EPRI

Keywords:
Radioactive waste management
Low level radioactive wastes
Waste characterization
Waste classification
Scaling factor determination

EPRI TR-107201
2691
Final Report
November 1996

Low Level Waste Characterization Guidelines

Prepared by
Low Level Radioactive Waste Characterization Guidelines Committee

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Low Level Waste Characterization Guidelines

TR-107201
2691

Final Report, November 1996

Prepared by
Low Level Radioactive Waste Characterization Guidelines Committee

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LOW LEVEL RADIOACTIVE WASTE CHARACTERIZATION GUIDELINES COMMITTEE

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REPORT SUMMARY

Proper characterization of all low level radioactive waste streams is an essential element of an effective waste management program. This report presents an overview of the waste characterization process, with emphasis on representative sampling, the use of laboratory data to develop scaling factors, determination of radionuclide content and waste classification, and future radwaste manifest requirements. These guidelines will direct utilities through the characterization and documentation process.

Background

NRC regulation of low level waste (LLW) through 10CFR61 involves complicated rules. More than 14 years of utility compliance experience has resulted in a mass of characterization data, ways to use these data effectively, and practical experience in characterization processes. EPRI assembled a committee to develop LLW characterization guidelines that would summarize industry knowledge and experience and make it readily available to utility radwaste managers for use in implementing or improving their LLW programs.

Objectives

- To provide procedural guidance for characterizing LLW, including practical and effective methods of characterization resulting from industry experience.
- To reduce utility costs of performing LLW characterization.

Approach

A committee of 22 nuclear utility representatives and industry experts, developed a guidelines document to provide defensible, cost-effective methods for characterizing properly packaged low level radioactive wastes.

Results

The key elements of the guidelines include

- Discussion of regulatory requirements, summarizing 10CFR61, radioactivity concentration averaging, the uniform shipping manifest, and DOT shipping requirements

- Description of the basis for characterization, explaining why generators must characterize waste and how results can impact disposal facility capacity
- Identification of waste streams and sampling requirements, including industry experience in sampling and classification
- Explanation of the use of radionuclide and waste stream trending analyses, statistical and calculational methods for establishing defensible scaling factors, and practical techniques for characterizing the final waste packages
- A summary of industry experience in applying scaling factor and sample assay results to the characterization and classification of packaged waste, including dose-to-curie methods, direct-assay methods, and the use of computer codes
- Guidance on maintaining supporting documentation and records of waste characterization

EPRI Perspective

These guidelines provide an approach to characterizing properly packaged waste from plant streams and evaluating existing waste characterization programs or implementing a comprehensive characterization program. Key aspects of the guidelines include the identification of data requirements throughout the characterization process and the use of constant or generic scaling factors to improve accuracy of reporting and decrease the cost of sample analysis. Equipped with these guidelines, utilities can avoid over-reporting of key nuclide activities and make informed decisions that take into account ultimate management of the waste. This guideline document is one of several EPRI has developed on LLW management over the past three years, including Solid Low Level Waste Management Guidelines (TR-104583), Mixed Waste Management Guidelines (TR-103344), Mixed Waste Characterization Guidelines (TR-104401), and Interim On-Site Storage of Low Level Waste: Guidelines for Extended Storage (TR-106925). This document expands on other EPRI publications, including TR-104583 and TR-104401 as well as NP-7386, Radwaste Desk Reference Volume 1: Dry Active Waste.

Interest Categories
Low level radioactive waste management
Radiation protection technology

Keywords
Radioactive waste management
Low level radioactive wastes
Waste characterization
Waste classification
Scaling factor determination

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ABSTRACT

Soon after the implementation of 10CFR61 in 1982, EPRI and other groups provided guidance for utility radwaste classification programs supplementing that given by the NRC. In the intervening 14 years, most plants have changed radwaste personnel, the NRC issued changes in regulations and technical positions, industry has developed new waste processing and stabilization methods, a mass of waste characterization data has accumulated, and utilities have gained practical experience in characterization issues and identified additional issues.

To summarize the changes and additional experience in waste classification, EPRI instituted a radwaste project to issue Low-Level Radioactive Waste Characterization Guidelines. Developed by a committee of 22 utility specialists, the guideline goals are: 1) to reduce over reporting of key nuclide activities; 2) reduce costs of waste characterization; 3) improve reporting consistency among generators; and 4) provide a summary of up-to-date practical and defensible utility practices in low level waste characterization. The Guidelines are for use by plant personnel to support LLW characterization efforts. They provide a technical basis for the use of generic scaling factors for a variety of nuclides in a number of waste streams. Use of constant or generic scaling factors discussed in the document will improve the accuracy of reporting while decreasing the cost of sample analysis. The report also contains a considerable reference list and bibliography, and appendices that detail various aspects in waste characterization, including some analytical procedures that could be applied to waste samples.

The Guidelines Document describes the basis for the regulation of 10CFR61 and the requirements of the regulation itself along with the ancillary NRC and DOT regulations and technical positions. It has sections on representative sampling of different waste streams and on issues and experiences of sample analysis. An important section is on the use of laboratory data to develop scaling factors, to identify operational waste streams (i.e., those with common scaling factors), and to identify and to verify constant or generic scaling factors and significantly reduce subsequent sampling and analyses for those factors. It contains discussions of practical experience in use of the data, in determination of radionuclide content and classification of packaged waste, in future radwaste manifest requirements and in record keeping, including Quality Assurance needs.

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INTRODUCTION

BACKGROUND

After the U.S. Nuclear Regulatory Commission (NRC) detailed requirements for characterization of low-level radioactive waste (LLW) in 1982, implemented in 1983, nuclear utilities instituted programs to characterize their wastes intended for disposal. EPRI and other industry groups provided guidance for these programs that supplemented that provided by the NRC. Since 1982, nuclear utilities have gained considerable experience in handling radwaste that resulted in solutions to some characterization issues and identification of additional questions. Most plants had changes of plant radwaste personnel, the NRC issued changes in regulations and guidance for classification and the industry developed new waste processing and stabilization methods.

EPRI assembled a committee of practiced utility and contractor radwaste personnel to develop a LLW Characterization Guidelines Document that would summarize industry knowledge and experience and make it readily available to radwaste personnel and managers as guidance for implementing or improving their own programs.

PURPOSE AND OBJECTIVES

The purpose of these Guidelines is to provide the nuclear power industry with practical information for implementing an effective plant program to properly characterize LLW streams. This document is intended to:

- reduce over-reporting;
- improve consistency among generators;
- reduce costs;
- justify a defensible approach for low level waste characterization;
- consider new technologies for waste reduction; and,
- facilitate transition to the Uniform Manifest.

USING THESE GUIDELINES

These Guidelines are written for use by plant personnel in support of LLW characterization efforts. The intended audience for this document includes those individuals responsible for writing and auditing LLW characterization programs at the plants and includes: Radwaste Managers; Chemistry Managers; and Health Physics Managers.

The Guidelines document is structured according to the basic elements of characterizing LLW at a nuclear utility. Figure 1-1 shows a flow diagram of the processes relating to characterizing LLW. References to individual report sections that deal with these specific issues are noted on the figure. Key elements of individual report sections are outlined as follows:

Section 2 - Regulatory Requirements: Summarizes what generators "have to do" relative to LLW characterization. The summary discusses: 10CFR Part 61; the NRC Branch Technical Position on *radioactivity concentration averaging;* amendments to 10CFR parts 20 and 61 regarding the *uniform shipping manifest,* and 49CFR parts 171-177 on DOT shipping requirements.

Section 3 - Basis for Characterization: Describes why generators must characterize waste and how results can impact disposal facility capacity; relates characterization to disposal facility operating life; and identifies "critical" radionuclides for characterization. A summary of the population radiation exposures that are the bases of the NRC regulations and technical positions regarding the radionuclide concentrations and total activity in disposal sites.

Section 4 - Identification of Waste Streams: Identifies the types of waste generated at a nuclear utility and presents an overview of waste streams addressed in this document. And summarizes practical experience at operating plants in defining waste streams for purposes of waste classification.

Section 5 - Laboratory Considerations: A guide, gained from experience, of considerations for selecting and contracting with a vendor laboratory for analyzing waste stream samples. It includes guidance on sampling and analysis quality control and assurance important for reliable results.

Section 6 - Representative Sampling: A summary of industry experience and practices in the difficult task of obtaining samples for analysis that are representative of the waste stream contents.

Section 7 - Use of Laboratory Data: A summary of experience in:

- validating laboratory results;

- selecting applicable scaling factors;
- use of radionuclide and waste stream trending analyses;
- scaling factor updating;
- waste stream identification and consolidation;
- plant and waste stream constant scaling factors;
- industry generic scaling factors; and,
- use of computational modeling for scaling factor determination.

Section 8 - Determining Curie Content: A summary of industry experience in applying scaling factor and sample assay results to the characterization and classification of packaged waste, including that returned from waste processors. It includes discussion of dose-to-curie methods, direct-assay methods and the use of computer codes.

Section 9 - Waste Manifest: A summary of information requirements and changes to the manifesting process as a result of adopting the uniform waste manifest.

Section 10 - Record Keeping: Guidance on maintaining supporting documentation and records of waste characterization.

This Guidelines document pertains to the radiological characterization of LLW. For guidance in characterizing the hazardous component of a potential mixed waste, refer to EPRI's *Mixed Waste Characterization Guidelines* (EPRI TR-104401).

