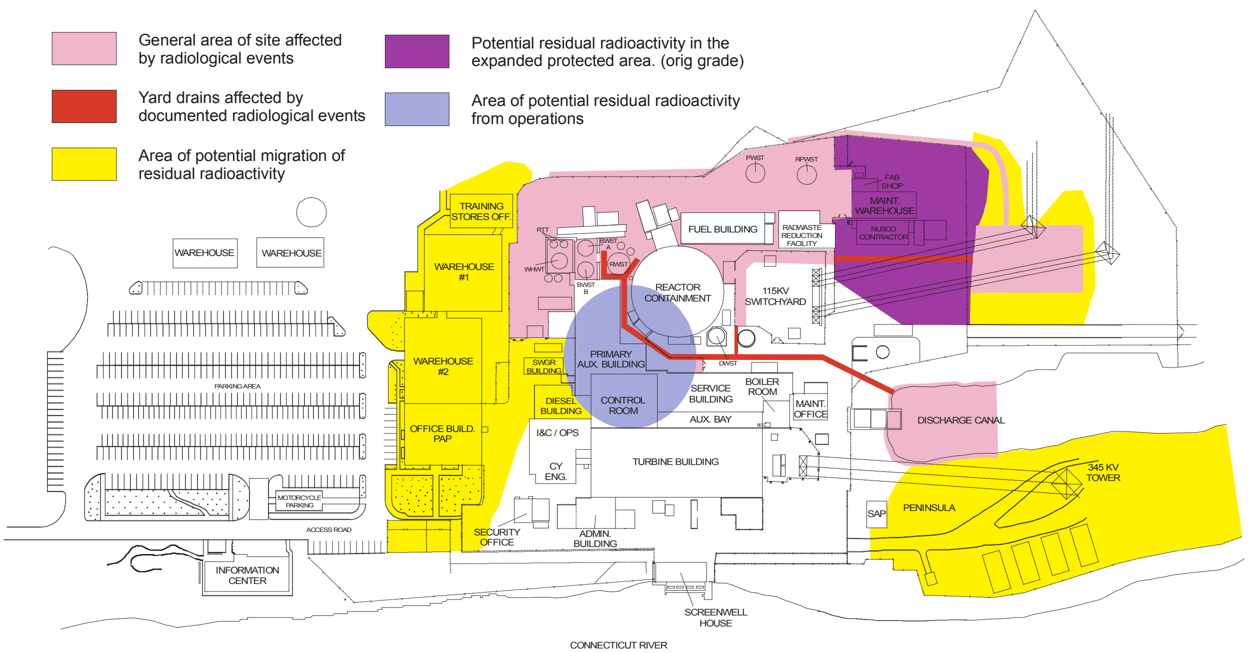

Figure A-4
Results of CY Historical Site Assessment

A.2.3.2 Soil Characterization Campaign

Approximately 200 sample locations generating approximately 1000 samples, primarily in the tank farm and down gradient areas, were chosen. Figure A-5 shows the subsurface soil sampling locations and where the sample results indicated remediation needed to be conducted. The dots on the figure correspond to subsurface soil sample locations. Most locations were outside buildings. The soil under certain buildings was sampled by coring through floors and sampling the soil below with "Direct Push" sampling equipment.

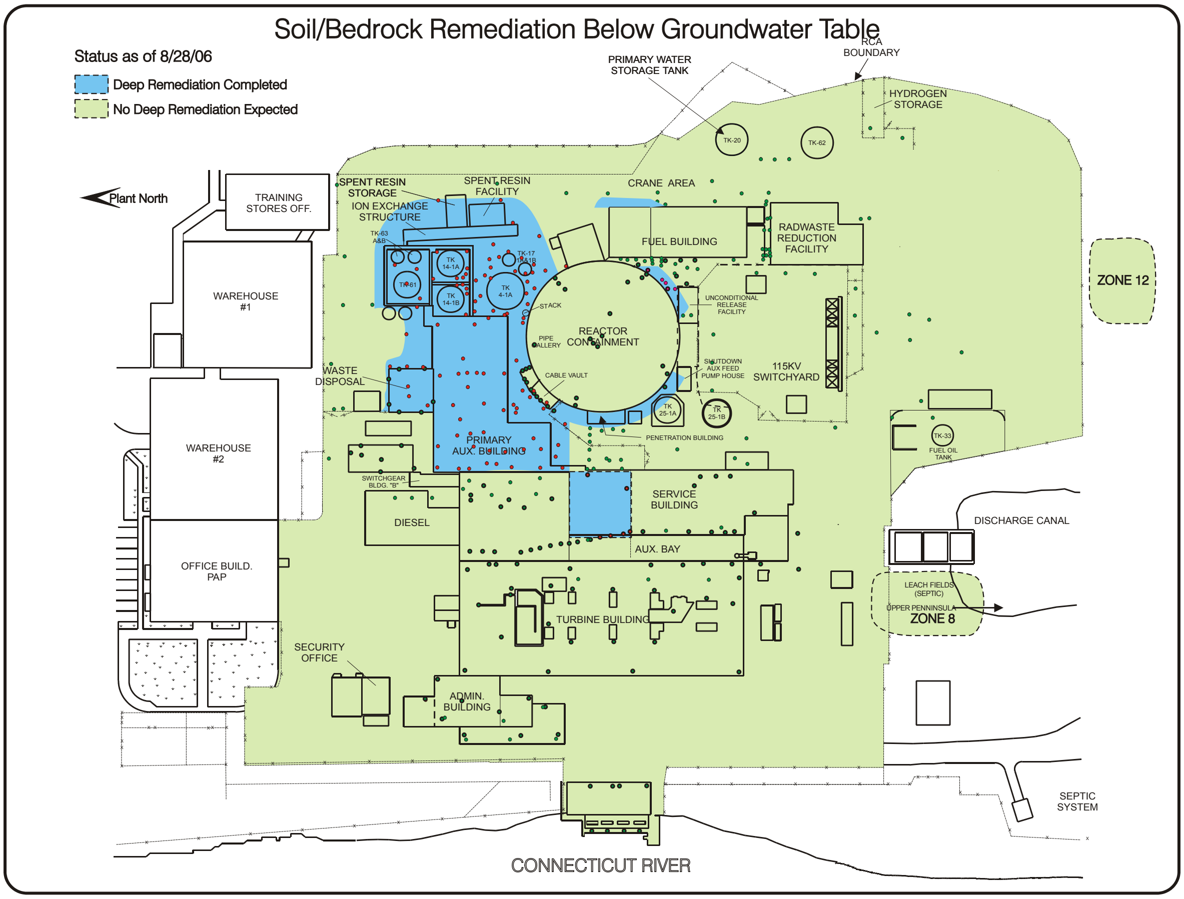

Figure A-5
Results of CY Site Soil and Bedrock Characterization

The results of this campaign indicated the highest levels of radionuclides were present under the Primary Auxiliary Building (PAB) Tank Farm shown to the northeast of the Containment Building in Figure A-5. In addition to high levels of Sr-90, significant levels of most of the other radionuclides of interest were present in the soil below the tank farm. A summary of the measured concentrations is given in Table A-26. It can be seen from Table A-26 that most radionuclide concentrations in the soil drop quickly as the sample distance below grade is increased. The radionuclides where concentrations in the soil deeper than 9 feet (3 meters) below plant grade were more than 10 % of average concentration above 9 feet below grade were H-3, Co-60, Ni-63, Sr-90 and Cs-137. This result is consistent (with the exception of Ni-63 which exhibited greater mobility than expected) with the relatively high mobility expected for these radionuclides for the sand backfill in this area.

