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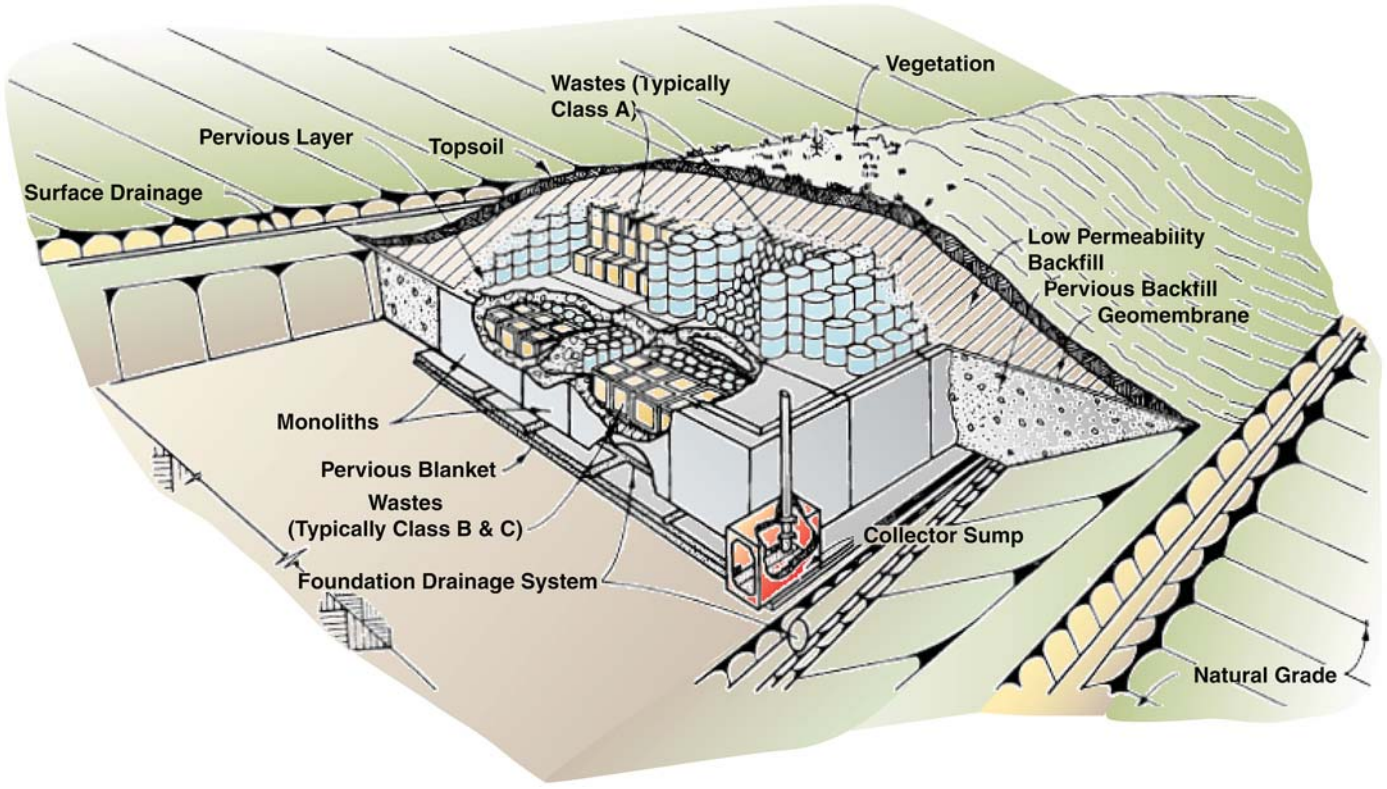
Developing Alternative Low Level Waste Disposal Criteria Per 10 CFR 61.58

Recent Experience and Current Practices



Investigation of Low Level Radioactive Waste Disposal Regulations and Practice

Recent Experience and Current Practices

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Interim Report, November 2009

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REPORT SUMMARY

The NRC identified, as a priority, expanding the guidance available for the development of alternative disposal criteria pursuant to Paragraph 61.58 of Part 61 of Title 10 of the US Code of Federal Regulations. This report constitutes the first part of a two-year project to examine the feasibility of alternative criteria and provide a model evaluation demonstrating the formulation of such criteria. The report also examines the radionuclides listed in 10 CFR §61.55 with respect to their properties and individual disposal risk, and discusses how recent information and alternative disposal credits might change their disposal concentration limits.

Background
The US regulations governing the disposal of low level radioactive waste are unchanged since their publication in December 1982. The Low Level Waste Policy Act of 1980, which defined LLW as a State Responsibility and directed the States to form regional compacts for the development of new disposal sites that would replace existing sites, directed formulation of the regulations. The industry derived the regulations to enhance public safety of disposal site operation as practiced in 1982, without imposing undue restrictions or increased costs on disposal site operation. Since the formulation of 10 CFR 61, there has been a significant evolution, both in the US and abroad, in the understanding of risks associated with specific radionuclides, as well as in disposal technology to address those risks. This report develops specific recommendations for the implementation of a model evaluation of alternative disposal criteria per 10 CFR §61.58.

Objectives
To develop specific recommendations relating to the implementation of a 10 CFR §61.58 model evaluation alternative disposal criteria.

Approach
The project team reviewed NRC guidance documents and surveyed past literature to develop a better understanding of the basis for the original 10 CFR 61 criteria. They also collected and reviewed IAEA reports and presentations to gain an understanding of the IAEA disposal models. The project team also reviewed past presentations by disposal site developers and compact authorities to get a snap shot of the technologies put forward for the proposed (but never realized) disposal sites. The report examines disposal site developments, including current and proposed compact sites, to gauge the general consensus on minimum requirements for near surface facilities.

Results

This report provides a benchmark position for the development of alternative disposal criteria based on current technical information and disposal practices. It examines the radionuclides documented in 10 CFR 61 individually, including their place in the overall source term, and how they impact classification. The following are several key conclusions drawn to date:

- The industry generally regards low level radioactive waste as waste with radioactivity dominated by short half-life activity not requiring permanent isolation.
- The concentration limits tabulated in 10 CFR 61.55 are out of date with respect to current ICRP dose conversion factors.
- Engineered barriers and manmade structures do not contribute and are not intended to contribute significantly to public protection beyond 500 years after site closure. By definition, waste suitable for near surface burial does not result in significant risk to the general public following its isolation period.
- Experience with disposal site development since the publication of 10 CFR 61 provides engineered provisions far in excess of that required by the regulation.
- Intruder scenarios used to form the basis for radionuclide limits in 10 CFR 61.55 do not recognize existing disposal practices or site-specific land use.
- Very long-lived radionuclides including, those listed in Table 1 of 10 CFR 61.55, generally do not contribute significantly to activity inventories in low-level radwaste disposal sites and should not be a controlling factor after a reasonable isolation period.

EPRI Perspective

This report is of interest to persons actively involved in the processing, management, and disposal planning of LLRW in power plants, and to persons involved in the design and licensing of new reactor facilities currently announced by a number of plant operating companies. Any implementation of proposals depends on a similarly favorable interpretation and development of a guidance document by the NRC, which would provide a technical basis for State regulators to follow. Currently, operating disposal sites may or may not have much incentive to revisit their existing licensing basis. However, it is expected that with proper guidance, any subsequent disposal site development would follow an alternative process. It is important for the industry to take a pro-active position in the resolution of LLW disposal to reduce or preclude storage of materials on-site and to reconstruct the regulatory framework to facilitate disposal in existing venues as well as the development of new disposal venues.

Keywords

Low level radioactive waste
Radwaste disposal criteria
Radionuclide concentration limits
Alternative disposal criteria
International Atomic Energy Agency (IAEA)
10 CFR Part 61

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EXECUTIVE SUMMARY

Brief History of U.S. Low Level Waste Disposal

More than 25 years have gone by since passage of the Low Level Waste Policy Act (LLWPA) in 1982 and the promulgation of 10CFR61 in 1983. This Act and regulation were devised to resolve disparities in the burdens assumed by States who at the time were hosting LLW disposal facilities. The disposal criteria developed for the regulation were generic criteria intended to accommodate as many as 8-10 new disposal sites dispersed regionally throughout the country. The facility requirements imposed by the regulation were minimal based on the premise that more stringent requirements would be too costly and, perhaps, also to reduce the dependence on State led license review efforts. Little envisioned in the Act came to fruition. The "regional" compacts were formed with all possible combinations including a number of states electing to "go it alone". During the first 20 years following the passage of the LLWPA, the only disposal site that emerged as a regional site was that of the Northwest Regional Compact which took over the already operating site at Richland, Washington which had been licensed prior to 10CFR61.

By 1992, it was generally recognized that the industry did not generate enough waste to support the number of sites envisioned by the Act. The Energy Policy Act of 1992 dropped several of the provisions of LLWPA. In particular, it dropped the provision that the States had to take title to the waste if they failed to meet the deadlines for new disposal site development. The process had failed and any incentive there might have been for the compacts to develop disposal sites were essentially eliminated when the State of South Carolina opted to allow the Barnwell disposal site to re-open and accept waste from all States. However, since July 2008, the Barnwell site reverted back to only accepting wastes from members within their Atlantic Compact.

Current Applicability of Original Bases of 10 CFR 61

Based on an evaluation of the technical basis for 10 CFR 61, it has been determined that the current U.S. waste classification system is not applicable in today's disposal context. The criteria is flawed in that it does not make accommodation for new information (e.g. improved characterization of source term, improved understanding of health impacts from radiation exposure, etc), does not provide protection credit for improved site designs, and contributes to orphaning materials that would otherwise be suitable for near surface disposal.

With the NRCs expressed interest in revisiting disposal criteria, including those relating to the classification system, it is important for the industry to be active participants in the review process both to address the current impasse but also to facilitate the process for future sites. Effectively, it had always been the NRC's position that wholesale revision of 10CFR61 would be

either insurmountable or that if it was revised that the outcome would be too unpredictable to undertake the risk. Therefore, attention at this time is focused on 10CFR §61.58. This paragraph allows for licensees to propose alternative disposal criteria based on a site specific evaluation. We believe that such an evaluation would enable the use of current evaluation technology, current dose conversion factors, updated scenarios for dealing with intruders, and proper regard for disposal technologies used including engineered structures and deeper disposal depth. These additions would obviate the current classification system in 10CFR §61.55 and could readily accommodate all of the waste currently regarded as low level radwaste. Implementation of these recommendations would represent a risk-informed approach to the regulation of the disposal of LLW.

Issues Discussed in This Report

This report focuses on issues related to classification and acceptability of disposal. A fundamental concern regarding regulations associated with the disposal of LLW is that there is no discrete definition of low level waste. Instead, it is defined by exclusion. The Waste Policy Act of 1980 provides a discrete definition of high level waste and by product material. Low level waste is then defined as not high level waste and not by product material. This definition by exclusion leaves a broad range of materials and waste types with an equally broad range of disposal protection requirements. In the context of a nuclear power plant, most of the waste that falls into the low level waste category are wastes dominated by short half life activity. This waste would not require permanent isolation which makes it candidate for near surface disposal. Some wastes from other fuel cycle and industrial activities are generally included in the low level waste category but do not conform to the half-life criteria. Such waste requires special treatment in disposal and does not fit well in the one size fits all classification system. Additionally, wastes that are classified as greater than Class C (GTCC), currently have no disposal pathway.

Additional attention is given to the nuclides that are listed in 10 CFR 61.55 with respect to their properties and the appropriateness of the express limits. It is observed that since the publication of 10 CFR 61 new statements of primary dose conversion factors from ICRP result in significant increases in some radionuclide limits. Class A limits for Ni-63 and Sr-90 are low by factors of 15 and 6, respectively, when the new ICRP dose factors are incorporated into the original dose rate estimation models.

A summary comparison of the IAEA system of waste classification is made with the NRC system of classification. The IAEA separates low level waste and intermediate level wastes (ILW) based mainly on material types. NRC does not define an intermediate level waste category. Most of the waste that IAEA would call intermediate level continues as LLW in the NRC scheme. Waste exceeding NRC Class C disposal categories may or may not be suitable for near surface disposal but is generally excluded without evaluation. The broadness of the IAEA categories and basic evaluation approach would accommodate all of the wastes we view as LLW including activated hardware without orphaning a portion of the stream.

Finally, operational improvements in older disposal sites including Barnwell and Richland and design enhancements in newer sites have obviated the intruder bases that were the foundation of the 10 CFR 61 limits. All of the sites are using some form of engineered structures to ensure

disposal site stability above and beyond that provided by the waste form. All of the disposal sites have invoked a system of layering with engineered, intruder resistant, covers. The new Waste Control Specialists (WCS) site in Andrews, TX has cover systems ranging from 25 to 45 feet thick. All wastes received at the site, regardless of disposal class, will be subjected to the same cover requirements. Intrusion is limited to drilling into the site. The intruder construction and agriculture scenarios are not probable. Estimated exposures afforded by the site are orders of magnitude lower than the performance objectives of 10 CFR 61.

This report is focused on exploring some of these issues and bringing them to light in the present discussions. The recommendations of this report will be used to construct a likely scenario for developing a new disposal site which will in turn be used to define inventory and concentration limits for the site. The aim is to lay out the issues and put them on the table for continuing discussions with regulators.

Key Conclusions

- Low level radioactive waste is generally regarded as waste whose radioactivity is dominated by short half-lived activity not requiring permanent isolation.
- The concentration limits tabulated in 10 CFR 61.55 are out of date with respect to current ICRP dose conversion factors.
- Engineered barriers and manmade structures do not contribute and are not intended to contribute significantly to public protection beyond 500 years after site closure. By definition, waste suitable for near surface burial does not result in significant risk to the general public following its isolation period.
- Experience with disposal site development since the publication of 10 CFR 61 provides engineered provisions far in excess of that required by the regulation.
- Intruder scenarios used to form the bases for radionuclide limits in 10 CFR 61.55 do not recognize existing disposal practices or site-specific land use.
- Very long lived radionuclides including those listed in Table 1 of 10 CFR 61.55 generally do not contribute significantly to activity inventories in low-level radwaste disposal sites and should not be a controlling factor after a reasonable isolation period.

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INTRODUCTION AND CONCLUSIONS

The NRC has undertaken an internal assessment of their regulation of Low-Level waste disposal. (1) This comes in response to observations by the Advisory on Nuclear Waste as well as the US General Accounting Office that the current regulatory approach was "not risk-informed". The corner-stone of the NRC regulation of low level waste disposal is the three tiered classification system provided in 10 CFR 61.55. (2) The classification system was developed to accommodate the broad range of site conditions attendant to 8-10 new regional disposal sites. It did not account for any particular site, although conditions and operating practices at the Barnwell, SC site and others served as important references. It was also developed without a clear definition of what constitutes low-level waste. The existing definition of LLW continues as a definition of exclusion. Low-level waste is defined as waste that is not high level waste or transuranic (TRU) waste or a few other categories. Therefore, low-level waste is highly dependent on the definitions of the other waste categories.

In the NRC's strategic assessment of low-level waste, high priority was placed on seven tasks, two of which (Tasks 4 and 6) directly relate to the disposal of nuclear power plant generated radioactive wastes. These are 1) Update Branch Technical Position on Concentration Averaging and Encapsulation and 2) Develop Guidance Document on Alternate Waste Classification (10 CFR 61.58). The first item was addressed in EPRI Report 1016120, "An Evaluation of Alternative Classification Methods for Routine Low Level Waste from the Nuclear Power Industry" published in 2007 and in EPRI Report 1016761, "Proposed Modification to the NRC Branch Technical Position on Concentration Averaging and Encapsulation (BTP): Technical Bases and Consequence Analysis" published in 2008. Highlights of this study were presented to the NRC at a meeting in June of 2008 and the study itself was provided to the NRC in early 2009. Considerations for the second NRC task to "Develop Guidance Document on Alternative Waste Classification (10 CFR 61.58), will be addressed by this report.

The central premise of 10 CFR 61 (2) was drawn from a directive of the Low Level Waste Policy Act of 1980 to promulgate regulations for the development of a series of regional disposal sites envisioned by the Act. (3) In support of the regulations, model sites were theorized representing the four distinct US regions with varying populations, annual rainfall, transportation distances, etc. Using the anticipated source terms for radioactive waste generated within the respective regions, model performance assessments were carried out to determine the impacts on various population groups such as facility operators, waste transporters, people living along transportation routes, and persons living in the vicinity of the site. (4)

There were four basic considerations to the NRC's evaluation:

- Protection of occupationally exposed workers and the public during operation of the facility
- Long-term environmental protection
- Protection of an inadvertent intruder
- Changes imposed by the regulatory initiatives should have a modest impact on disposal costs and disposal site operations

Up to this point there are no specific requirement for stabilization or for segregation of unstable wastes. Wastes were assumed to be placed in unlined trenches with minimal cover. The study determined that at least some of the wastes required additional protection including deeper disposal and specific stabilization. Due to cost considerations, rather than place a blanket requirement for cover depth , lower activity wastes were allowed to be disposed without stabilization under 2 meters of cover. Higher activity wastes were split into two categories based on activity content. All of this waste would be stabilized with an additional requirement for 5 meters of cover on the highest activity grouping. The unstable waste must be segregated from the stabilized higher activity wastes. This led to the 3 tier classification system (Class A, Class B, and Class C) identified in 10 CFR 61.55.

Since the initial development of 10 CFR 61, nuclear power plant waste generation has been significantly reduced due to proactive measures by the industry to better manage cost and minimize worker exposure. Along with improved management of routine, processing waste, improved performance also has been achieved with expendable activated components such as fuel shrouds, control rods, and instrument assemblies. Never-the-less these wastes and others resulting from component replacements continue to be problematic. Additionally, the extension of plant operating life will likely increase the radioactive content of future plant wastes that serve for the life of the plant such as discarded reactor internals. Currently, there are limited disposal and processing options for irradiated hardware that are classified as Class B and Class C. As a result, a large fraction of the nuclear power industry must store this waste on-site until a disposition pathway becomes available. Due to the radioactive content, unique configuration, and varying radiological risk of this waste stream, some nuclear plants are challenged with providing adequate storage capability. As the nuclear power industry prepares itself for extended operating life and plant power up-rates to meet increasing electricity needs, it is likely that more irradiated hardware will be generated.

The lack of a consistent and risk-informed classification system has resulted in adverse impacts to both the industries that utilize nuclear materials and to the public. The convenience of defining and utilizing classification limits based on three subcategories of low level radioactive waste have resulted in the inadvertent orphaning of parts of certain waste streams. For example, there is currently no mechanism for disposal or processing of Class B and Class C irradiated hardware for the majority of the nuclear industry. Additionally, there is no disposal pathway for wastes classified as Greater Than Class C (GTCC). As a result, these two waste streams must remain indefinitely at the site of generation. Institutional controls over these storage facilities to ensure proper isolation of the waste from the public will be required for the duration of time that these wastes remain a radiological hazard (up to 500 years). Cs-137 is the major source of health risk for the first 300 years after which, Ni-63 dominates the risk for the next 300-500 years.

However, the EPRI analysis has shown that these risks can be adequately managed in a near surface disposal facility that relies on appropriate engineered and natural barriers. Even some of today's LLW disposal sites utilize practices and technologies that provide adequate protection of these risks. Because 10 CFR 61 subdivides LLW into subcategories, it inadvertently permits state authorities to license disposal facilities for subcategories of LLW regardless of consideration for whether the protection afforded by the actual facility design could control additional radiological hazards.

However, it is recognized that there are some benefits for having a generic, concentration based classification criteria that envelop the conditions that may be encountered at various disposal sites. If the proposed disposal sites followed criteria as presented in 10 CFR 61.55, it could be demonstrated that the protection levels sought in the regulations would be met through site selection. The classification system would assure safe operation by stabilizing and isolating the highest activity wastes. Furthermore, a standardized system of classification would reduce confusion both for the generator as well as for the disposal site operator. The generator would not have to be concerned with varying disposal requirements and the disposal site operator would be able to automatically determine how to disposition each package.

While the classification system may offer some convenience, as mentioned above, it is not generally recognized as "risk informed". To be risk informed it would have to be subjected to some type of formal risk assessment.

The phrase "risk-informed decision-making" has been defined for NRC (in Reference 1.1) as follows:

> *A "risk-informed" approach to regulatory decision-making represents a philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to health and safety.* (5)

This description is broad and does not add clarity to how a risk-informed approach would apply to the issue of waste classification Most of the radionuclide concentration limits are based on exposures to a hypothetical, subsistence intruder (Resident-Farmer) who excavates into the site and exposes the waste in order to construct a foundation for a house. The intruder remains on site to build a house, spreads the excavated material around the area of residence and maintains gardens and livestock for food and milk consumption on the site. The intruder analysis is a deterministic analysis that evaluates the risk as a true occurrence. It also assumes that exposure to a human will happen as described in the scenario. The intruder scenario is also completely dependent on the hypothetical disposal model previously described where un-stabilized waste is buried below a thin 2 meter cover. The exposure and subsequent dose to the intruder was then used to develop the generic concentration limits. The convenience of developing a set of concentration limits based on the most restrictive disposal site conditions found from each region in the U.S. and based on the most restrictive exposure scenario precludes the use of a risk-based analysis that focuses on site specific conditions and end-land-use scenarios. In order to risk-inform 10 CFR 61, efforts to understand exposure and dose to the various populations of concern require re-focusing the analysis on specific site parameters, specific disposal configurations, and specific end-land-use scenarios.

A "risk-informed" approach would lead to a site-specific evaluation and the setting of site specific disposal conditions independently. Such an evaluation could be conducted in accordance with the regulation following paragraph 10 CFR 61.58. The paragraph is reproduced here for convenience.

> *§ 61.58 Alternative requirements for waste classification and characteristics.*
>
> *The Commission may, upon request or on its own initiative, authorize other provisions for the classification and characteristics of waste on a specific basis, if, after evaluation, of the specific characteristics of the waste, disposal site, and method of disposal, it finds reasonable assurance of compliance with the performance objectives in subpart C of this part.*

This provision would allow an applicant to redefine concentration limits corresponding to a specific site and disposal configuration as long as the general performance requirements of 10 CFR 61 are met (i.e. 25 mrem/year whole body, 75 mrem/year to thyroid, and 25 mrem/year to any other organ).

While the 10 CFR 61.58 provision stands out in the regulation, it has not been used to develop an alternative disposal criteria for any commercial disposal site development. Classification in accordance with 10 CFR 61.55 is required for transfer of radioactive wastes per Appendix G of 10CFR20. (6) Agreement States potentially licensing a disposal site must treat compliance with 10 CFR 61.55 as a "matter of compatibility"[1]. (7). Notwithstanding these impediments, the NRC has, up to the time of this writing, eschewed directly taking on a revision of 10 CFR 61. It believes that clearer guidance toward implementing an application in accordance with 10 CFR 61.58 would facilitate new site development.

An additional issue relating to the radionuclide concentration limits in 10 CFR 61.55 is the supporting analyses conducted to define the limits relied upon in ICRP publications of radiation risk effects that have been superseded by later publications. This particularly affects some beta emitting radionuclides that figure prominently in LLRW disposal classification determinations. Section 2 of the report examines relative risks associated with the radionuclides limited by 10 CFR 61 and specifically addresses the dose conversion factors associated with Sr-90 and Ni-63, the two leading radionuclides driving classification for filters and irradiated hardware.

As part of a longer term project to investigate and develop guidance related to alternative, waste classification models, this report investigates additional related issues. These include:

- Redefinition of "What is LLRW?" in the context of utility generated radioactive waste and its relation to 10 CFR 61.
- Investigate and clarify the basic protection requirements for LLRW

[1] SA-200, "Compatibility Categories and Health and Safety Identification for NRC Regulations and Other Program Elements", US Nuclear Regulatory Commission, Office of State Programs. "Matter of Compatibility" is a formal designation in the Agreement State requirements which refers to items that are generally not flexible for interpretation. That is, the State would have to obtain NRC approval of any alternative approach would meet the objective of the requirement.

- Investigate current and proposed disposal practices of US disposal sites including plans relating to undeveloped compact Sites
- Investigate LLRW categories and guidance Provided by the International Atomic Energy Agency (IAEA)
- Investigate NRC guidance on alternate protection requirements

This Report

Section 2 of this report addresses the basic protection objectives of 10 CFR 61. It addresses the definition of LLRW. Also included in the sections is discussion of specific radionuclides limited by 10 CFR 61, including their properties and behaviors, relative risks and relevance to the disposal safety.

Section 3 of the report further delves into alternative classification systems including that defined by the International Atomic Energy Agency and some European countries. A contrast is drawn between those systems and that of 10 CFR 61.

Section 4 of the report addresses the use of engineered barriers in disposal site designs. Current practices of operating disposal sites are examined along with various facility concepts advanced in support of various compact disposal sites. NRC regulations do not make allowance for more aggressively engineered facilities even though nearly all proposed facilities followed models that eclipsed expectations inferred by 10 CFR 61.

Section 5 of the report provides additional analysis of the specific importance of radionuclides listed in 10CFR61. Each radionuclide is examined individually to discuss its production, the relative source term, measurement and general state of knowledge about the radionuclides.

Section 6 of the report provides a consolidated reference list keyed to reference numbers used throughout the report.

Section 7 of the report provides a list of acronyms used in the report.

Appendixes A and B provide a detailed line by line comparisons between between the original intruder dose rate calculation in the 10 CFR 61 Environmental Impact Statement and the updated IMPACTS methodology report for Ni-63 and Sr-90. The calculations were performed using the equations and parameters from the two reports.