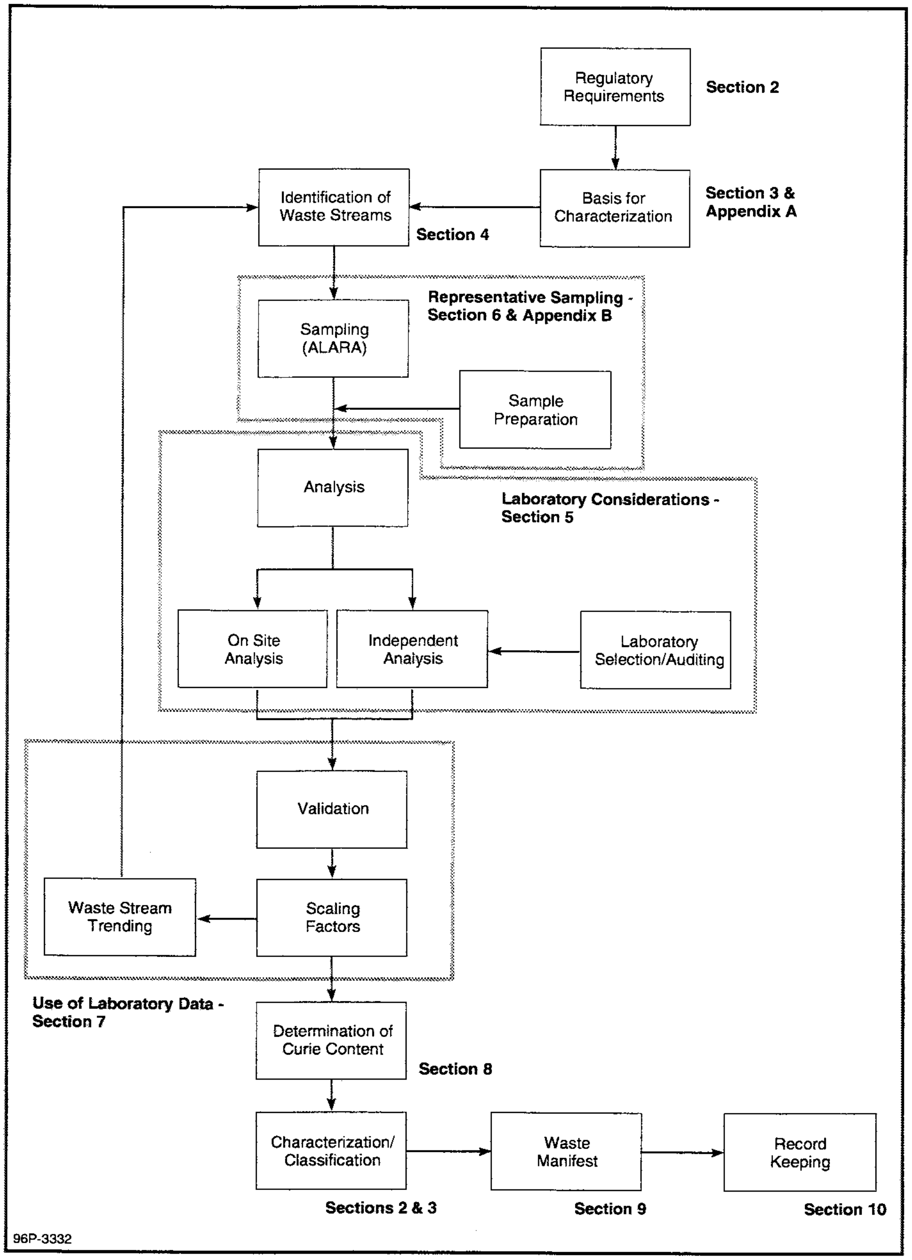

Figure 1-1 LLW Characterization Flow Diagram

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REGULATIONS RELEVANT TO LLW CHARACTERIZATION

The Nuclear Regulatory Commission (NRC) places requirements and guidance on characterizing and classifying packaged low-level radioactive waste (LLW). Additionally, the U.S. Department of Transportation (DOT) sets requirements on the radioactivity within and the radiation from packaged wastes shipped either to processors or for disposal. These regulations have evolved over the years and changes to the regulations will continue, as State Compacts ready their disposal sites.

This section summarizes the current regulations and guidance that are relevant to characterizing and classifying low-level waste for disposal. Many of these requirements have been in effect since 1983 and are generally familiar to utility radwaste managers and operating personnel. Others are relatively new or are scheduled for future enactment. Some new guidance clarifies or supersedes prior requirements. Five primary documents are discussed as most relevant to the waste classification process. They are:

1. CFR Part 61, ***Licensing Requirements for Land Disposal of Radioactive Waste***, published in the Federal Register 27 December 1982 and effective 27 December 1983,
2. ***Low-Level Waste Licensing Branch Technical Position on Radioactive Waste Classification***, 11 May 1983,
3. ***Issuance of Final Branch Technical Position on Concentration Averaging and Encapsulation, Revision in Part to Waste Classification Technical Position***, issued 17 January 1995,
4. **SECY-94-273**, ***Final Amendments to 10 CFR Parts 20 and 61 on Low-Level Waste Shipment Manifest Information and Reporting***, published in the Federal Register 21 April 1992 and effective 1 March 1998,
5. 49 CFR Parts 171-177, ***Hazardous Substances, Final Rule***, 26 September 1989; ***Emergency Response Communication Standards, Final Rule***, 27 June 1989; ***Training for Hazardous Materials-Transportation***, 10 March 1983, and Proposed Rule, 26 July 1989; ***Transportation Regulations, Comparability with Regulations of the International Atomic Energy Agency***, 10 March 1983, and The Final Rule, September 1995.

NRC REGULATIONS FOR WASTE CHARACTERIZATION

The NRC regulation, 10CFR Part 61.55, "Waste Classification", provides the basis for regulating radioactive content of low-level wastes (LLW) packaged for disposal. The part regulates LLW disposal in currently available disposal facilities and also serves as the base requirements for disposal in the proposed State Compact sites. It provides minimum requirements for shallow land LLW disposal to facilitate handling at the disposal site and to provide protection of public health and safety during operation of the disposal site and after its closure. Part 55 sets limits on the concentration of specific radionuclides expected in LLW that are based on conservative population exposure scenarios described in the environmental impact analyses for the regulation. The principal scenarios deal with groundwater transport for the soluble nuclides and inadvertent intruder activities (construction and agriculture) for the other nuclides. The NRC issued two Branch Technical Position documents (BTP; items 2 and 3). They clarify aspects of the regulation and provide approaches that waste generators may take in classifying their waste, and contain one amendment to the regulation defining information that must be on the waste shipping manifest.

10 CFR Part 61.55, Waste Classification

1. The regulation divides low-level waste into four Classes, A, B, C, and Greater-Than-Class-C, on the basis of concentrations of specific radionuclides present in the waste, with Class A waste containing the lowest activity concentrations. These classes pertain to permitted packaging and disposal practices.

2. The regulation divides the radionuclides into two categories:

   — long-lived nuclides, those having half-lives greater than about 200 years and alpha-emitting and transuranic (TRU) nuclides; and,

   — short-lived nuclides.

   The two categories are assessed independently.

3. 10CFR61.55 contains a table for each radionuclide category (known as Tables 1 and 2, respectively, for the long- and short-lived nuclides), giving the allowed upper limit for each listed nuclide for each class of waste (A, B &C). *Note: Table 1 nuclides have only Class A and C classifications.*

4. Waste classification must analyze each of the listed nuclides independently. If the concentration of any radionuclide exceeds the allowed limit for a given waste class, the entire waste becomes the next highest class.

5. A "sum-of-the-fractions" rule requires that, independently, for all the nuclides in each table, the known concentration be divided by the allowed concentration in the table (fractional limit) and that the fractions for that table be summed. If the sum of these fractions exceeds 1.0, the waste is elevated to the next highest class.

6. Concentrations of some radionuclides whose activity levels are low or which do not emit easily measured radiation {x-rays, beta particles or alpha particles and DTM (difficult-to-measure) nuclides} may be determined by indirect methods such as the use of scaling factors that relate their concentrations to another that is more easily measured. There must be validation that the scaling factors accurately predict the true concentrations.

10CFR61.55 did not provide prescriptive details on issues such as:

- waste sampling methods and frequency requirements;
- allowed methods of determining contents of packaged waste;
- determination and accuracy of scaling factors; or,
- averaging nuclide concentrations over waste containers.

The NRC provided guidance on these items through specific Branch Technical Position papers (BTP) issued in 1983 and 1995.

Low Level Waste Licensing Branch Technical Position on Radioactive Waste Classification, May 1983,

This first BTP on waste classification described overall procedures that were acceptable to the NRC staff for compliance with 10CFR61. The guide suggested verification frequencies for waste stream characterizations, addressed the issue of waste stream homogeneities and described methods for characterization. It also provided guidance on the minimum information that should be included on shipment manifests. It specified that all wastes shipped for disposal must be classified. This specification presumably included waste shipped to processors although processors were not anticipated at the BTP issuance. The 1995 revision, in part, clarifies the issue and removes any classification requirement for wastes shipped to processors. Some of the guidance in the document has been superseded or clarified by that in the revised BTP, issued in January 1995.

1. ***The NRC identified four acceptable methods of waste characterization:***

Materials Accountability -- a mass balance technique where a given amount of activity is tracked throughout the processes in the facility and a known amount ends up in the waste stream. *This method is not usually applicable to utility wastes!*

Classification by Source -- another mass balance technique where classification is known through control of the activity source and processes through the facility. *Except for identifying radionuclides that would not normally be in the waste, this method is not usually applicable to utility wastes!*

Gross Activity Measurements -- direct measurement of the gross activity of a packaged waste, *e.g.*, by gamma-ray scans or dose-rate measurements, used in conjunction with verified scaling factors for that waste.