Table A-26
Soil Concentrations in CY Tank Farm Area

Radio-nuclide	K_d (Sand per Ref. C-3) (cm3/g)	Concentration Ranges at Depths Below Plant Grade (Nominal Water Table Elevation 10 ft Below Grade) pCi/g (Bq/g)					
		Sand Between RWST & Pad	0 to 3 ft (0 to 1 m)	3 to 6 ft (1 to 2 m)	6 to 9 ft (2 to 3 m)	9 to 12 ft (3 to 4 m)	Average of All Depths
H-3	N/A	N/A	8.6 - 643 (0.3 -24)	10.1 (0.37)	96 (3.6)	21 (0.78)	8.1 (0.30)
C-14	5	N/A	0.28 to 1.6 (0.01 to 0.06)	1.16 (0.04)	<MDA	<MDA	0.16 (0.006)
Fe-55	220	N/A	62.3 (2.31)	<MDA	<MDA	<MDA	<MDA
Co-60	60	33,800 (1,250)	28.0 to 147 (10.4 to 5.4)	42.4 (1.57)	0.31 to 42.7 (0.01 to 1.6)	0.86 to 13.7 (0.03 to 0.51)	0.25 to 9.0 (0.01 to 0.3)
Ni-63	400	N/A	40.4 (1.5)	48.9 (1.8)	<MDA	7.54 (0.28)	<MDA
Sr-90	15	N/A	1.51 (0.056)	2.22 (0.08)	0.55 (0.02)	0.68 to 5.42 (0.025 to 0.02)	1.15 (0.043)
Cs-134	280	3,390 (125)	0.28 (0.01)	<MDA	<MDA	<MDA	<MDA
Cs-137	280	159,000 (5,663)	8.4 to 702 (0.31 to 26)	9 to 61 (0.4 to 2.3)	0.78 to 11.1 (0.03 to 0.4)	0.95 to 14.2 (0.035 to 0.53)	0.23 to 2.15 (0.01 to 0.08
Eu-154	N/A	N/A	1.76 (0.065)	1.1 (0.04)	<MDA	<MDA	<MDA
Pu-238	550	N/A	0.23 (0.01)	2.2 (0.08)	<MDA	0.013 (0.005)	0.04 (0.001)
Pu-239	550	N/A	0.23 (0.009)	0.8 (0.03)	<MDA	<MDA	<MDA
Pu-241	550	N/A	7.38 (0.27)	24.3 (0.09)	<MDA	1.24 (0.05)	1.97 (0.07)
Am-241	1,900	286 (10.6)	2.64 to 5.2 (0.1 to 0.19)	2.46 (0.09)	1.2 (0.044)	0.11 (0.004)	0.03 (0.001)
Cm-243	4,000	N/A	0.33 (0.01)	0.63 (0.02)	<MDA	<MDA	<MDA

Note: All analysis results for Mn-54, Nb-94, Tc-99, Ag-108m, Eu-152 and Eu-155 were less than MDA

Due to elevated levels of H-3 and Sr-90 in groundwater and the need to reduce these concentrations to the U.S. EPA Maximum Contaminant Levels (MCLs), Connecticut Yankee needed to remediate any soil that exceeded the following concentrations for those radionuclides:

- H-3 - 3.3 pCi/g
- Sr-90 - 0.065 pCi/g

This resulted in a great deal of remediation in the tank farm area and in areas hydraulically down gradient of the tank farm as discussed in the next section. A review of Table A-26 shows that for the tank farm area:

- Only the relatively small amount of sand between the RWST and its support pad, and some of the shallow samples indicated levels that exceeded the VLLW limits proposed in this report.
- For the other depths samples, down to 12 ft (4m), radionuclide contamination levels were above the site release limits (i.e., would require remediation) but below the proposed VLLW limits.

A.2.3.3 Areas at CY Remediated due to Groundwater Contamination

The soil hydraulically down gradient of the tank farm was contaminated due to the radionuclide migration in groundwater. Remediation in these areas was performed due to the Sr-90 contamination (and to a lesser extent H-3 contamination) present and the need to meet the EPA Maximum Contaminant Level limits identified by the NRC/EPA MOU and the State of Connecticut.

Remediation in the area consisted of removal of all the soil down to bedrock (approximately 25 feet [8.3 meters] below the plant grade). It was also noteworthy that soil above the water table in this area was not contaminated. This information supported the theory discussed above that Sr-90 contamination was being spread by groundwater from the PAB tank farm area to down gradient areas, increasing the amount of remediation needed.

Table A-27
Concentrations Down Gradient of CY Tank Farm Area

Radio-nuclide	Concentrations at Depths Below Plant Grade (Water Table Elev. 10 ft (3 m) Below Grade) pCi/g (Bq/g)					
	3 ft (0.9 m)	6 ft (1.8 m)	9 ft (2.7 m)	12 ft (3.6 m)	15 ft (4.5 m)	18 ft (5.5 m)
H-3	<MDA	<MDA	<MDA	<MDA	<MDA	0.70 (0.026)
Co-60	<MDA	<MDA	<MDA	<MDA	0.039 (0.001)	0.03 (0.001)
Sr-90	<MDA	<MDA	<MDA	0.057 (0.002)	0.54 (0.02)	0.37 (0.014)
Cs-137	<MDA	<MDA	<MDA	<MDA	0.078 (0.003)	<MDA

Note: All analysis results for C-14, Fe-55, Pu-241, and all other alpha and gamma emitting radionuclides not listed above were less than MDA

Tables A-27 and A-28 show the radionuclide concentrations in the areas down gradient of the CY tank farm. It can be seen in these tables that most of the deep samples have levels of Sr-90 above the site release limits (i.e., would require remediation) but well below the possible proposed VLLW limits.

Table A-28
Soil Radionuclide Concentration Adjacent to the CY Discharge Tunnels

Average Depth of Sample Below Plant Grade ft (m)	Soil Radionuclide Concentration (Range for Each Below Grade Depth)	
	H-3 in pCi/g (Bg/g)	Sr-90 in pCi/g (Bg/g)
2 (0.6)	0.016 to 2.31 (0.0006 to 0.085)	0.009 to 0.015 (0.0003 to 0.0006)
6 (2)	<0.023 to 1.02 (<0.0009 to 0.038)	<0.002 to 0.014 (0.00007 to 0.0005)
10 (3.1)	0.014 to 2.24 (0.0005 to 0.083)	<0.010 to 0.020 (<0.0004 to 0.0007)
14 (4.3)	0.019 to 1.73 (0.0007 to 0.064)	<0.009 to 0.026 (<0.0003 to 0.001)
18 (5.5)	0.016 to 1.74 (0.0006 to 0.064)	0.011 to 0.102 (0.0004 to 0.004)
22 (6.7)	<0.021 to 2.39 (0.0008 to 0.088)	0.014 to 0.184 (0.0005 to 0.007)
26 (8)	0.034 to 1.24 (0.001 to 0.046)	0.015 to 0.122 (0.006 to 0.0005)