Appendix C provides for easy reference a reproduction of the performance assessment summary from the WCS license application report. The summary indicates the types of scenarios considered and demonstrates the generally negligible public exposures from the proposed operation.

Conclusions

Low level radioactive waste is generally regarded as waste whose radioactivity is dominated by short half-lived activity. Such waste would be acceptable for general release following a finite period of isolation and would not require permanent isolation.

The concentration limits tabulated in 10 CFR 61.55 are out of date with respect to current ICRP dose conversion factors and do not reflect any protection factor afforded by currently implemented disposal practices at operating and planned disposal sites. Changes made in the dose conversion factors for Ni-63 and Sr-90 if reflected in the current regulation would result in Class A limits a factor 16 and 6 higher, respectively. The impact of these radionuclides would be evaluated even lower if the concentration limits were set on the same basis as that of Cs-137. The system of classification imposed by the regulation artificially splits the stream of low-level radioactive waste which can be disposed in a common venue.

Engineered barriers and manmade structures do not contribute and are not intended to contribute significantly to public protection beyond 500 years after site closure. By definition, waste suitable for near surface burial does not result in significant risk to the general public following its isolation period.

Experience with disposal site development since the publication of 10 CFR 61 provides engineered provisions far in excess of that required by the regulation. Applying these designs and operating practices, the sites can readily accommodate radioactive wastes generated through nuclear power plant operation without the need for differentiation by 10 CFR 61 Classification.

Very long lived radionuclides including those listed in Table 1 of 10CFR 61.55 generally do not contribute significantly to activity inventories in low-level radwaste disposal sites and should not be a controlling factor after a reasonable isolation period.

Future Research

This work was undertaken as part of a two year program to provide industry input and perspective in support of NRCs efforts to revisit current 10CFR61 disposal criteria. The initial NRC plan was to develop guidance for the developing alternative disposal criteria as allowed by §61.58 of the regulation. This year's work explored issues surrounding disposal criteria including how disposal criteria are developed or that reflect other approaches already defined. Future tasks include the development of a model approach for the disposal of LLW accounting for the technologies that are currently being used and accounting for the current understanding of factors effecting public exposures and that employs a risk informed basis. This approach will be provided in an EPRI report that will be release to the NRC for their consideration and use in the formulation of regulatory guidance and changes that may evolve pursuant to their internal reassessment. Effectively, the NRC has given the industry a unique opportunity to participate in the regulatory process.

2
BASIC PROTECTION REQUIREMENTS FOR LLRW

What is Low Level Radioactive Waste?

In order to design basic protection requirements for a low-level radioactive waste (LLRW) disposal facility, a complete understanding of the constituents of this waste is required. Low-level radioactive waste (LLRW) for lack of a better definition is legally defined as waste that is not "high level radioactive waste, spent nuclear fuel, or byproduct material." (3) The lack of a firm understanding of the constituents of LLRW makes it difficult to define protection requirements that would encompass a full range of known and predicted risks. To adequately incorporate risk-informed concepts into 10 CFR 61, a firmer definition of LLRW, based on the actual risk it poses and not by its generation source, will need to be developed.

Since LLRW is defined by exclusion, a starting point, then, would be to identify what is not LLRW. High Level Waste (which is not LLRW) is waste which requires "permanent" isolation such as that provided in geologic structures (9). It can therefore be assumed that LLRW is waste that does not require permanent isolation and would render itself harmless within some defined period of time (in the context of other environmental risks). The Low-Level Waste Policy Act (LLWPA) and 10 CFR 61 were constructed around the premise that sufficient isolation of this waste was achieved with near surface burial. The main purpose of the LLWPA was to delegate responsibility for LLRW disposal under the assumption that it was acceptable for near surface disposal. The LLWPA left the NRC the responsibility to classify which waste is LLRW. (8).

Table 2-1 lists what is commonly considered to be LLRW and what is commonly considered to be not LLRW. It should be noted that some of the "not" LLRW may also be acceptable for near surface disposal subject to a specific determination of the suitability and the conditions of disposal. Since these are not LLRW in the context of the LLWPA, acceptance for disposal would be determined by the relevant disposal authority.

Table 2-1
LLRW and Not LLRW

Not Low-Level Radioactive Waste	Low-Level Radioactive Wastes
Spent Nuclear Fuel	Nuclear Power Plant Generated Wastes (excluding Nuclear Fuel)
Wastes resulting from the reprocessing of spent nuclear fuel	Process Wastes (Resins, Filter materials, DAW)
Byproduct Material	Expendable hardware
Mill Tailings 11e(2)[2]	Decommissioning Wastes including most activated hardware
Waste Incidental to Fuel Manufacturing 11e(2)	Government
Discrete source of Ra-226	Dry Solids
Accelerator Wastes	Trash
Other Sources developed from NARM	Absorbed liquids
Transuranic (TRU) Wastes – Wastes Containing Concentrations of TRU not exceeding 10CFR61 Disposal Limits	Biological
Greater than Class C Wastes	Solidified chelates
Chemically Hazardous LLRW Wastes (Mixed Wastes)	Sealed Source
NORM, Naturally Occurring Radioactive Material	Industrial Generated Wastes
Some Sealed Sources	Miscellaneous solids and absorbed liquids
	Solidified oils
	Resins and filter wastes
	Biological wastes
	Discarded manufactures products
	Hospitals and Medical Facilities
	Laboratory Wastes
	Biological
	Academic
	Laboratory Wastes
	Dry solids
	Biological

[2] Public Law 83- , The Atomic Energy Act of 1954, Mill tailings and waste incidental to extraction and concentration for source material content would be defined as byproduct material. (37)

Medical applications represent a large segment of the non-utility generated wastes. These types can be generated within any of the non-utility groups, since both the government and universities maintain hospitals and medical research programs. Furthermore, a segment of industrial generation supports the development of radio-pharmaceuticals, x-ray equipment along with other products applying radioisotopes. With the exception of sealed sources used for some cancer treatments and radiographic imaging, the preponderance of medical related radioactive waste contains only short half-life radionuclides.

Breakdowns of activity in non-utility generated wastes are available through the Manifest Information Management System (MIMS) database (10). The activity cut-off for reporting to MIMS is 0.01 mCi. This is generally higher than that needed to actually track long-lived radionuclides which are usually restricted to very low concentrations. Medical applications tend to focus on a few isotopes used for cancer treatment, as diagnostic tracers, and as medical sources. Sources are often restricted from LLRW disposal or are subject to special rules.

Mixed Waste can be disposed as LLRW waste if it is processed to eliminate organic hazards. Contaminated heavy metals including lead are sometimes allowed, subject to special provisions.

The Current U.S. Waste Classification System

A cornerstone of the 10 CFR 61 has been the system of waste classification defined in the regulation. The regulation divides the LLRW stream into three Classes; A, B, and C, on the basis of the radioactivity content of a particular set of intermediate and long lived radionuclides. Disposal requirements are progressively increased with waste class. Class A, with the lowest radioactivity, has minimum disposal constraints; Class A waste need not be stabilized and may be directly buried with 2 meters of cover. Class B which is triggered by higher activity levels of radionuclides with half-lives up to 100 years requires stabilization of the waste form and 2 meters of cover. Class C is triggered by higher activity levels of the longer lived radionuclides and requires longer term isolation which is provided by stabilization and additional cover thickness of 5 meters on the waste emplacement. (4) Tables 1 and 2 of 10 CFR 61.55 are reproduced below as Table 2-2 and Table 2-3, respectively. Disposal limits for the longest lived radionuclides are listed in Table 2-2. Shorter lived radionuclides are listed in Table 2-3. Classification is determined on the basis of a sum of fractions calculation comparing activity content of the waste with the concentration limits. The calculations for each table are done independently since the hazards imposed by the radionuclides in the two tables are not concurrent. The most restrictive determination controls classification. (2)

Table 2-2
10 CFR 61 Table 1 Concentration Limits for Very Long-Lived Radionuclides in Class C Waste

Radionuclide	Concentration curies per cubic meter**
C-14	8
C-14 in activated metal	80
Ni-59 in activated metal	220
Nb-94 in activated metal	0.2
Tc-99	3
I-129	0.08
Alpha emitting transuranic nuclides with half-life greater than 5 years	100*
Pu-241	3,500*
Cm-242	20,000*

*Units are nanocuries per gram.
**Class A limits are 1/10 the Class C limit. Class B limits are not applicable

Table 2-3
10 CFR 61 Table 2 Concentration Limits for Radionuclides with Intermediate Half-Lives

Radionuclide	Concentration, curies per cubic meter		
	Class A Limit	Class B Limit	Class C Limit
Total of all nuclides with less than 5 year half-life	700	(*)	(*)
H-3	40	(*)	(*)
Co-60	700	(*)	(*)
Ni-63	3.5	70	700
Ni-63 in activated metal	35	700	7000
Sr-90	0.04	150	7000
Cs-137	1	44	4600

*There are no limits established for these radionuclides in Class B or C wastes. Practical considerations such as the effects of external radiation and internal heat generation on transportation, handling, and disposal will limit the concentrations for these wastes. These wastes shall be Class B unless the concentrations of other nuclides in Table 2 determine the waste to the Class C independent of these nuclides.

Nuclides selected for specific disposal limits were those with sufficiently long half lives to persist in the disposal environment. The 10 CFR 61 Table 1 radionuclides, which included only Class A or Class C limits were those with very long half-lives for which stabilization would provide no benefit. Recognizing that at some point in time the nuclides would find their way into the general environment, the Class A limits were set low enough to assure that the risk posed by unconstrained release would be acceptable. The Class C limits were set a factor of 10 higher to account for the impeded access to the material associated with the deeper disposal. The factor of 10 increase in the limit could also be at least partially justified by the dilution that would be associated with excavation through the deeper cover and reduced emplacement efficiency associated with the containers. (4)

As an additional concession in the original framing of the criteria, Cs-137 concentration limits were set on the basis that the waste would be diluted by a factor of 20 across the segregated (lower activity) trench. While this may be realistic in context of the actual radioactivity content, it fails to recognize that Cs-137 dominates risk directly following the end of the institutional control period when intruder risks are most applicable. Additionally, restriction limits created for other radionuclides did not account for a similar, realistic dilution factor resulting in an overstatement of their risk relative to cesium 137. Therefore, the failure to apply a consistent level of protection that accounts for a more realistic dilution factor, raise doubts on whether the protection criteria related to the listed radionuclides are commensurate with their level of risk. It is believed that this dilution credit was a response to disposal site practice and concerns expressed at that time about the economic impact of complying with a more restrictive rule. A

similar dilution credit was not taken in setting the Class A limits for other radionuclides such as Ni-63 and Sr-90. The effect is to increase the relative importance of these nuclides without a commensurate impact on the relative risk.

How Does Utility LLRW Conform to This Definition?

Utility LLRW is the dominant source of radioactivity in LLRW accounting for almost 90% of the overall activity, when considered in the context of all sources. Radioactivity in utility LLRW is dominantly short half lived activity with most of it disappearing after a few hundred years. Based upon the data collected by EPRI for the 2003-2006 time period, waste generation by volume from the current fleet of nuclear power plants accounts for about 26,000 m^3 (923,000 ft^3) per year. (11) Average annual generation by plant type, including key nuclide activities, are summarized in Table 2-4. The values in Table 2-4 represent stream wide averages as shipped. These were determined from shipping records provided by approximately 65% of EPRI member utility plants. The records were drawn from the four year period from 2003 through 2006. The values do not include activated metal waste. These will be examined separately. No effort was made to segregate the material by 10CFR61 class.

Table 2-4
Annual Generation Rates of Radionuclides Important to Classification per 1000 MWe (Values Exclude Activated Hardware)

	BWR	PWR	Units
Volume	17264	8603	ft^3
H-3	5.9E+05	1.9E+05	mCi
C-14	7.0E+02	1.0E+03	mCi
Co-60	1.3E+05	1.6E+04	mCi
Ni-63	6.2E+03	5.5E+04	mCi
SR-90	2.3E+02	1.4E+02	mCi
Cs-137	1.7E+04	2.9E+04	mCi
TRU*	1.6E+01	3.8E+00	mCi

*Excludes Pu-241 and Cm-242

Wastes included in Table 2-4 generally contain a mixture of contaminant radionuclides including fission products, activation products, and TRU. Not included in Table 2-4 are Technetium 99 (Tc-99) and Iodine 129 (I-129). These generally appear at such low levels that they are not reliably measured using radiochemical procedures. There is no evidence that any radiochemistry laboratory has reported anything other than a detection limit for I-129. In the case of Tc-99, while it is sometimes reported as a statistically significant value in the radiochemistry reports, it is also clear that unrealistically high values have been included in the database. These high values significantly bias any determination of risk derived from the database results. These radionuclides are discussed specifically in relation to their specific risks. It has been demonstrated that there cannot be sufficient release of these isotopes in solid waste streams to

figure prominently in disposal risk. Investigations sponsored in the past by EPRI included mass spectroscopy measurements of I-129, Tc-99 and C-14 in samples drawn from test columns installed at six PWR and four BWR power plants. (12) Scaling factor ratios with Co-60 and Cs-137 derived from these measurements were compared with industry averages reported from routine sampling and radiochemical analysis. These comparisons indicated that estimated scaling ratios based on radiochemistry results for these radionuclides were 3 to 5 orders of magnitude higher than those determined through the controlled experiments. Even using higher values, however, Tc-99 and I-129 still do not figure strongly in the risk assessment of utility generated wastes and their average concentrations remain well below the Class A limits. C-14 values, while exhibiting significant discrepancies between the Battelle measurements and industry reported averages, were not consistently lower in the case of PWR plants.

Key Point: I-129, and Tc-99 do not represent a strong source of disposal risk because there are not sufficient quantities of these radionuclides in the solid waste streams to figure prominently in the risk assessment for a disposal site.

Table 2-4 activities were decayed for varying time periods out to 1000 years, which would be the likely maximum time period that any benefit could be derived from near surface disposal. Concentrations were calculated from the decayed values using the volumes provided.

Figure 2-1 provides the results of this calculation as average concentration versus years following disposal. It is observed at the concentrations in this waste that Ni-63 and Cs-137 dominate activities out to about 300 years. After that time period, C-14 becomes dominant. It is also noted that C-14 concentration is practically unchanged from the day of disposal out to 1000 years. (Three hundred years is marked with a vertical line on the graph for convenience.)

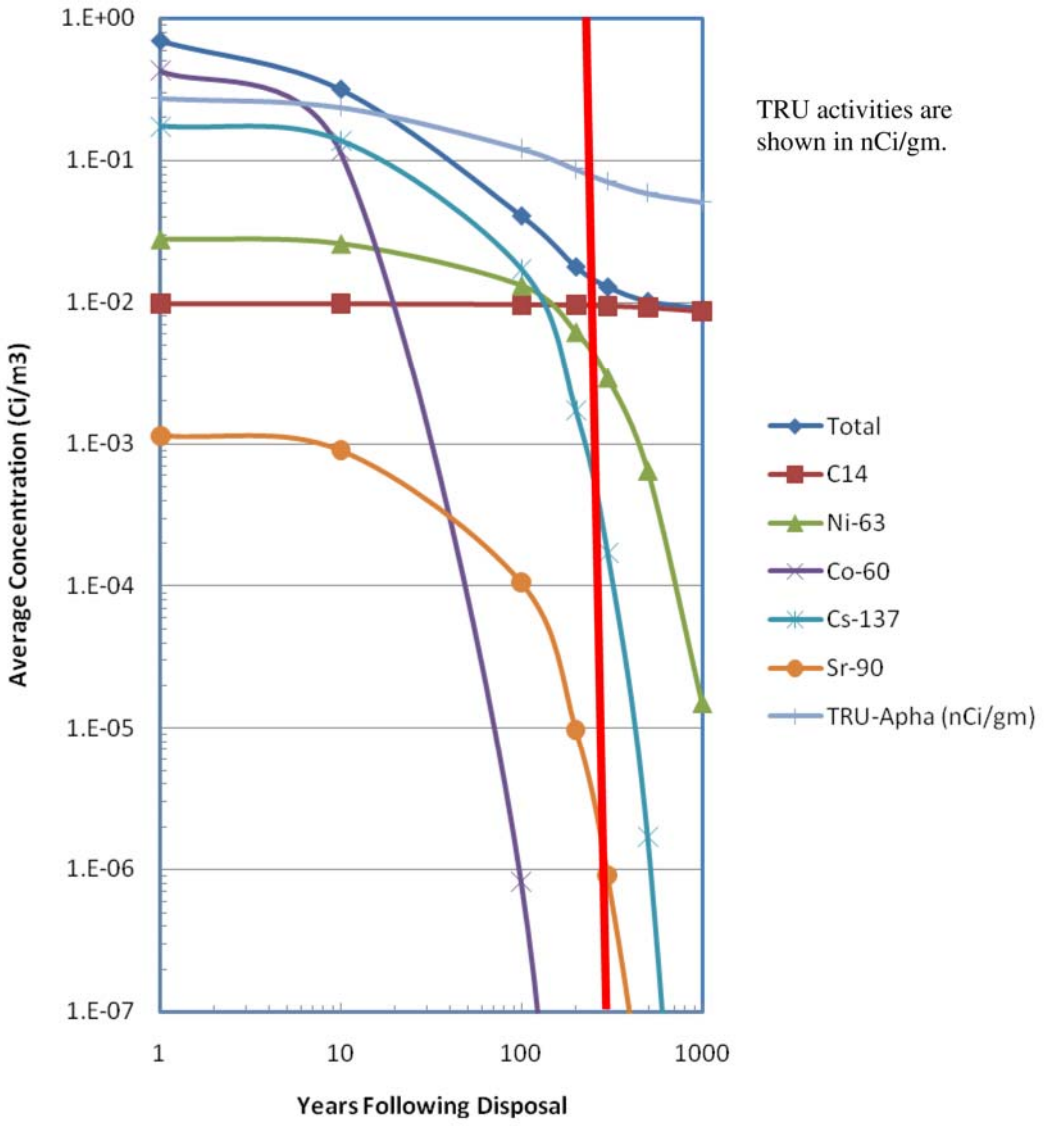

Figure 2-1
Dominant Long Lived Activity Through Intermediate Time Spans

Just viewing the concentrations versus time provides an incomplete picture. The 10 CFR 61 concentration limits were derived to correspond to a common level of risk (i.e. that concentration that would result in 500 mrem/year to an inadvertent intruder). This assumes that the Class A concentration limit for each of the listed isotope would result in a 500 mR/yr dose rate to an intruder if that isotope was the only isotope present. The actual activity divided by the concentration limit would represent the fraction of the 500 mR/yr as a measure of relative risk. The concentration values calculated for Figure 2-1 are divided by the Part 61 Class A limiting concentration and plotted on Figure 2-2. Note that in these waste streams, there are no other radionuclides that figure prominently in the classification calculation. For the first 300 years disposal risk is driven almost entirely by Cs-137 and Ni-63. After 500 years, only C-14 and TRU continue to stand as dominant risk contributors. Even then, both C-14 and TRU in this mixture are always less than about 10% of the Class A limits. We should remind ourselves that

we are evaluating risk on the basis of an intruder scenario which is primarily a short term exposure issue and both C14 and TRU should be evaluated in the context of long term risks. Furthermore, in the evaluation of long-term risk, the concentration in each waste stream at the time of its disposal has little bearing. The overall inventory in the disposal site at the time of closure forms the basis of the risk evaluation since transport within the disposal site must be effectively complete (i.e. completely mixed) before transport off the disposal site would be at its maximum level. Once we move past the time frames associated with the intruder scenarios (up to 300 years for acute exposures to concentrated activity), the limiting condition for assessment of offsite risk occurs when the activity in the disposal site is homogeneously mixed. Figure 2-2 provides a representation of the same radionuclides shown in Figure 2-1. In this case, the concentrations are divided by their corresponding 10 CFR 61 Class A limit to provide a measure of relative risk. At 300 years, average Cs-137 concentrations are less than 1% of the Class A limits. Ni-63 risks are comparable to those of TRU and C-14 but are no longer significant at 500 years. It should also be noted that the 10 CFR 61 Class A limit overstates the risk of Ni-63. This will be discussed in more detail in the Section 5.

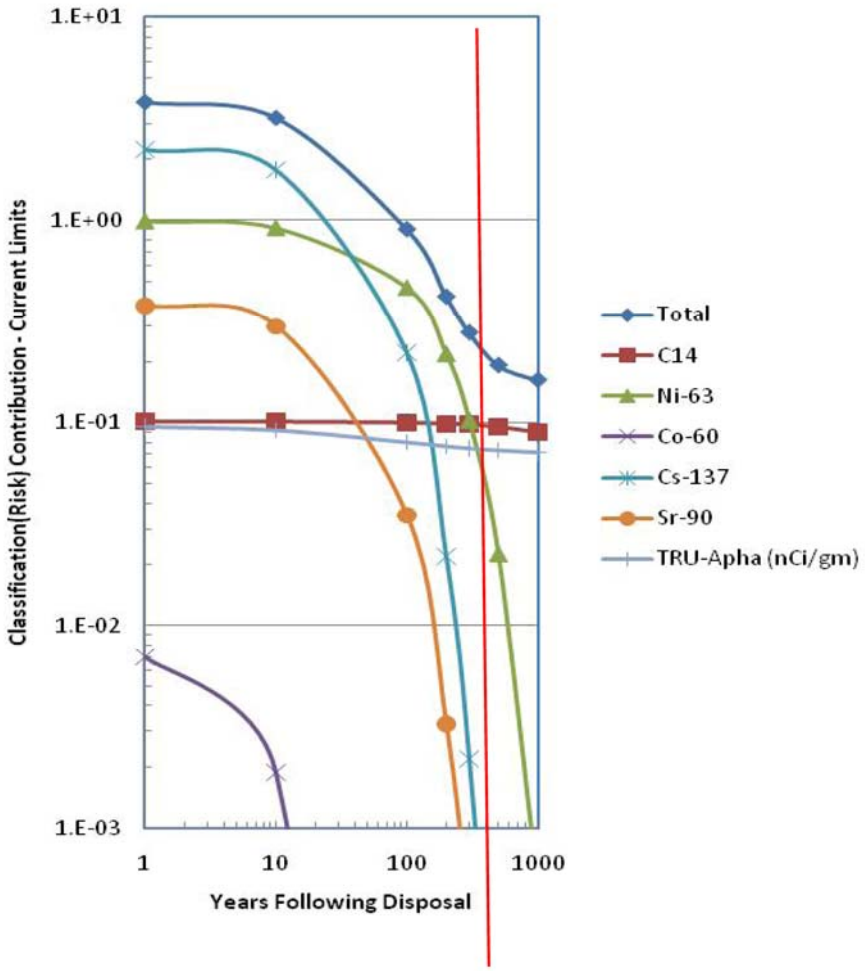

Figure 2-2
Relative Risk Through Intermediate Time Spans (Concentration / Class A Limit)

Key Conclusions

One conclusion that can be drawn from this discussion of the radionuclides included in Tables 1 and 2 of §61.55 are that in the short term, Cs-137 dominates the risk for the first 100 years. If Ni-63 was evaluated on the basis as Cs-137 it would not be a significant contributor to risk at any time in the disposal life cycle. This would also be true for Sr-90, though its contribution to risk is small to begin with on the basis of current source term analysis. Therefore, alternative disposal limits should be evaluated for both Ni-63 and Sr-90.

Table 1 radionuclides in streams other than in activated metal are not released in sufficient quantity to show up as important risks during the period of viability of the disposal site. Even after the disposal site is assumed to be uncontrolled and the engineered barriers are not effective, the overall contribution of Tc-99 and I-129 are negligible. After 500 years, C-14 along with transuranics in source term make the most significant contributions to public health risks but these are well within the 10CFR61 and 10CFR20 limits. Section 5 of this report examines risks relating to specific radionuclides. As will be seen, C-14 activities may be over reported by an order of magnitude.

3
RADIOACTIVE WASTE CLASSIFICATION SCHEMES—GUIDES AND DEFINITIONS FROM THE INTERNATIONAL ATOMIC ENERGY AGENCY AND NRC

The studies and reports sponsored by the NRC in support of the promulgation of 10 CFR 61 provided an international benchmark for investigation and planning for the disposal of low level radioactive waste. This action was completed nearly a decade before there was wide spread international attention to the issue. While there are substantial variations in approaches taken by various countries for the disposal of LLRW, there is also a clear linkage to the landmark research and development of the U.S. regulators.

History of IAEA Guidance

The classification systems for radioactive waste in the US and the International Atomic Energy Agency were developed for similar purposes but under very different conditions.

NRC

The U.S. Nuclear Regulatory Commission (NRC) derives its authority to license commercial radioactive waste disposal from the Atomic Energy Act of 1954. (37) This Act governs the production and use of source, special nuclear material and by-product material for defense and peaceful purposes. The Energy Reorganization Act of 1974 created the current NRC as a regulatory-only authority and gave the Department of Energy (DOE) responsibility for the development and production of nuclear weapons, commercial nuclear power, and other energy-related work. (38) DOE took over authority for the regulation of defense nuclear facilities. Since the NRC's authority to license comes from the Atomic Energy Act, it does not include Naturally Occurring or Accelerator Produced Radioactive Material (NARM) which is the responsibility of the individual States or the DOE if the NARM is produced in support of defense activities (37) (38).