Direct Measurement of Individual Radionuclides -- direct measurement of the nuclide contents of the waste through quantitative radiochemical analyses of a sample of the waste and the assumption of homogeneity throughout the package.

Utilities generally use combinations of the latter two methods to classify their wastes, using sampling and radiochemical analyses for concentrations of a homogeneous waste or scaling factor determinations used in conjunction with a direct measurement on the packaged waste, *e.g.*, a dose-to-curie method.

2. *Determination and verification of radionuclide concentrations and correlations*

These should be done:

- Routinely for the gamma-ray emitting nuclides in Class B and C wastes when using either of the latter two classification methods and whenever process or plant operational variations could cause a change (particularly upward) in waste classifications ... at least for every batch of waste processed.
- Periodically for the DTM radionuclides when using the Gross Activity Measurement technique and whenever there is a change in plant or waste processing operations that could alter the scaling factor correlations by a factor of ten. .... at least annually for Class B and C wastes and biannual (*twice a year*). *Amended to be biennial (every other year) in the 1995 BTP* for Class A wastes.

3. *Concentration volumes and masses*

Single waste streams of resins, filter media, solidified liquids , and soil, are considered as homogeneous for the purpose of determining an average concentration of activities in the waste container.

DAW is considered homogeneous when compacted within shipping containers.

Waste activities should be averaged only over the volumes of the homogeneous waste within the container and not over the container volume for the purpose of nuclide concentration averaging. Solidified wastes are exceptions and the volume of the solidified mass may be used.

This BTP did not address the issue of concentration averaging over mixtures of waste from different streams in a container which is discussed in detail in the January 1995 BTP.

4. *Reporting on manifests*

The waste class must be identified on each container. Exceptions are bulk shipments of Class A waste, *e.g.*, contaminated soil.

The manifest must list the concentrations of ^{3}H, ^{14}C, ^{99}Tc, and ^{129}I in the waste.

The manifest must list the concentration of any radionuclide listed in the 2 tables of 10CFR61.55 present in concentrations considered "significant", defined as greater than one percent of the Class A limit. For nuclides having half lives less than 5 years, except for ^{242}Cm, the nuclide needs to be listed only if its content exceeds 7 Ci/m^{3}.

The manifest must list any other nuclides in the waste whose concentration is above 7 Ci/m^{3}.

The manifest should list any other nuclide required by the DOT regulations in 49CFR173.433(f).

The manifest must contain the following information beyond waste classification details:

- A physical description of the waste -- the specific waste stream.
- The waste volume -- the disposal container volume.
- The principal chemical form -- the chemical in which the activity is contained.
- The solidification agent -- given in general terms.

> These shipping manifest requirements will be superseded by the requirements in the amendments to 10CFR61 to be implemented 1 March 1998.

5. *Irradiated hardware*

Classification of irradiated hardware may combine a direct-measurement technique for determining the average radiation dose rate from a component in conjunction with a dose-to-curie calculation and scaling factors, determined either through activation-analysis computations or through sampling.

Branch Technical Position on Concentration Averaging and Encapsulation, January 1995

This BTP expands upon, additionally defines, and replaces guidance provided in the section of the previous BTP on "Concentration Volumes and Masses". It also provides corrections to the 1983 BTP changing the *biannual requirement for analysis of Class A waste streams to biennial, and removing the requirement to classify wastes shipped to processors.* The BTP describes a rather complex set of criteria and rules for determining the classification of packages containing wastes from multiple streams. The rules differ for gamma-emitting nuclides and the others. They also differentiate between "conventional" waste streams and contaminated or irradiated metal hardware and mixtures of cartridge filters introducing definitions and rules for averaging over "discrete" components.

1. *Mixing of homogeneous waste types or streams*

Mixing of homogeneous streams is not considered mixing if done *for a designed collection of homogeneous waste types from a number of sources within a licensee's facility, for purposes of operational efficiency or occupational dose reduction.* In these cases concentration averaging can be over the entire volume of the waste in the container, providing the individual contributors are within a factor of 10 of the average concentration of the mixture. Otherwise the classification must be based from analyses of the concentrations of the highest contributor to the mixture.

Utility waste streams usually qualifying as homogeneous are filter sediments and powdered demineralizer resins, evaporator bottoms and DAW. Most mixtures of bead demineralizer resins qualify as homogeneous under the criteria as resulting from *purposes of operational efficiency or occupational dose reduction.*

The classification of the homogeneous streams in utility wastes is usually controlled in Class A and B wastes by the beta-emitting nuclide ^{63}Ni or the gamma-emitting nuclide ^{137}Cs. Averaging of different streams of homogeneous wastes must follow the rules for the nuclides that dominate.

2. *Solidified and absorbed liquids*

Activities of these wastes may be averaged over the volume or mass of the solidified waste if the binder is approved for the stability requirement. If not, *e.g.*, in the case of absorbed liquid, the activity must be averaged over the volume of the preabsorbed liquid.

3. *Mixing of activated materials or metals*

Activities may be averaged over the volume of a single component, even if cut into sections, providing the entire component is included in the container. Different components are considered as discrete pieces. An individual section or piece is also defined as discrete if:

- The volume of the piece is less than 0.01 ft^3 or 0.00028 m^3 (typically less than 10 lb., 4.5 kg) and has nuclide activities greater than 700 Ci ^{60}Co (not applicable for Class C), 1 mCi ^{94}Nb, or 3 mCi (30 Ci for Class C) ^{137}Cs.
- The piece is a section cut from a larger component where the entire component is not included in the container and whose activities exceed 8 Ci ^{3}H (not applicable for Class C), 1 Ci ^{14}C (10 Ci for Class C), 4 Ci ^{59}Ni (40 Ci for Class C), 7 Ci ^{63}Ni (1500 Ci for Class C), or 3 mCi TRU with half lives greater than five years (30 mCi for Class C). Averaging is always allowed if the total gamma-emitting activity of a discrete piece of component is less than 1 mCi.

> Averaging is allowed for contents of a container where the pieces are all from a single component although the component itself is too large to fit inside a single container.

Concentrations may be averaged over mixtures of components or discrete pieces only if the classification-controlling concentrations of the individual components or discrete pieces are within a factor of ten of the average for DTM nuclides and a factor of 1.5 for the controlling gamma emitters, *e.g.*, ^{60}Co, ^{94}Nb, ^{137}Cs. These comparisons may be done independently and apply only for waste classification-controlling radionuclides.

Else it is necessary, and always possible, to base a container classification on the highest classification of any discrete piece, section or component within the container.

> Much irradiated hardware is either Class C or Greater-Than-Class C. The classification-controlling nuclides in these items are usually ^{63}Ni in the beta-emitting category and ^{94}Nb as a gamma emitter. Either or both of these nuclides can affect the averaging rules.

4. *Contaminated materials*

For a single contaminated component, activities may be averaged over the volume of the component.

An individual section or piece is defined as discrete if:

- The volume of the piece is less than 0.01 ft^3 or 0.00028 m^3 (typically less than 10 lb., 4.5 kg) and has nuclide activities greater than 700 Ci ^{60}Co (not applicable for Class C), 1 mCi ^{94}Nb, or 3 mCi (30 Ci for Class C) ^{137}Cs.
- The piece is a section cut from a larger component where the entire component is not included in the container and whose activities exceed 8 Ci ^{3}H (not applicable for Class C), 1 Ci ^{14}C (10 Ci for Class C), 4 Ci ^{59}Ni (40 Ci for Class C), 7 Ci ^{63}Ni (1500 Ci for Class C), or 3 mCi TRU with half lives greater than five years (30 mCi for Class C). Averaging is always allowed if the total gamma-emitting activity of a discrete piece of component is less than 1 mCi.

Activities for a mixture of discrete contaminated components may be averaged over the entire volume if the concentrations of the DTM nuclides in each discrete piece do not differ from the average by more than a factor of ten or the activity of the gamma emitters in each component do not differ from the average by more than a factor of 1.5, or their activities are less than 700 Ci ^{60}Co (not applicable for Class C), 1 mCi ^{94}Nb, or 3 mCi (30 Ci for Class C) ^{137}Cs.

Else it is necessary, and always possible, to base a container classification on the highest classification of any discrete piece, section or component within the container.

5. *Mixing of cartridge filters*

Activities for a mixture of filter cartridges may be averaged over the entire volume if the concentration of the non-gamma emitters in any filter do not differ from the average by more than a factor of 10 or the activity of the *classification-controlling* gamma emitters in each filter do not differ from the average by more than a factor of 1.5, or their activities are less than 700 Ci ^{60}Co (not applicable for Class C), 1 mCi ^{94}Nb, or 3 mCi (30 Ci for Class C) ^{137}Cs.

Else it is necessary, and always possible, to base a container classification on the highest classification of any individual (discrete) filter within the container.

> The exterior (envelope) volume of a cartridge may be used for classification as may the non-radioactive mass (including the end cap). The volume of the cartridge air space cannot be used if the filters are compacted or sheared before packaging.

In normal LLW, the classification-controlling nuclide is ^{137}Cs. Cartridge filters generally do not efficiently collect this nuclide and their classification is usually controlled by either of the beta-emitting nuclides, ^{14}C or ^{63}Ni. Thus, in this event the activities may be averaged over discrete items (e.g., filters) whose concentrations differ from the average within a factor of ten.