A.2.3.4 Locations of Radwaste Discharge Line Failure

Table A-29
Soil Concentrations from Radwaste Discharge Line Failure

Radio-nuclide	Concentration of Detected Radionuclides at Depths Below Plant Grade (Average Water Table Elevation 10 ft [3m] Below Grade) pCi/g (Bq/g)				
	pCi/g (Bq/g) @ 6 ft (1 m)	pCi/g (Bq/g) @ 9 ft (1 m)	pCi/g (Bq/g) @ 12 ft (2 m)	pCi/g (Bq/g) @ 15 ft (3 m)	pCi/g (Bq/g) @ 18 ft (5 m)
H-3	N/A	9.74 (0.36)	N/A	N/A	1.95 (0.072)
C-14	N/A	0.59 (0.021)	N/A	N/A	<MDA
Fe-55	N/A	N/A	N/A	N/A	34.5 (1.27)
Co-60	3.3 (0.12)	1.39 (0.05)	0.26 (0.01)	0.28 (0.01)	0.18 (0.007)
Ni-63	N/A	8.57 (0.32)	N/A	N/A	9.25 (0.34)
Sr-90	N/A	0.81 (0.03)	<MDA	0.068 (0.003)	<MDA
Cs-134	0.16 (0.006)	0.40 (0.015)	0.15 (0.006)	<MDA	0.037 (0.0014)
Cs-137	41.1 (1.52)	6.49 (0.24)	17.8 (0.66)	1.46 (0.054)	1.56 (0.058)
Pu-238	N/A	0.18 (0.007)	N/A	N/A	N/A
Pu-241	N/A	20.3 (0.75)	N/A	N/A	N/A
Am-241	0.22 (0.008)	0.27 (0.01)	<MDA	<MDA	<MDA

Sampling of the soil under the Service Building in the area of this pipe failure showed radionuclide levels above the soil screening concentrations for H-3 and Sr-90 (Values given in Table A-9) in the soil above the normal water table. Radionuclide concentrations to a depth of 15 ft (5m) were above the site release limits and required remediation after removal of the Service Building but below the possible VLLW limits proposed in this report.

A.2.4 Saxton Plant Soil Characterization

The Saxton plant removed contaminated soil in a Soil Remediation Project early in the decommissioning project (1994). The goal of this project was to reduce site Cs-137 levels to an average of less than 0.04 Bq/g (1 pCi/g). 55 additional surface sample locations and 42 additional subsurface samples locations were sampled in a soil characterization conducted in 1999. The results of this sample and analysis campaign are given in Table A-30.

Table A-30
Saxton Soil Radionuclide Concentrations Outside of Containment Vessel

Bore-hole #	Survey Depth ft (m)	Cs-137 pCi/g (Bq/g)		Co-60 pCi/g (Bq/g)	
		Concentration	Minimum Detectable Activity	Concentration	Minimum Detectable Activity
10	1 to 5 (0.3 to 1.5)	<MDA to 3.7 (0.14)	0.2 to 0.3 (0.007 to 0.011)	<MDA	0.3 to 0.5 (0.011 to 0.019)
10	6 (1.8)	7.9 (0.29)	0.4 (0.015)	<MDA	0.5 (0.019)
10	7 (2.1)	28.9 (1.07)	0.7 (0.026)	<MDA	0.9 (0.033)
10	8 (2.4)	17.0 (0.63)	0.4 (0.015)	<MDA	0.6 (0.022)
10	9 to 12 (2.7 to 3.7)	<MDA to 2.8 (0.10)	0.2 to 0.3 (0.007 to 0.011)	<MDA	0.3 to 0.5 (0.011 to 0.019)
11	1 to 5 (0.3 to 1.5)	0.2 to 5.5 (0.007 to 0.20)	0.2 to 0.3 (0.007 to 0.011)	<MDA	0.3 to 0.4 (0.011 to 0.015)
11	6 (1.8)	109.9 (4.07)	0.3 (0.011)	<MDA	0.4 (0.015)
11	7 (2.1)	17.4 (0.64)	0.4 (0.015)	<MDA	0.5 (0.019)
11	8 to 13 (2.4 to 4)	<MDA to 1.6 (0.06)	0.2 to 0.3 (0.007 to 0.011)	<MDA	0.3 to 0.4 (0.011 to 0.015)
13	0 to 7 (2.1)	0.1 to 5.3 (0.004 to 0.2)	0.1 to 0.4 (0.004 to 0.015)	<MDA	0.2 to 0.5 (0.007 to 0.019)
13	8 (2.4)	12.7 (0.47)	0.3 (0.011)	<MDA	0.5 (0.019)
13	9 (2.7)	66.2 (2.45)	0.2 (0.007)	<MDA	0.3 (0.011)
13	10 (3)	8.9 (0.33)	0.2 (0.007)	<MDA	0.3 (0.011)
13	11 to 14 (3.4 to 4.3)	0.2 to 0.4 (0.007 to 0.015)	0.2 to 0.3 (0.007 to 0.011)	<MDA	0.3 to 0.4 (0.011 to 0.015)

The results of the 1999 campaign indicated that the 1994 remediation project was largely successful with no surface samples indicating concentrations approaching the Saxton Soil DCGLs. Cs-137 was the only plant related radionuclide indicated in any sample. Only an area on the north side of the Containment Vessel (CV) indicated the need for remediation to a depth of at least 10 feet (~ 3 meters) with a maximum soil concentration of 0.34 Bq/g (9.3 pCi/g) at that depth. For all the areas remediated the radionuclide concentrations were well below the possible VLLW limits proposed in this report.

A.2.5 Summary of Soil Remediation Waste Activity levels for Decommissioning Plants

The characterization results discussed in this section indicate that a high percentage of the radioactive waste resulting from soil remediation during decommissioning for the plants reviewed would be below the possible VLLW limits proposed in this report.

References Appendix A

A-1 EPRI Report # 1015502, Concrete Characterization and Dose Modeling During Nuclear Power Plant Decommissioning, March 2008

A-2 EPRI Report # 1019228, Characterization and Dose Modeling of Soil, Sediment and Bedrock During Nuclear Power Plant Decommissioning, November 2009

B
DISPOSAL OF WASTE UNDER NRC ALTERNATE WASTE DISPOSAL PROCEDURES

Due to the high volume of low level radioactive waste which can result for the decommissioning process, options which present lower costs for waste disposal or beneficial onsite use of the waste have been explored. The following are two examples of a currently available option for access to lower cost disposal of waste for very low level waste through the NRC exemption process.

B.1 Alternate Waste Disposal Procedures for Yankee Rowe

The information in this section is from Reference A-1.

In December 2004 and supplemented in February 2005, Yankee Atomic Electric Company (YAEC) requested approval from the NRC for disposal of demolition debris containing small quantities of licensed radioactive material from the YNPS site, in accordance with the provisions of 10 CFR 20.2002 (Alternate Waste Disposal Procedures). YAEC proposed to transfer a portion of the solid waste generated from the decommissioning of buildings and structures to the Waste Control Specialists (WCS), LLC Facility in Andrews, Texas. WCS is a Subtitle C RCRA hazardous waste disposal facility permitted under Texas law, RCRA, and TSCA. At the time of submittal, WCS held a radioactive material license issued by the State of Texas, and had been receiving certain exempt radioactive materials since 2001. The site has since received a license from the NRC to receive all classifications of Low Level Radioactive Waste.

As a result of the environmental assessment of the proposed disposal option (April 2005), the NRC concluded that the “proposed action would not have a significant effect on the quality of the human environment”. The NRC approved YAEC’s request for disposal of demolition debris at the WCS site on May 6, 2005; however, company management decided against using this disposal option due to the timing for receiving Texas regulatory approval.

The following sections summarize the elements of the YAEC application to the NRC:

- A description of the waste material including the physical and radiological properties important to dose assessment
- A description of the proposed disposal facility and proposed manner and conditions of waste disposal
- An assessment of the potential dose to workers involved in transportation of the waste to the site and in waste handling at the site

- An evaluation of dose to members of the public after closure of the facility as a consequence of the proposed waste disposal

B.1.1 Description of Waste Material

The waste material included structural steel, soils, and concrete, as well as asphalt pavement or other similar solid materials. The origin of the waste material was the demolition and removal of structures and paved surfaces at the YNPS plant site following decontamination. The physical form of the bulk dry solid waste ranged from the size of sand grains up to solid structures with a volume of several cubic feet. The mass of demolition debris was estimated at approximately 60 million pounds, with an assumed density of 1.0 gram per cubic centimeter during shipment and 1.5 grams per cubic centimeter after compaction in the disposal cell at WCS. A breakdown of the waste by source and type is as follows: 5 million pounds of steel, 15 million pounds of soil and asphalt, 30 million pounds of concrete originating from the reactor support structure (RSS), and 10 million pounds of concrete originating from other structures.