The Nuclear Waste Policy Act of 1982 defined the relationships of the federal authorities with respect to the disposal of HLW and directed the development of Yucca Mountain as the national repository for HLW. This Act also established financial surety arrangements for the closure of LLW disposal facilities. (9).

In 1980, Congress passed the Low Level Radioactive Waste Policy Act which established responsibility for disposal of Low Level Radioactive Waste (LLRW) from commercial sources with the States. NRC retained the authority to license this process although authority for licensing and regulation could be passed to the States under provisions of the Atomic Energy Act. (3) (37). The States made little progress in actually developing new LLW disposal sites so in 1985, Congress passed the Low Level Radioactive Waste Policy Amendments Act which established the Compact process for siting new disposal facilities for LLW. (39).

The 1985 Amendments Act also directed the NRC to establish rules for excluding waste with low concentrations of activity from regulatory control. The Below Regulatory Concern (BRC) policy directive was rescinded by Congress in the Energy Policy Act of 1992. (40).

IAEA

The International Atomic Energy Agency (IAEA) was established in 1957 as part of the United Nations "Atoms for Peace" program. The IAEA has no regulatory authority in any nation other than as granted by its Members. The IAEA's three main areas of work are Safeguards and Verification, Safety and Security and Science and Technology. IAEA promotes safe and secure uses of nuclear energy by establishing guidelines using the best available research and technology. (41)

Definition of LLW

The differing authorities for NRC and IAEA have led to the development of different definitions for radioactive waste in general and LLW in particular. This creates difficulties when attempting to make comparisons between activity concentrations used to establish boundaries or limits for waste classifications.

NRC

The definition of LLRW under NRC regulations is exclusionary; as discussed earlier in this report, LLW is defined by what it is not. Therefore the definition of LLW is dependent on a series of other definitions that must be evaluated and excluded.

10 CFR 61.2 defines (LL) Waste as "…those low-level radioactive wastes containing source, special nuclear, or byproduct material that are acceptable for disposal in a land disposal facility. For the purposes of this definition, low-level radioactive waste means radioactive waste not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or byproduct material as defined in paragraphs (2), (3), and (4) of the definition of *Byproduct material* set forth in 10 CFR 20.1003 of this chapter." (2). In addition, due to the source from which NRC derives its regulatory authority which limits its range of jurisdiction, LLRW can only be waste meeting the above definition that is derived from the nuclear fuel cycle and its licensees.

Therefore LLRW is any waste from the nuclear fuel cycle that contains radioactive material and is not any of the above. The definition inherently does not contain a lower limit and it can therefore be construed that any amount of radioactive material, even at levels that could be insignificant from a radiation dose or risk perspective or be at levels found in natural background would still meet this definition and be subject to regulation. Some LLRW may also contain nuclides and activities at or above the levels found in some High Level Wastes (HLW) but are not considered HLW due to the production mechanism. (5)

IAEA

IAEA recommends a classification system that identifies LLRW as "Waste, which because of its low radionuclide content does not require shielding during normal handling and transportation**".** Within the ILW and LLW classification, IAEA also differentiates between short and long-lived waste and alpha bearing waste. There is a recognition that dose rates alone are not the sole criteria for determination of waste classification. (42) IAEA does not distinguish the source of radioactive waste by production mechanism or regulatory authority, merely by radionuclide content. IAEA recommendations also include an exemption level for radioactive waste. Waste containing radionuclides below a certain level are 'excluded' from treatment as radioactive waste and are free to be released into the processes normally used for the disposal of any other type of waste. (42) (43). LLW under IAEA therefore is waste that meets a specific set of criteria based on activity, hazard or any other criteria established by the regulatory authority of the generator.

"Low-level waste (LLW): waste that contains material with radionuclide content above clearance levels, but with limited amounts of long-lived activity. It requires robust isolation and containing for periods out to a few hundred years [typically 300]. It includes a very broad band of materials that includes very high activity waste with short half life that requires shielding and some long lived material at low activity levels. Such waste would require up to around 300 years of control but would not be hazardous [to an intruder] beyond that period of time. The radionuclides with the waste will decay to activity levels that are acceptably low from a radiological safety view point, within a time period during which institutional controls can be relied upon. Such waste can be determined by radiological performance assessment of the disposal system chosen." (44)

Classifications of Radioactive Waste

It is necessary to put any discussion of the differences between NRC and IAEA LLW classifications into perspective with regard to all radioactive waste in order to fully understand the differences. This section contains a general description of the definitions of radioactive waste classes under NRC and IAEA programs.

NRC

NRC waste classifications are derived from the authorizing legislation applicable to each process or type of waste. As such, the classifications leave open categories or processes that are not under the authority of NRC and so are not included even though they may have similar physical, chemical and radiological characteristics to wastes that are defined and regulated by NRC.

High Level Waste

High Level Radioactive Waste (HLW) is defined in the Nuclear Waste Policy Act of 1982 (as amended) as follows:

> "The term "high-level radioactive waste" means–(A) the highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and (B) other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation" (9).

NRC defines HLW as "(1) The highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; (2) Irradiated reactor fuel; and (3) Other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation" (5) (2).

There are no specific activity levels or concentration limits that define HLW."

Transuranic Waste

Transuranic wastes are wastes containing more than 100 nCi/g (4kBq/g) of alpha-emitting, transuranic[3] radionuclides with half-lives greater than 20 years excluding high-level waste.

Uranium Mill Tailings

Mill tailings are the residues from the mining and chemical processing of ores to obtain source material (uranium or thorium). (5) (45).

NORM/NARM

Naturally Occurring and Accelerator-produced Radioactive Material waste is radioactive waste produced by concentrating naturally occurring radionuclides through non-fuel cycle processes (e.g. the oil & gas industry) or are produced in accelerators (5).

Low Level Waste

NRC defines sub-classes of LLW in 10CFR61. The sub-classes are based on evaluations of disposal site performance objectives and designate concentrations at which the waste form must provide additional stability. Classes are defined in 10CFR61.55 by the concentration of specific long-lived (Table 1) and short-lived radionuclides (Table 2) as shown in Tables 2.2 and 2.3, respectively. (2)Classification is determined using a sum of fractions method applied independently to each of the tables.

[3] Nuclides whose atomic number is greater than that of uranium (92).

Class A

Class A waste contains the lowest concentration of radionuclides and requires minimal additional stability beyond the waste itself as packaged for disposal. It is to be segregated from other wastes unless stabilized to the same criteria. It requires 2 meters of cover.

Class B

Class B waste contains higher concentrations of radionuclides and must meet additional stability requirements. It also requires 2 meters of cover.

Class C

Class C waste contains the highest concentrations of radionuclides and must meet all of the previous criteria plus have additional measures implemented at the disposal facility to protect against inadvertent intrusion. This waste requires stabilization and 5 meters of cover.

Greater Than Class C (GTCC)

GTCC waste is waste that contains radionuclides in concentrations that are above the Class C limits and may be considered not suitable for disposal in a shallow-land facility.

IAEA

IAEA waste classifications are entirely independent of the path through which the waste was created and are meant to be all-encompassing. There are no categories defined by exclusion nor is there any waste that is determined to be not suitable for disposal. IAEA does not propose specific activity concentrations for any category. These are left to the discretion of the competent regulating authority. (42). IAEA's waste classification system is based on both increasing activity concentrations and increasing longevity of the radioisotopes. It is meant to identify situations were increased levels of control or isolation are needed to achieve performance objectives due to concentration, longevity or some combination.

IAEA Waste Classification Scheme

- Highest Activity
  - High level Waste (HLW)
- Low and Intermediate Activity
  - Long-lived waste(LILW-LL)

    Half-life >30 years

    400-4000 Bq/g long lived alpha emitters
  - Short-lived waste (LILW-SL)
- Exempt waste (EW)

Exempt Waste (Clearance)

IAEA considers waste that contains radioactive material in quantities that are so small that they pose an insignificant risk when released to the environment may be exempted from disposal as LLW and from further regulatory control. The activity concentration considered to be an insignificant risk is based studies to determine clearance levels and limiting annual doses to the public to less than 0.01 mSv/year (1 millirem/year). Activity concentrations are nuclide-specific but range from 0.1 Bq/g to 10^4 Bq/g (2.7 pCi/g to $2.7x10^5$ pCi/g). (42) (43).

Very Low Level Waste

Waste with activity concentrations that are not significantly above clearance levels (<100X the clearance level), may be considered for exemption or exclusion from disposal as LLW depending on the nature of the waste and the intended pathway for release. (43) (44). Some of these may be considered for disposal in ordinary landfills.

Low / Intermediate Level Waste

IAEA defines Low / Intermediate Level waste as waste containing radioactivity concentrations above clearance levels and decay heat generation below 2 kW/m^3. Older definitions categorized LLW as waste that does not require shielding during normal handling and transportation. ILW would then be waste that required shielding but did not need additional provisions for the dissipation of decay-heat. The typical measurement used to divide the two classes was a contact dose rate of 2 mSv/hr (200 mR/hr). (42).

This definition has been superseded in more recent documents after studies evaluating the dose from radionuclides given various exposure pathways. The studies showed additional consideration should be given to the longevity of the waste (how long the waste will remain hazardous). Considerations of shielding remain the dividing line between LLW and ILW however, the specific value may be dependent on the waste form selected.

Short Lived

Short-lived Low/Intermediate Level waste is radioactive waste with low concentrations of long-lived radionuclides. Administrative controls can be effective in reducing hazards to the public such as maintenance of institutional controls on the disposal facility until sufficient decay of the shorter-lived nuclides has occurred. The longer-lived radionuclides that will not decay during the institutional control period are maintained at low concentrations consistent with the performance objectives of the disposal facility. Waste form and packaging may be important to the classification as there are many alternatives that will affect the ultimate performance of the waste. (42).

Long Lived

Long-lived Low / Intermediate Level waste is radioactive waste with concentrations of long-lived radionuclides such that a high degree of isolation is required in order to meet the performance objectives. Again, the specific concentration of long-lived radionuclides is dependent on the waste form and disposal facility characteristics. IAEA reports that many countries implementing this waste classification system have used a limit of 400 Bq/g (~10 nCi/g) for long-lived alpha emitters for this type of waste in surface disposal facilities. (42).

High Level Waste

High Level Waste contains large concentrations of both short and long-lived radionuclides and generates a significant quantity of heat from radioactive decay over several hundred years. Again, specific radioactivity concentrations limits are left to the discretion of the national authority. However, IAEA recognizes activity levels from 5E+04 TBq/m^3 to 5E+05 TBq/m^3 (1.35E+06 C/m^3 to 1.35E+07 C/m^3) with heat generation of 2 kW/m^3 to 20 kW/m^3 for periods of up to ten years after discharge as representative of HLW with the lower limits used as the approximate distinction between HLW and ILW. (42).

Low Level Waste Activity Concentration Limits

IAEA does not recommend specific radionuclide concentrations for waste classification. However, as part of guidance to develop activity limits for low level waste disposal in near-surface facilities, IAEA derived activity concentrations for theoretical facilities and disposal conditions for illustrative purposes. These values are shown in Table 3-1 and Table 3-2. (46)

Table 3-1
IAEA Illustrative Activity Concentrations for LLW (Trench Disposal in Sand Geo-sphere under Arid Conditions)

Concentrations for the Trench Disposal Facility with a Sand Geosphere Under Arid Conditions					
Radio-nuclide	**Operational Period**		**Post Closure Period**		**NRC Limit**
	Limiting Concentration (Bq/kg of Waste)	**Limiting Scenario**	**Limiting Concentration (Bq/kg of Waste)**	**Limiting Scenario**	**Class C Value from 10 CFR 61.55 (Bq/kg @ 1g/cc)**
H3	3.00E+06	Liquid release (public)	7.00E+04	Leaching scenario	
Be10	1.00E+07	Liquid release (public)	N/A	N/A	
C14	4.00E+06	Liquid release (public)	7.00E+04	Leaching scenario	2.96E+08 2.96E+09 (in activated metal)
Na22	1.00E+06	Liquid release (public)	N/A	N/A	
Ca41	3.00E+06	Liquid release (public)	1.00E+05	Leaching scenario	
Mn54	8.00E+06	Liquid release (public)	N/A	N/A	
Fe55	4.00E+07	Liquid release (public)	2.00E+19	On-site residence (soil)	
Ni59	3.00E+08	Liquid release (public)	1.00E+07	On-site residence (soil)	8.14E+09 (in activated metal)
Ni63	1.00E+08	Liquid release (public)	1.00E+07	On-site residence (soil)	2.59E+10
Co60	1.00E+06	Liquid release (public)	1.00E+09	On-site residence (soil)	
Zn65	7.00E+06	Liquid release (public)	N/A	N/A	
Sr90	2.00E+04	Liquid release (public)	1.00E+04	On-site residence (soil)	2.59E+11
Zr93	2.00E+07	Liquid release (public)	1.00E+05	Leaching scenario	
Nb94	2.00E+06	Liquid release (public)	3.00E+03	On-site residence (soil)	7.40E+06 (in activated metal)

Table 3-1 (continued)
IAEA Illustrative Activity Concentrations for LLW (Trench Disposal in Sand Geo-sphere under Arid Conditions)

Concentrations for the Trench Disposal Facility with a Sand Geosphere Under Arid Conditions					
Radio-nuclide	**Operational Period**		**Post Closure Period**		**NRC Limit**
	Limiting Concentration (Bq/kg of Waste)	**Limiting Scenario**	**Limiting Concentration (Bq/kg of Waste)**	**Limiting Scenario**	**Class C Value from 10 CFR 61.55 (Bq/kg @ 1g/cc)**
Tc99	8.00E+04	Liquid release (public)	1.00E+03	Leaching scenario	1.11E+08
Ru106	8.00E+05	Liquid release (public)	N/A	N/A	
Ag110m	2.00E+06	Direct irradiation (worker)	N/A	N/A	
Sn121m	1.00E+07	Liquid release (public)	N/A	N/A	
Sb125	3.00E+06	Liquid release (public)	N/A	N/A	
Sn126	7.00E+05	Liquid release (public)	N/A	N/A	
I129	9.00E+02	Liquid release (public)	2.00E+01	Leaching scenario	2.96E+06
Cs134	1.00E+06	Liquid release (public)	1.00E+18	On-site residence (soil)	
Cs137	1.00E+06	Liquid release (public)	8.00E+04	On-site residence (soil)	1.70E+11
Ce144	1.00E+07	Liquid release (public)	N/A	N/A	
Pm147	6.00E+07	Liquid release (public)	N/A	N/A	
Sm151	1.00E+08	Liquid release (public)	2.00E+08	On-site residence (soil)	
Eu152	4.00E+06	Direct irradiation (worker)	N/A	N/A	
Eu154	3.00E+06	Direct irradiation (worker)	N/A	N/A	

Table 3-1 (continued)
IAEA Illustrative Activity Concentrations for LLW (Trench Disposal in Sand Geo-sphere under Arid Conditions)

Concentrations for the Trench Disposal Facility with a Sand Geosphere Under Arid Conditions					
Radio-nuclide	**Operational Period**		**Post Closure Period**		**NRC Limit**
	Limiting Concentration (Bq/kg of Waste)	**Limiting Scenario**	**Limiting Concentration (Bq/kg of Waste)**	**Limiting Scenario**	**Class C Value from 10 CFR 61.55 (Bq/kg @ 1g/cc)**
Tl204	1.00E+07	Liquid release (public)	N/A	N/A	
Pb210	5.00E+03	Liquid release (public)	N/A	N/A	
Ra226	5.00E+03	Liquid release (public)	8.00E+02	On-site residence (soil)	
Ac227	2.00E+04	Liquid release (public)	N/A	N/A	
Ra228	4.00E+04	Liquid release (public)	1.00E+08	On-site residence (soil)	
Th232	3.00E+04	Liquid release (public)	6.00E+02	On-site residence (soil)	
U234	4.00E+04	Liquid release (public)	7.00E+02	Leaching scenario	
U235	4.00E+04	Liquid release (public)	7.00E+02	Leaching scenario	
U238	4.00E+04	Liquid release (public)	2.00E+03	Leaching scenario	
Np237	2.00E+03	Liquid release (public)	2.00E+04	On-site residence (soil)	
Pu238	1.00E+05	Liquid release (public)	2.00E+05	On-site residence (soil)	3.70E+03
Pu239	1.00E+05	Liquid release (public)	1.00E+04	Leaching scenario	3.70E+03
Pu240	1.00E+05	Liquid release (public)	7.00E+04	Leaching scenario	3.70E+03
Pu241	6.00E+06	Liquid release (public)	2.00E+06	On-site residence (soil)	1.30E+05
Am241	2.00E+05	Fire release (worker)	8.00E+04	On-site residence (soil)	3.70E+03

Table 3-2
IAEA Illustrative Activity Concentrations for LLW (Vault Disposal in Clay Geo-sphere under Temperate Conditions)

Concentrations for the Vault Disposal Facility with a Clay Geosphere Under Temperate Conditions					
Radio-nuclide	**Operational Period**		**Post Closure Period**		**NRC Limit**
	Limiting Concentration (Bq/kg of Waste)	**Limiting Scenario**	**Limiting Concentration (Bq/kg of Waste)**	**Limiting Scenario**	**Class C Value from 10 CFR 61.55 (Bq/kg @ 1g/cc)**
H3	1.00E+12	Gas release (public)	3.00E+06	Bathtubbing	
Be10	1.00E+20	Drop and crush	N/A	N/A	
C14	8.00E+09	Gas release (public)	1.00E+09	Road Construction	2.96E+08 2.96E+09 (in activated metal)
Na22	4.00E+11	Drop and crush (worker)	N/A	N/A	
Ca41	3.00E+14	Drop and crush (worker)	1.00E+09	Bathtubbing	
Mn54	1.00E+12	Drop and crush (worker)	N/A	N/A	
Fe55	1.00E+12	Drop and crush (worker)	2.00E+17	Bathtubbing	
Ni59	9.00E+13	Drop and crush (worker)	2.00E+09	Bathtubbing	8.14E+09 (in activated metal)
Ni63	1.00E+20	Drop and crush	2.00E+09	Bathtubbing	2.59E+10
Co60	4.00E+11	Drop and crush (worker)	1.00E+09	Bathtubbing	
Zn65	2.00E+12	Drop and crush (worker)	N/A	N/A	

Table 3-2 (continued)
IAEA Illustrative Activity Concentrations for LLW (Vault Disposal in Clay Geo-sphere under Temperate Conditions)

Concentrations for the Vault Disposal Facility with a Clay Geosphere Under Temperate Conditions					
Radio-nuclide	**Operational Period**		**Post Closure Period**		**NRC Limit**
	Limiting Concentration (Bq/kg of Waste)	**Limiting Scenario**	**Limiting Concentration (Bq/kg of Waste)**	**Limiting Scenario**	**Class C Value from 10 CFR 61.55 (Bq/kg @ 1g/cc)**
Sr90	1.00E+20	Drop and crush	5.00E+04	Bathtubbing	2.59E+11
Zr93	1.00E+20	Drop and crush	4.00E+08	Road Construction	
Nb94	5.00E+11	Drop and crush (worker)	9.00E+04	Road Construction	7.40E+06 (in activated metal)
Tc99	1.00E+20	Drop and crush	1.00E+07	Bathtubbing	1.11E+08
Ru106	4.00E+12	Drop and crush (worker)	N/A	N/A	
Ag110m	3.00E+11	Drop and crush (worker)	N/A	N/A	
Sn121m	1.00E+14	Drop and crush (worker)	N/A	N/A	
Sb125	2.00E+12	Drop and crush (worker)	N/A	N/A	
Sn126	5.00E+11	Drop and crush (worker)	N/A	N/A	
I129	2.00E+13	Drop and crush (worker)	5.00E+03	Bathtubbing	2.96E+06
Cs134	6.00E+11	Drop and crush (worker)	6.00E+12	Bathtubbing	
Cs137	2.00E+12	Drop and crush (worker)	8.00E+04	Bathtubbing	1.70E+11

Table 3-2 (continued)
IAEA Illustrative Activity Concentrations for LLW (Vault Disposal in Clay Geo-sphere under Temperate Conditions)

Concentrations for the Vault Disposal Facility with a Clay Geosphere Under Temperate Conditions					
Radio-nuclide	**Operational Period**		**Post Closure Period**		**NRC Limit**
	Limiting Concentration (Bq/kg of Waste)	**Limiting Scenario**	**Limiting Concentration (Bq/kg of Waste)**	**Limiting Scenario**	**Class C Value from 10 CFR 61.55 (Bq/kg @ 1g/cc)**
Ce144	8.00E+10	Drop and crush (crane operator)	N/A	N/A	
Pm147	2.00E+17	Drop and crush (worker)	N/A	N/A	
Sm151	2.00E+16	Drop and crush (worker)	2.00E+11	Road Construction	
Eu152	8.00E+11	Drop and crush (worker)	N/A	N/A	
Eu154	7.00E+11	Drop and crush (worker)	N/A	N/A	
Tl204	6.00E+14	Drop and crush (worker)	N/A	N/A	
Pb210	9.00E+13	Drop and crush (worker)	N/A	N/A	
Ra226	4.00E+05	Gas release (public)	5.00E+04	Road Construction	
Ac227	2.00E+12	Drop and crush (worker)	N/A	N/A	
Ra228	1.00E+20	Drop and crush	5.00E+08	Bathtubbing	
Th232	3.00E+14	Drop and crush (worker)	4.00E+04	Road Construction	

Table 3-2 (continued)
IAEA Illustrative Activity Concentrations for LLW (Vault Disposal in Clay Geo-sphere under Temperate Conditions)

Concentrations for the Vault Disposal Facility with a Clay Geosphere Under Temperate Conditions					
Radio-nuclide	**Operational Period**		**Post Closure Period**		**NRC Limit**
	Limiting Concentration (Bq/kg of Waste)	**Limiting Scenario**	**Limiting Concentration (Bq/kg of Waste)**	**Limiting Scenario**	**Class C Value from 10 CFR 61.55 (Bq/kg @ 1g/cc)**
U234	3.00E+14	Drop and crush (worker)	2.00E+06	Road Construction	
U235	4.00E+12	Drop and crush (worker)	7.00E+05	Road Construction	
U238	3.00E+13	Drop and crush (worker)	2.00E+06	Road Construction	
Np237	3.00E+12	Drop and crush (worker)	3.00E+05	Road Construction	
Pu238	3.00E+14	Drop and crush (worker)	1.00E+07	Road Construction	3.70E+03
Pu239	7.00E+14	Drop and crush (worker)	2.00E+05	Road Construction	3.70E+03
Pu240	3.00E+14	Drop and crush (worker)	2.00E+05	Road Construction	3.70E+03
Pu241	1.00E+20	Drop and crush	2.00E+07	Road Construction	1.30E+05
Am241	2.00E+13	Drop and crush (worker)	5.00E+05	Road Construction	3.70E+03

Exclusion levels which can be considered the lower bounds of LLW (actually Very LLW) are shown in Table 3-3. (43).