6. *Waste in high-integrity containers*

Concentration averaging should only be over the volume and mass of the waste rather than that of the container, providing that the waste meets the criteria for a homogeneous waste stream.

7. *Encapsulation of solid material*

Concentration averaging may be over the volume or mass of the encapsulated waste providing that:

> The volume and attributes of the waste comply with the requirements in Appendix C of the BTP, *i.e.*:
>
> - A minimum amount of encapsulation is present to "increase the difficulty of an inadvertent intruder moving the waste by hand",
> - The maximum amount of encapsulation is 0.2 m^3 unless a specific rationale is provided,
> - The maximum amount of gamma-emitting activity is that which, if credit is taken for a 500-year decay period, would result in a dose rate less than 0.02 mrem/hr on the surface of the encapsulating media,
> - The classification of the encapsulated package does not exceed Class C,
> - The discrete activity source is reasonably centered in the encapsulation, and
> - The structural form meets the requirements of 10CFR61.56 for Class B and C waste.

8. *Mixing of dissimilar streams (different waste types)*

Averaging is allowed if the classification of the mixture is not lower than the classification resulting from any individual component of the mixture. Excluded from this requirement are small check sources (less than 100 μCi).

9. ***Other provisions may be authorized upon request to the NRC.***

SECY-94-273, Final Amendments to 10 CFR Parts 20 and 61 on Low-Level Waste Shipment Manifest Information and Reporting, Effective 1 March 1998

The purpose of these amendments, discussed more fully in Section 9, is to:

1) *improve the quality and uniformity of information contained on manifests that are required to control transfers of LLW ultimately intended for disposal at a land disposal facility;*

2) *establish a set of NRC forms, that serves as information needed to meet NRC, DOT, State, and Compact information requirements;*

3) *require LLW disposal site operators to electronically store container-specific manifest information; and,*

4) *require disposal site operators to be capable of submitting reports of stored manifest information on a computer-readable medium (e.g., magnetic disks or tapes).*

The rule requires that the disposal site operator be able to track the origin, transport, disposition, and characteristics of individual disposal containers of waste.

Licensees are not required by NRC to comply with the manifesting requirements of the amendments Appendix G to 10 CFR part 20 when they ship:

a) *LLW for processing and expect its return (i.e., for storage under their license) prior to disposal at a licensed facility;*

b) *LLW that is being returned to the licensee who is the "waste generator" or "generator", as defined in the amendment; or*

c) *radioactively contaminated material to a "waste processor" that becomes the processor's "residual waste."*

The final product of waste sent to a processor with an expected return to the generator (thus neither classified nor listed on a uniform manifest) must be classified before shipment for disposal. Classification of this product can present some particular difficulties.

DOT REGULATIONS FOR SHIPMENT OF RADIOACTIVE AND OTHER HAZARDOUS MATERIALS.

49 CFR Parts 171-177, Hazardous Substances, Final Rule, 26 September 1989; Emergency Response Communication Standards, Final Rule, 27 June 1989; Training for Hazardous Materials-Transportation, 10 March 1983, and Proposed Rule, 26 July 1989; Transportation Regulations, Comparability with Regulations of the International Atomic Energy Agency, 10 March 1983, and Final Rule, September 1995.

DOT amended sections of 49CFR implementing updated regulations for the transportation of radioactive and other hazardous waste prescribing categories for radioactive waste that depend upon the waste content. Procedures establish DOT radioactive material typification as LSA-I, LSA-II, or LSA-III using an A_2/g limit for each radionuclide and a sum-of-the fractions rule similar to that in 10CFR61.55. Regulation 49CFR 173.403 lists the A_2/g limits and 49CFR173.435 lists the A_2 values. Utility wastes that might have all nuclide concentrations at the upper limits of 10CFR61.55 Class B or C wastes would exceed the limits for LSA material. However, nuclide concentrations in most utility-generated LLW, except for an occasional irradiated hardware piece, are well within LSA limits.

DOT requirements must be satisfied for any shipment of waste.

For additional information on DOT shipping requirements, refer to *1995 Revisions to DOT and NRC Transport Regulations and their Impact on Nuclear Power Plants* (EPRI TR-106926).

3

BASIS FOR LLW CHARACTERIZATION

INTRODUCTION

The discussion in this section provides an understanding how and why the Nuclear Regulatory Commission (NRC), developed a LLW classification system. The NRC's decisions were based on protecting the public from: *Concentrations of radioactivity which may be released to the general environment in groundwater, surface water, air, soil, plants or animals must not result in an annual dose exceeding an equivalent of 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public. Reasonable efforts should be made to maintain releases of radioactivity in effluents to the general environment as low as reasonably achievable* (10CFR 61, Licensing Requirements for Land Disposal of Radioactive Waste issued 1982,). This regulation, implemented in December 1983, contains the fundamental requirements for the radiological characterization of wastes from nuclear power plants and other licensed radioactive waste generators.

Important methods for preventing the release of radionuclides and reducing potential exposures to the public from a disposal facility are:

- the low level waste form;
- types of engineered barriers; and/or
- deep burial at the disposal facility.

Limiting the inventory of long-lived mobile anions such as ^{14}C, ^{129}I, ^{99}Tc, in a disposal facility is key to meeting the dose limits set in 10CFR61 via the agricultural and ground water scenarios.

Performance assessment models are mathematical models which have been developed to demonstrate compliance with the objectives of 10CFR61. They are used to quantitatively predict potential exposures to an individual from a LLW disposal site. Performance assessments are required of a disposal facility, as a license requirement to determine when (and if) a maximum exposure to the public will occur. The time-frames for exposure may go well beyond the 300 and 500 year intruder barrier that is often thought of as the time-period of most importance for exposure. The NRC requires that dose calculations go beyond 500 years, recognizing the long half-lives of mobile anions. Therefore, it is quite common for performance assessment models to identify the peak projected exposure out to the 10,000 year time-frame.

To ensure that a disposal facility is used to the capacity that it can safely sustain, in terms of quantity of nuclides disposed, LLW characterization should be as accurate as possible and practicable. Being overly conservative in estimating the long-lived nuclides sent to a disposal facility in LLW can reduce the operating life of the facility and severely limit the quantity of LLW it can receive during its operations. Accurate waste characterization will contribute to proper performance assessment analyses, since that information is derived by the radionuclide inventory at the disposal facility (see Appendix A, Performance Assessment and LLW Characterization).

LLW must be characterized to determine it's classification for disposal purposes. LLW is classified as: Class A; Class B; Class C and greater-than-Class C. The classification scheme depends on the concentrations of short-lived and long-lived nuclides in the waste. These concentrations were defined by exposure limits set to limit the dose that a member of the public could receive (on an annual basis) from a shallow land LLW disposal site.

The following is a brief overview of the characterization requirements and their implications to low level radioactive waste disposal siting.

WASTE CLASSIFICATION REQUIREMENTS

Classification requirements in 10CFR61 are based on determining the specific protective measures necessary, to reduce the exposure to the public. The protective measures to achieve this are:

1) waste form stability; and,
2) disposal features (i.e., intruder barrier).

For each waste class this translates into the following requirements:

Class A	- no waste form requirement (except for liquids) no site design requirement except that class A waste must be segregated from B & C if it is not stabilized.
Class B	- waste must be in a stabilized form. Either the waste itself or it's package must be stable for 300 years.
Class C	- stable waste form (same as class B above) plus site design requirement that acts as an intruder barrier i.e., waste form, a structure or overpack around the waste, or burial of the waste deeper [>3 meters].

Greater Than Class C - no waste form requirements.

Note: In 1985 Congress made the Department of Energy responsible for disposal of Greater than Class C wastes.

Simply put, NRC arrived at it's waste classification by first establishing allowable dose limits to a maximally exposed member of the public, on an annual basis. These dose limits are:

- 25 mrem/yr. whole body
- 75 mrem/yr. to thyroid
- 25 mrem/yr. to any other organ of the body
- 500 mrem/yr. to an inadvertent intruder (this exposure limit is not set in regulations but is in the environmental impact statement that is the basis for regulations)

Exposure scenarios were then identified. Scenarios are postulated pathways that humans could come in contact with the waste or that the radionuclides in the disposal facility could be transported to humans.

Mathematical models representing exposure conditions in the exposure scenarios were then developed. And finally, waste concentrations that would likely result in projected doses-to-the-public which were lower than the limits were calculated.

As part of this process, NRC decided that the following radionuclides were <u>important to LLW disposal</u>. Each waste-steam must be characterized for its content and concentration of these nuclides.

^{3}H
^{14}C
^{60}Co
^{59}Ni
^{63}Ni
^{90}Sr

^{94}Nb
^{99}Tc
^{129}I
^{137}Cs

Transuranics
^{238}Pu
^{239}Pu
^{241}Pu
^{241}Am

^{242}Cm
^{243}Cm
^{244}Cm

Nuclides in **bold** are long lived and of particular interest to disposal facility development because of their potential dose to the public from the groundwater and or to the inadvertent intruder.

EXPOSURE PATHWAY

Inadvertent Intruder

NRC allowable dose limits for an inadvertent intruder is set at 500 mrem (maximum one-time exposure). The classification system is set up to protect the inadvertent intruder disturbing the disposal facility from receiving more than the 500 mrem dose. Inhalation of beta activity is the main exposure pathway for an intruder. For this reason, Class C waste requirements are:

- a stable waste form;
- a site design requirement that acts as an intruder barrier i.e., waste form and a structure or overpack around the waste or, deeper burial of the waste [>3 meters].