The majority of the steel estimated for disposal at the WCS facility consisted of plate steel from the containment shell and plate steel walkways from within the containment structure. The remainder of the steel components included beams, pipes, and other framework structures with similar contamination levels as the plate steel. The estimated soil and asphalt originated from various areas on site. In addition to RSS concrete, concrete considered for shipment to WCS originated from the Spent Fuel Pool (SFP), the Ion Exchange Pit (IXP), the Primary Auxiliary Building (PAB), and miscellaneous slabs. This material was estimated at 10 million pounds.

At the time of application, YAEC was continuing to characterize the remaining contaminated structures and soils on the site. The structures had undergone extensive remediation and only low levels of residual contamination remained. Structural materials were expected to have only low levels of surface contamination. Concentrations of any rebar encased in the concrete were assumed to be much less than the surface contamination levels. Typical radionuclide fractions for each waste form were used in the initial characterization. The fractions used for concrete characterization are summarized in Table B-1 below, and represent conservative estimates of isotopic distributions for these waste categories. Concentrations of H-3 in each category were evaluated separately from other radionuclides and are shown in Table B-1 in units of pCi/g.

Table B-1
Isotopic Distributions Used in 20.2002 Application

Nuclide	Percent of Detected Activity (Note 1) (H-3 in pCi/g)	
	RSS Concrete	Other Concrete
H-3	198 pCi/g	89.6 pCi/g
C-14	8.2	<0.1
Fe-55	ND	ND
Ni-63	ND	72.4
Sr-90	ND	<0.1
Co-60	17.8	1.7
Cs-134	0.2	<0.1
Cs-137	73.8	25.8
Pu-241	ND	ND
All others	ND	0.1

Note 1: H-3 concentration (pCi/g) is independent of the concentration of other nuclides. Nuclides other than H-3 are shown as a percentage of detected activity. In the table ND = Not detected

As part of the 10 CFR 20.2002 application, an evaluation was performed to determine volumetric contamination limits (pCi/g) applicable to the shipment of waste. All material to be shipped to WCS would then be monitored using the YAEC truck monitoring system. The truck monitor consists of 6 collimated high purity germanium detectors (HPGe) capable of detecting concentrations in the waste that are well below the volumetric concentration limits established in the waste evaluation. The methods used for establishing concentration limits for each radionuclide are discussed further below.

Extensive direct measurement surveys were performed on the RSS following remediation, and prior to demolition. Characterization results indicated that the RSS would meet NRC Final Status Survey (FSS) unrestricted release requirements if it were to remain on site. Surveys indicated that in general, contamination levels were below 5000 dpm/ l00 cm^2 with small areas that exceeded that level. The mass of concrete debris from demolition of the RSS estimated for disposal at WCS was 30 million pounds.

B.1.2 Description of the Disposal Facility

As stated above, in addition to the recently issued NRC disposal license, WCS is a Subtitle C RCRA hazardous waste disposal facility permitted under Texas law, RCRA and TSCA. A license to treat, process and store low-level radioactive waste was issued to WSC in November 2006. An in depth description of the site is provided in attachments to the WCS low-level waste disposal license application, which may be viewed at: http://www.wcstexas.com/. A brief description of the important features is included below.

Natural Site Features of the WSC Site

The most important engineered feature at the facility relative to radioactive material confinement is the 5-meter thick, low permeability, erosion resistant cover to be constructed at cell closure. This final cover is to be constructed of compacted red bed clay in conjunction with a 40-mil HDPE liner, integrated with a similar liner along the sides and bottom of the cell.

Other facility design features and operating procedures provide shorter-term confinement of radioactive materials and limit the potential for radiation exposure during receipt and placement of materials in the cell. These procedures minimize release of material through the use of mechanized equipment for transfer and disposal, and through dust suppression measures.

The YAEC application indicated that disposal of radioactive materials at the WCS site is regulated under the State of Texas, Texas Department of Health (TDH) or Texas Commission of Environmental Quality (TECQ), and that WCS operations were regulated in accordance with the State of Texas radiation protection standards. Specifically, "no owner or operator may operate in a manner such that any member of the public would receive an annual TEDE in excess of 100 millirem per year. In addition, no person may release radioactive material for unrestricted use in such a manner that the reasonable maximally exposed individual would receive an annual TEDE greater than 10 millirem per year".

WCS operations are conducted in accordance with its Radiation Safety Program and ALARA, which includes radiation, contamination, and airborne radioactive material surveys, and personnel dosimetry, bioassay, and environmental monitoring programs. The environmental monitoring program includes air, ground water, surface water, and soil analysis, as well as ambient radiation level assessment in the environs of the facility. The waste material from the YNPS would not be isolated or dedicated to a single burial cell at the WCS facility, but would be interspersed with other radioactive and non-radioactive waste material.

B.1.3 Potential Doses to Transportation and Facility Workers

YAEC conducted a dose analysis using two transportation scenarios to demonstrate that the Maximally Exposed Individual (MEI) dose equivalent would not exceed a "few millirem/yr". This standard of a "few mrem/yr" (e.g., five (5) millirem/yr) to a member of the public prior to license termination is defined in NRC Regulatory Issue Summary 2004-08. As required by the NRC, in the scenarios, the transportation workers and workers at the WCS site were not treated as occupational radiation workers, but as members of the public. Evaluations of both internal and external dose hazards to the transportation worker were performed.

YAEC evaluated two shipping scenarios, one in the original application and one in a supplemental application. In the original scenario, the waste would be the transported in intermodal containers from YNPS to a rail site in Palmer, MA. Once in Palmer, the intermodals would be relocated to a rail transport car where they would be transferred to the WCS site in

Texas. In the supplemental scenario, the waste is off loaded to a gondola rail car at a facility in Worcester, MA. (Reference 6-2).

B.1.3.1 Initial Scenario

YAEC assumed that all the waste material would be shipped to and received at the WCS facility in one year, starting in 2005. Each shipment was assumed to correspond to the maximum road weight of 45,000 lb. The analysis assumed 36 shipments per week, with a total waste mass of approximately 84 million pounds. The shipments would occur in strong-tight containers after verification that Department of Transportation (DOT) external loose surface contamination limits were met. No internal dose hazards were assumed for this transportation scenario.

Conservative average activity concentrations for each container were calculated using the penetrating gamma dose rates associated with cobalt-60 and cesium-137, and an appropriate geometry model based on container dimensions. Calculations of gamma exposures to the MEI were performed for concrete rubble, soil, and scrap iron, with scrap iron yielding the highest exposures. Worst-case exposures were used in the Microshield© dose rate model, for three receptors:

- A worst-case 1 meter (3 feet) receptor point at the center of the length of the container for workers loading and securing the shipment.
- A typical 1.5 meter (5 feet) receptor point at the center of the length of the container for miscellaneous worker tasks such as inspection and off loading.
- A theoretical "driver" receptor point of 3 meters (10 feet) for transport to the TSD facility.

Dose rates to each receptor per pCi/g for each of the dominant gamma emitting radionuclide are shown in Table B-2 below.