Table 3-3
IAEA Activity Concentrations for Exclusion

Radio-nuclide	Activity Concentration (Bq/g)		Radio-nuclide	Activity Concentration (Bq/g		Radio-nuclide	Activity Concentration (Bq/g	
H3	100		Mn56	10	*	Se75	1	
Be7	10		Fe52	10	*	Br82	1	
C14	1		Fe55	1000		Rb86	100	
F18	10	*	Fe59	1		S-85	1	
Na22	0.1		Co55	10	*	Sr85m	100	*
Na24	1	*	Co56	0.1		Sr87m	100	*
Si31	1000	*	Co57	1		Sr89	1000	
P32	1000		Co58	1		Sr90	1	
P33	1000		Co58m	10000	*	Sr91	10	*
S35	100		Co60	0.1		Sr92	10	*
Cl36	1		Co60m	1000	*	Y90	1000	
Cl38	10	*	Co61	100	*	Y91	100	
K42	100		Co62m	10	*	Y91m	100	*
K43	10	*	Ni59	100		Y92	100	*
Ca45	100		Ni63	100		Y93	100	*
Ca47	10		Ni65	10	*	Zr93	10	*
Sc46	0.1		Cu64	100	*	Zr95	1	
Sc47	100		Zn65	0.1		Zr97	10	*
Sc48	1		Zn69	1000	*	Nb93m	10	
V48	1		Zn69m	10	*	Nb94	0.1	
Cr51	100		Ga72	10	*	Nb95	1	
Mn51	10	*	Ge71	10000		Nb97	10	*
Mn52	1		As73	1000		Nb98	10	*
Mn52m	10	*	As74	10	*	Mo90	10	*
Mn53	100		As76	10	*	Mo93	10	
Mn54	0.1		As77	1000		Mo99	10	
Mo101	10	*	Sn125	10		Cs129	10	
Tc96	1		Sb122	10		Cs131	1000	

Table 3-3 (continued)
IAEA Activity Concentrations for Exclusion

Radio-nuclide	Activity Concentration (Bq/g)		Radio-nuclide	Activity Concentration (Bq/g		Radio-nuclide	Activity Concentration (Bq/g	
Tc96m	1000	*	Sb124	1		Cs132	10	
Tc97	10		Sb125	0.1		Cs134	0.1	
Tc97m	100		Te123m	1		Cs134m	1000	*
Tc99	1		Te125m	1000		Cs135	100	
Tc99m	100	*	Te127	1000		Cs136	1	
Ru97	10		Te127m	10		Cs137	0.1	
Ru103	1		Te129	100	*	Cs138	10	*
Ru105	10	*	Te129m	10		Ba131	10	
Ru106	0.1		Te131	100	*	Ba140	1	
Rh103m	10000	*	Te131m	10		La140	1	
Rh105	100		Te132	1		Ce139	1	
Pd103	1000		Te133	10	*	Ce141	100	
Pd109	100		Te133m	10	*	Ce143	10	
Ag105	1		Te134	10	*	Ce144	10	
Ag110m	0.1		I123	100		Pr142	100	*
Ag111	100		I125	100		Pr143	1000	
Cd109	1		I126	10		Nd147	100	
Cd115	10		I129	0.01		Nd149	100	*
Cd115m	100		I130	10	*	Pm147	1000	
In111	10		I131	10		Pm149	1000	
In113m	100	*	I132	10	*	Sm151	1000	
In114m	10		I133	10	*	Sm153	100	
In115m	100	*	I134	10	*	Eu152	0.1	
Sn113	1		I135	10	*	Eu152m	100	*
Eu154	0.1		Ir192	1		Pa230	10	
Eu155	1		Ir194	100	*	Pa233	10	
Gd153	10		Pt191	10		U230	10	
Gd159	100	*	Pt193m	1000		U231	100	

Table 3-3 (continued)
IAEA Activity Concentrations for Exclusion

Radio-nuclide	Activity Concentration (Bq/g)		Radio-nuclide	Activity Concentration (Bq/g		Radio-nuclide	Activity Concentration (Bq/g	
Tb160	1		Pt197	1000	*	U232	0.1	
Dy165	1000	*	Pt197m	100	*	U233	1	
Dy166	100		Au198	10		U236	10	
Ho166	100		Au199	100		U237	100	
Er169	1000		Hg197	100		U239	100	*
Er171	100	*	Hg197m	100		U240	100	*
Tm170	100		Hg203	10		Np237	1	
Tm171	1000		Tl200	10		Np239	100	
Yb175	100		Tl201	100		Np240	10	*
Lu177	100		Tl202	10		Pu234	100	*
Hf181	1		Tl204	1		Pu235	100	*
Ta182	0.1		Pb203	10		Pu236	1	
W181	10		Bi206	1		Pu237	100	
W185	1000		Bi207	0.1		Pu238	0.1	
W187	10		Po203	10	*	Pu239	0.1	
Re186	1000		Po205	10	*	Pu240	0.1	
Re188	100	*	Po207	10	*	Pu241	10	
Os185	1		At211	1000		Pu242	0.1	
Os191	100		Ra225	10		Pu243	1000	*
Os191m	1000	*	Ra227	100		Pu244	0.1	
Os193	100		Th226	1000		Am241	0.1	
Ir190	1		Th229	0.1		Am242	1000	*
Am242m	0.1		Cm248	0.1		Cf253	100	
Am243	0.1		Bk249	100		Cf254	1	
Cm242	10		Cf246	1000		Es253	100	
Cm243	1		Cf248	1		Es254	0.1	
Cm244	1		Cf249	0.1		Es254m	10	

Table 3-3 (continued)
IAEA Activity Concentrations for Exclusion

Radio-nuclide	Activity Concentration (Bq/g)		Radio-nuclide	Activity Concentration (Bq/g		Radio-nuclide	Activity Concentration (Bq/g	
Cm245	0.1		Cf250	1		Fm254	10000	*
Cm246	0.1		Cf251	0.1		Fm255	100	*
Cm247	0.1		Cf252	1				
Naturally Occurring Radionuclides								
K40		10						
All other radionuclides of natural origin		1						
* Half-life of less than 1 day.								

Basis for Development of Activity Concentrations for Classification

NRC and IAEA base waste classification concentrations on similar performance objectives and include the principle of ALARA (As Low As Reasonably Achievable). Both systems evaluate long-term exposures based on disposal site performance and short-term exposures as a result of inadvertent intrusion and recognize the effect of waste form and engineered barriers on the radionuclide source term for the exposure events. NRC's bases for LLW activity concentrations are identified in NUREG 0782. IAEA describes their philosophy and basic site development protocols in Safety Guide WS-R-1, Safety Series 111-F, Safety Series 111-G-1-1 and Safety Series 111-G-3.1. In both cases, low-level waste is considered to be hazardous for 300 to 500 years after which radioactive decay will render the waste relatively harmless. (4) (47)

Disposal Site Performance

Disposal site performance is measured as a function of exposure per year to a member of the public. Evaluations are based on the environmental conditions of the disposal site taking into account various, physical, geographical, geological and hydro-geological factors.

NRC's long-term performance objectives for shallow land disposal of LLRW are 25 millirem/year (0.25 mSv/yr) to a member of the public. (4)

IAEA's recommendations for performance objectives are 30 millirem/yr (0.3 mSv/yr) for long-term exposures. (46).

IAEA recommends evaluation of each disposal site to establish radionuclide concentrations that will meet the performance objectives. (48). NRC performed evaluations based on several regions in the United States and then chose the most restrictive environment as the reference facility from which to establish concentrations for classification. (4).

Inadvertent Intruder

In both IAEA and NRC guidance, exposures to inadvertent intruders are to be developed based on an assessment of the risk of intrusion considering both the likelihood of the event and the probability of failure of the waste form and engineered barriers.

Current NRC guidance/regulation does not evaluate the probability of an inadvertent intruder event or the long-term performance of engineered barriers. Essentially the event is accepted as a certainty and all barriers are assumed to fail. The NRC performance objective for exposure as a result of inadvertent intrusion is 500 millirem/yr (5 mSv/yr). (4).

IAEA follows ICRP guidelines which recommend that additional protective measures should always be considered if the estimated exposure exceeds 10,000 mR/yr (100 mSv/yr) and are not considered necessary below 1,000 mR/yr (10 mSv/yr). Between these two values, the competent regulatory authority is left to decide on the appropriate additional protective measures based on a variety of applicable factors. (46) (47).

Waste Form and Engineered Barrier

As shown above, IAEA recommendations for waste classification are not directly comparable to NRC's LLRW sub-classifications identified in 10CFR61. IAEA has provided similar guidance for the derivation of activity limits for near-surface disposal based on performance objectives in the form of limits on exposure to the public that are very close to NRC's performance objectives for 10CFR61.

NCRP Alternative Classification System

In 2002, the National Council on Radiation Protection and Measurements (NCRP) published recommendations on an alternative classification system for radioactive and hazardous wastes. This method proposed classification based on risk rather than on the source of the waste or presence of a hazardous constituent. The report illustrates our current hazardous / radioactive waste classification system as being arbitrary and inconsistent with respect to the protection of the public and not particularly cost effective or efficient. The report also clearly states the obvious that the current system makes absolutely no sense to a non-technical outsider, i.e. a member of the public and therefore does not engender trust from those who are asked to live with the risk. (5)

NCRP proposed a classification system that evaluates hazards based on similar health risk and then provides for consistent levels of protection based on the evaluated risk. Risk would be determined by the probability and potential severity of harm using generally accepted technical practices for risk assessment. Hazardous / radioactive wastes with low risk assessments would be treated the same and disposed using methods of protection commensurate with the risk. Higher risk wastes would be given more stringent controls. (5)

The NCRP method makes a lot of sense as long as there is agreement on how to develop and determine the levels of risk. Risk assessment is typically based on scientific studies of the effects of a hazardous material on a population and the use of statistical tools to develop probabilities of occurrence or incidence of a deleterious effect. While this serves to establish a consistent numerical standard for risk, it does not necessarily lead to a consistent and publicly accepted definition of 'acceptable' risk. To date, there has been little agreement among various groups as to what constitutes an "acceptable" risk. "Acceptable" is in the eye of the beholder introducing a type of Heisenberg Uncertainty into the definition. What is "acceptable" will change depending on who is making the observation at the time and the observation itself may change the view of 'acceptability' to others. There is no indication that this issue will be solved any time in the near future.

Assuming that the definitions of risk and acceptability can be agreed to and the measurements mathematically (or statistically) established, the NCRP approach would provide a clear and consistent method for waste classification and disposal. For radioactive waste, the majority of the very low-activity DAW and resins could be sent for disposal in RCRA Subtitle C landfills. Some current High Level Waste and TRU waste could be acceptable for near-surface disposal along with Low Level Waste. Conversely, some NORM materials and mill tailings would require more stringent disposal methods and isolation because of the concentrations of long-lived radionuclides.

Comparison to Other National Programs

The French program for radioactive low level radioactive waste disposal is the best example of the implementation of the IAEA disposal system, although there are some differences. The French program allows for surface disposal for those wastes whose activity is dominantly (main elements) comprised of short half-life radionuclides or for wastes with both short-lived and long-lived radionuclides at concentrations approximately 100X the IAEA clearance levels. Short half-life is considered to be approximately 30 years or less. The disposal facility for this waste is assumed to guarantee isolation for at least 300 years. After that time residual long lived activity in the site is assumed to be low enough so as to not represent unacceptable exposure to a future intruder to the site. The French do not currently permit unconditional release of waste with activity at or below approximately the IAEA clearance levels but there is a process for case-by-case evaluation for disposal under restricted conditions. (49)

Very Low Level wastes can be disposed at the Morvilliers Centre disposal facility which is similar to a RCRA Subtitle C hazardous waste landfill. Waste sent to this facility is packaged in sealed or open containers depending on surface contamination levels. There are specifications for dryness, void space and chemical hazards similar to those in effect at US RCRA sites. (49) Waste in this category is similar to the lower activity DAW and contaminated secondary side resins from PWR's generated in the United States.

Low Level and Intermediate Level Waste are disposed at the Centre de l'Aube facility. In general, activity limits for disposal in Centre de l'Aube follow those currently included in 10 CFR 61 with some variations. All wastes are required to be "conditioned" prior to disposal. Conditioning generally takes the form of mixing the waste with a type of grout to stabilize the

physical structure of the waste and eliminate void spaces. Disposal activity concentrations are based on the final package as conditioned and are weight-based rather than volume-based for the majority of 10 CFR 61.55 nuclides. (49)

There are two levels of specific activity limits that mirror the two layers of protection provided in disposal. The first layer of protection is provided by the waste form and container. The second layer is that provided by the disposal technology including the concrete vaults as well as the overall site parameters. Wastes exceeding approximately 0.1 Ci/T (3700 Bq/g) of Co-60 or Cs-137, or 5 nCi/gm (185 Bq/g) of long lived alpha is required to be placed in disposal facility containments. The second set of limits represents the maximum concentrations of activity allowed in the disposal site. Waste packages whose specific activity exceeds 3510 Ci/T (1.3E+08 Bq/g) of Co-60, 8.9 Ci/T (3.3E+05 Bq/g) of Cs-137, 0.00324 Ci/T (1.2E+02 Bq/g) of Nb-94, 5.4 Ci/T (2.0E+05 Bq/g) of H-3, 86.4 Ci/T (3.2E+06 Bq/g) of Ni-63 or 100 nCi/g (3.7E+03 Bq/g) of long-lived alpha are not acceptable for disposal at this site. (49)

Activity concentrations are determined from the conditioned and packaged waste form and includes the weight of the conditioning material and the concrete containers. Essentially all waste generated from reactors in the United States with the exception of what is currently defined as High Level Waste and TRU waste would be acceptable for disposal using this method of classification and disposal. (49)

4
IMPORTANCE OF MORE RISK BASED PROTECTION REQUIREMENTS

Need for Protection

The NRC Performance Assessment Working Group (PAWG) report, NUREG/CR-1573, identifies examples of engineered barriers as including surface drainage systems, cover systems, concrete vaults, HICs, backfills, in-fills, etc. Identifying water intrusion as the principal mechanism for transport of activity from the site, engineered barriers are considered in the NUREG primarily with respect to limiting water contact with the buried waste.

> "Although engineered barriers may be used to improve facility performance, it is nonetheless expected that the disposal characteristics of the site itself will meet the suitability requirements of 10 CFR 61.50." (28)

It was recognized in the Environmental Impact Statement for 10 CFR 61 that at some point in time all that would be left is the site itself and its characteristics would govern the release of radioactivity. Citing another study completed in 1990, the working group reported that most of the long lived activity (half lives greater than 100 years) disposed in LLRW resides in waste disposed as Class A. (27)

The PAWG looked at engineered barriers from the perspective of the level of protection that could be relied upon to remain physically distinct and stable long after the 500 years that the intruder barriers are required to perform in accordance with 10 CFR 61.52 which states:

> "Wastes designated as Class C pursuant to §61.55 must be disposed so that the top of the waste is a minimum of 5 meters below the top surface of the cover or must be disposed with intruder barriers that are designed to protect against an inadvertent intruder." (2)

Notwithstanding that the 500 year requirement is only applied to Class C wastes, and already having placed the bulk of the risk in wastes that are not subject to extended protection, the PAWG discounts the importance of intruder barriers. Two issues arise in this discussion: 1) How can the class A waste be attributed to more significant long term protection need? 2) Isn't a near surface facility primarily suited for short lived activity where the disposal concentration limits on long term activity are designed to obviate the need for protection after 500 years? Granted, it's an added benefit if some assurance of protection extends beyond 500 years, it would be difficult to guarantee such protection. (50)

Long-lived Activity in Class A Wastes

In general, long lived activity of concern in LLRW disposal may be represented by those radionuclides listed in 10 CFR 61.55 Table 1. These radionuclide limits are based on the premise that at some point in time they will be released to the general environment. There is nothing in the site restrictions or provided by barriers that could prevent their release to the general environment. Indeed, the half-lives are sufficiently long that an entire geological era could pass by before they reach equilibrium with their daughter products. The basic limit for transuranic radionuclides of 10 nCi/gm was set on the premise that in terms of radio-toxicity and half-life, Pu-239 is comparable to Ra-226. Since radium is widely dispersed throughout the earth's crust, it was viewed as reasonable to set a limit for transuranics approaching the upper bound of concentrations of natural radium. The value that was arrived at was 10 nCi/gm (for pure Pu-239) (29). This roughly corresponds to a locally high value which would result in a dose rate to an individual of ~200 mrem/year (11).[4] It should be emphasized that this is the dose rate that would arise if all of the waste was at the concentration limit and no dilution occurred preceding exposure. Actual average concentration of transuranics are more than an order of magnitude lower and would be significantly diluted before the scenario would be realized. (4)

Given the assertion of 10 CFR 61.52 that the barriers are provided to protect against Class C waste for a period of 500 years, it should be noted that the concentrations of radionuclides listed in Table 4-1 are hardly changed after 500 years. While the probability of the intrusion may be reduced, the impact of the intrusion would be basically the same. The real protection from delaying the intrusion applies to radionuclides listed in 10 CFR 61.55 Table 2. Protection applicable to Table 1 radionuclides is only achieved by setting the disposal limits low enough so that when intrusion does occur the impact is inconsequential or within acceptable parameters. This was done in the case of the transuranic radionuclides. While the risk of intrusion is reduced with respect to these isotopes the consequence is unchanged. Since we are dealing with a very long time frame the probability that the waste would be intruded upon within the risk time frame is relatively high. The protection gained by the intruder would be dependent only on the dilution of the source material during the delay. Assuming that there was no migration of activity from the disposal site, opportunity for dilution is provided the addition of lower activity fill during the emplacement of the waste and by mixing with the cover when the waste is excavated. The minimum impact would be achieved by assuring that there is sufficient dilution in the disposal operation to assure that the longest lived radionuclides are rendered inconsequential at the time of disposal and indefinitely beyond.

This brings us back to purpose of barriers. Based on the premises upon which 10 CFR 61 was formulated, we are allowed 300 years for stabilization (as a first level intruder barrier) and additional 200 years provided by the barrier imposed by deeper cover. The best we can get out the barriers is therefore 500 years. This time is sufficiently long to eliminate risks associated with Table 2 radionuclides. Based on the numerous examinations of source term that have taken place over the last 30 years, the radionuclides most important to intruder risk turn out to be those in Table 2. Co-60, which is only limited in Class A wastes, is quickly depleted. Cs-137 which dominates the intruder exposures in all short term scenarios is fully depleted at 500 years. This is similarly true for Sr-90 although its impact is much less acute. The only radionuclide in 10 CFR 61.55 Table 2 that would remain hazardous for more than 500 years is Ni-63. As we have

[4] Based on IMPACTS and RESRAD cases for Intruder agriculture scenario

seen in the previous section, the limit for Ni-63 overstates its risk. We agree that intruder barriers provide no protection related to very long lived activity. We do not agree that there is (or should be) substantial risk from very long lived activity beyond the reliable isolation provided by the barriers.

The reference in the Roles Report to significant quantities of long-lived activity in Class A wastes appears to apply to C-14 in non-utility wastes. (27) This activity in the associated waste streams is already diluted to an acceptable level for unmonitored release. Irrespective of the bases and opinions expressed in the PAWG report, recent disposal developments have put a lot of emphasis on stabilization and intruder barriers. (28) Nearly every new development in the United States proposed or implemented since the publication of 10 CFR 61 has relied heavily on enhanced barriers including concrete structures and thicker, highly engineered structures. The following discussion highlights some of these developments.

Disposal Site Planning – Standards Set by Compact Groupings

The disposal concept analyzed in the EIS for 10 CFR 61 for Class A waste was that of an unlined trench with shallow cover. (4) The waste would be placed randomly and covered with original soil with minimum compaction. Public protection is achieved primarily through judicious site selection. The radioactivity concentration limits provided in 10 CFR 61 are based principally on intrusion into the Class A trench and are dependent on the specification of a 2 meter soil cover over the waste. For Class C wastes an additional factor of 10 in the disposal limits is allowed to account for a 5 meter disposal depth. The regulation mirrored the dominant disposal practices in place prior to the promulgation of LLWPA. The aim of the regulation was to provide a uniform basis for the development of a series of regionally distributed LLRW disposal sites. Figure 4-1 provides a representative example of random un-stabilized disposal as practiced prior to 10CFR61. The particular example shown is taken at a disposal site once operated by the Department of Energy. Final cover depth was 1.2 meters (~4 feet). (29)

Figure 4-1
Random Placement (Kick-and-Roll) Disposal of Un-stabilized Waste (circa 1970)

The Low Level Waste Policy Act directed States to form regional compacts that would be responsible for coordinating the development of a regional site. Decisions related to locating the site and site design would be made by the compact committee in accordance with the regulations of the State that would host the site. Disposal sites that were in operation prior to the issuance of 10 CFR 61 sites in existence prior to 10 CFR 61 were allowed to continue in operation in accordance with their licenses. (39)

Newly formed compact organizations began independent evaluations of disposal site options and generally opted for more conservative designs than what was used by NRC as a base line. Even though none of the originally proposed compact sites were developed, the design concepts and disposal planning associated with them is worth noting. Most of the activity related to compact disposal sites occurred during the late 1980's and early 1990's. Both Chem-Nuclear and US Ecology provided consulting and design support to several of the compact authorities. In nearly all of the documented cases the compacts opted for more conservative designs than what would have been required to meet the 10 CFR 61 criteria.

The Low-Level Waste Policy Act of 1980 encouraged the formation of regional compacts to develop new disposal sites by 1986 at which time mandated access to the existing disposal sites would be phased out. (3) The formation of the compacts took substantially more time than was anticipated in the Act as various States vied to enter groupings that gave them the most advantage and least probability to be the host of the site. The Low-Level Waste Policy Amendments Act of 1985 extended the time for development of the regional sites to 1992. At the same time it extended the mandated access to the existing disposal sites to 1992. (39) The greatest concentration of activity by the compacts toward the development of the regional sites occurred during that period from 1986 through 1992. By that point some of the regional sites were proceeding aggressively while others had disintegrated when the selected host state opted out of the compact. None would be ready to operate by 1992. (40)

With the exception of the US Ecology Richland disposal site which had been co-opted by the Northwest regional compact, operating disposal sites were closed June 1992 leaving unfilled demand. The State of South Carolina elected, following the closure of the Barnwell disposal site, to re-open it to out-of-compact waste. The States of North Carolina and Michigan were locked out since each had reneged on its commitment to act as a compact host State. The re-opening of Barnwell removed the some of the urgency built into the process for developing new disposal sites. In addition, with so many sites under consideration, it was realized (by most of the involved parties) that given the high development costs including litigation and delays most of the proposed sites could not be economically operated. There just wasn't enough waste generated to establish a reasonable volume based cost. (51)

As the compact disposal site development proceeded, member States developed new regulations to expand design requirements for the disposal sites. Much of the development effort was directed at establishing the regulatory review process to be followed by the compact authorities, establish design criteria, and select a suitable design concept. The literature provides few examples of completed performance assessments.

Up to this point two new disposal sites have been conceived and licensed since the publication of 10 CFR 61. These include the EnergySolutions' Clive, Utah disposal site and the WCS disposal site near Andrews, Texas. The EnergySolutions site was developed entirely outside of the

compact process and is not subject to compact authority. The site was licensed for the acceptance of B and C waste but with the final issuance of the license the Governor of the State of Utah restricted the site to receive Class A waste only. The WCS site was also conceived independently of the compact process but the site was adopted by the Texas compact after its effort to develop a site near El Paso fell through. A license was finally issued in 2009 to WCS after more the 12 years in process. Both the EnergySolutions Site and the WCS site are discussed along with other designs in more detail in the following paragraphs.

The Chem-Nuclear Concept Site

Chem-Nuclear was contracted to develop and operate three of the newly proposed compact disposal sites. These included, Central Midwest (Illinois), Appalachian (Pennsylvania), and Southeast (North Carolina) compacts. (52)

The concept site proposed by Chem-Nuclear Systems, Inc. is built on multiple barriers to limit the release of radionuclides from the disposal site. These included site characteristics, Waste Form, and engineered barriers. The engineered barriers included overpacks, closed disposal cells and multiple layer covers. All of the waste would be placed in a concrete overpack (7-10 inches thick) with a capacity of 150-400 cubic feet of waste. The concrete overpacks are sealed with grout and placed in reinforced concrete modules which would hold up to 200 of the overpacks. The modules are back filled with sand to take up any free space and provided with a concrete cap. The modules can be located above or below grade. The final cover extends from 7 to 10 feet (2.13 meters to 3.05 meters) above the module cap. The reinforced concrete modules and engineered cover are common to all waste classes.

The performance appraisal methodology proposed a design life of 560 years including the operational period. Three successive periods were defined following site closure including 100 years of active institutional control, followed by 200 years of passive institutional control, and 200 additional years as an intruder control period. During the last period the vaults and overpacks are assumed to provide intrusion resistance.

The State of Nebraska (Central Midwest)

The Nebraska disposal facility was planned as an above ground facility following the concept of Centre de l'Aube in France. (53) The Class A disposal facility consists of 20 separate reinforced concrete structures or cells. Class B and C waste disposal cells are comparable to those used for Class A except that the cells are more compartmentalized to aid placement and retrieval. All of the disposal cells are backfilled with sand before final capping. The multilayered closure cap consist of a clay layer, a concrete layer, another layer of sand, a geotextile filter, a native soil cover and rock armor to prevent erosion on the sides. Thickness at the center of closure cap is ~12 feet (3.8 m) in addition to approximately 1 additional meter for the concrete roof section of the disposal structure. The Nebraska facility designed by US Ecology was located on a rock plateau and was one of the first to propose an above ground facility. (54)

Ward Valley – California

Among the compact developed sites, the Ward Valley site advanced the furthest through State licensing and public acceptance. The site is located in the Mojave Desert near the California-Nevada border. The site is remote and one of the driest locations in the US providing ideal conditions for LLRW disposal. It was planned to follow closely to the concept of 10CFR61 and closely parallel to that of the US Ecology Site in Richland Washington. The Ward Valley site as proposed included a separate trench for Class A unstable wastes. Higher activity, stabilized Class A waste would be included in a BC waste trench. The BC facility did not include supplemental concrete structures or special liners on the trench. However, cover design for these trenches were significantly augmented over that proposed by 10CFR61. As described by Dressen in 1992, the site will use a "...three layer cover system, in which the outermost layer will be a 2-foot intrusion barrier and capillary break constructed of sand, gravel, and cobbles. The next layer will be a 2-foot thick layer of site soil amended with clay to provide a permeability contrast to enhance the capillary break. Finally a 22-foot layer (6.71 meters) of site soils will be placed on top." (55)

State of New York

The State of New York sponsored the investigation by Roy F. Weston Co. and Burns and Roe, of a broad range of disposal options. Their evaluation excluded near surface facilities without engineered barriers owing to a State regulatory constraint. Most of the forty options considered include some form of concrete structure either as a modular disposal unit, vault, or a monolith. Other options included deeper disposal options including mines and tunnels. The highest ranked option in their examination was the use of surface located modular disposal units. (56)

Chem-Nuclear Systems was providing design for 3 compact facilities including Central Midwest, Southeast, and Appalachian all designs included enhanced technology features described in advance disposal technologies, The PA and NC facilities were designed to follow Centre De L'Aube in France. (57)

Texas/Hueco Bolson

The disposal facility proposed by the Texas radioactive waste disposal authority provided for the disposal of Class A waste below grade in concrete containers followed by placement of a multi layered earthen cover. Waste will be placed so that a minimum total cover thickness of 5 meters exists. Class B and C wastes would be disposed in much the same way. The cover systems are constructed with a slope of 3% on all sides. The multilayered cover included both permeable and impermeable layers to divert and limit water infiltration. (58) A performance assessment for the Hueco-Bolson site was performed by the University of Texas (59) using PATHRAE. The maximum external gamma dose calculated was estimated to a farmer working adjacent to the site was 9.28 mrem/yr immediately following site closure. The maximum dose from contaminated well water occurring at 12,300 years was 4.52 mrem/yr to the thyroid and 3.11 mrem/yr whole body. The study reported both scenarios as well within the criteria of 10CFR61 and extremely unlikely.