Table 3-1 Shows concentration limits for nuclides in waste classes defined in 10CFR61.55. Table 3-2 indicates exposure scenario, nuclide and dose pathway.

Agricultural/Groundwater

Biological uptake is limited to 25 mrem whole body and 75 mrem to the thyroid annually in humans, 10CFR61 requires new disposal facilities conduct performance assessments. A performance assessment projects the annual dose to the public via drinking groundwater, crop irrigation and livestock pathways i.e., bioaccumulation via human consumption of these materials at the site of the disposal facility. Long lived mobile anions such as ^{14}C, ^{129}I and ^{99}Tc are the nuclides with the greatest potential for meeting or exceeding the above limits.

By carrying out a performance assessment, we can calculate the inventory of a radionuclid that will ensure that the dose to the public is equal to or below regulatory limits. (Note, the inventory will differ from site to site depending on specific site characteristics.) With an understanding of the projected dose delivered by a particular nuclide inventory, one can control the dose to the public by limiting the amount/inventory of the nuclides in the disposal facility. This should insure that the dose limits are not exceeded.

> Note: the availability of space is not the factor which limits the amount of waste that a disposal facility can receive. Rather it is the inventory of certain critical nuclides ^{14}C, ^{129}I, ^{99}Tc that can limit it's operating life.

Table 3-1
Radionuclides in 10CFR61.55 Important to Waste Classification

Isotope	**Half-life(yr.)**	**Emission**	**Energy(MeV)**	**A**	**B**	**C**
^{14}C	5730	Beta	0.156	0.8	--	8
^{99}Tc	2.13×10^5	Beta	0.293, 0.09	0.3	--	3
^{129}I	1.6×10^7	Beta	0.15, 0.039	0.008	--	0.08
^{241}Pu	14.35	Beta	4.8, 0.02, 0.15	350*	--	3500*
^{242}Cm	.446	Alpha	6.1, 0.04	2000*	--	20,000*
Other TRU	>5	mostly Alpha	4.7, 5.6	10*	--	100
^{3}H	12.3	Beta	0.019	700	--	---
^{60}Co	5.27	Beta/gamma	0.32, 1.33, 1.17	40	--	---
^{63}Ni	100.1	Beta	0.066	3.5	700	7000
^{90}Sr**	28.82	Beta	0.546	0.04	150	70000
^{137}Cs	30.17	Beta/gamma	0.512, 0.66	1	44	46000

* TRU Limits in nCi/g
** Accompanied by 90Y decay (2.28 Mev β)

Table 3-2
Nuclide Exposure Scenarios of Interest

Radionuclide	Scenario	Dose Pathway
^{3}H	Intruder-Agriculture	Food Uptake
^{14}C	Intruder-Agriculture	Food Uptake
^{99}TC	Intruder-Agriculture	Food Uptake
^{129}I	Intruder-Agriculture	Food/Inhalation
All Transuranics	Intruder-Construction	Inhalation
^{60}Co	Intruder-Agriculture	Direct Gamma
^{63}Ni	Intruder-Agriculture	Food Uptake
^{90}Sr	Intruder-Agriculture	Food Uptake
^{137}Cs	Intruder-Agriculture	Direct Gamma

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IDENTIFICATION OF WASTE STREAMS

During the normal operation of a nuclear power plant, radionuclides are formed and/or released from the fuel by three methods. These are:

1. Neutron activation of primary coolant (^{14}C and ^{3}H).
2. Neutron activation of reactor structural materials, corrosion products, and other fine particulate substances circulated through the core by primary coolant.
3. Escape of fission products into primary coolant through defects or failures in fuel cladding, and release of trace (tramp) uranium impurities from the exterior of the fuel element.

Most of the radionuclides formed as described above are released into the primary coolant, and transported into the numerous auxiliary and support systems connected to the primary system. Since these systems do not provide absolute containment to the primary water, there is leakage from system components such as valves, fitting, pumps, *etc.* Through these release mechanisms, radionuclides generated in the fuel region are transported and released to various systems and areas throughout the plant. To minimize the potential radiation exposure to operating personnel caused by the deposition of these radionuclides throughout the plant, special systems are employed to collect, process, and handle liquid leakage containing radionuclides, and other solid material which comes into contact with these radionuclides thereby becoming contaminated. The wastes resulting from these processes are referred to as low level radioactive waste (LLW).

Proper characterization of LLW generated at nuclear power plants requires knowledge of several factors:

- waste sources and characteristics,
- effects of plant design and operational parameters on these sources and characteristics, and
- effects of waste treatment/processing.

This section is intended to summarize the waste streams covered by these guidelines and briefly identify some of the factors that will impact the characterization of these wastes.

Refer to EPRI's *Mixed Waste Characterization Guidelines* (EPRI TR-104401) and *Nuclear Utility Mixed Waste Stream Characterization Study* (EPRI TR-104400) for identification of and characterization guidance for potential mixed waste streams.

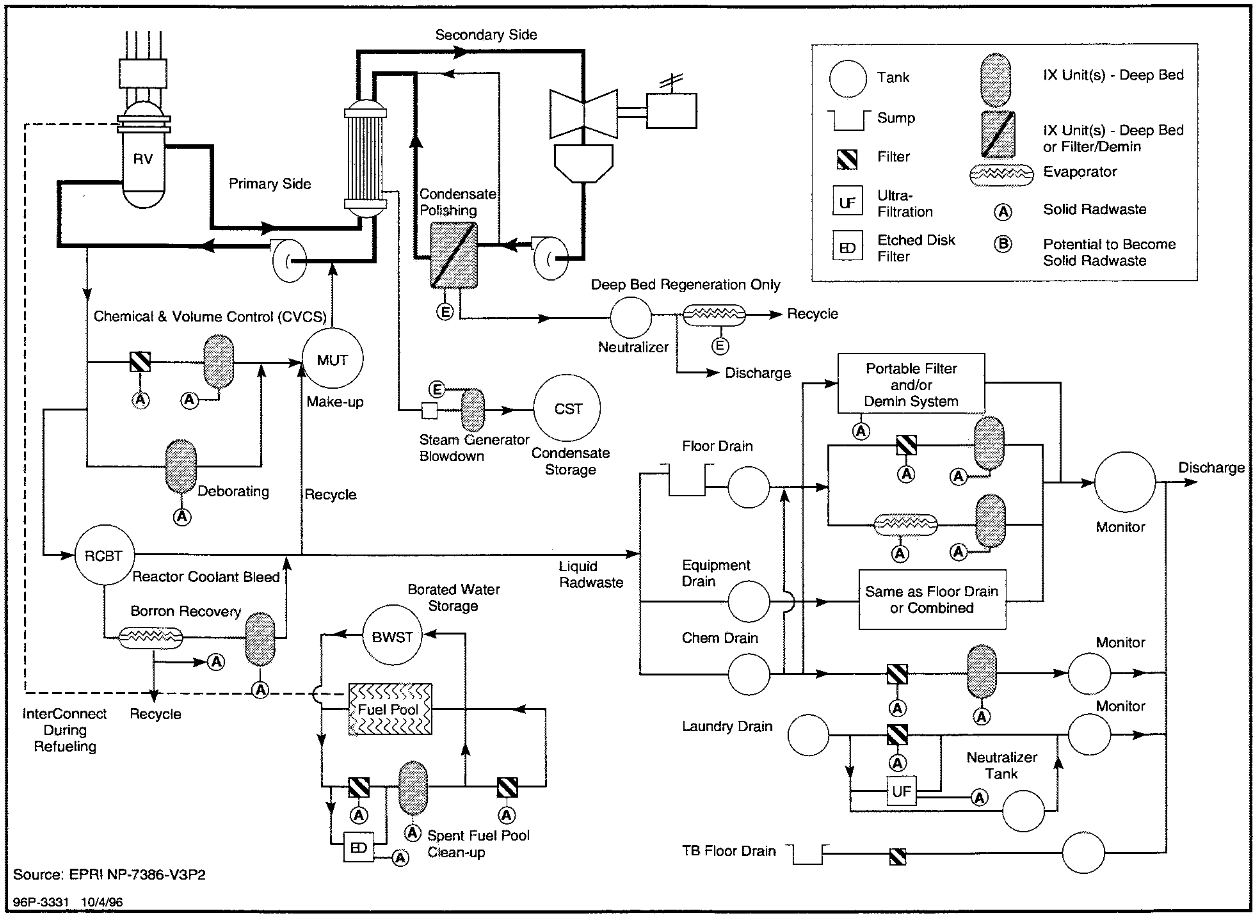
SUMMARY OF TYPICAL LLW STREAMS

LLW from power reactors generally consists of two major categories depending upon point of origin within the plant: "Wet Waste" and "Dry Active Waste (DAW)." Wet wastes originate primarily in liquid processing systems, and consist of ion-exchange resins, filters, sediments, and evaporator concentrates. DAW is primarily contaminated trash, ventilation filters and scrap/components. The activity concentrations associated with these wastes are primarily controlled by the type of fuel, the amount of fuel leakage, the materials of construction and primary chemistry control. The type of fuel and the construction materials seldom change, but fuel leakage and impurities in the coolant (*i.e.*, corrosion products), and chemical additives can vary. The activity concentration of waste from a given reactor can be expected to vary. Small changes in fuel performance can significantly affect activity concentrations while the variation in corrosion products is small since they result directly from construction materials. Each of the major waste categories are discussed below.

Note: The term "waste stream" is often used interchangeably with waste type. For example, resins are often referred to as a waste stream when, in fact, there may be several different resin waste streams generated at a plant. For purposes of waste characterization, the term "waste stream" is defined as any waste product or mixture of products where the DTM nuclide concentrations can be inferred by use of a single set of scaling factors (see Section 7).