Table B-2
Dose rates During Transportation of YNPS Waste to WCS

Receptor	Co-60 dose rate (uR/h)	Cs-137 dose rate (uR/h)
Loader	0.863	0.165
Off-loader/Inspector	0.553	0.106
Driver	0.238	0.046

A "time and motion" study was performed to determine annual exposure times for transportation and landfill workers. It was determined that based on a 1 pCi/g concentration and the time/motion analysis, that the "Landfill Driver" was the MEI with an exposure time of 936 hours and 0.517 mrem/yr and 0.099 mrem/yr dose rates for Co-60 and Cs-137, respectively. The exposure time was conservatively divided between 2 workers instead of the assumed 10 worker staff at the WCS facility. YAEC determined that based upon the landfill driver MEI scenario, radionuclide concentrations of 20 pCi/g for Co-60, and 100 pCi/g for Cs-137 would equate to a 5 mrem/yr exposure. For waste containing mixtures of Co-60 and Cs-137, the unity rule would apply.

Operating experience at the WCS facility indicates that there would be no internal dose hazards associated with proposed disposal activities and that on-site monitoring would be used to demonstrate and control compliance with all applicable limits.

B.1.3.2 Supplemental Scenario

In a supplement to the initial request, YAEC provided an evaluation of an additional transportation option for the transfer of the wastes to the WCS site. YAEC indicated that it was anticipated that both transportation plans would be implemented.

The supplemental option involved offloading of the material to a gondola rail car at a facility in Worcester, MA, which is about 2.5 hours from the YNPS site. The purpose of the facility is to receive and dump loads of materials into rail cars for transport. The proposal indicated that the waste material from the YNPS would be hauled to the Transload Facility in trucks and loaded into gondola rail transport cars. The material would be discharged through an open directional hopper directly into a strong tight container within the gondola railcar. Loads would be groomed and adjusted, as needed, using a backhoe attached to the rear of the hopper. The strong tight container would be sealed and surveyed for transport. The Transload Facility crew consists of a crane operator and one or two ground individuals. The approximate time to load a gondola railcar was assumed to be 60-90 minutes. It was further assumed that approximately 510 rail cars would be needed to ship the 84 million pounds of material at 75% capacity for each load. It was anticipated that 50 truck loads per week would be transferred to the facility by eight truck drivers. It was estimated that each of the eight drivers would require 656 hours to transfer the 84 million pounds of waste to Worcester.

The initial transportation exposure analysis provided dose rates from an intermodal-type container for three configurations. YAEC determined that the exposure rate calculation for the intermodal container was a reasonable representation of the rail car scenario.

Two receptors were considered in the supplemental analysis: a driver who would be 3 meters (10 feet) from each load, and, a waste handler assumed to be 1.5 meters (5 feet) from each load. Each driver was assumed to be exposed for 656 hours and each waste handler was assumed to be exposed for 765 hours. An exposure analysis for Co-60 and Cs-137 indicated that the handler at the transload facility was the MEI for this scenario (0.423 mrem/y and 0.081 mrem per year for Co-60 and Cs-137 respectively). However, the exposure calculated for this worker was less than for the MEI in the original scenario.

The potential for internal exposures from inhalation was considered. YAEC concluded that the operation at the Transload Facility is similar to onsite operations at WCS and therefore potential for internal exposure could be eliminated on the same basis.

Both YAEC and the NRC concluded that the supplemental transportation scenario is bounded by the analysis performed for the original scenario where the maximally exposed individual is the landfill driver at the WCS facility.

B.1.4 Potential Doses to the Public After Site Closure

The RESRAD computer code was used to calculate the projected dose from the proposed disposal activity to future residents at the disposal site using the Resident Farmer Scenario. Each radionuclide identified in the License Termination Plan was included at a concentration of 1 pCi/g to determine the calculated dose equivalent to the maximally exposed individual. Soil, steel, and concrete were modeled separately.

The results of the calculations indicated that the only radionuclide with a calculated dose greater than the RESRAD lower cutoff value of 1.0E-30 mrem/yr was Pu-238. Doses from Pu-238 from the three media categories were on the order of 4E-07 to 2E-06 mrem/pCi/g. The dose is due primarily to radon production from the decay of Pu-238. Using the radionuclide ratios in Table B-1, and the activity limits stated in Section B.1.2, the maximum concentration of Pu-238 in the Yankee Rowe waste would be 0.1 pCi/g. This means that the maximum dose to a member of the public from the Yankee Rowe waste after the closure of the WCS site would be 2E-07 mrem/yr or more than 7 orders of magnitude below the NRC criteria of 5 mrem/yr. This illustrates that the dose to the workers at the WCS site is by far the factor that limits the radionuclide concentrations allowed under the NRC exemption of the Yankee Rowe waste.

YAEC concluded that after applying the concentration limits calculated for Cs-137 and Co-60 (based on the MEI individual from the transportation scenario) to the RESRAD results using the isotopic distributions for each of the waste categories, “it is extremely unlikely that the waste stream contemplated in this analysis could result in a dose that could approach the ‘few millirem’ criteria. Therefore, this pathway need not be considered further in the YAEC dose analysis.

B.2 Alternate Waste Disposal Procedures for Connecticut Yankee

In the same manner as the Yankee Rowe case given above, Connecticut Yankee Atomic Power Company (CYAPCO) requested and received approval from the NRC of alternate waste disposal procedures in accordance with the provisions of 10 CFR 20.2002. This approval was exemption that would allow CYAPCO to dispose of demolition debris from the Haddam Neck Plant (HNP also known as CY) decommissioning activities at the US Ecology Idaho Facility in Grand View, Idaho. As the schedule of a required license change for the US Ecology Idaho site did not support the CY decommissioning schedule, it was not possible for CY to ship the subject waste to the Idaho facility.

The US Ecology Idaho site is located near Grand View, Idaho in the Owyhee Desert. Grand View is approximately 40 miles south-southeast of Boise, Idaho. The material in this section is taken from the following reference: CYAPCO letter NRC, CY-04-168, Haddam Next Plant Request for Approval of Proposed Procedures in accordance with 10 CFR 20.2002, September 16, 2004.

The following Sections describe disposal site characteristics, the waste material, the radiological assessment and conclusions contained in the CYAPCO request. The main conclusion of the analysis in the request was that the potential dose to workers involved in the transportation and placement of the waste at the site and to members of the public after closure of the US Ecology Idaho facility as a consequence of the proposed CYAPCO waste disposal would be no more than a few millirem per year Total Effective Dose Equivalent (TEDE).

B.2.1 Disposal Site Characteristics

The following describes the features and permits of the disposal facility of importance in radiological assessment in place when the request was submitted to the NRC in 2004. It provides a summary of the geographical and physical environment of the facility, including the engineered features, the permits under which the site operates, including radioactive material disposal limits, site operations, including radiation monitoring, and post-closure plans.

B.2.1.1 Environment and Facility Design

The most significant natural site features that appear to limit the transport of radioactive material are the low precipitation rate and the long vertical distance to groundwater. The precipitation rate in this arid location is 0.184 meters per year. The depth to groundwater accommodates a 3.6-meter thick cover, a 33.6-meter thick disposal zone, and a 61-meter thick unsaturated zone between the base of the disposal cell and groundwater.

A number of engineered features designed to enhance confinement performance have been incorporated in the facility. The most important from the standpoint of radioactive material confinement is the 3.6-meter thick, low permeability, erosion resistant cover to be constructed at cell closure. This final cover is to be constructed of compacted soil in conjunction with a 40-mil HDPE liner. The HDPE cover liner is to be integrated with a similar liner along the sides and bottom of the cell. (The confinement effectiveness of the HDPE liner is ignored in this analysis to assure that projections of potential radiation dose are conservatively maximal.)