WCS/Texas

The Andrews disposal site recently licensed as the State of Texas compact disposal site represents the most recent example of disposal site standards. The disposal site includes separated facilities for commercial wastes specifically focused from private sector waste generators including nuclear power plants in Texas, and federal wastes. The facilities are generally referred to as the CWF (Compact Waste Facility) and FWF (Federal Waste Facility). The CWF and FWF disposal facility layouts are similar in many respects, but the FWF is larger and deeper than the CWF. The FWF is divided into two separate areas. In the FWF Canistered Class A, Class B, and Class C waste will be placed into either cylindrical or rectangular concrete canisters. The concrete canisters will be placed in arrays or layers in the Canister Disposal Unit (CDU). A portion of the Federal waste that is stable Class A bulk waste and satisfies certain restrictive conditions will be designated for disposal in the Non-Canister Disposal Unit (NCDU). (60)

The CWF facility includes a composite cover system is designed to complement the lithology of the site. A cover system of this thickness, at 25 to 45 feet (7.62 meters to 13.72 meters) thick, provides "environmental protection from surface erosion mechanisms, allows extra thicknesses of various cover layers to be specified, and simplifies constructability. The infiltration barrier subsystem of the cover is placed deep in the cover system layer as well, and is enhanced with additional low permeability red bed clay fill layers to further prevent infiltration and to increase cover system resilience to settlement." (60)

The final cover consists of three complementary systems:

1. Performance Cover System,
2. Bio-barrier Cover System, and
3. Evaporation-transpiration Cover System.

Each system is identified by its principal feature and is a composite of multiple layers. The performance cover system is designed to meet the performance criteria. The other systems are added to enhance the function of the performance cover system. A general schematic section of the three (61) All containerized waste including Class A, B and C is required to be disposed in concrete canisters within the Commercial or Federal Waste Facilities.

A multi-level performance assessment was performed in support of the license application for the WCS site. Table 4-1 summarizes the scenarios examined and provides dose estimates from the facilities as a percentage of the acceptance criteria. The dose rate limits conform, generally, with those traditionally used for operators and members of the public. Separate criteria were devised for the State of Texas application for intruder evaluations. A value of 100 mrem/year was set in lieu of the 500 mrem/yr followed by the NRC. In all of the cases examined the estimated dose rates were less than 10% of the 100 mrem/year criterion. Dose rates for the short term intruder scenarios were the highest for the CWF facility. This is consistent with the general premise that these wastes, disposed under the compact program are dominantly short activity wastes. Wastes disposed in the other facilities, including the federal facility are controlled by the long term scenarios (i.e. off site exposures out to 50,000 years). The thicker covers used in the WCS trench designs obviated the 10 CFR 61 EIS (NUREG-0782) intruder agriculture and

construction scenarios. The controlling intruder scenarios examined was that of an adjacent resident and a drilling scenario. The drilling scenario was postulated to bore into waste stacked at the most restrictive concentrations and gaseous radionuclides within the disposal site were assumed to available for release through the bore hole.

Table 4-1
Summary of WCS License Application Exposure Scenarios*

	Acceptance Criterion	%of Limit	% of Limit
	(mrem/yr)	FWF and CWF	CWF Only
Normal Operations, Worker	5,000	7.20%	4.80%
Normal Operations, Site Boundary Individual	25	0.10%	0.00%
Normal Operations, Nearest Resident	25	0.00%	0.00%
Institutional Control, Worker	5,000	0.01%	0.00%
Institutional Control, Site Boundary Individual	25	4.80%	0.05%
Institutional Control, Nearest Resident	25	0.02%	0.00%
Post-Institutional Control, Intruder Driller	100	6.70%	6.70%
Post-Institutional Control, Intruder Resident	100	4.60%	4.60%
Post-Institutional Control, Adjacent Resident	25	13.60%	6.40%
Accidents, Worker	5,000	13.40%	3.20%
Accidents, Site Boundary Individual	100	2.50%	2.20%
Accidents, Nearest Resident	100	0.00%	0.00%

*Table 8.33 from Section 8 of the WCS license application is provided in Appendix C for reference.

EnergySolutions - 2001

The EnergySolutions containerized waste facility (CWF) was developed to handle all classes of LLRW shipped in containers and liners. Higher activity Class A waste would be subject to the same disposal provisions as Class B and C packages. This includes placing the waste in cylindrical concrete canisters or caissons at the lowest level of the above grade facility. The waste containers are limited to 5.5 feet diameter to fit within the canisters. After the packages are added, the canister layer is backfilled with flowable sand. Sand is built up to 1 foot on table of the layer. An additional 1 foot layer of clay is added along with another foot of side before the second layer of canisters is started. Space between the canisters is reserved for the placement of large components. The added sand and clay layers are repeated on top of the second layer. The entire construction is then covered with six meters of shredded and compacted bulk LLRW. The bulk layer represents material that would be well below the Class A limits and would not in itself represent an intruder hazard. Higher activity Class A waste containing most of the Cs-137 would be confined to the containerized waste. (62)

The design reflects the concept of layering as discussed in the 10 CFR 61 EIS. This concept uses the lowest activity waste as a buffer and shield of the higher activity wastes.

NUREG/CR-5041, Volume 1 – Below Ground Vault System

At the request of the NRC Division of Low-Level Waste Management, the U.S. Army Engineer Waterways Experiment Station (WES) and the U.S. Army Engineer Division, Huntsville (HNDED) have developed general design criteria and specific design review criteria for the below-ground vault (BGV) alternative method of low-level radioactive waste (LLW) disposal. A BGV is a reinforced concrete vault (floor, walls, and roof) placed underground below the frost line, and above the water table, surrounded by filter blanket and drainage zones and covered with a low permeability earth layer and top soil with vegetation. (63)

Eight major review criteria categories have been developed ranging from the loads imposed on the BGV structure through material quality and durability considerations. These include:

- Loads and Load Combinations
- Structural Design and Analysis
- Construction Material Quality and Durability
- Construction and Operations
- Quality Assurance
- Structural Performance Monitoring
- Filter and Drainage Systems
- Waste Cover Systems

Specific design review criteria have been developed in detail for seven of the eight major categories. Interestingly, these evaluations looked only at the facility from a civil structural perspective. It does not appear that any particular attention was paid as to how these designs might impact the performance assessment.

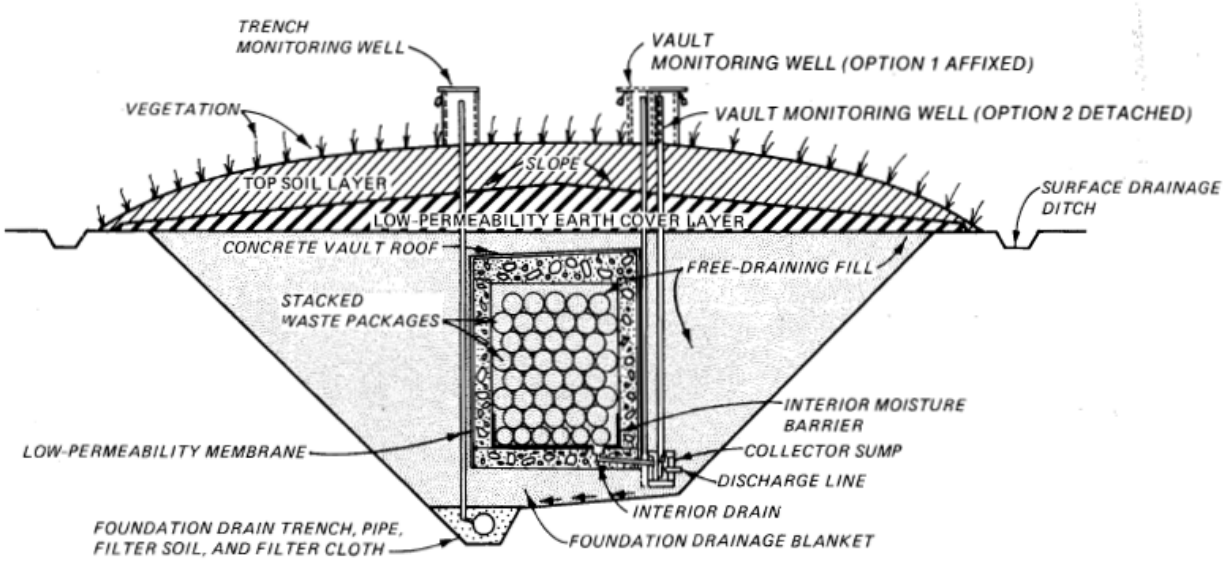

Figure 4-2
NUREG/CR-5041 Below-Ground Vault Conceptual Design

The model referenced in the NUREG mirrors very closely the below grade proposals made for some of the compact sites. Features included inner concrete vault structures equipped with internal moisture barriers, moisture collection, free draining fill, concrete vault roofs, low permeability covers to redirect infiltration and monitoring wells for the trench and the vaults as well.

NUREG/CR-5041, Volume 2 – Earth Mounded Concrete Bunker (EMCB)

Major components of an EMCB are shown in Figure 4-3. An EMCB consists of a below-grade reinforced concrete bunker placed in an excavation, below the freeze line, on a pervious foundation blanket with pervious fill material placed adjacent to the walls and roof of the bunker and covered with a low permeability material and an above grade tumulus consisting of waste packages covered with pervious material, a low-permeability soil-layer, pervious layer, and capped with a topsoil with vegetation or a rock protection surface. The EMCB includes a drainage system with pipes that are connected to a monitored sump. Typically, Class C and B waste would be placed in the below-grade concrete bunkers while Class A waste that meets the stability requirements of 10 CFR Part 61.56(b) would be placed in the tumulus. The design guidance provided in this report is based on the assumption that the 10 CFR Part 61.50(a)(7) site suitability technical requirement has been met and the actual location of the EMCB does provide sufficient depth to the water table such that ground water intrusion , Perennial or otherwise, into the waste will not occur. (64)

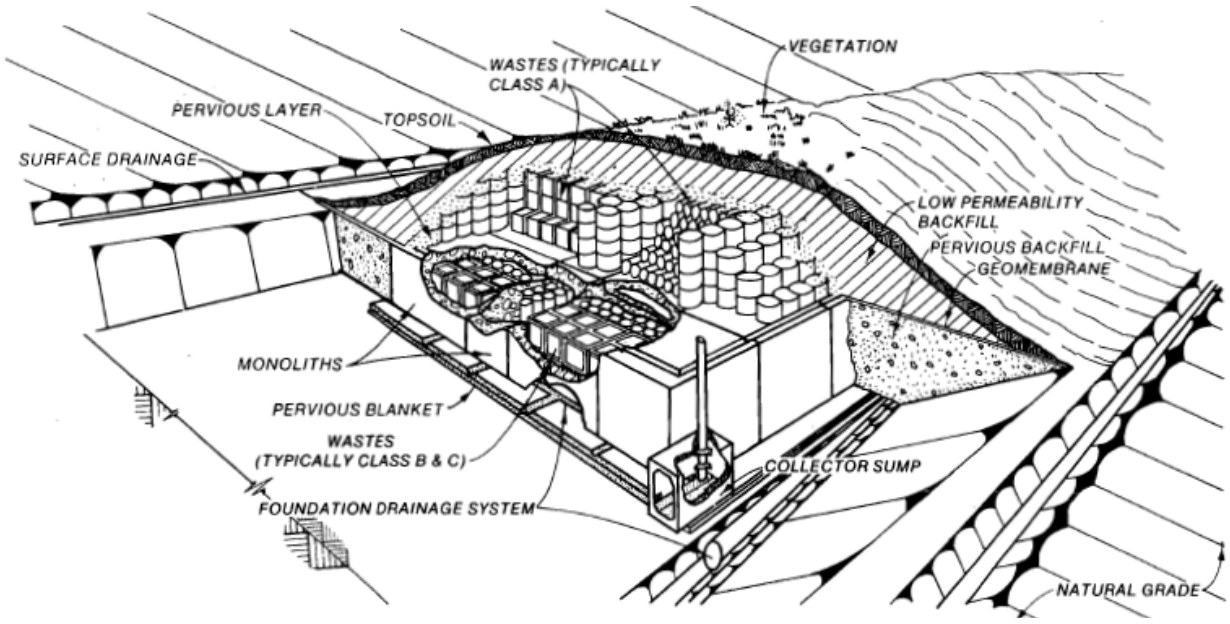

Figure 4-3
NUREG/CR-5041 Earth Mounded Bunker Conceptual Design

Figure 4-3 provides a perspective view of an earth mounded concrete bunker. The figure shows Class A waste located and stacked on top of the bunker structures. Class B and Class C wastes are placed below in concrete structures backfilled with concrete to form a monolithic structure.

The facility is provided with a drainage network within and around the structure which along with cover design prevents contact of water with the wastes. The above ground vault system evaluated by NRC is comparable, if not identical, to the earth mounded systems used by the French since the early 1980s. The concepts are effectively identical. (49)

Centre de le Manche – ANDRA France

Among the regional facilities proposed at Eastern sites nearly all of them followed versions of the French model of above ground vaults. The le Manche facility was being phased out in the early 1990's to be replaced by a new disposal facility at l'Aube. The new facility was similar to le Manche in concept but offered some enhancements. As in the design used at La Manche, L'Aube includes concrete cells to contain the waste. This design includes is a liquid collection system that includes a permeable layer underlaid with an impermeable clay layer. Liquids intruding in the facility are collected through an underground gallery. (49)

Recommendations of the Performance Assessment Working Group (PAWG) – NUREG -1573

The basic concept upon which the limits of 10 CFR 61 were defined included minimum barriers to intrusion into a Class A facility. Un-stabilized waste was assumed buried under two meters of soil cover. The only provision that limited intrusion was the 100 year period of institutional controls. The site was further assumed to be located in a region where there was sufficient

rainfall and appropriate soil conditions to support agricultural activity. The Class A disposal limits were established to provide an adequate level of protection for that condition. None of the proposed disposal sites approximated that condition. Of the sites licensed since10CFR 61 was published, significant additional engineered barriers are provided to protect the site from intruders and to prevent water transport through the site. Furthermore operating disposal sites have enhanced disposal practices in view of this trend. (50)

Overall protection requirements for radioactive wastes in near surface facilities assumes isolation of the waste for limited time period generally viewed as on the order of 300 to 500 years. This is considered to be a conservative time frame for the reliance on engineered barriers. Figure 4-4 provides an overview of the function of an near surface facility. Once past 500 years, the dominant radionuclides contributing to public risk are those with longer half-lives.

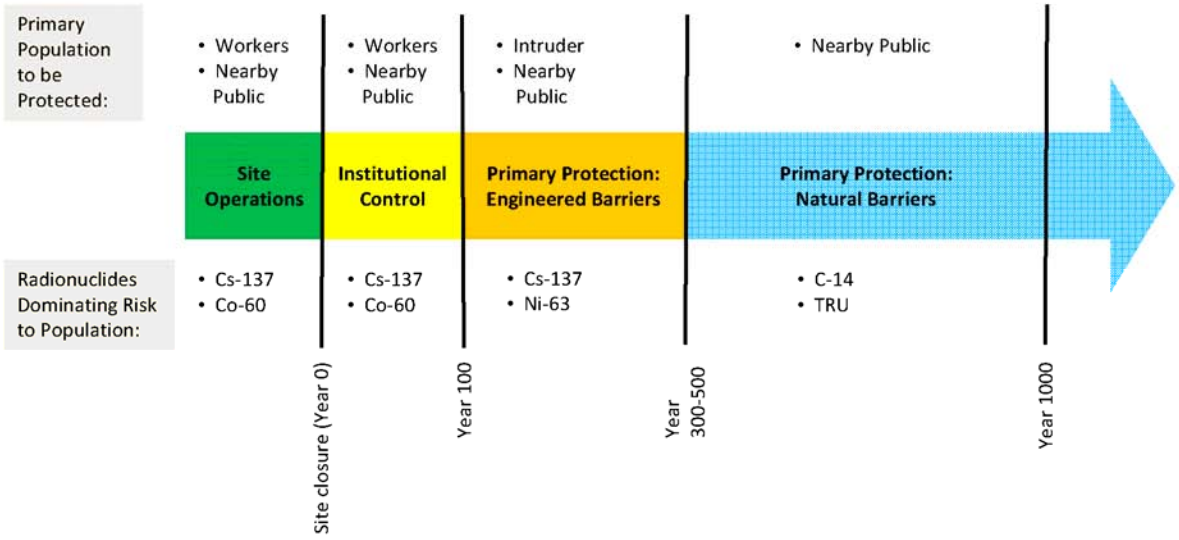

Figure 4-4
Waste Disposal Time Line

Once past 500 years, protection is provide by natural barriers only. These can include natural process resulting in environmental dilution. No further containment by the facility is assumed.

5
RISK ASSESSMENT AS APPLIED TO INDIVIDUAL RADIONUCLIDES

Several factors play into the determination of risk associated with each radionuclide in disposed radioactive waste. These include how much is produced, their half-lives, transportability within the generating facility, the types of emissions associated with the radionuclides (alpha, beta, gamma), their transportability in the disposal site, volatility, biological behavior, etc. Some radionuclides may be produced in great quantity with significant dose potential, but because of a short half-life would never actually be released from the facility. Other radionuclides could have a sufficiently long half-life but would have little propensity to leave the point of their creation due to low diffusivity or attraction to surfaces. Each radionuclide is evaluated according to its own properties. As observed in the previous section, nuclides important to solid LLRW represent an extremely small subset of the more than 700 radioisotopes that have been identified across the periodic table. Most of the isotopes have been ruled out based on half-lives that preclude disposal risk or no appreciable production pathway. For example, radionuclides with half lives less than 5 years would decay to insignificant concentrations in less than the institutional control period, which is defined as—yrs after the closure of a disposal site.

In this section, those radionuclides specifically called out in 10 CFR 61, are investigated in the context of our current understanding of the risks associated with them. In some cases, we have a better understanding of the LLRW source term that enables us to make a better estimate of the disposal activity level activities than what was assumed during the development of 10 CFR 61. In other cases, updated ICRP/NCRP dose conversion factors result in reduced exposure risks.

Historically (before the reorganization of the Atomic Energy Commission), radioactive waste disposal was provided at AEC operated disposal sites. The general criteria defining waste acceptable for disposal at these facilities was a transuranic content less than 10 nCi/gm which provided an enveloping basis for alpha based radiation exposure in the general environment. (4) This limit remains current and has been internationally adopted. With the formation of the NRC and the privatization of LLRW disposal, this limit was retained in the commercial facilities. When 10 CFR 61 was published, limits were established for additional radionuclides considered important in the disposal environment. The most mobile of these radionuclides include H-3, C-14, Tc-99, and I-129. They are often referred to as “the four phantoms” since they are most often either undetectable or not significant relative to the classification limit. These along with transuranics (TRU) were singled out in 10 CFR 20 to define the minimum reporting requirements for waste sent for disposal. These radionuclides are always required to be listed for disposal reporting (6). The limits on TRU are consensus limits. While there may be some issues in how the limits are applied, there is general agreement to the original limit even if the basis is not well known. The following paragraphs discuss the limits imposed by 10 CFR 61 and their general applicability to risks imposed by LLRW disposal.

Inventory versus Concentration

While concentration limits are defined in 10 CFR 61 for radionuclides important to disposal. Not all of the radionuclides are strictly concentration limited. Very long lived radio-isotopes do not generally present a hazard to the public while the isolation provided by the disposal site remains intact. After that point, activity in the disposal site will start to migrate within and out of the disposal site. Radionuclide migration out of the site requires the assumption that constraints on migration provided by the waste form and engineered structures have long since broken down. For the purposes of this evaluation, engineered structures are assumed to be in tact for the first 300 years after site closure.

"The Four Phantoms", H-3, C-14, Tc-99 and I-129

One aspect of the radioactive waste disposal regulations that is worth examination is the special status that has been granted to tritium, Carbon-14, Technetium-99 and Iodine-129. These four nuclides are not only specifically listed in 10 CFR 61 along with another dozen or so there but are also singled out in 10 CFR 20 with a specific manifesting requirement. (6) The reporting requirements for these radionuclides represent additional program costs that are not justified by their radiological risk. The primary reason that these radionuclides were singled out was to track total onsite inventory "to assure that the performance objective for long term environmental protection is not exceeded". (4) The performance objective is as stated in 10 CFR 61 is 25 mrem/year whole body and 75 mrem/year to any organ. Since the concern is based on total inventory and not concentration, short term acute exposure is not at issue. Potential migratory impacts (e.g. groundwater transport) depend on the total inventory.

Additional details relating to each of these radionuclides follows below.

Tritium

Tritium (H-3) has proven to be mobile in the burial site environment. Beyond that, it has a relatively short half-life of slightly more than 12 years and very low decay energy of 18KeV. (13) It has a minimal role in the pathway analysis because it is expected to have substantially decayed during the period of institutional control and, due to the meager decay energy, would result in very little dose even if it did not decay. If dose is used as an indicator of risk, H-3 would not belong in this grouping. However, due to its high mobility, it would contribute to short term gaseous and liquid transport from the disposal site. (14)

Production of Tritium

The primary pathway for H-3 production is neutron reactions with lithium, boron, and deuterium in reactor coolant. The reaction with Boron accounts for about 90% of the production from activation. Estimated annual production in a pressurized water reactor (PWR) is around 400 Ci/yr from coolant activation and about 150 Ci/yr from fission. Production in BWRs is much lower since Boric Acid is not used for reactivity control. This also makes LiOH unnecessary for pH control. Some H-3 is produced and released through absorption in the boron carbide (B_4C)

used in BWR control rods. Of this activity, only a small portion of the H-3 is collected in solid waste streams. Most H-3 generated in nuclear power plants is released directly to the atmosphere through various venting systems or in liquid release pathways. In BWRs, H-3 is released via the condenser air ejector system to the plant vent stack. This relatively high removal flow keeps on-site levels of H-3 relatively low. In PWRs, since the reactor coolant system continually recycles, inventory levels of H-3 in plant systems build up to equilibrium levels corresponding to lower vent rates from storage tanks. On some level, all of the H-3 produced on annual basis is either released or decays. Based on our survey of LLRW shipments, PWRs, on average, can account for about 34 Ci/yr per 1000 MWe in LLRW. BWRs, on average can account for about 10 Ci/yr per 1000 MWe. The remainder of the H-3 would be released through liquid or gaseous pathways. (15)

Measurement of Tritium

H-3 is a pure beta emitter and is measured by liquid scintillation counting. Combustion gases from solid samples are passed through a catalyst which converts the tritiated gas to HTO. The HTO is mixed with a liquid scintillation medium and counted. (15)

H-3 at many PWR plants is routinely measured in reactor coolant using plant equipment. H-3 in solid waste streams is measured periodically in off-site laboratories. The laboratory measurements can be used directly or coupled with key radionuclides in a scaling factor. All of the radiochemical results show wide variability in H-3 concentration within individual waste streams and across waste streams. Scaling factor correlations generally fail leaving a necessity to apply some enveloping condition. (11)

H-3 is rarely actually measured in DAW, the waste constituent that makes up the vast majority of the radioactive waste disposal volume. The waste H-3 activity content is typically calculated based on hypothetical waste moisture content and the H-3 content of reactor coolant. While this method of waste activity determination is certainly defensible, it hardly adds to the waste activity data reliability. Even for wet wastes such as secondary resins, the waste is unlikely to have been in contact with or transferred to the waste container with reactor coolant water. (11)

How Good Are Our Estimates?