Wet Waste

To understand where "wet wastes" are generated in a power plant, Figures 4-1 and 4-2 depict typical liquid processing systems found in a PWR and BWR, respectively. The result of this processing is a combination of 1) very clean water that is returned to the power plant; 2) water which meets discharge limits in the power plant license - this water is discharged to the environment; and 3) "wet waste" which is characterized, processed, packaged (containerized) and shipped for disposal to a licensed LLW disposal facility.



Source: EPRI NP-7386-V3P2

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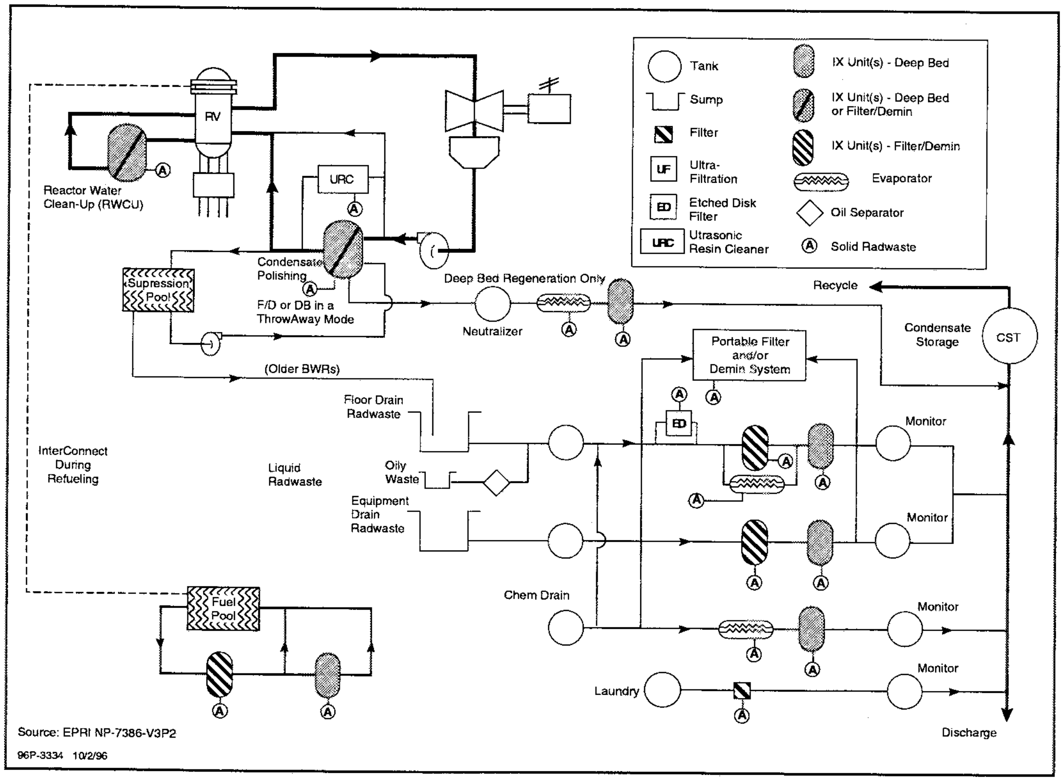
Figure 4-1 Typical PWR Liquid Waste Processing System (General System Flow)



Source: EPRI NP-7386-V3P2

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Figure 4-2 Typical BWR Liquid Waste Processing System (General System Flow)

Ion Exchange Media (PWR and BWR)

Ion exchange (IX) media may consist of bead resins, powdered resins, carbon, and/or zeolytes. Bead and powdered resin are essentially the same organic material, the only difference is their relative particle size. Bead resins are manufactured as small spheres with a relative size of 20 to 50 mesh. Powdered resins are produced by simply grinding the bead material to a smaller particles size, normally in the range of 50 to 100 mesh.

Ion exchange resins are used for the purification of liquid streams within their cycles. Each type of plant incorporates a reactor cleanup system for removal of ionic species within the reactor water for corrosion protection of the nuclear fuel and reactor components. In addition, most plant designs use condensate demineralizers for purification of the condensate/feedwater stream prior to its delivery to the reactor or steam generators. Finally, ion exchange materials are used extensively for the processing of liquid wastes collected within the nuclear plant.

Bead Resins: Bead resins are defined as small porous beads of organic material used to process various liquid waste streams through a combination of absorption and/or adsorption of soluble ionic material and through filtration of insoluble material.

Sources:	Demineralizers (deep bed type)
Purpose:	IX of dissolved contaminants
Side Benefit:	Some filtration of suspended solids (~ 60-70% for iron)
Waste Processing Method:	Dewatering, possible solidification
Containers:	HIC, liner
Typical dose rates:	10 mrem/hr - 300 R/hr

Resins may be disposed of as a separate bed or combined with other resin beds, typically in the spent resin storage tank, prior to disposal (refer to Section 6).

Powdered Resin: Powdered resins are used in many BWR facilities. Powdered resin results from grinding up bead resins. Therefore, their chemical composition is identical to bead resin. The powdered resin is particularly well suited to purification of liquid streams containing minute amounts of dissolved solids and minor amounts of suspended materials. Powdered resin is normally deposited on a filter septum as an "overcoat" and used where the ionic loading is low and filtration of suspended solids is desired. For example, they are used as condensate demineralizers at fresh water plant sites, while sea water plants require deep beds of bead resins. The result is an excellent filtration media which possesses some ion exchange capacity.

Sources:	Filter demineralizers
Purpose:	IX of dissolved contaminants
Side Benefit:	Filtration of suspended solids

Waste Processing Method:	Dewatering, solidification
Containers:	HIC, liner
Typical dose rates:	10 mrem/hr - 300 R/hr

Cartridge Filters (Primarily PWRs):

Cartridge-type filter elements are used extensively in PWR plants. There is a wide range of filters construction. Some are spiral wound elements. Others use single surface media, pleated to increase its overall surface area. BWRs often use cartridge filters in the processing of reactor cavity liquids and for spent fuel pool vacuuming.

Cartridge filters are assemblies which provide mechanical filtration in nuclear plant liquid waste processing systems to remove insoluble material.

Sources:	In-plant filter system/portable underwater vacuum systems
Purpose:	Filtration of suspended solids
Waste Processing Method:	Dewatering, encapsulation, absorbed
Containers:	HIC, liner, drum
Typical dose rates:	100 mrem/hr - 80 R/hr

Evaporator Bottoms:

Original U.S. plant designs included evaporators for the processing of liquid solutions containing high levels of dissolved solids. Specifically, BWRs were supplied with evaporators for treatment of floor drains and bead resin regenerant solutions. Due to relatively poor experience with this equipment in terms of maintenance and personnel exposure, the majority of these units have been permanently retired. However, a few BWRs and zero discharge PWRs supplied with crystallizers have operated their waste evaporators to reduce the volume of solid radwaste.

Many PWRs continue to operate their Boric Acid Evaporators for processing boric acid from the reactor cooling water. PWRs use boric acid to control fuel burnup (release of energy from fuel). For example, when new fuel replaces old fuel in the reactor core, and the power plant is brought back to full power, boric acid is added to ensure that the desired level of output is maintained. Over time, as more of the fuel is spent, the concentration of boric acid in the reactor is decreased by a feed and bleed method to facilitate the fission reaction. This allows the utility to maintain the desired power level. The borated letdown reactor water can be treated for discharge by evaporation and disposal of the dried or solidified concentrates. Alternatively, the water can be decontaminated by ion exchange, then evaporated with recovery of the boric acid for use in the next fuel cycle.

Where still in use, evaporators concentrate liquids. The resulting concentrated solution (12-50) weight percent solids is normally solidified or dried as part of the final processing for disposal.

Sources:	Evaporators
Purpose:	Concentrate solids
Waste Processing Method:	Solidification, drying
Containers:	Drums
Typical dose rates:	0.2-2 R/hr

Miscellaneous Sediment (PWR and BWR):

Deposits often collect in the bottom of sumps, vessels, and tanks in nuclear plants. These deposits are composed of settled suspended solids that are present in the liquids contained in these components.

Sources:	Sumps, tanks
Waste Processing Method:	Dewatering, drying, solidification
Containers:	HIC, liner, drums
Typical dose rates:	0.01-2 R/hr

Note: Disposal facilities will not accept sediments. These wastes must be processed (*e.g.*, dried or solidified before disposal).

Oil and Miscellaneous Liquids (PWR and BWR):

Used oil and miscellaneous liquids (e.g., antifreeze), for the most part, result from changing the oil used to lubricate pumps and motors.

Sources:	Sumps, equipment drains, lube oil leakage
Waste Processing Method:	Solidification, incineration
Containers:	Drums
Typical dose rates:	Generally <5 mrem/hr

Dry Active Waste (DAW)

Dry active waste (DAW) is composed of a wide range of materials (see Table 4-1). Contained within this general class of waste are such items as; failed parts and components, contaminated clothing items, cleaning materials such as mops and wiping materials, and in unusual circumstances, contaminated soil. These waste materials are often sub classified by how they will be processed for disposal.

Refer to EPRI's *Solid Low Level Waste Management Guidelines* (EPRI TR-104583) for guidance on managing DAW.