Together, the low precipitation rate, the thick, low-permeability cover, and the thick unsaturated zone minimize the potential for long term infiltration, dissolution, and transport of constituents to groundwater. The thick cover also minimizes the potential for exposure of waste material radionuclides by erosion or intrusion and minimizes release of radon gas to the atmosphere (although the dose due to the release of radon is shown to be insignificant in this analysis).

Other facility design features and operating procedures provide shorter term confinement of radioactive materials and limit the potential for radiation exposure during receipt of material and emplacement of materials in the cell. These include a closed facility with filtered ventilation exhaust for transfer of incoming waste material from the shipping conveyance to US Ecology Idaho waste transfer vehicles, mechanized equipment for disposition of waste material in the cell, and the application of an asphaltic spray (to control resuspension of the material into the air) over newly deposited material at the end of each day's operations.

The total capacity of the cell which would receive the CY waste is approximately two million cubic yards (1.5 million cubic meters). The surface area of the cell is approximately 88,000 square meters. The material that CY proposes for disposal if occupying the full depth of this cell would have a surface area of approximately 900 square meters. This means that the CY material would occupy approximately 1 % of the total volume of this disposal cell.

B.2.1.2. Permits

The US Ecology Idaho site is a Subtitle C RCRA hazardous waste disposal facility permitted under the authority of the Idaho Hazardous Waste Management Act, Chapter 44, Title 39, of the Idaho Code. The site operates under permit IDD073 114654. A Class I Permit Modification was approved in 1999, and a Class II Permit Modification was approved in 2001. The latter permit modification also accommodates recent changes to Idaho law and regulations regarding the disposal of radioactive material, as described below. In accordance with its regulations and permit conditions, the site had been receiving certain radioactive materials exempt from Nuclear Regulatory Commissioning licensing requirements, including U.S. Army Corps of Engineers FUSRAP program materials, for a number of years.

Disposal of radioactive materials at the US Ecology Idaho site is regulated under the Rules of the Department of Environmental Quality, IDAPA 58.01.10, "Rules Regulating the Disposal of Radioactive Materials Not Regulated Under the Atomic Energy Act of 1954, As Amended." These regulations establish radiation protection standards and permit conditions for disposal of these materials at a permitted disposal facility under the authority of the Idaho Hazardous Waste Management Act, Chapter44, Title 39, Idaho Code.

Under the Idaho Department of Environmental Quality general protection standards, all owners and operators disposing of radioactive materials are required to conduct operations in a manner consistent with radiation protection standards contained in 10 CFR Part 20. In addition, no owner or operator may operate in a manner such that any member of the public would receive an annual Total Effective Dose Equivalent (TEDE) in excess of 100 millirem per year. In addition, no person may release radioactive material for unrestricted use in such a manner that the reasonably maximally expose individual would receive an annual TEDE greater than 15 millirem per year, excluding natural background.

The facility owner or operator was also required to comply with each of the following permit conditions:

- Department-approved waste acceptance criteria for radioactive material;
- A Department-approved closure program that provides reasonable assurance that the radon emanation rate from the closed disposal unit will not exceed twenty (20) picocuries per square meter per second averaged across the entire area of the closed disposal unit and meets the general radiation protection standard for the public (TEDE of 15 millirem per year); and

- A Department-approved environmental monitoring program that monitors air, ground water, surface water and soil for radionuclides and ambient radiation levels in the environs of the facility, and which demonstrates that no member of the general public is likely to exceed a radiation dose of 100 millirem per year from operations conducted at the site.

B.2.1.3 Operations

US Ecology Idaho accepts only wastes that conform to waste acceptance criteria approved by the Idaho Department of Environmental Quality, as required in DAPA 58.01.10. This is implemented in the form of a two-step pre-acceptance protocol. In the first step, the generator prepares a chemical and physical characterization of the waste stream on a US Ecology Idaho standard form. The second step is an evaluation performed by US Ecology Idaho to determine the acceptability of the waste. No waste is shipped until the waste is determined to be acceptable by US Ecology Idaho.

Waste acceptance criteria applicable to the material intended for disposal are as follows:

- Acceptable Dose Rate at receipt to insure that the yearly dose criteria stated in the following paragraph are maintained.
- The sum of the concentrations of all radionuclides present in the waste does not exceed 2,000 pCi/g (This limit has subsequently been raised to 3,000 pCi/g).

US Ecology Idaho was required by condition of its Department of Environmental Quality permit to operate in a way that assures that the highest potential dose to a worker handling radioactive material is 400 millirem TEDE per year, and that assures that the highest potential dose to a member of the public is 100 millirem TEDE per year from operations or 15 millirem TEDE per year from release of radioactive materials for unrestricted use.

To meet these requirements, US Ecology Idaho conducts its operations in accordance with its Radioactive Material Health and Safety Manual and other operating procedures. These procedures include measures for minimizing release of material in receipt and handling. Transfers of as-received materials from shipping conveyances to US Ecology Idaho vehicles are performed in a closed structure with bag-filtered ventilation exhaust. Workers use mechanized equipment to transfer and deposit material in the disposal cell. Materials placed in the cell are covered each day with asphaltic spray to minimize the potential for release of radioactive materials to the atmosphere.

To assist in demonstrating compliance with these requirements, US Ecology Idaho also operates a radiation monitoring program approved by the Idaho Department of Environmental Quality, as required in IDAPA 58.01.10. The program includes:

- Periodic collection of grab air samples with analysis for radon progeny,
- Periodic deployment and collection and analysis of passive track-etch detectors with analysis for radon concentration,

- Periodic deployment and collection of passive dosimeters at locations around the perimeter of the cell with analysis for direct radiation exposure

The follow samples are analyzed for Isotopic uranium and thorium, Ra-226 and Gross Alpha and Gross Beta radioactivity (Prior to allowing shipment of CY material, analyzes of the following samples for gamma radionuclides was to be instituted).

- Periodic collection of grab air samples during material transfer operations
- Periodic collection of continuous air samples from the admin/lab area
- Periodic collection of soil samples from locations downwind of the disposal area
- Periodic collection of groundwater samples from two monitoring wells with analysis for gross

B.2.1.4 Post-Closure Plan

As required by the Idaho Department of Environmental Quality in IDAPA 58.01.10, US Ecology Idaho maintains an approved closure plan, submitted as part of its permit application. The plan conforms to all standard closure and post-closure requirements applicable to RCRA disposal facilities, including post-closure monitoring and financial assurance.

The plan provides reasonable assurance that the radon emanation rate from the closed disposal unit will not exceed twenty (20) picocuries per square meter per second averaged across the entire area of the closed disposal unit and reasonable assurance that the general radiation protection standard for the public (TEDE of 15 millirem per year) will not be exceeded. It should be noted that this standard for post closure exposure to a member of the public is set below the NRC standard for unconditional release of an NRC licensed facility which is 25 millirem per year TEDE.

B.2.2 Description of Waste

B.2.2.1 Physical Properties

The waste material (the demolition debris) intended for disposal includes flooring materials, concrete, rebar, roofing materials, structural steel, soils associated with digging up foundations, and concrete and/or pavement or other similar solid materials. Soils remediated for the purpose of meeting the final status survey requirements of the Haddam Neck Plant License Termination Plan (i.e. exceed the Derived Concentration Guideline Levels (DCGL) in the LTP) would generally not be disposed of at the US Ecology facility as the concentrations of the key gamma radionuclides at the DCGL levels are approximately a order of magnitude over the averages determined in later in this evaluation. Large quantities of material at the DCGLs would therefore increase the dose to site workers.

The demolition debris proposed for disposal at the US Ecology Idaho facility will originate from the demolition and removal of structures and paved surfaces at the HNP plant site, after the structure/surface has been decontaminated to remove areas that are highly contaminated.