The amount of H-3 present in a power plant at any given time is dependent on the turnover rate of reactor coolant and the fixed production rate of H-3 from activation of reactor coolant along with constituents of the coolant. Based on estimates made by Westinghouse (16) the reactor coolant turnover rate is equivalent to four volumes per year. The reactor coolant volume (equivalent 1000 MWe unit) is approximately 261 m^3 (~69,000 gallons). Using an annual H-3 production rate of 400 curies, the expected average concentration of H-3 in reactor coolant is less than 1 μCi/cc. This concentration should represent an upper bound on H-3 concentration in LLRW streams. Note that since H-3 is incorporated in water, there is no selectivity in any of the processes that would result in concentrating H-3. Excluding DAW for the moment, if we assume that half of the volume of PWR process waste (~13.5 m^3) is effectively water at the reactor coolant concentration, we get a total of 13.5 Ci/year released into solid wastes or about 1/4 of the reported activities in PWR LLRW. To further investigate this incongruity, reported H-3

concentrations from the EPRI survey of shipping data were examined. The EPRI study compiled records from 41 PWR units over a 4 year period. The PWRs together accounted for approximately 38,000 MWe. Figure 5-1 shows the distribution of reported H-3 concentrations in waste activities by numbers of packages in each grouping. For the majority of packages the H-3 concentration remains below the reactor coolant concentration. However, for a significant number of packages the reported concentrations actually exceed the upper bound of expected concentrations. (11)

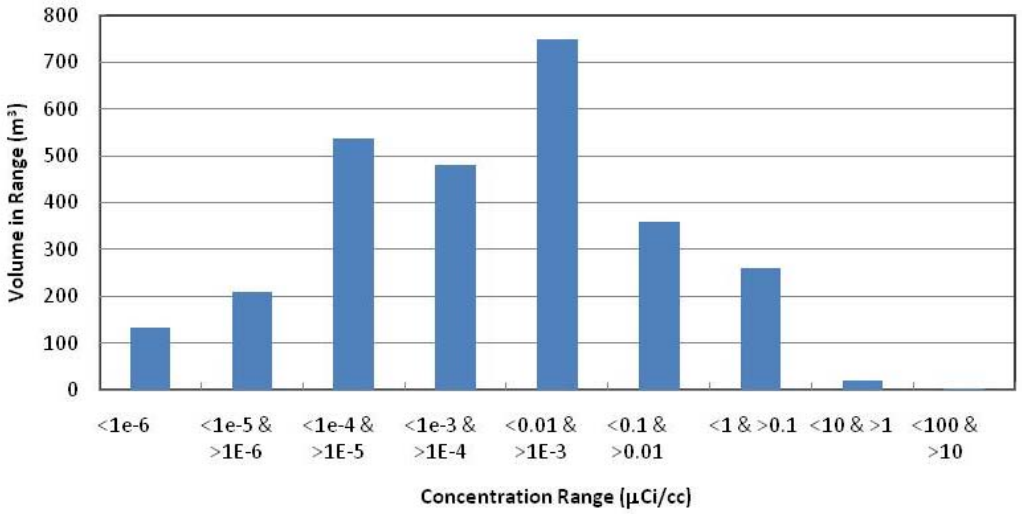

Figure 5-1
Distribution of H3 Concentration by Package Volume

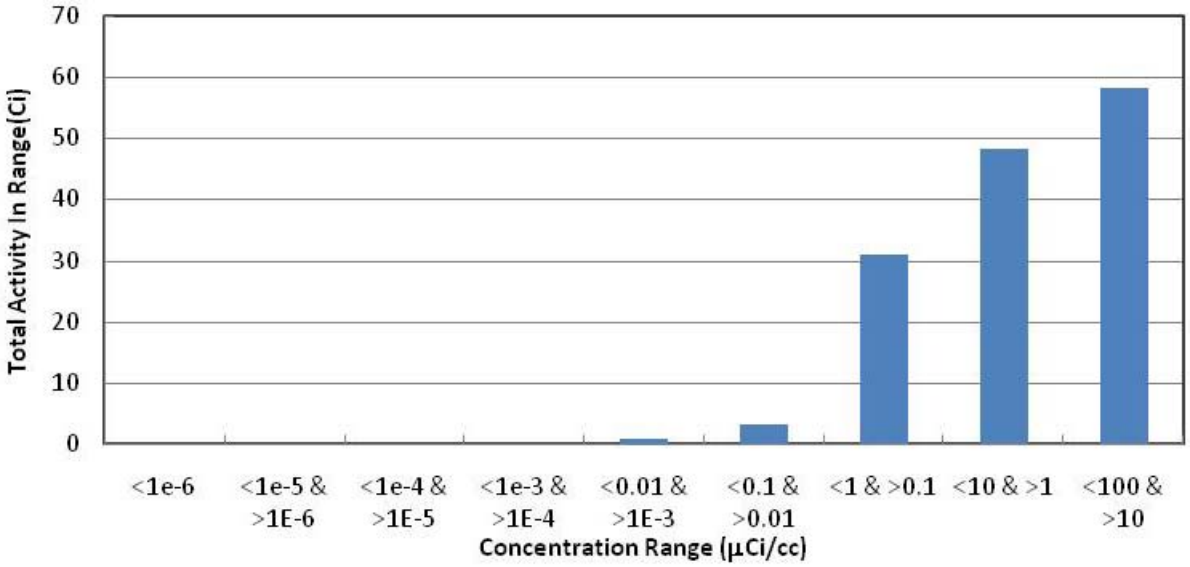

Figure 5-2
Distribution of H3 Activity by Concentration Range

While the number of high concentration packages represents a relatively small percentage of the total volume, it represents 97% of the total activity reported going to disposal sites as shown in Figure 3-2. Capping the concentration at a possibly more realistic level of 1 μCi/cc would reduce the activity total by 80%. This would leave about 18 curies total or less than 1 curie per year per 1000 MWe. Even if the concentration was capped at 10 μCi/cc the reported activity disposed in solid waste would be reduced by nearly 60%.

How Much Is Really a Problem?

H-3 activity does not figure significantly in any intruder scenario or long term exposures associated with LLRW disposal. (4) The disposal limit for H-3 in class A waste is 700 Ci/m3. There is no disposal limit for H-3 in Class B or C waste. (2) Average concentrations in power plant wastes typically are less than 0.01 Ci/m3. (11)

Can It Be Generically Determined?

As noted above, most plants do not measure H-3 directly, rather, the most common practice is to refer to the measured concentration in reactor coolant and estimate the moisture content of the waste. Some plants may continue to use scaling factors to estimate H-3 although this has never been advocated. What is more likely is that H-3 is being scaled up in dose to activity analysis along with the entire spectrum of radionuclides. Since H-3 doesn't (ever) significantly impact classification, little attention is paid to the result. In any event the H-3 concentration should never exceed the reactor coolant concentration.

The above discussion is centered on process wastes from PWR power plants. These wastes would be the most likely to contain important quantities of H-3. As we have seen, the overall amounts of H-3 that could feasibly be envisioned in LLRW is negligible and would never figure significantly in disposal site performance assessment.

Regulatory Considerations for Tritium

Because of the potential for tritium rich wastes from other non nuclear power plant generators, it is appropriate to maintain a reporting requirement for tritium. In the context of NPP wastes, H-3 is not a classification determinant and since wastes in liquid form are restricted- there is little potential for H-3 to be a factor if reasonably characterized.

Carbon-14

Carbon-14 (C-14) has a relatively long half life of 5730 years, but low decay energy of 156 keV. (13) C-14 is not reliably detected in radiochemical analysis of reactor waste streams. It is widely used in research which makes it a significant radioactive constituent of waste from academic institutions. One of difficulties with C-14 is the expected chemical form. Most nuclides are presumed to be present in oxide form; however, oxides of carbon are gases. If carbon is present as methane, a similar dilemma exists because gases are not expected to behave reliably in solid wastes, even resins or filters. (17) (18) The behavior in the disposal cell is equally uncertain. (19) All of these features contribute to the uncertainty in C-14 waste content data and pathway impacts. In addition, it is not a large dose contributor.

Production of Carbon-14

Estimates of annual production of C-14 in light water reactors are on the order of 22 Ci/yr in fuel, and 5.6 Ci/yr in BWR coolant and 2.3 Ci/yr in PWR coolant based on 1000 MWe plants. (15). Of this generation a significant amount is partitioned to the gaseous release pathways, perhaps more than 90% since C-14 is expected to be in dominantly volatile forms. (19) Following the publication of 10CFR61, it was noted that estimated risks from the disposal of LLRW were dominantly from C-14. These risks were traced to numerous conservative assumptions resulting from the "poor state of knowledge of C-14 waste forms and environmental transport" (20). In 1990, EPRI organized a workshop ("C-14 Workshop") to investigate issues with C-14. (21) The workshop discussed inventory, waste characteristics in the disposal site, leaching rates from the waste, and the partitioning of releases between liquid and gaseous phases. Based on those discussions, it was observed in the workshop that reasonable projections of C-14 generation had been made.

Measurement of Carbon-14

The most widely used method for measuring C-14 at concentrations typical of nuclear power plant waste streams is liquid scintillation counting. Sensitivities as low as 10^{-4} Bq/gm have been reported. (17)

Annual Generation

Based on the results of our survey of shipped LLRW gross annual quantities of C-14 shipped to disposal are on the order of 50 and 30 Ci/yr per 1000 MWe more for BWRs and PWRs, respectively. Comparable solid waste release rates reported in the C-14 workshop were 0.7 Ci/yr for a PWR and 0.5 Ci/yr for a PWR. (11)

How Good Are Our Estimates?

Assuming that we understand the generation rates and cross-sections associated with C-14, reported solid waste sources of C-14 account for an order of magnitude more C-14 than what is expected to be actually generated. As in the case of H-3, the average is significantly impacted by individual reports of substantial quantities.

How Much Is Really a Problem?

The Class A disposal limit for C-14 is 0.8 Ci/m^3. Using a source estimate of ~1 Ci/GW- year, this would result in a disposal site concentration of 0.03 Ci/m^3 (before any dilution from mixing with soil or grout filler or structural displacements). The 20 year generation of C-14 would result in approximately 1080 curies. Although the actual generation rate is higher than originally assumed, the overall C-14 in NPP wastes constitutes a small fraction of the overall C-14 when non-utility wastes are included in the mix. Based on activities currently reported, C-14 in NPP wastes does not correspond to a significant risk in either the "basis" intruder scenario or in any long term scenario. If results are scaled to actual generation, the concentration would be less than 1% of the Class A limits and C-14 could be considered insignificant under NRC's criteria for other nuclides. (22)

Account for Overestimate

Although the reported generation rates are higher than what is expected, it is believed that the concentrations are over-reported. There are two factors contributing to the over-estimation of C-14, one is the lack of a consistent key radionuclide for defining a scaling factor. Since C-14 is dominantly produced from dissolved nitrogen in reactor coolant, and has volatile components, there is no mechanism for correlation between it and other radionuclides. Scaling factors will always be overly conservative. Furthermore, if dose to activity methods are used in conjunction with scaling factors, C-14 could also be significantly increased in particular packages without a justifying scaling basis. (11)

Regulatory Considerations for C-14

It is appropriate to limit the disposal of C-14 in LLW. However, the requirement for mandatory reporting imposed by 10CFR20 may not be necessary because the overall C-14 in NPP wastes constitutes a small fraction of the overall C-14 when non-utility wastes are included in the mix. Based on activities currently reported, C-14 in NPP wastes does not correspond to a significant risk in either the "basis" intruder scenario or in any long term scenario. Even with the use of overly conservative scaling factors, C-14 accounts for less than 1% of the Class A limit. Therefore, it should be appropriate to not list C-14 if less than 1% of the Class A limit. Mandatory reporting of this radionuclide represents an unnecessary, costly burden.

Technetium-99

Technetium-99 (Tc-99) has a long half life of 213,000 years and has a short lived precursor (meta-stable Tc-99m) that is widely used in nuclear medicine. It has a decay energy of 293 KeV. Tc-99 is expected to be mobile in the disposal facility environment but is not a large dose contributor to the analysis. The short lived form used in nuclear medicine doesn't contribute substantially to Tc-99 inventory due to the longer half life of the daughter. (15) Data on Tc-99 in reactor waste streams show that it is often not present at levels that can be detected by radiochemical analysis. As a consequence, the reported quantities are most often based on minimum detectable levels. This decreases the reliability of the database for determining the waste inventory. (11)

Production of Technetium-99

Tc-99 is formed both as a fission product and as an activation product. Its primary parent nuclide is Mo-99. Mo-99 decays directly to Tc-99 11.4% of the time and to Tc-99m 86.6% of the time. Tc-99m converts to Tc-99 by isomeric transition. Fission product precursors to Mo-99 include: Rb-99, Sr-99, Y-99, Zr-99, Nb-99, Mo-99, as well as, Tc-99m. Mo-99 has the longest half-life of the precursors with a 2.7477 day half-life. Elemental molybdenum (24.75% Mo-98) is a common contaminant in stainless steels and is sometimes used as an additive to promote strength in steels. (15)

Measurement of Technetium -99

Since Tc-99 is a pure β-emitter, the most common method of measurement is by direct β counting using a thin window gas flow proportional counter. The major alternative method is liquid scintillation counting (LSC). A more sensitive method is thermal ionization emission mass spectrometry (TEMS). Approximate sensitivities for these methods are: β-proportional counting 0.02 Bq/gm; LCS - < 0.5 Bq/gm, TEMS - 6 x 10^{-4} Bq/gm. Sensitivities for radioactive measurements are based on minimum interference. (23)

Annual Generation

The total annual generation of fission product Tc-99, based on ORIGEN2 calculations, is 246 Ci/year in a PWR and 348 Ci/year in a BWR. If a fuel failure rate of 0.25% is assumed the amount of fission product Tc-99 available for release each year through all pathways is 0.62 and 0.96 Ci for PWRs and BWRs, respectively. The remainder is retained in the fuel. (15) To put these activities in perspective, if all of the Tc-99 available for release was limited to PWR primary resins and BWR reactor waste cleanup resins, the corresponding concentrations would be well within the Class A limits. If the activity was averaged over the entire stream, the effective concentration would be less than 1% of the class A limit or low enough to obviate the need for reporting it (except as required by 10CFR20).

How Good Are Our Estimates?

Industry estimates of Tc-99 are based principally on scaling factors derived from detection limit values. Because of the requirement to report Tc-99, values go into inventory that has little relation to the actual activity. While Tc-99 is reported with relative frequency in laboratory data, the values reported have high uncertainties and generally don't appear particularly more reliable than the LLDs. Furthermore as the overall activity of the sample goes down, the detection of Tc-99 is reduced and the resulting scaling factors increase. Using data from the 1992 EPRI sample data compilation, industry average scaling factors for Tc-99/Co-60 were determined for representative waste groupings. These were used together with average plant annual Co-60 generation rates to estimate annual Tc-99 release rates in LLRW. (11) Values compiled are listed in Table 5-1.

Table 5-1
Estimated Technetium-99 Release Rates in LLRW

Stream	PWR		BWR		Source
	Tc/Co	Tc-99 (Ci)	Tc/Co	Tc-99 (Ci)	
High Activity Resins/Media	1.72E-05	6.9E-03	5.44E-06	3.79E-02	Primary/RWCU
Low Activity Resins/Media	3.50E-04	7.2E-03	3.50E-05	1.23E-02	Radwaste Resins
High Activity Filters	1.35E-05	7.9E-05			Letdown Filters
Low Activity Filters	1.77E-04	7.8E-04			Radwaste Filters
DAW	5.00E-04	5.1E-02	1.02E-04	4.94E-02	DAW
Totals		6.6E-02		1.0E-01	

It is interesting to observe that the scaling factors are higher in the lower activity wastes. DAW which represents the lowest overall activity, but has relatively high volume appears to contain more than half of the overall Tc-99 in BWRs and almost 80% in PWRs. Since (according to the NRC disposal model (4)) DAW is subject to the least restrictive disposal, unstabilized with minimum cover, we are left with the illusion of placing the largest inventory of Tc-99 with the least protection. Since Tc-99 is inventory limited, if the result was to be believed it would be contrary to the intent of the regulation. That being said, it needs to be recognized that this is not actually the case and that the overall quantities of Tc-99 are negligible to begin with. It should also be noted that these estimates don't reflect the substantial over-reporting that occurs through the process. Figure 5-3 shows Tc-99 activities reported by BWR and PWR power plants. (11) The bulk of the values are well within the expected generation rates. As noted above, expected values are less than 0.1 curie per year. In Figure 5-3, 52 of 162 total reports exceed the expected report values. The highest reported values ranging from 10 to 60 curies for an individual plant report. In fact the top five entries represent five times more activity than all of the remaining entries combined.

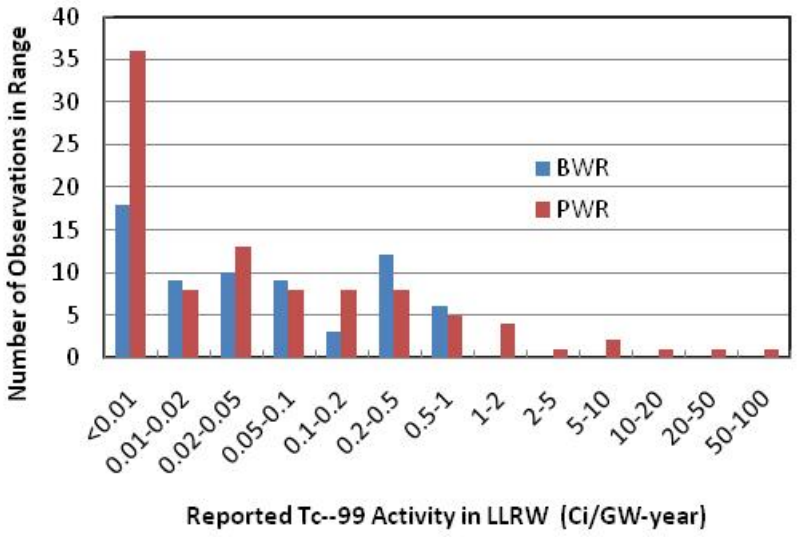

Figure 5-3
Reported Tc-99 Activity in LLRW

Furthermore, studies sponsored by EPRI in 1992 where mass spectrometry measurements were made using resin samples collected through controlled experiments at BWR and PWR power plants demonstrated that the actual scaling factor for Tc-99 ranged from a factor 100 to 1000 lower than those routinely derived from radiochemical methods. In other words, the scaling factors that are being used are a factor of 100 to 1000 greater than actual. (24)

How Much Is Really a Problem?

Despite the high priority implied for the determination and reporting of Tc-99 in power plant LLRW, there is no evidence that sufficient amounts are generated to present a short term or long term hazard to the general public. Conventional methods of determination result in substantial over-reporting and false impression that the dominant risk is in the least protected disposal. Mandatory reporting of Tc-99 should be based on source term estimates or generic scaling factors derived from mass spectrometric measurements.

In our current scenario, we have 100 plants operating producing about 1 million cubic feet per year of LLRW. The Class A disposal limit for Tc-99 is 0.3 Ci/m3. To approach this concentration in the disposal site the streams would have to carry more than 8000 Ci/per year or about 80 curies per plant. Even under the worst fuel conditions, with all of the Tc-99, being released through solid waste, average concentration would not exceed the minimum reporting level set by the NRC (i.e. 1% of the Class A limit).

Regulatory Considerations for Tc-99

It is appropriate to limit the disposal of Tc-99 in LLW. However, Tthe requirement for mandatory reporting imposed by 10 CFR 20 may not be necessary because there is no evidence that sufficient quantities are generated to present a short term or long term hazard to the general public. It should be appropriate to not list Tc-99 if the concentration is less than 1% of the Class

A limit. Additionally, for Tc-99, if there is a history of low concentrations in individual waste streams, it should not be necessary to continue efforts to measure it in waste streams not directly tied to reactor coolant.

Iodine-129

Iodine-129 (I-129) has a long half life of 15.7 million years and low decay energy of 150 keV. (13) Similar to the situation with carbon, iodine is expected to exhibit significant volatility. It is expected to behave similarly to I-131 which is easily detected by gamma spectroscopy. I-131 is routinely released in gaseous reactor effluents, but rarely reported in solid reactor wastes due to its short half-life. Production of I-129, on an atom basis, is approximately one fifth that of iodine-131 from fission of U-235. This, coupled with its long half life, accounts for the low activity levels of I-129. Similar to the situation with T-99, I-129 is often not present in reactor wastes at levels that can be detected by radiochemical analysis. (15) As with Tc-99, this generally results in the use of detection limits to determine reactor waste activity content and systematic over-reporting. In addition, the pathway analysis indicates that the dose contribution from I-129 is low. In NUREG/CR-1759, a discussion is provided on the I-129 pathway dose conversion factor (PDCF) and the effect of atomic ratio of stable I-127 and I-129 in the environment. It states that at levels reported in environmental studies in the vicinity of nuclear facilities, it is not possible to exceed thyroid dose guidelines. It then states that the whole body dose may be the limiting factor for I-129 and advises that the PDCF should be used judiciously. (25) Effectively, the impacts analysis applied the whole body dose limit to in place of the organ dose limit which increased the apparent impact of I-129 by a factor of 3.

Production of Iodine-129

I-129 is formed primarily as a fission product. Some additional formation can occur as a result of neutron capture by stable fission product Tellurium 128. Due to its long half life I-129 does not reach equilibrium during the fuel cycle. Precursors to I-129 include In-129, Sn-129, Sb-129, and Te-129. All precursors have relatively short half lives. Te-129 has the longest at 33.4 days. Other nuclides have half lives on the order of minutes. (13)

Measurement of Iodine-129

Low level gamma counting using a well type, high resolution germanium detector in a low background anticoincidence shield can achieve a sensitivity of .01- .1 Bq/gm, assuming that background activity is sufficiently low so as to not interfere with the I-129 peak. Similar results can be expected with β counting by liquid scintillation. The more aggressive methods including neutron activation analysis and mass spectrometry methods can achieve detection limits on the order of 10^{-8} Bq (26).

Annual Generation

As noted above I-129 is generated primarily as a direct result of fission. Total annual production per 1000 MWe, is about 0.83 and 0.68 Ci/yr for a BWR and PWR power plant respectively. Assuming the fuel failure is limited to 0.25% (usually the limiting operating condition) we should expect the release to reactor coolant to be on the order of 0.002 Ci/year. Since there is no reliable measurement of I-129, it is usually scaled to Cs-137 (if Cs-137 is measured) on the basis of limit of detection value. Both cesium and iodine are considered to be mobile in aqueous environment with iodine being more volatile. It is expected, and readily observed that in fuel failure events based on the presence of short-lived iodine isotopes in reactor coolant that iodine is released proportionately with cesium. Because of its relative volatility, iodine is more likely to be depleted relative to cesium through gaseous releases. (15) Figure 5-4 plots the production ratio of I-129 to Cs-137 versus fuel exposure for a PWR power plant. The ratio increases slowly with time but falls into the range between 2.5-3 x 10^{-7}. Excluding gaseous losses, this would be the expected ratio in most waste streams.

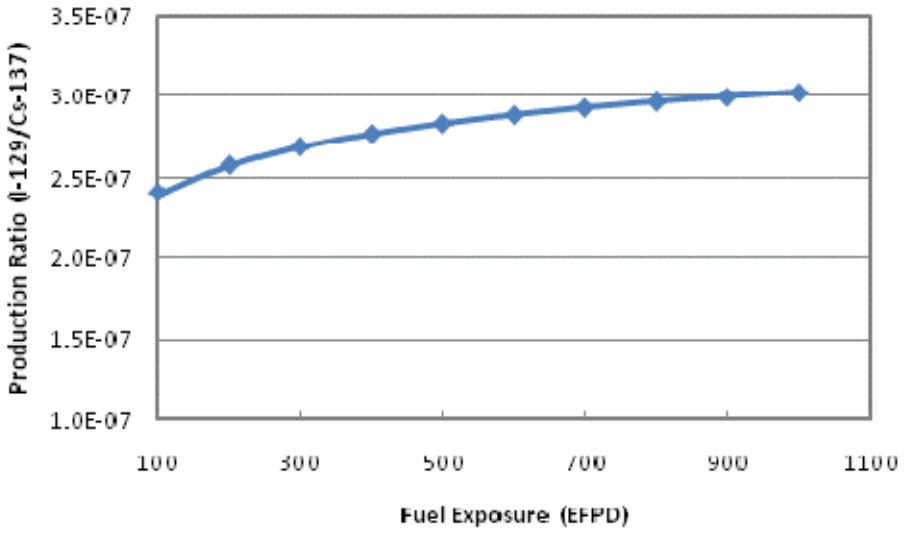

Figure 5-4
I-129/Cs-137 Production v. PWR Fuel Exposure

Studies sponsored by EPRI in the early 1990's examined the scaling factor relations between I-129, Tc-99, and C-14 with key nuclides of Cs-137 and Co-60. Test ion exchange columns were set up on side streams drawn from the primary coolant treatment systems on six PWR and four BWR power plants. The ion exchange columns were devised as scaled down versions of the primary demineralizer treatment systems for both plant types. Resins collected from these tests were then analyzed by Battelle Pacific Northwest Laboratories using plasma mass spectroscopy with a sensitivity about 10,000 times that of radiochemical methods. (12) Scaling factors observed I-129/Cs-137 were on the order of 1.1E-7 for PWRs and 9.5E-7 for BWRs. These ratios were about 1000 times lower for PWRs and 100 times lower for BWRs than those estimated in prior examinations of radiochemical data. Primary resins account for the bulk of radioactivity in process wastes included in LLRW and these scaling factors would generally

bound those in other lower activity streams. If we apply these scaling factors to the gross annual cesium 137, the total expected I-129 in LLRW (including all 100 plants) would be on the order 1 mCi/year. (12)

In the NRC Performance Assessment Working Group report (PAWG), NUREG - 1573, it was noted that pursuant to results shown in disposal records annualized by the NRC (27) (Roles report) there tended to be more very long-lived activity in class A wastes as stated below.

> *The Final Environmental Impact Statement (FEIS) for Part 61 (NRC, 1982) clearly recognized that in time a disposal site's natural characteristics must be relied on to isolate waste. A later study of LLW disposed of in the United States from 1987 through 1989 (Roles, 1990) shows that although most of the activity in initial waste inventories resides in Class C waste, Class A waste typically contains the largest quantity of long-lived radionuclides (radionuclides with half-lives greater than 100 years). This means that within about 1000 years after disposal, the higher activity short- and intermediate-lived radionuclides of B/C Class waste will have decayed to the point where most of the remaining activity will be from Class A waste. The remaining radionuclides in Class A waste will have such long half lives that it is unreasonable to assume that any physical barrier can be designed to function long enough to influence, through radioactive decay, the amount of long-lived radionuclides eventually available for release. (28) P.3-12*

Since Class A wastes are disposed without stabilization there was no benefit to be gained by the addition of engineered intruder barriers since this activity would persist beyond any conceivable barrier. Long term mitigation could only be achieved through site selection. Later source term studies and improved understanding of the how these radionuclides are generated have shown, reasonably so, that the conclusion drawn from the Roles report cannot be supported.