Table 4-1
Examples of DAW Composition

Percent	Description of Waste
31.6	Plastic bags
19.8	Paper coveralls
17.9	Plastic booties
14.5	Various commercial plastic sheeting
6.3	Misc. paper, rope, smears, tape, cotton liners, face shields
5.2	Masslin cloth, paper towels
3.2	Recoverable items, protective clothing
1	Mop heads

Source: EPRI Report TR-104583

The following provides some insight on common groupings of DAW. Dry active wastes often have very low levels of contamination, *i.e.*, <50 mrem/hr. These wastes constitute approximately 80% of the volume of waste generated from a nuclear power plant.

Incinerable Trash:

Materials within this class lend themselves to incineration. Their composition does not produce incineration difficulties or major amounts of secondary waste in the incineration process.

Common Materials:	Polyethylene, polyurethane, cloth, paper, wood, etc.
Waste Processing Methods:	Incineration, compaction, supercompaction
Containers:	Drums, steel boxes, sea-land containers
Typical dose rates:	Generally <50 mrem/hr

Non-Incinerable Trash:

Materials in this group are seen as undesirable from the standpoint of incineration, because their incineration produces corrosive gases or large secondary waste problems.

Common Materials: Polyvinyl chloride, nylon reinforced plastic, rubber, filters, glass, etc.
Waste Processing Methods: Compaction, supercompaction
Containers: Drums, steel boxes, sea-land containers
Typical dose rates: Generally <50 mrem/hr

Contaminated Metals:

Materials within this group are most often the result of maintenance activities and component modifications and replacements.

Common Materials: Tools, pipes, valves, components, scaffolding
Processing Method: Decontamination, metal-melt, supercompaction
Containers: Drums, steel boxes, sea-land containers
Typical dose rates: Generally <50 mrem/hr

Other Non-Incinerable Materials:

This group contains miscellaneous materials.

Common Materials: Spent grit blast waste, rubble, large quantities of contaminated dirt.
Waste Processing Method: "Metal melt", Supercompaction
Containers: Drums, steel boxes, sea-land containers
Typical dose rates: Generally <5 mrem/hr

Activated Hardware:

Materials within this class are associated with the reactor. They are the product of neutron activation of materials in proximity to the nuclear fuel.

Common Materials: Reactor components, fuel assembly hardware, control-rod blades, *etc.*
Source: Reactor core components
Waste Processing Method: Segmentation
Containers: HIC, liner, spent fuel trash liner
Typical dose rates: 200-40,000 R/hr

> Note that characterization of activated hardware is outside the scope of this LLW Characterization Guideline document. Please refer to EPRI NP-5497.

Guidance pertaining to proper characterization of these waste streams, with the exception of activated components, is contained in the following sections of this Guidelines document.

POST-PROCESSED WASTE

The treatment or processing of a waste stream may change the physical, chemical, and radiological characteristics of the waste (Section 4 summarizes typical processing techniques for the various waste streams). Therefore, it is important that the generator understand the process by which the waste is treated and the impacts to waste characterization. Table 4-2 summarizes the potential impacts to radiological characterization caused by various processing options.

Table 4-2
Impacts to Waste Characterization from Different Processing Methods

Processing Method	Potential Impacts to Characterization
Compaction/Supercompaction	Concentration of radionuclides
Incineration	Concentration of most radionuclides; removal (volatilization) of ^{14}C, ^{3}H, ^{137}Cs, *etc.*
Solidification/Stabilization	Dilution of radionuclides
Metal Melt	Concentration of most radionuclides; removal (volatilization) of ^{14}C, ^{3}H, *etc.*; self-shielding
Decontamination	Reduction of most radionuclides; spectrum shifts possible
Dewatering	Reduction of ^{3}H
Vitrification	Concentration of most radionuclides; removal (volatilization) of ^{14}C, ^{3}H, *etc.*; self-shielding
Encapsulation	Dilution of radionuclides
Catalytic Extraction Process (CEP)	Concentration of most radionuclides; removal (volatilization) of ^{14}C, ^{3}H, *etc.*; self-shielding

Waste generators should consider the following when characterizing waste where the radiological properties may be altered due to off-site treatment or processing:

- Strict accounting systems are typically used by waste processors to track radionuclides - radioactivity in equals radioactivity out. There is no attempt to calculate losses due to volatilization, decay, or residuals that may get trapped in filters, etc. The activity reported on the shipping papers to the processor is the activity reported post-processing when waste is shipped for disposal or returned to the generator. Due to the relatively small fraction of activity that is lost via these mechanisms (i.e., the activity lost is very small compared to the total activity and taking into consideration the accuracy of characterization), the regulators consider this practice to be a practical and acceptable way of reporting.
- Waste Processors have accounted for effluent releases to the environment in their license application. Effluent releases are monitored, accounted for, and regulated.
- As advanced technologies become available to process higher activity waste streams, the issue of not accounting for effluent releases in waste packages shipped for disposal or back to the generator becomes more important. Currently, DAW being incinerated contains relatively small quantities of volatile isotopes. As technologies such as Vitrification, metal melt, CEP, etc. are used to process higher activity waste, the quantities of volatile radionuclides important to waste characterization and performance assessment will increase.
- One way for generators to avoid double reporting (i.e., activity released to the environment that is still reported as being in the waste shipped from the processor after treatment) and obtain a more accurate representation for disposal is by characterizing the waste after processing. One waste processor interviewed as part of this effort indicated that they are willing to work with generators to allow for characterization after processing. Potential advantages to characterization of waste post-processing include:
  - Accounts for decay (particularly for waste being stored prior to processing/disposal)
  - Accounts for nuclide removal during processing
  - Improved homogeneity of the waste
  - Improved geometry of the waste
  - Higher activity wastes are easier to quantify

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LABORATORY CONSIDERATIONS

Measuring DTMs (difficult-to-measure) radionuclides in representative samples of various LLW streams is a key element of the waste stream characterization process. For most utilities, this means selecting and using a commercial radioassay laboratory. Realistic scaling factors, and the subsequent classification of radioactive waste, depend on the joint efforts of the utility and it's contract laboratory to collect properly a representative sample, to preserve chemically and to measure quantitatively the radionuclides of interest within the sample. Many considerations enter into the laboratory selection process to ensure accurate and consistent quantitative results. This section provides a summary of the important considerations related to the selection and operation of a laboratory engaged in the radioassay of 10CFR61 waste stream samples. These considerations include:

- Operational and technical considerations in laboratory selection.
- Sample collection, preservation and preparation techniques.
- Radiochemical procedure considerations.
- Independent third party quality control analyses and validation of results.

LABORATORY SELECTION

Commercial radioassay laboratories are faced with quality and operational requirements that depend on the type of client, their expectations and the applicable regulatory specifications. For a laboratory, no single quality or operational policy is applicable for all work performed. Therefore, the trend is for the laboratory to perform to contract specifications that outline the operational and quality performance desired by the client. The contract specifications should be based on "data quality objectives" related to the usability of the analytical results.

Not all laboratories can analyze waste stream samples for the difficult-to-measure radionuclides because of their analytical capability or NRC (or State) material license limitations. For those laboratories performing radioassays of 10CFR61 waste stream samples, realistic analytical results and scaling factors depend upon key practices inherent in their operational procedures. Use of these practices, in conjunction with the operational and quality factors required by 10CFR50 Appendix B, should ensure consistent, reproducible and quality analytical results.

Contract Specifications

The most efficient manner to ensure satisfactory laboratory services for 10CFR61 sample processing is for the utility to define their expectations in the scope of work and technical specifications of a formal contract. Utilities normally issue a "Request for Quotation" (RFQ) to commercial laboratories being considered for bids. The RFQ defines the scope of work and technical specifications. Successful bidders must be on the utility's approved vendors list which typically involves a quality assurance audit and, in some cases, the successful passing of proficiency tests related to QC materials. Normally, the utility audit team is comprised of a certified quality assurance assessor and a technical specialist familiar with radiochemistry. The audit team reviews the quality assurance and operational aspects of the laboratory services to determine whether the laboratory can meet the scope of work outlined in the RFQ. A contract is issued to the successful bidder reiterating the scope of work defined in the RFQ.

Consider the following when selecting a commercial laboratory:

To facilitate the acquisition of good laboratory services, the quality and operational performance specifications should be clearly defined. Listed below are some basic items for consideration:

- operational performance criteria such as turn-around times, *etc.*;
- operational issues such as designated utility and laboratory points of contact, scheduling of sample submittal, shipping container requirements, NRC (or State) license and DOT related specifications, chain of custody - sample receipt acknowledgment specifications, *etc.*;
- operational and quality assurance programs that are consistent with ANSI, ASTM, ANS guidance and 10CFR50 Appendix B;
- quality performance criteria for precision and bias;
- minimum nuclide detection levels for each sample matrix;
- qualifications of key technical staff;
- maintenance of an internal quality assurance program that has at least five percent QC for 10CFR61 nuclides and media;
- participation in an external, National Institute of Standards and Technology (NIST) traceable, third-party quality assurance program for 10CFR61 nuclides and media;
- schedule of on-site QA audits;

- validated radiochemical procedures that address typical nuclide interferences and sensitivities for 10CFR61 media and nuclides;
- sample preparation that ensures complete nuclide solubilization and/or sample digestion;
- initial isotopic gamma spectrometric analysis upon receipt and subsequent to sample digestion to determine losses in the sample digestion/leaching processes;
- radiochemical procedures that use radiotracers or chemical carriers to determine the radiochemical recovery for each difficult-to-measure nuclide in a sample;
- detection levels that include interferences (increased background) from contaminants in the sample;
- reporting format for analytical results, i.e., utility ID, laboratory ID, collection location, sample description, volume/weight of sample analyzed, date received and analyzed, analyte concentration, total propagated uncertainty that incorporated uncertainties of interference and detection level for each measurement;
- reporting time requirements for routine radioassays and unusual results;
- requirements for backup aliquants and long-term storage for sample re-analysis; and
- schedule for issuance of routine QA reports and for feedback of QA/QC issues that affect the quality of the analytical results.