The physical form of this demolition debris was that of bulk material of various sizes ranging from size of sand up to occasional monoliths with a volume of several cubic feet. CYAPCO, for the purpose of calculations, assumed the material to be a homogeneous mixture with a specific density of 1 gram per cubic centimeter during shipment and 1.5 grams per cubic centimeter after compaction in the disposal cell at US Ecology. The material will be dry solid waste containing no absorbents or chelating agents.

B.2.2.2 Estimated Waste Volume

It is estimated that the mass of demolition debris originating from the decommissioning of the HNP would total approximately 100 million pounds. A breakdown of this waste by source is shown in Table B-3. With an assumed density of 1.5 grams per cubic centimeter(after compaction at the disposal site), the estimated volume of material disposed at the US Ecology Idaho facility would be approximately 40,000 cubic feet. This represents approximately 9 percent of the annual volume of waste received at the US Ecology Idaho facility at that time. It was conservatively assumed that all the HNP material would received in one year although it is anticipated that waste will be shipped to the US Ecology facility starting in 2004 thru 2006.

Table B-3
Estimated CY Waste Quantities Proposed for Exemption under 10 CFR 20.2002

Source of Waste	Estimated Weight (lbs)
Containment Walls	40,000,000
Containment Floors and Internal Structures	20,000,000
Residual Heat Exchanger (RHR) Pit Floors	1,000,000
RHR Pit Walls	2,000,000
Waste Disposal Building Walls	2,500,000
Waste Disposal Building Floors	500,000
Remainder of Primary Auxiliary Building (w/o RHR Pit)	7,000,000
Spent Fuel Pool Walls and Floor	1,000,000
Remainder of Fuel Building (Above Grade Portion)	8,000,000
Service Building (Above Grade Portion)	8,000,000
Miscellaneous Structures/Soil/Asphalt	10,000,000
Total	100,000,000

The material would not be isolated or dedicated to a single burial cell at the US Ecology Idaho facility. Rather, it will be co-mingled with other radioactive and non radioactive waste material. The material will be covered at the end of each workday with an asphaltic spray to lockdown contamination, in accordance with US Ecology Idaho facility requirements.

B.2.2.3 Radiological Characterization of Waste

B.2.2.3.1 Background

As discussed in Appendix A, Connecticut Yankee performed a great deal characterization the radiologically contaminated buildings on site. The demolition plans when the partially decontaminated waste was to be sent to US Ecology Idaho were to scabble off surface concrete where contamination levels are high and to dispose of this material at radioactive waste disposal facilities other then the US Ecology, Idaho facility. Areas of concrete where high neutron flux has caused significant activation of the concrete also were not proposed for disposal at the US Ecology Idaho facility. After dispositioning the surface contaminated material containing the highest levels of radioactivity, the remainder of the building and structures would be demolished and the debris shipped to the US Ecology facility near Grand View, Idaho. For the purpose of determining the radioactivity level of material to be shipped to the US Ecology facility, concrete core sampling is most appropriate as these portions of the applicable buildings will be demolished in total. The demolition process results in mixing to the surface and volumetric contamination with the remainder of the wall and floor material. This makes the average concentration in the total thickness of the wall or floor appropriate in determining the overall radioactivity content of the waste material. It is also appropriate to use average values as the dose limits are in terms of annual exposures. Any variation of the waste shipments would be incorporated in the average of all shipments made during a year.

Structural material other then concrete were expected to have only low levels of surface contamination and were therefore bounded by the characteristics of the concrete intended for disposal. Any rebar incased in concrete was also expected to be much less than the surface contamination levels as it was located below the depth to which most of the surface contamination was located and therefore could be treated as the same as the concrete.

B.2.2.3.1 Concrete Characterization Results

The characterization sample results for the contaminated building at CY were given in Section A.1.3. Whereas many of the samples did not show any detectable activity for most radionuclides, the average scaling factor calculated from the surface wafers were used to calculate the average activity for all radionuclides except H-3, C-60 and Cs-137. For these radionuclides the average of the sample results was used to characterize the waste.

The characterization samples for from inside the containment liner show measurable levels of the C-14. As is shown later in this section, C-14 has fairly significant calculated post closure dose for the US Ecology Idaho site. This radionuclide was detected in concrete outside of the containment liner in the containment wall or in other buildings on site. It was postulated that a gaseous diffusion mechanism resulted in the shallow permeation of the C-14 into containment interior concrete. Therefore, the average sample results for C-14 inside the containment liner will be applied for all concrete inside the containment liner.

B.2.2.3.2 Miscellaneous Structures, Soil and Asphalt

There are other relatively small structures which were in the Radiological Controlled Area at CY but had very low contamination levels. These included the Cable Vault and the Radwaste Reduction Facility. It is planned that the above ground portions of these building above plant grad elevation be disposed at the US Ecology facility. These building have a very low contamination history and either have very small or no contaminated areas. There was also quantities of slightly contaminated soil that will be displaced to allow access for removal of foundations. Quantities of slightly contaminated asphalt was also removed from the site to meet non-radioactive site closure criteria. As previously discussed soil with radionuclide concentrations near the LTP DCGLs would not be disposed of at US Ecology, Idaho as these levels would be inconsistent with the concentrations in other type of waste proposed for disposal there. Waste concentrations determined for the containment walls outside the liner were applied to this class of waste materials.

B.2.2.3.3 Average Concentration of Waste to be shipped to US Ecology

CYAPCO applied the concentrations given in the tables in Section A.1.3 to the estimated weight of waste from each corresponding plant building. As the core samples were fairly shallow, the sample results were distributed over the complete thickness of the concrete samples to determine the average concentration of the waste to be disposed of at US Ecology. The results of that calculation are shown in Table B-4 along with the post closure dose calculated for each of the radionuclide concentrations. These values are used later to determine expected yearly dose to transportation and US Ecology site workers involved in disposal of the CY material.

It can be seen from a review of Table B-4 that the primary radionuclides that affect dose to personnel either transporting the waste or working with its disposal at US Ecology were C-60 and Cs-137. All other gamma emitting radionuclides were present in much lower levels and therefore need not be included in calculating worker dose. The alpha and beta emitting radionuclides were not a direct dose concern and can only be an inhalation or ingestion hazard during placement in the disposal cell. The controls, discussed earlier, present at the US Ecology facility and the relative low concentrations would preclude any significant dose from these radionuclides to the workers.

For the purposes of determining potential dose to a member of the public after the closure of the US Ecology site, the activities of other radionuclides were determined by the use of scaling factors based on actual CY characterization sample data. A review of the sample data shows that the scaling factors determined are conservative as many are based on sample results that indicate no detectable activity at Minimum Detectable Activity concentration rather then actual values above MDA.