The Roles report tabulated data collected by the disposal in a monumental effort to assimilate and evaluate the industry wide waste stream source term. (27) Since our interest at this point is only in the I-129 summary values these are listed in Table 5-2. Included in the values are the sums of activities reported in Class A, B and C wastes at three operating disposal sites (Barnwell, Beatty, and Hanford. Estimated values in Table 5-2 were determined using scaling factors from Robertson (12) with Cs137 values tabulated from the Roles report (27). Effectively the reported values over the three year period are about 200 times that of the estimated values. It is also noted that that the reported I-129 values in solid wastes exceed the expected maximum release rates without taking into account releases through liquid and gaseous pathways.

Table 5-2
I-129 Disposal Values from Roles Report

Year	I-129 (Ci)*	Cs-137(Ci) *	I-129 Best Estimate (Ci)	Ratio Reported to Estimated I-129
1987	1.02E+00	7.15E+03	2.76E-03	370
1988	1.19E+00	1.41E+04	5.44E-03	219
1989	5.57E-01	1.64E+04	6.33E-03	88

*Sum of A, B and C Activities from 3 disposal sites (18)

The other issue with these numbers is the distribution of activity between waste classes. Using the same data drawn from the report and summing by class over the 3 year period we obtain the results shown in Table 5-3.

Table 5-3
Sum of Activities over 3 years by Classification (Roles Report)

	I-129			Cs-137		
Disposal Site	**A**	**B**	**C**	**A**	**B**	**C**
Barnwell	6.86E-01	1.68E-01	1.66E-01	7.50E+02	3.54E+03	2.86E+03
Richland	1.04E+00	6.22E-02	8.70E-02	5.99E+02	6.60E+03	6.89E+03
Beatty	4.57E-01	2.22E-02	7.75E-02	8.86E+02	4.29E+03	1.12E+04

Effectively, for Cs137, Class B and C wastes account for 89-95% of the total activity. For I-129, Class B and C wastes account for only 32 % of the activity at Barnwell and less than 20% at the other two sites. This shows that (according to the Roles report) one of radionuclides that we're most concerned about in relation to long term protection, whose risk is inventory based, (and the reason why we don't credit engineered barriers for waste stabilization) is actually in the unstable Class A trench. (27) This would be a concern if we hadn't already established that the I-129 estimates themselves are a factor of 1000 too high.

Regulatory Considerations for I-129

It is appropriate to limit the disposal of I-129 in LLW. However, the requirement for mandatory reporting imposed by 10 CFR 20 may not be necessary. It should be appropriate to not list I-129 if concentrations are less than 1% of the Class A limit. If there is history of low concentrations of I-129 in individual waste streams, it should not be necessary to continue efforts to measure it in waste streams not directly tied to reactor coolant.

Limits on Transuranics

Establishing Limits on Transuranics

The basic limit for transuranic radionuclides of 10 nCi/gm was set on the premise that in terms of radio-toxicity and half-life, Pu-239 is comparable to Ra-226. Since radium is widely dispersed throughout the earth's crust, it was viewed as reasonable to set a limit for transuranics approaching the upper bound of concentrations of natural radium. The value that was arrived at was 10 nCi/gm (for pure Pu-239). (29) This roughly corresponds to the radiation level associated with a locally high value of radium 226 which would result in a dose rate to an individual of ~200 mrem/year[5]. (11) It should be emphasized that this is the dose rate that would arise if all of the waste was at the concentration limit and no dilution occurred preceding exposure. Actual average concentration of transuranics are more than an order of magnitude

[5] Based on IMPACTS and RESRAD cases for Intruder agriculture.

lower and would be significantly diluted before the scenario would be realized. Given that naturally occurring radioactivity already exists in pockets that could produce this level of risk, the addition of small pockets associated with LLRW disposal would not significantly add to present or future risks. Implicit in the limit is the perspective that the limit and future exposure risks are sufficiently low that further isolation would be no longer required once short half-life radioactivity is decayed away.

The original limit of 10 nCi/gm for transuranics was set prior to the publication of 10CFR61. It was a basic criteria for waste disposed in sites operated by the NRC in the 1960's and later in the commercial sites originally licensed in the 1970's. Adherence to the value was retained in developing the 10CFR61 disposal limits although it was recognized that there was significant margin in the wastes being disposed that would allow an increase in the value that would be averaged out by other wastes in the disposal site.

Production of Transuranics

Plutonium (Atomic Number=94) is second in the series of trans-uranium nuclides formed from the neutron activation of uranium (Atomic Number=92) isotopes. Plutonium isotopes of interest in low-level waste disposal include Pu-238, Pu-239, Pu-240, Pu-241, and Pu-242 because of their long decay half-life and general abundance. Neptunium (Atomic Number=93) is formed as an intermediate isotope between Uranium and Plutonium so production of Plutonium is dependent on the neutron absorption and decay properties of Neptunium as well as Uranium. In naturally occurring Uranium there are three isotopes in general abundance. These are U-234, U-235 and U-238. Other isotopes have comparatively short half-lives and have long disappeared from original matter. In general U-234 does not contribute significantly to Plutonium production since it activates to U-235 which is already orders of magnitude more abundant than U-234. (13)

Nuclear properties for key Plutonium isotopes are shown in Table 5-4. (13)

Table 5-4
Plutonium Isotopes Nuclear Properties

Nuclear Property	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242
Half-life (a)	87.74	24,100	6560	14.35	3.76×10^5
Specific Activity (GBq/gm)	634	2.29	84.1	3831	0.14
Decay Constant (sec^{-1})	2.5E-10	9.1E-13	3.3E-12	1.5E-9	5.8E-14
Neutron Capture Cross Section (barns)	540	269	290	360	19
Resonance Integral (barns)	18	742	.05	1010	<0.2
Fission Cross-Section (barns)	18	742	0.05	1010	<2.5
Decay Product	U-234	U-235	U-236	Am-241	U-238
Q-Value (MeV/decay)	5.491	5.244	5.159	0.021	4.898

Formation of Am-241, Am-242m and Am-243

Americium (Atomic Number=95) is the third in the series of transuranic nuclides formed from the neutron activation of Uranium. Am-241 is formed during the decay of Pu-241 with the emission of a beta. Am-241 then undergoes neutron activation to Am-242 and Am-242m in the competing reactions Am-241 (n,γ)Am-242 and Am-241(n,γ)Am-242m. Am-242 decays by emission of a beta to Cm-242. Am-243 is formed through the neutron capture reaction Am-242m(n,γ)Am-243. Properties are summarized in Table 5-5. (13)

Table 5-5
Properties of Am-241, Am-242m and Am-243

Property	Value		
Nuclide	^{241}Am	^{242m}Am	^{243}Am
Half-Life	432 a	152 a	7380 a
Specific Activity (MBq/gm)	1.27E5	3.6E5	7.38E3
Neutron Capture Cross Section (barns)	63+560 b	1700 b	74+4 b
Resonance Integral (barns)	200+1300 b	200 b	1700+110 b
Fission Cross Section (barns)	3.2 b	7000 b	<0.08 b
Decay Product	Np-237 (ch)	Am-242 (ch)	Np-239 (ch)
Q-Value (MeV/decay)	5.482	0.048	5.267

Formation of Cm-243, Cm-244, Cm-245, and Cm-246

Cm-244, Cm-245 and Cm-246 are formed in multiple neutron capture and decay series from U-238 to Pu-239 to Am-243 and finally Am-244. Am-244 decays to Cm-244. Cm-245 and Cm-246 are then formed in additional neutron capture reactions. Curium does not occur naturally to any appreciable extent. Properties are summarized in Table 5-6. (13)

Table 5-6
Nuclear Properties of Cm-243, Cm-244, Cm-245 and Cm-246

Nuclear Property	Value			
Nuclide	^{243}Cm	^{244}Cm	^{245}Cm	^{246}Cm
Half-Life (a)	28.5	18.11	8500	4370
Specific Activity (MBq/gm)	1.91E+6	2.99E+6	6.35E+3	1.14E+4
Neutron Capture Cross Section (barns)	130	15 b	200 b	1.2
Resonance Integral (barns)	220	~650	100	110
Fission Cross Section (barns)	610,160	1.0, 13	210, ~790	0.2,10
Decay Product	Pu-239 (99.76%)	Pu-240	Pu-241	Pu-242
Q-Value (MeV/decay)	6.168	5.901	5.623	5.373

Because of their long half-lives Cm-243, Cm-244, Cm-245 and Cm-246 activity builds continuously over the length of exposure.

Measurement

With exception of Pu-241, plutonium isotopes are counted by alpha spectrometry, either a Frisch grid ionization chamber or a solid state surface barrier diode. Pu-241, which decays primarily by β,γ emissions, is measured by liquid scintillation counting. Reported measurement efficiencies (detection limits) range from 10^{-5} Bq in reactor coolant to 10^{-2} Bq in high activity resins). (30)

Americium and curium isotopes are separated together from other α-emitting isotopes and counted by α spectrometry, either a Frisch grid ionization chamber or a solid state surface barrier diode. Reported measurement efficiencies (detection limits) range from 10^{-5} Bq in reactor coolant to 10^{-2} Bq in high activity resins. Because to the similarity of emissions between Plutonium 239 and 240 and between Curium 243 and 244, these radionuclide pairs are reported as a single value representing the combined activity of the pair. (30)

Annual Source Term in LLRW/Release from Intact Fuel

Using the EPRI database of shipping records, total TRU activity excluding Curium 242 and Plutonium 241 were summed by year along with the associated volume to calculate average nanocuries per gram for the reporting plants. All waste streams were included so the total indicates the overall average as it would appear in the disposal site. The data is shown in Table 5-7. (11)

Table 5-7
Bulk Concentration of TRU in NPP LLRW (nCi/gm based on 1 gm/cc)

Year	BWR	PWR
2003	0.020358	0.02223
2004	0.02747	0.016837
2005	0.01784	0.013725
2006	0.046162	0.005641

Effectively, average TRU concentrations for all of the years of the survey were well below 1% of the 10CFR61 Table 1 Class A limit of 10 nCi/gm. This is below NRC specified reporting limit of (1% of the Class A limit) when taken in the aggregate.

Distribution of TRU Activity by Classification

Transuranic radionuclides, like other dominantly long half life radionuclides are progressively more difficult to measure as overall activity of the sample is reduced. The EPRI shipment database was queried to determine the total activities reported by classification. Included in the

sum were the principal Transuranics (Pu-238, Pu-239, Pu-240, Am-241, Cm-243, and Cm-244). The result showed that half of the total Transuranic activity is reported in Class A waste. Overall the concentrations of Transuranics were less than 1% of the 10 CFR 61.55 Table 1 Class A limit of 10 nCi/gm. Distribution of TRU activity by disposal class is represented in a pie chart in Figure 5-5. Concentrations of these radionuclides in the Class C wastes still averaged less than the Class A limit. Transuranics are generally not limiting in disposal classification for NPP produced LLRW.

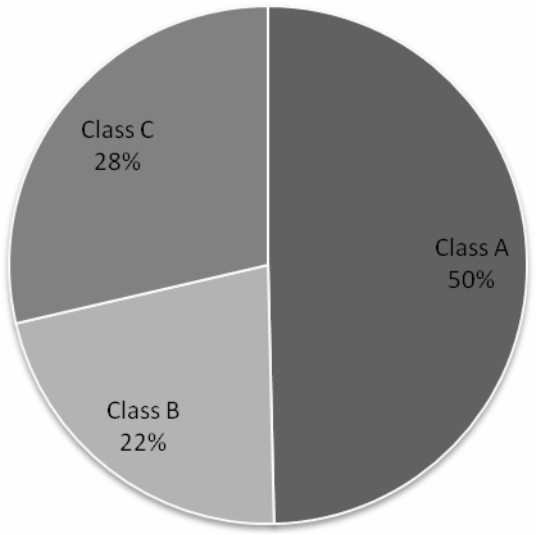

Figure 5-5
Distribution of Transuranic Inventory by Disposal Class

Long term risk associated with transuranics is dependent on the total inventory in the disposal site. (27) The observation that 50% of the inventory is in Class A waste reinforces the observation that transuranics are systematically overestimated in low activity NPP wastes. Effectively, the already acceptably low risks associated with long term disposal could be further reduced by more precise reporting.

How Good Are Our Estimates?

All of the known transuranic radionuclides fall into the general periodic table grouping of actinides. (13) Those most strongly tied to NPP LLRW (Uranium through Curium) have similar chemical properties including high electro-positivity and multiple oxidation states. Because of the electro-positivity, there is a strong potential for the atoms to bind together. Atoms removed from the fuel by whatever mechanism would tend to adhere to surfaces including fuel cladding and reactor internal surfaces. In addition, because of this electro-positivity there is always some residual contamination of uranium on the external surfaces of fuel assemblies after manufacturing. Manufacturing specifications limit the contamination on external surfaces of the fuel. Some additional uranium is present as a trace contaminant in zircaloy cladding. The radionuclides are produced in predictable proportion and nuclide ratios in various streams tend to be very uniform throughout an individual plant and generally uniform across the industry. While

it would be difficult to quantify the amounts of external fuel, it would take only a few grams of exposed fuel in each plant to produce the estimated inventory of Transuranics. (15) Improved fuel performance and operating practices since the publication of 10 CFR 61 have significantly reduced the overall release rates of Transuranics and effectively removed them as a factor in waste classification.

The principal radionuclides tracked in LLRW represent those most abundant and most readily measured. All TRU radionuclides decay through long chains to stable isotopes of lead, bismuth or polonium. (13) During the period of isolation, 10 CFR 61.55 Table 2 radionuclides continue to dominate the risk equation. (4)

Regulatory Considerations for TRU

It is appropriate to limit transuranic activities in LLRW. The historically used value of 10 nCi/gm provides an internationally accepted basis upon which risk from radioactive waste are tied to naturally occurring risks. Since wastes exceeding the disposal limit are already assigned a disposal treatment there is no need to revisit the limit. That being said, it is important to appropriately balance the the expression of risk associated with nuclear power plant generated waste with trace TRU alpha contamination with other potentially low level wastes with substantially greater alpha activity.

10 CFR 61.55 Table 2 Radionuclides

Based on the discussion above, none of the 10 CFR 61.55 Table 1 radionuclides figure significantly into disposal risk for power plant generated LLRW when considered in the context of their production rates and actual waste content. This leaves 10 CFR 61.55 Table 2 radionuclides as the primary contributors to disposal risk. There are basically five radionuclides singled out in Table 2. Of these, H-3 and Co-60 have limits far exceeding any practical activity expectation in power plant generated LLRW. Their half-lives are short enough that they decay away within the first 100 years following site closure. This discussion will focus on the remaining nuclides i.e. Nickel-63 (Ni-63), Strontium-90 (Sr-90), Nickel-59 (Ni-59), and Niobium-94 (Nb-94).

Nickel 63

Production of Ni-63

Ni-63 is formed primarily as an activation product of Ni-62 which constitutes about 3.8% of stable nickel. Nickel-63 is not formed as a fission product. The applicable formation reaction is given by: $Ni\text{-}62 + n \rightarrow Ni\text{-}63 + \gamma$. Ni-62 is a stable isotope found in stainless steel and Inconel alloys used in the reactor core. The nickel is released by surface corrosion and activated while suspended in reactor coolant in the core region. Ni-63 decays by β emission to Cu-63. No photon emissions are associated with Ni-63 decay. (15)

Measurement of Ni-63

Ni-63 is measured by direct β counting using a gas flow proportional detector or by liquid scintillation counting (LSC). Direct β counting sensitivity is ~10^{-3} Bq, and for LSC, the sensitivity reported by AECL was 0.5 Bq/g for a 2000 second counting period. Ni-63 is generally reliably measured in radiochemistry samples in stable proportions to Co-60. (31) (32)

Annual Generation

Annual generation rates of Ni-63 per 1000 MWe are on the order of 160 Ci/year in process wastes and about 1300 curies per year in activated metal wastes. These values should reasonable apply to both PWRs and BWRs. Observed ratios between Ni-63 and Co-60 in PWR operational wastes are typically higher in PWRs due to the widespread use of Inconel in steam generator tubing. This accounts for somewhat higher concentrations in the PWR LLRW. Average Ni-63 concentration, excluding activated metal is ~0.6 Ci/m^3. (15) Ni-63 can impact classification in mechanical filters but rarely goes above the Class B limit. It does not contribute significantly to classification in other waste streams. The impact of Ni-63 on mechanical filters is driven by the restrictive averaging criteria applied to mechanical filters rather than a clear correlation with risk. Of the 160 curies attributed to Ni-63, mechanical filters only account for about 1% of it. Assuming no other constraints 99% of the Ni-63 could be present in Class A waste. (11)

How Good Are Our Estimates

Activity estimates of the Ni-63 are perhaps more reliable than other DTM radionuclides included in Tables 1 and 2 of 10 CFR 61.55. Elemental nickel is present in known concentrations in parent materials. (15) The ratio of Ni-63 to Co-60 is relatively constant through all waste streams. The concentration of Co-60 is used as the primary index for estimating the concentrations of most of the DTM radionuclides. We would expect to know Ni-63 about as well as we know Co-60 which is directly measured. (11)

How Much Is Really a Problem?

10 CFR §61.55 sets a limit of 3.5 Ci/m^3 for Ni-63 in Class A Wastes. The value is increased by a factor of 10 for activated metal which is considered to be inherently stable. Both limits are increased by a factor of 20 to Class B. The limits set in 10 CFR 61 were based on analyses performed in support of the original EIS which were based on early IMPACTS studies presented in NUREG/CR-1759. An update to the IMPACTS analysis methodology was presented in NUREG/CR-4370 which reduced the fundamental dose conversion factors (DCFs) as well as parameters governing uptake factors. However, this update was not incorporated into 10 CFR 61. Using the updated analysis, limits for Ni-63 could be increased by about a factor of 15 over those listed. This would increase the Class A limit to about 40 Ci/m^3 under the conditions assumed in the regulation for Class A disposal. A detailed comparison of the calculations is provided in Appendix A.

Regulatory Considerations for Ni-63

Disposal concentration limits should be increased to reflect current ICRP dose conversion factors. Since most Ni-63 is contained in stable waste forms (i.e. activated metals), Ni-63 in other waste types can be averaged across the disposal cell.

Strontium 90

Production

Sr-90 is formed as a fission product. While Sr-90 is the most likely product at this isotopic weight, Sr-90 precursors formed include Br-90, Kr-90, and Rb-90. Nuclear properties for Sr-90 are shown in Table 5-8. (13)

Table 5-8
Sr-90 Nuclear Properties

Nuclear Property	Value
Halflife (a)	29
Yield ^{235}U	0.0591
Yield ^{239}Pu	0.0211
Thermal Neutron Capture (barns)	0.8
Resonance Integral	N/A
Specific Activity (TBq/gm)	5.1

Sr-90 decays by β emission to Y-90 (yttrium). No photon emissions are associated with Sr-90 decay. A decay diagram is shown in F**.** (33)

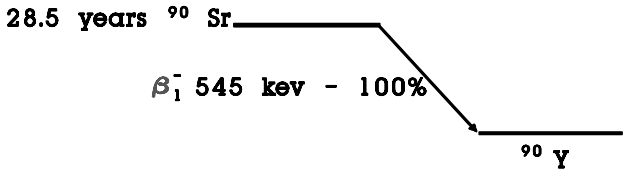

Figure 5-6
Sr-90 Decay Diagram

β energy is commonly taken as 100% 545 kev.

Measurement

Purified Sr-90 solution can be measured by direct β counting using a thin window gas flow proportional counter. An alternative method is liquid scintillation counting (LSC). Approximate sensitivities of these methods are: β proportional counting α 10^{-3} Bq/gm; LSC - ~.5 Bq/gm. Most frequently, however, Sr-90 is not measured directly. Instead, it is purified and stored to allow it to decay to equilibrium with its Y-90 daughter. The daughter can be measured by liquid scintillation or by gamma spectrometry. (34)

Annual Generation

Since Sr-90 is produced as a fission product, there is no Sr-90 production in activated metals. It is only considered in context of process wastes. It is produced in proportion with Cs-137 in fuel, however, due to its transport properties it is strongly retained within the fuel matrix and would not be expected to be released except in the case where bare fuel is exposed to coolant. Estimates of annual production based on reported disposal activities is about 0.38 Ci/year/1000 MWe and 0.32 Ci/year/1000 MWe for BWRs and PWRs respectively. (15) Overall concentrations are well below the 10CFR61 Class A limit when considered without additional dilution. Highest concentrations are found in PWR high activity resins. However, even within these wastes the weighted average concentration of Sr-90 is less than the current Class A limit.

How Good Are Our Estimates?

Sr-90 is reasonably reliably measured in laboratory samples. Scaling factors to Cs-137 are not as reliable since Cs-137 is frequently obscured or very low in gamma spectrum measurements performed on waste samples. Where cesium is not reliably measured, Sr-90 may be scaled to Co-60 which produces scaling ratios about as good as those based on Cs-137. (11)

How Much Is Really a Problem?

10CFR §61.55 sets a limit of 0.04 Ci/m^3 for Sr-90 in Class A Wastes. The limit for Sr-90 is increased by a factor of 3750 to Class B. The limits set in 10CFR61 were based on analyses performed in support of the original EIS which were based on early IMPACTS studies presented in NUREG/CR-1759. (25) An update to the IMPACTS analysis methodology was presented in NUREG/CR-4370 which reduced the fundamental dose conversion factors (DCFs) as well as parameters governing uptake factors. However, this update was not incorporated into 10 CFR 61. Using the updated analysis, limits for updated could be increased by about a factor of 7 over those listed. This would increase the Class A limit to about 0.3 Ci/m^3 under the conditions assumed in the regulation for Class A disposal. (14) A detailed comparison of the calculations is provided in Appendix A.

Regulatory Considerations for Sr-90

Disposal concentration limits should be increased to reflect current ICRP dose conversion factors. Because Sr-90 is a fission product it is only present in process wastes. Concentrations vary widely across the disposal cell. The limit for Sr-90 should, following the same argument used for setting Cs-137 concentration limit, be set by taking into account the overall dilution in the disposal cell.

Treatment of Radionuclides in Activated Metals

Activated metals produced from nuclear power operation include replacement components used for control and instrumentation in the reactor core as operational wastes, and reactor internals as decommissioning wastes. (35) Fuel assembly hardware could also be treated as operational wastes but at the present time there is no functioning industry for separating the assembly hardware from the fuel. While there is some uncertainty as to whether activated metals should be treated as an intermediate level waste (i.e. not LLRW), activated metals dominantly contain short half-life activity. Very long half-life activity including C-14 and Nb-94 are present in the base metals at trace levels and generally do not exceed limits set for near surface disposal.

Niickel-59

Production

Nickel 59 (Ni-59) is produced overwhelmingly from neutron activation of Ni-58. The high natural abundance of the target isotope, the large amounts of nickel used in reactor internals, and the relatively high capture cross-section leads to high concentrations of Ni-59 particularly in reactor internals. The applicable formation reaction is given by:

$$\text{Ni-58} + \text{n} \rightarrow \text{Ni-59} + \gamma$$

Ni-59 decays purely by electron capture to Co-59. The decays are accompanied by low energy x-rays and electrons characteristic of cobalt. Most of the transition energy is carried away by a mono-energetic neutrino. Ni-59 has exceptionally high transition energy for its half-life, 1.073 MeV. However, Ni-59 produces a continuous spectrum of gamma rays up to 1.07 MeV in about one out of every 1000 decays. (15)

Measurement

Ni-59 x-rays are measured using a high purity germanium diode equipped with a beryllium window. (31) (32)

Annual Generation

Ni-59 is produced in a nearly constant proportion to Ni-63 of around 0.0075. Because of its generally weak emissions, it is not limited in the 10CFR §61.55 except for activated metals where it can exist in significant quantity. Using the activated metal estimate for Ni-63 as the basis, approximately 10 Curies/year/1000 MWe would be expected. Ni-59 is never classification limiting in activated metals and would not be an important contributor to long term exposures in a LLRW disposal site. (11)

Regulatory Considerations for Ni-59

The disposal limit set for Ni-59 is based on chronic unshielded exposure. At the point in time where Ni-59 dominates, it should be assumed that the Ni-59 is dispersed.

Niobium-94

Production

Production of Nb-94 in fuel is generally insignificant. It is not produced directly as a fission product. The parent nuclide Nb-93 is produced through the decay of Zr-93 which results as a fission product. Nb-94 is formed by the activation of stable Nb-93. Niobium is often present as an impurity in specialized alloys for reactor internals. Natural niobium is comprised 100% of Nb-93. Niobium may also have been used as a stabilizer in certain austenitic nickel-chromium stainless materials to prevent carbide precipitation. For the most part, however, niobium is present only as a trace element ranging from about 40 to 200 ppm. Concentrations are just sufficient to approach class C limits concurrent with Ni-63 in activated metals. (15)

Nb-94 is most important for decommissioning waste due to its long half life and energetic gamma emissions. The reaction Nb-93 (n,γ) Nb-94 has a thermal cross section of 1.15 barn and a resonance integral of 8.5 barns. (13)

Nb-94 decays by beta emission with a half-life of about 2.0E+04 years to Mo-94 which is stable. The decay is accompanied by the emission of two photons. The meta-stable Nb-94m has a half life of 6.26 minutes, and converts to Nb-94. (13)

Measurement

Because of the strengths of its photon emissions and its relative abundance in most waste streams Nb-94 can be generally measured by γ spectrometric methods. In cases where the Nb-94 activity is very low or masked by other nuclides chemical purification may be necessary. Minimum detectable activity with cobalt 60 interference is ~0.15 Bq/ml.