Table B-4
Average CY Waste Concentrations and Post Closure Dose Calculation Results

Radionuclide	Avg Waste Concentration pCi/g	Post Site Closure Dose mrem/yr	Waste Concentration Allowed at 5 Mrem/yr Post Closure Dose pCi/g
H-3	261.88	2.74 E-03	4.8 E+05
C-14	3.68	1.13	14.6
Mn-54	**1.67E-03**	1.05 E-27	8.0E+24
Fe-55	0.14	<1 E-30	>7.0 E+29
Co-60	0.28	4.69 E-22	3.0 E+21
Ni-63	1.69	<1 E-30	>8.5 E+30
Sr-90	0.03	<1 E-30	>1.5 E+29
Nb-94	**1.25E-03**	1.25 E-03	5.0
Tc-99	6.49E-03	1.44 E-03	22.5
Ag-108m	**2.04E-03**	1.18 E-03	8.6
Cs-134	4.89E-03	2.88 E-28	8.5 E+25
Cs-137	0.97	6.67 E-27	7.3 E+26
Eu-152	5.01E-03	7.85 E-26	3.2 E+23
Eu-154	3.81E-03	2.29 E-25	8.3 E+22
Eu-155	3.85E-03	<1 E-30	>1.9 E+28
Pu-238	3.69E-03	7.40 E-09	2.5 E+06
Pu-239	1.23E-03	<1 E-30	>6.2 E+27
Pu-241	5.09E-02	<1 E-30	>2.5 E+29
Am-241	6.58E-03	<1 E-30	>3.3 E+28
Cm-243	1.11E-03	<1 E-30	>5.5 E+27

Note: Values in bold type are based on the Minimum Detectable Activity (MDA) (i.e., the radionuclide was not detected at the MDA concentration)

B.2.2.4 Radiological Assessments

B.2.2.4.1 Transport Worker Dose Assessment

The Transportation Scenario Maximally Exposed Individual (MEI) dose equivalent was projected not to exceed a few (e.g., five (5)) millirem/yr for the CY waste proposed for disposal at US Ecology Idaho. The transportation workers and worker at the US Ecology site are treated as members of the public as the US Ecology site is not licensed by the NRC. Evaluations of both internal and external dose hazards to the transportation worker are discussed below.

Each conveyance was to be a strong-tight container and verified to be in compliance with Department of Transportation (DOT) external loose surface contamination limits prior to shipment. Therefore, there are no internal dose hazards associated with the Transportation Scenario.

The conservative average activity concentrations given in Table B-4 were used to calculate penetrating gamma dose rates external to the conveyance used to transport the material. The geometries modeled bounded any variations in the actual conveyances (e.g., intermodals) that could be used and locations of transportation workers.

The resultant worst case dose rate was 1.025 E-3 mR/hour. Therefore, a worker would need to spend in excess of 4,878 hours per year, in contact with side of the conveyance to exceed a dose equivalent of five (5) millirem per year. It is qualitatively judged to be non-credible that the Transportation Scenario Maximum Exposed Individual (MEI) (e.g., transportation worker, or any other member of public interacting with the transportation activity) would exceed these occupancy times. When a realistic maximum occupancy time of 1000 hours per year (50 % of the normal 2000 hour work year actually spend transporting CY waste) and worker location in respect to the package used, the expected dose is less than 1 millirem/yr for the CY waste concentrations.

As the limit of the 10 CFR 20.2002 exemption is 5 mrem/yr the concentrations of the primary gamma radionuclides would be 1.4 pCi/g for Co-60 and 4.9 pCi/g for Cs-137.

B.2.2.4.2 Disposal Facility Worker Dose Assessment

The Disposal Site–During Material Placement Scenario MEI dose equivalent was projected not to exceed a few millirem/yr for the CY waste proposed for disposal at US Ecology Idaho.

In support of their operating permit issued by the State of Idaho, US Ecology Idaho maintains a Radiation Protection Program including routine performance of radiation, contamination, and airborne radioactive material surveys as previously described in this section. The facility had conducting disposal activities involving materials similar to those proposed for disposal from CY, except that they are contaminated with source material, which has been exempted under 10 CFR 40. These source material isotopes (i.e., ^{238}U and ^{232}Th) are present in concentrations greater than, and have Derived Air Concentration (DAC) and Annual Limit on Intake (ALI) values several orders of magnitude more restrictive than the primary isotopes of concern for the CY. Despite this much larger internal dose hazard, the site had no significant internal dose exposures up to the time of the CY exemption request (information is not available for later time periods). Therefore, operating experience indicates that there would be no internal dose hazards associated with the disposal activities described herein, and on-site monitoring will be used to demonstrate and control compliance with all applicable limits.

The conservative average activity concentrations shown in Table B-4 were used to calculate penetrating gamma dose rates to the worker in the vicinity of the placed material prior to it being covered. The geometry modeled assumes a dose receptor point centered 18” above a representative slab of material (i.e., after placement) and bounded any after placement scenario. This calculation conservatively did not take credit for the non-radioactive material that would be co-mingled with the CY waste.

The resultant dose rate to the receptor point is 1.452 E-3 mR per hour. Therefore, a worker would need to spend in excess of 3,444 hours per year at this point to exceed a dose equivalent of five (5) millirem per year. It is qualitatively judged to be non-credible that the Disposal Site – During Material Placement Scenario MEI (e.g., disposal activity worker, or any other member of public interacting with the disposal activity) would exceed this occupancy time. When a realistic maximum occupancy time above the disposal cell of 1000 hours/yr (50 % of the normal work year spent directly above the disposal cell) is used the dose to the disposal facility worker is 1.45 millirem/yr.

As the limit of the 10 CFR 20.2002 exemption is 5 mrem/yr the concentrations of the primary gamma radionuclides would be 0.97 pCi/g for Co-60 and 3.4 pCi/g for Cs-137. It should be noted that this dose modeling scenarios used by CY for the transportation and landfill workers were much more conservative than those described in Section B.1.3.1 perform by YAEC which determined limits of 20 pCi/g for Co060 and 100 pCi/g for Cs-137.

B.2.2.4.3 Resident/Farmer Dose Assessment

The RESRAD computer code was used to calculate the projected effect of the disposal of the CY waste at US Ecology Idaho. Each isotope of concern was included at a soil concentration of one (1) pCi/g, such that the resultant calculated dose equivalent to the maximum exposed individual (Resident Farmer) could be evaluated in terms of mrem/year per pCi/g activity concentration. These results were then scaled to average concentrations for the various radionuclides given in Table B-4. The post closure dose for each of the radionuclides at the CY average concentrations is shown in the last column of Table B-4. It can be seen in Table B-4 that the total expected dose to a member of the public from the CY waste, post site closure, is 1.133 millirem/yr. At the maximum dose allowed for the 20.2002 exemption request of 5 mrem/yr, the maximum concentrations would be restricted by a C-14 concentration of 16.2. Using the CY scaling factor to the primary gamma radionuclides, the maximum allowable concentration for Co-60 would be 1.2 pCi/g and for Cs-137 4.3 pCi/g although if C-14 were not present in the waste, the allowable concentrations considering only post closure dose would be many orders of magnitude higher .

In addition to other conservatism in the post-closure dose calculation discussed else in this appendix, the values shown in Table B-4 are very conservative for the following reason: It can be seen that more than 99 % of the post closure dose results from the radionuclides C-14, Nb-94, Tc-99 and Ag-108m. A review of the RESRAD code runs for the CY waste shows that the dose from these radionuclides results from the groundwater pathway. A further review of code runs shows that the RESRAD default distribution coefficient (Kd) for these radionuclides which was used is a value of zero (0). This means that these radionuclides are assumed to have mobility in and below the disposal cell which is equivalent to water. A review of NRC guidance

(NUREG/CR-6697, Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes, Attachment C, Table 3.9-1) shows positive Kd values for all of these radionuclides. If the NRC guidance values for Kd had been used in lieu of the RESRAD defaults, the mobility of these radionuclides would be significantly retarded and post closure dose reduced to even less significant levels. This factor explains the large difference between the CY results those for the YAEC post closure dose analysis described in Section B.1.4.

B.3 Summary Disposal of Waste Under NRC Alternate Waste Disposal Procedures

This appendix shows examples of an alternate disposal approach for waste with very low radionuclide concentrations utilizing the NRC 10 CFR 20.2002 exemption process. It can also be seen in the above that the disposal site characteristics and/or the assumptions used in determining dose to a member of the public from this type of alternate waste disposal at a non-NRC regulated, RCRA site can have a large effect on the radionuclide concentrations allowed by the exemption.

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