Annual Generation

Nb-94 outside of activated metals is not included in the classification determination. There is some reporting of Nb-94 activity in these wastes but the amounts are generally not significant. In activated metals, the production is dependent on the initial elemental composition of the base metal. NUREG/CR-3474 reports 90 ppm as the average trace elemental niobium concentration in type 304 stainless steel which is the most prominent material in reactor internals. (36) Based in this concentration, Nb-94 is produced at a rate of about 0.002 x the concentration of Ni-63. Ni-63 represents a strong index since its concentration in metal is well known. Production of 1300 Ci/year/1000 MWe of Ni-63 would correspond to a production of 2.6 Ci/year/1000 MWe of Nb-94. The undiluted concentration in the disposal site would be well below the concentration limit for Class A wastes. In addition since the Nb-94 is contained in activated metal it is considered stable and therefore intruder protected for at least 300 years.

How Good Are Our Estimates?

The estimates of Nb-94 are dependent on the assumed trace metal concentration in primary materials. Most certified material test reports, if indicated at all, show niobium as a limit of detection rather than a real number. Some materials, no longer widely used, had a small constituent concentration of niobium. These materials generally do not meet the 10CFR61 concentration limits. Concentrations, as estimated, are dispersed in the material, local concentrations are typically very low and do not represent an acute dose source as might be typical with cobalt 60 or Cs-137. (11)

Regulatory Considerations for Nb-94

Currently Nb-94 is treated as a key gamma nuclide in averaging criteria. This is not appropriate since key gamma are generally used to identify radionuclides requiring shielding for handling and in short term intruder scenarios. The disposal limit set for Nb-94 is based on chronic unshielded exposure. At the point in time where Nb-94 dominates, it should be assumed that the Nb-94 is dispersed.

Summary

The 10 CFR 61 system of classification was defined primarily to address wastes from nuclear power plants, set a practical basis for disposal and provide suitable protection to the public. At the time of publication of 10 CFR 61, less than half of the current fleet of plants were in operation. Those that were in operation were still on the learning curve for improving fuel performance, and managing operations for the minimization of personnel exposures and radioactive waste generation. Limits were derived using a limited knowledge base that has grown geometrically since that time. The amounts of wastes, and the source terms for specific radionuclides that we are finding today obviate many of the assumptions that were applied in deriving the limits. In the case of the radionuclides listed in Table 1 of 10 CFR 61.55, none of them rise to the level of limiting classification in low level wastes when considered in the context of disposal site inventory. Most of them, even when regarded on a concentration basis, barely figure into the classification calculation.

Of those radionuclides included in 10 CFR 61.55 Table 2 only Cs-137 , with a 30 year half-life, emerges as contributing significantly to exposures in the disposal time frame of 500 years. The concentration limits of other radionuclides singled out in Table 2 including Ni-63 and Sr-90 have been overstated with respect to their dose conversion factors. A reassessment would result in increased limits which would diminish their overall importance which in turn would magnify the dominance of Cs-137.

There is little left today of the original bases for the disposal limits outlined in 10 CFR 61. Overall quantity estimates are much lower and, in addition, fuel performance has improved to a level that many of the radionuclides, including transuranics don't figure strongly into the disposal risk calculation based on the actual source term. The basic disposal model outlined in the 10 CFR 61 EIS has been largely abandoned for tightly engineered facilities and augmented disposal practices. Of the radionuclides listed in 10CFR61, we find that Cs-137 governs the short term and C-14 and TRU govern the long term. Limits set for the latter are sufficiently low to preclude significant public exposures without specific protection.

6
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7
ACRONYMS

AEC	Atomic Energy Commission
ALARA	As Low As Reasonably Achievable
ANDRA	Agence Nationale pour la Gestion des Dechets Radioctifs (French National Agency for Radioactive Waste Management)
BGV	Below Ground Vault
BRC	Below Regulatory Concern
BWR	Boiling Water Reactor
CDU	Canister Disposal Unit
CFR	Code of Federal Regulations
CWF	Containerized Waste Facility
CWF	Compact Waste Facility
DAW	Dry Active Waste
DCF	Dose Conversion Factor
DOE	Department of Energy
DTM	Difficult to Measure
EFPD	Effective Full Power Days
EIS	Environmental Impact Statement
EMCB	Earth Mounded Concrete Bunker
EPRI	Electric Power Research Institute
EW	Exempt Waste
FEIS	Final Environmental Impact Statement
FSAR	Final Safety Analysis Report
FWF	Federal Waste Facility
HIC	High Integrity Container
HLW	High Level Waste
HNDED	U.S. Army Engineer Division, Huntsville
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
ILW	Intermediate Level Waste
IMPACTS	NRC Computer Code for Analysis of Potential Radiological Impacts
LILW-LL	Low and Intermediate Level Waste-Long Lived

LILW-SL	Low and Intermediate Level Waste-Short Lived
LL	Low Level
LLD	Lower Limit of Detection
LLRW	Low Level Radioactive Waste
LSC	Liquid Scintillation Counting
LLW	Low Level Waste
LLWPA	Low Level Waste Policy Act
NARM	Naturally Accelerator Produced Radioactive Material
NCDU	Non-Canister Disposal Unit
NCRP	National Council on Radiation Protection
NORM	Naturally Occurring Radioactive Material
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
PAWG	Performance Assessment Working Group
PDCF	Pit Disassembly and Conversion Facility
PWR	Pressurized Water Reactor
RCRA	Resource Conservation and Recovery Act
TEMS	Thermal Ionization Emission Mass Spectrometry
TRU	Transuranic
WCS	Waste Control Specialists
WES	Waterways Experiment Station

A
10CFR61 MODEL COMPARISON FOR NI-63

Intruder Agriculture - Dose Rate Comparison		**14.8**	**Ratio Original/Update**	
10 CFR 61 Basis Limit	**3.5**	**51.7**		

Parameter Values and Formulas	Original Analysis	Updated Analysis	Parameter Description	Units (as Applicable)
Report Reference	NUREG/CR-1759	NUREG/CR-4370		
Formula	Ni-63	Ni-63		
t-1/2	100	100	halflife	Yrs
•	0.00693	0.00693	decay Constant (yrs-1)	yrs^{-1}
TDEL	100	100	Years following closure	Years
C_n	1	1	Initial Concentration	(Ci/m3)
EMP	0.75	0.75	Emplacement Efficiency	
SEF	0.88	0.88	Surface disposal Efficiency	
$H = H_{air}+H_{food}+H_{dg}$	2.0E+01	1.4E+00	Total Dose Rate	mrem/yr
$H_{w/o}$ Ingestion	2.0E-01	1.7E-01	Dose Rate Without Ingestion	mrem/yr
$H/H_{w/o}$	1.E+02	8.E+00	Ratio With/Without	
$H_{air} = \Sigma(C_n*I_{air}*PDCF-3)$	5.27E+00	4.92E-01	Dose Rate From Inhalation	mrem/yr
H_{air} w/o Ingestion	2.03E-01	1.66E-01	Inhalation without Ingestions	mrem/yr
H_{dg}	0	0	Direct Dose Rate	mrem/yr
$H_{food} = \Sigma(C_n*I_{food}*PDCF-4)$	1.51E+01	8.87E-01		mrem/yr
H_{food} w/o Ingestion	0	0		
$I_{air} = f_0*f_d*f_D*f_s$	6.6495E-12	6.6495E-12	Air Interaction Factor	
f_0	0.5	0.5	Decay Factor	Dimensionless
f_d	0.165	0.165	Site Dilution Factor	=EMP*SEF*0.25
f_D	1	1	Dispersibility Multiplier	Dimensionless

10CFR61 Model Comparison for Ni-63

Parameter Values and Formulas		Original Analysis	Updated Analysis	Parameter Description	Units (as Applicable)
f_s		8.06E-11	8.06E-11	Site Selection Factor Air Uptake (SW)	Dimensionless
$I_{food} = f_0*f_d*M_0*0.5$		0.0006105	0.0006105	Food Interaction Factor	Dimensionless
M_0		0.0148	0.0148	Radionuclide Specific Leach Fraction	Dimensionless
t_c		3.60E-05	3.60E-05	Contact time Fraction (SW Site)	Dimensionless
Define PDCF-3				Pathway Dose Conversion Factor for Chronic Airborne pathways	
C		1.00E+12	1.00E+12	Ci to pCi	pCi/Ci
$C*f_{15}*DCF2$		3.05E+10	2.49E+10	Inhalation (air)	mrem/yr/Ci
C*DCF5		0	0	Direct Radiation(air)	mrem/yr/Ci
$C*(D_1*PT+(D_2/CY)*PTP)*DCF1$		7.61E+11	4.92E+10	Food Ingestion (air)	mrem/yr/Ci
$C*D_1*f_{18}*f_{14}*f_{15}*DCF2$		1.59E+06	3.42E+03	Inhalation(soil)	mrem/yr/Ci
$C*D_1*f_{18}*DCF4$		0	0	Direct Radiation (area)	mrem/yr/Ci
$C*D_1*f_{18}*f_{14}*DCF5$		0	0	Direct Radiation (air)	mrem/yr/Ci
PDCF3		7.92E+11	7.40E+10		mrem/yr/Ci/m3
PDCF3		3.05E+10	2.49E+10	w/o Ingestion	mrem/yr/Ci/m3
Define PDCF-4				Pathway Dose Conversion Factor for Food/Soil Pathways	
C*(PT/D)*DCF1		2.48E+04	1.45E+03	Food Ingestion (soil)	mrem/yr/Ci/m3
PDCF4		2.48E+04	1.45E+03	Pathway Dose Conversion Factor	mrem/yr/Ci/m3
f_{14}		8.59E-09	8.59E-09	Resuspension Factor	m-1
f_{15}		8.03E+03	8.03E+03	Inhalation Rate	m3/yr
f_{18}		16	16	Areal Mass Available	kg/m3
PT		9.0902175	4.029292	$P_1+P_2+P_3$	kg/yr
PTP		478.4325	212.068	$P_1P+P_2P+P_3P$	kg/yr
D		1600	1600	Soil Density(kg/m2)	kg/m2
D_1	3.79E+02	3.79E+02	Soil Deposition by Fallout		m3/kg
D_2	3.58E+02	3.95E+02	Foliar Deposition by Fallout		m
CY	1	1	Crop Yield per Unit Area (kg/m2)		kg/m2

Parameter Values and Formulas		Original Analysis	Updated Analysis	Parameter Description	Units (as Applicable)
DCF1	4.36E-06	5.77E-07	Dose Conversion Factor (Ingestion)		mrem/yr per pCi
DCF2	3.80E-06	3.10E-06	Dose Conversion Factor (Inhalation(		mrem/yr per pCi
DCF4	0	0	Dose Conversion Factor External Exposure (Volume Source)		mrem/yr per pCi
DCF5	0	0	Dose Conversion Factor External Exposure (Air Immersion)		mrem/yr per pCi
$P_1 = f_1*f_2$	3.61	3.61	Soil-Plant- Man		yr-1
$P_2 = f_1*f_3*f_4*f_5$	4.78325	0.344394	Soil-Plant-Animal-Man		yr-1
$P_3 = f_1*f_3*f_6*f_7*365$	0.6969675	0.074898	Soil-Plant-Animal-Product-Man		yr-1
$P_1P = f_2$	190	190	plant-man		yr-1
$P_2P = f_3*f_4*f_5$	251.75	18.126	plant-animal-man		yr-1
$P_3P = f_3*f_6*f_7*365$	36.6825	3.942	plant-animal-product-man		yr-1
f_1	0.019	0.019	soil to plant transfer (nuclide Specific)		Dimensionless
f_2	190	190	Plant Consumption		kg/yr
f_3	50	36	rate of plant consumption by animals		kg/da
f_4	0.053	0.0053	Feed and Water to Meat		day/kg
f_5	95	95	animal consumption		kg/yr
f_6	0.0067	0.001	Feed and Water to Milk		day/liter
f_7	0.3	0.3	Consumption of Milk by Man		liter/da
V	8.00E-04	8.00E-04	Settling Velocity		m/sec
S1	4.83E-02	4.38E-02	Fract on Foliage		day-1
S2	7.60E-04	7.60E-04	Frac.Activity Deposited in Root Zone		day-1
R	0.25	0.25	Fract fallout		
Z	240	240	Mass of Soil in Root Zone		kg/m2

B
10CFR61 MODEL COMPARISON FOR SR-90

Intruder Agriculture - Dose Rate Comparison		**6.9**	**Ratio Original/Update**	
10CFR 61 Basis Limit	**0.04**	**0.3**		

Parameter Values and Formulas	NUREG/CR -1759	NUREG/CR -4370	Parameter Description	Units (as Applicable)
Formula	Sr-90	Sr-90		
t-1/2	28.1	28.1	halflife	Yrs
•	2.47E-02	2.47E-02	decay Constant (yrs-1)	yrs^{-1}
TDEL	100	100	Years following closure	years
C_n	1	1	Initial Concentration	(Ci/m3)
EMP	0.75	0.75	Emplacement Efficiency	Dimensionless
SEF	0.88	0.88	Surface disposal Efficiency	Dimensionless
$H = H_{air}+H_{food}+H_{dg}$	4.5E+02	6.5E+01	Total Dose Rate	mrem/yr
$H_{w/o}$ Ingestion	2.7E+01	1.9E+00	Dose Rate Without Ingestion	mrem/yr
$H/H_{w/o}$	2.E+01	3.E+01	Ratio With/Without	
$H_{air} = \Sigma(C_n*I_{air}*PDCF-3)$	1.81E+02	9.43E+00	Dose Rate From Inhalation	mrem/yr
H_{air} w/o Ingestion	2.71E+01	1.90E+00	Inhalation without Ingestions	mrem/yr
H_{dg}	0	7.26E-04	Direct Dose Rate	mrem/yr
$H_{food} = \Sigma(C_n*I_{food}*PDCF-4)$	2.74E+02	5.59E+01		mrem/yr
H_{food} w/o Ingestion				
$I_{air} = f_0*f_d*f_D*f_s$	1.13E-12	1.13E-12	Air Interaction	

Parameter Values and Formulas	NUREG/CR -1759	NUREG/CR -4370	Parameter Description	Units (as Applicable)
			Factor	
f_0	8.49E-02	8.49E-02	Decay Factor	Dimensionless
f_d	0.165	0.165	Site Dilution Factor	=EMP*SEF*0.25
f_D	1	1	Dispersibility Multiplier	Dimensionless
f_s	8.06E-11	8.06E-11	Site Selection Factor Air Uptake (SW)	Dimensionless
$I_{food} = f_0*f_d*M_0*0.5$	6.90E-05	6.90E-05	Food Interaction Factor	Dimensionless
M_0	9.86E-03	9.86E-03	Radionuclide Specific Leach Fraction	Dimensionless
t_c	3.60E-05	3.60E-05	Contact time Fraction (SW Site)	Dimensionless
Define PDCF-3			Pathway Dose Conversion Factor for Chronic Airborne pathways	
C	1.00E+12	1.00E+12	Ci to pCi	pCi/Ci
$C*f_{15}*DCF2$	2.40E+13	1.68E+12	Inhalation (air)	mrem/yr/Ci
C*DCF5	1.76E+09	0	Direct Radiation(air)	mrem/yr/Ci
$C*(D_1*PT+(D_2/CY)*PTP)*DCF1$	1.36E+14	6.67E+12	Food Ingestion (air)	mrem/yr/Ci
$C*D_1*f_{18}*f_{14}*f_{15}*DCF2$	1.24E+09	2.28E+05	Inhalation(soil)	mrem/yr/Ci
$C*D_1*f_{18}*DCF4$	1.66E+11	0	Direct Radiation (area)	mrem/yr/Ci
$C*D_1*f_{18}*f_{14}*DCF5$	9.07E+04	0	Direct Radiation (air)	mrem/yr/Ci
PDCF3	1.60E+14	8.35E+12		mrem/yr/Ci/m3
PDCF3	2.40E+13	1.68E+12	w/o Ingestion	mrem/yr/Ci/m3

Parameter Values and Formulas	NUREG/CR -1759	NUREG/CR -4370	Parameter Description	Units (as Applicable)
Define PDCF-4			Pathway Dose Conversion Factor for Food/Soil Pathways	
C*(PT/D)*DCF1	3.96E+06	8.10E+05	Food Ingestion (soil)	mrem/yr/Ci/m3
PDCF4	3.96E+06	8.10E+05	Pathway Dose Conversion Factor	mrem/yr/Ci/m3
f_{14}	8.50E-09	8.50E-09	Resuspension Factor	m-1
f_{15}	8.00E+03	8.00E+03	Inhalation Rate	m3/yr
f_{18}	16	16	Areal Mass Available	kg/m3
PT	3.408755	14.815245	$P_1+P_2+P_3$	kg/yr
PTP	200.515	197.5366	$P_1P+P_2P+P_3P$	kg/yr
D	1600	1600	Soil Density(kg/m2)	kg/m2
D_1	3.79E+02	3.79E+02	Soil Deposition by Fallout	m3/kg
D_2	3.58E+02	3.58E+02	Foliar Deposition by Fallout	m
CY	1	1	Crop Yield per Unit Area (kg/m2)	kg/m2
DCF1	1.86E-03	8.75E-05	Dose Conversion Factor (Ingestion)	mrem/yr per pCi
DCF2	3.00E-03	2.10E-04	Dose Conversion Factor (Inhalation(	mrem/yr per pCi
DCF4	2.74E-05	0.00E+00	Dose Conversion Factor External Exposure (Volume Source)	mrem/yr per pCi
DCF5	1.76E-03	0.00E+00	Dose Conversion Factor External Exposure (Air Immersion)	mrem/yr per pCi
$P_1 = f_1*f_2$	3.23	14.25	Soil-Plant- Man	yr-1

10CFR61 Model Comparison for SR-90

Parameter Values and Formulas	NUREG/CR -1759	NUREG/CR -4370	Parameter Description	Units (as Applicable)
$P_2 = f_1*f_3*f_4*f_5$	0.04845	0.151335	Soil-Plant-Animal-Man	yr-1
$P_3 = f_1*f_3*f_6*f_7*365$	0.130305	0.41391	Soil-Plant-Animal-Product-Man	yr-1
$P_1P = f_2$	190	190	plant-man	yr-1
$P_2P = f_3*f_4*f_5$	2.85	2.0178	plant-animal-man	yr-1
$P_3P = f_3*f_6*f_7*365$	7.665	5.5188	plant-animal-product-man	yr-1
f_1	0.017	7.50E-02	soil to plant transfer (nuclide Specific)	Dimensionless
f_2	190	190	Plant Consumption	kg/yr
f_3	50	36	rate of plant consumption by animals	kg/da
f_4	6.00E-04	5.90E-04	Feed and Water to Meat	day/kg
f_5	95	95	animal consumption	kg/yr
f_6	1.40E-03	1.40E-03	Feed and Water to Milk	day/liter
f_7	0.3	0.3	Consumption of Milk by Man	liter/da
V	8.00E-04	8.00E-04	Settling Velocity	m/sec
S1	4.83E-02	4.83E-02	Fraction on Foliage	day-1
S2	7.60E-04	7.60E-04	Fraction Activity Deposited in Root Zone	day-1
R	0.25	0.25	Fract fallout	
Z	2.40E+02	2.40E+02	Mass of Soil in Root Zone	kg/m2

C
WCS PERFORMANCE ASSESSMENT SUMMARY

Performance Assessment Dose Summary (See below for pathway descriptors). (65)

	CWF	FWF-CDU	FWF-NCDU	Maximum/ Total	Criterion
	(mrem/yr)	(mrem/yr)	(mrem/yr)	(mrem/yr)	(mrem/yr)
Normal Operations, Worker					
PathwayA1,dust	0	0	1.80E+01		
PathwayA3,gases	0	0	1.10E-03		
PathwayS1,soil ing.	0	0	1.30E-01		
External gamma(a)	2.40E+02	9.50E+01	1.10E+01		
Total	2.40E+02	9.50E+01	2.90E+01	3.60E+02	5,000
Normal Operations, Site Boundary Individual					
PathwayA1,dust	0	0	2.40E-02		
PathwayA3,gases	0	0	4.60E-06		
PathwayS3,ext rad	0	0	1.40E-05		
PathwayG1,gw red bed	0	0	0		
PathwayG2,125zone	0	0	0		
PathwayG3,225 zone	0	0	0		
PathwayW2,surf water	0	0	4.90E-04		
Total	0	0	2.50E-02	2.50E-02	25
Normal Operations, Nearest Resident					
PathwayA1,dust	0	0	1.10E-04		
PathwayA3,gases	0	0	2.10E-08		
PathwayS3,ext rad	0	0	6.40E-08		
PathwayW2,surf water	0	0	2.20E-06		
Total	0	0	1.10E-04	1.10E-04	25

	CWF	FWF-CDU	FWF-NCDU	Maximum/ Total	Criterion
	(mrem/yr)	(mrem/yr)	(mrem/yr)	(mrem/yr)	(mrem/yr)
Institutional Control, Worker					
PathwayA6,gases	9.40E-03	3.90E-01	9.80E-05		
Total	9.40E-03	3.90E-01	9.80E-05	4.00E-01	5,000
Institutional Control, Site Boundary Individual					
PathwayA6,gases	1.20E-02	1.60E-04	1.10E+00		
PathwayG1,gw red bed	0	0	0		
PathwayG2,125 zone	0	0	0		
PathwayG3,225 zone	0	0	0		
Total	1.20E-02	1.60E-04	1.10E+00	1.20E+00	25
Institutional Control, Nearest Resident					
PathwayA6,gases	5.30E-05	5.20E-03	7.40E-07		
Total	5.30E-05	5.20E-03	7.40E-07	5.20E-03	25
Post-Institutional Control, Intruder Driller					
PathwayA6,gases	3.60E-03	2.90E-03	4.50E-07		
PathwayD3,mud pit	6.70E+00	1.80E+00	1.30E-02		
Total	6.70E+00	1.80E+00	1.30E-02	6.7E+00(b)	100
Post-Institutional Control, Intruder Resident					
PathwayA6,gases	9.80E-01	8.00E-01	1.20E-04		
PathwayG3,225 zone, 0-10,000 yrs	0	0	0		
PathwayG3,225 zone, 0-100,000 yrs	5.80E-01	1.10E+00	3.40E+00		
PathwayD3,mud pit	3.00E+00	7.90E-01	1.90E-02		
Total	4.60E+00	2.70E+00	3.40E+00	4.6E+00(c)	100
Post-Institutional Control, Adjacent Resident					
PathwayA6,gases	9.80E-01	8.00E-01	1.20E-04		
PathwayG3,225 zone, 0-10,000 yrs	0	0	0		
PathwayG3,225 zone, 0-100,000 yrs	5.80E-01	1.10E+00	3.40E+00		

	CWF	FWF-CDU	FWF-NCDU	Maximum/ Total	Criterion
	(mrem/yr)	(mrem/yr)	(mrem/yr)	(mrem/yr)	(mrem/yr)
Total	1.60E+00	1.90E+00	3.40E+00	3.40E+00	25
Accidents, Worker					
PathwayA8,dropped pkg	2.40E+01	6.70E+02	n/a(e)		
PathwayA9,fire	1.60E+02	1.80E+02	n/a(e)		
PathwayD1,ext rad	8.90E-01	8.90E-01	n/a(e)		
Total	1.6E+02(f)	6.7E+02(f)	n/a(e)	6.7E+02(f)	5,000
Accidents, Site Boundary Individual					
PathwayA8,dropped pkg	2.20E-02	2.50E+00	n/a(e)		
PathwayA9,fire	2.20E+00	2.50E+00	n/a(e)		
PathwayD1,ext rad	7.50E-03	7.50E-03	n/a(e)		
Total	2.2E+00(f)	2.5E+00(f)	n/a(e)	2.5E+00(f)	100
Accidents, Nearest Resident					
PathwayA8,dropped pkg	3.40E-06	3.80E-04	n/a(e)		
PathwayA9 fire	4.00E-03	4.50E-03	n/a(e)		
Total	4.0E-03(f)	4.5E-03(f)	n/a(e)	4.5E-03(f)	100

Notes:

(a) Average over all worker types, from Appendix 8.0-4, Worker Doses.

(b) Driller may drill at any of the three facilities, but only one inadvertent intruder is assumed.

(c) Inadvertent intruder resident may locate over any of the three facilities, but only one intruder is assumed.

(d) Accident severity to workers could be mitigated by requiring respirator while handling DU-oxide packages.

(e) No waste packages in FWF-NCDU. Impacts are bounded by the FWF-CDU accident.

Pathway Descriptors

A1	Airborne dust from open bulk waste cell
A3	Airborne gases from waste cell (H-3, C-14, Kr-85, I-129, radon)
A6	Gas emanation through finished cover (H-3, C-14, Kr-85, I-129, radon)
A8	Air releases associated with a dropped, breached container
A9	Air releases associated with a truck fire
S1	Worker inadvertent soil ingestion
S3	External radiation from off-site soil (contaminated by dust deposition)
G1	Leaching and groundwater transport through red beds to a well screened above the red beds
G2	Leaching and groundwater transport through the 125-foot zone to a well screened above the red beds
G3	Leaching and groundwater transport to a well screened in the 225-foot water-bearing zone
W2	Surface water transport of ground-deposited dust to a low-lying area
D1	External exposure from high activity waste packages during operations
D3	External exposure to inadvertent intruder well mud pit

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