Ensure that the laboratory defines the following parameters:

The laboratory should provide or suggest the following information:

- minimum volume and weight of sample to meet detection levels and NRC (or State) license considerations;
- sample collection procedures and sample container requirements for proper handling at the laboratory;
- minimum submittal time from date of collection;
- chemical preservatives to be added to the samples to prevent radionuclide losses;
- sample shipping requirements; and,
- sample submittal and record keeping information outlining laboratory's protocol and requirements for documenting unique sample information.

SAMPLE SIZE AND RECOMMENDED SAMPLE PRESERVATION

The size of a 10CFR61 waste stream sample collected by the utility is related to several factors. Among these are:

- waste stream representativeness;
- ease of collection;
- homogeneity of sample;
- worker (utility and laboratory) radiation exposure;
- DOT shipping requirements;
- laboratory's NRC (or State) license possession limits;
- analytical sensitivity requirements.

In addition, the laboratory typically requests an additional sample quantity so it can maintain a backup aliquot of the original or digested sample for re-analysis purposes. From a laboratory's perspective, it is essential to coordinate sample size requirements as well as the sample preservation techniques with the utility. Therefore, it is recommended that the scope of work within the utility-laboratory contract identify these items and knowledgeable staff be involved.

Radiation exposure levels experienced by the laboratory worker engaged in 10CFR61 sample processing varies according to media type, fuel condition and operating systems at the utility plant. Typically, only a very small percentage of samples received at the laboratory will have exposure rates of concern or require ALARA program implementation.

Improper sample preservation can have a dramatic impact on the accuracy of laboratory results and scaling factors.

To obtain meaningful quantitative analyte results, it is important that the utility properly collect and preserve samples. Since the laboratory has designed its sample preparation and radiochemical procedures around a given sample form and size, it is imperative that the laboratory provide guidance relative to sample collection and preservation techniques. For example, if samples are acidified with HNO_3 most likely ^{129}I and ^{14}C will be volatilized and lost from the sample. Likewise, if the utility sends too many filter swipes (dry active waste) for the sample digestion process, erratic and diverse results may result. In some cases, sample containers for various media are specified as a convenience to the laboratory to ease handling of samples and as contamination control measures.

Sample size for laboratory analyses depends on many parameters including media type, radioactivity and radiation exposure level, NRC material license limits, and sensitivity of radiochemical technique.

Listed below are some suggested sample collection, containerization and preservation practices. A typical NRC (or State) materials license has modest possession limits (1 mCi per nuclide for atomic numbers of 1-83; 100 µCi per nuclide for atomic numbers 84-96). Table 5-1 summarizes by media type the recommended minimum sample size that may be sent to a laboratory.

Table 5-1
Recommended 10CFR61 Sample Sizes

Sample Type	Required Analysis	Typical Sample Size (G Or Ml)
Resins, Salts, Ash	Gamma Scan, ^{3}H, ^{14}C, ^{55}Fe, ^{59}Ni, ^{63}Ni, ^{89}Sr, ^{90}Sr, ^{99}Tc, ^{129}I, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{241}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm	1.0
Cartridge Filters	Same As Above	1.0
Sediments	Same As Above	1.0
Diatomaceous Earth	Same As Above	1.0
Crud (Scrapings) (On Filter)	Same As Above	500
Evaporator Bottom, Reactor Coolant, Or Reactor Water	Gamma Scan, ^{55}Fe, ^{59}Ni, ^{63}Ni, ^{89}Sr, ^{90}Sr, and Transuranics	500
	^{3}H, ^{14}C, ^{99}Tc, and ^{129}I	250
Dry Active Waste (Smears Composite)	Same As Resins	N/A (Results Reported on a Per Sample Basis)

GENERIC:

- Utilize proper sample containers - this is part of the sample preservation process.

- Chemical preservations are required to prevent nuclide losses - by category
    - Volatile losses - ^{14}C and ^{129}I
    - Container wall losses - metals, etc.
- Perform an initial isotopic gamma analysis of the sample prior to shipment.

MEDIA TYPE:

RESIN, FILTERS, SEDIMENTS, SALTS and ASH

- Collect a representative and homogeneous sample.
- Deposit into pre-weighed 22 ml glass vial.
    - Avoid plastic vials, petri dishes, plastic receptacles or other containers that may acquire a static charge.
- Collect an appropriate sample size - sample size is typically ≤ 1 gram if highly radioactive.
- Estimate total radioactivity content and compare with Lab's NRC (or State) materials license and ALARA program limits.

CRUD (ON FILTER)

- Pass a 500 ml liquid sample through a slow filtering ashless filter.
- Sandwich crud filter between two clean filters.
- Place filters into petri dish and tape petri dish with electrical tape.

REACTOR COOLANT and EVAPORATOR BOTTOMS

- Collect two samples.
    - 500 ml sample in plastic coated glass container
        - Add 10 ml conc. HCl to prevent wall losses of metals and TRU
    - 250 ml sample in plastic coated glass container.
        - Add 10 ml of conc. NH_4OH (Carbon free) immediately (final pH of 7 - 10) to prevent losses of ^{14}C and ^{129}I.
        - Do not use NaOH, Na_2CO_3 or $(NH_4)_2CO_3$.

> Never acidify samples requiring ^{14}C or ^{129}I analyses.

DRY ACTIVE SMEARS

- Collect between 12 to 20 filter papers (ashless) per composite sample
- Sandwich composite sample between two clean ashless filters;
- Place sample into petri dish and tape petri dish with electrical tape - a very good seal is required to prevent loss of mass which translates to increased concentration.

Note: ^{3}H values analyzed via this method may not be representative.

SAMPLE PREPARATION BY THE CONTRACTOR LABORATORY

Initial sample preparation performed at the laboratory is pivotal to the entire radioassay process. For proper quantification of the difficult-to-measure nuclides, several preparation steps are normally completed to ensure consistent and dependable results. These include:

- an initial non-destructive gamma spectrometric analysis;
- complete nuclide solubilization of the sample aliquant;
- tracer or elemental carrier addition; and
- the determination of nuclide recovery from the sample digestion process.

> Initial sample preparation is pivotal to good analytical results.

Initial Gamma-ray Spectrometric Analysis

An initial non-destructive gamma-ray spectrometric analysis of the sample submitted by the utility is typically performed by most laboratories. This initial analysis serves several purposes:

- to ensure that the utility and the laboratory agree on the content of the gamma-emitting radionuclides for future scaling ratio applications;
- to ensure that subsequent aliquants of the sample submitted by the utility are representative and homogeneous; and,
- to establish chemical recovery or losses of radionuclides in the sample digestion and nuclide solubilization processes.

For certain waste streams, this initial isotopic gamma analysis will quantify the majority of the radionuclides present. However, due to the extreme radionuclide activity ratios that may be encountered and the elevated Compton continuum for low energy gamma emitting radionuclides, some of the less radioactive analytes may not be detected. The most noted is ^{144}Ce, having a rather low gamma-ray intensity for its principal 133 keV gamma ray.

Complete Sample Digestion and Nuclide Solubilization

To provide quantitative measurement for a given analyte (nuclide of interest), the chemical recovery for the radiochemical technique must be determined (See next section). A basic premise for chemical recovery is the assumption that there is no fractionation (separation) between the analyte and chemical or radiotracer carrier in the sample. In addition, the analyte and carrier must be in the same chemical form. Without complete nuclide solubilization or sample digestion, fractionation may exists and only a portion of the analyte is in solution or has the same chemical form as the carrier. Therefore, during sample processing, not all the analyte chemically follows the carrier through the entire process, resulting in normal chemical recovery but low analyte concentrations.

The laboratory must have complete sample digestion or nuclide solubilization processes such as the various sample digestion, fusion and leaching processes. Commonly used processes are:

- Pyrosulfate fusion (combination of $NaHSO_4$, Na_2SO_4 and H_2SO_4, heated over a blast furnace);
- potassium-pyrosulfate fusions;
- acid digestions (HF for siliceous materials, HNO_3, $HClO_4$ and HCl); and,
- alkaline leaching/digestions for ^{14}C, ^{129}I and ^{99}Tc.

Quantitative Recovery For Each Radionuclide and Sample - Carrier Addition

Proper scaling factor determinations require that the radionuclide quantification be predicated on the actual chemical recovery for the analytical technique and specific sample under consideration. Typically, there are three generic methods that are applied to determine the radionuclide recovery for a radiochemical procedure. These include radiotracer, gravimetric and instrumental recovery determination of the chemical element. In each case, a known quantity of a radioisotope or stable element "carrier" is added to the sample at the beginning of sample preparation or chemical analysis and the amount in the final product measured to determine the chemical recovery for each sample. In some cases, due to the inherent radioactivity of the sample or emissions of the radionuclides of interest for a particular element, the tracer recovery technique can be more complex than simply adding a single tracer. The radiotracer and chemical carrier used in the determination of the recovery must be standardized with documentation, typically to within 3-5% precision. Radiotracers, which are not affected by chemical interferences, should be counted to produce a 3% uncertainty. For carriers and instrumental analysis recovery determinations, the radiochemical technique should be sufficiently robust to eliminate any chemical interferences that may affect the final recovery determination. For gravimetric determinations, the recovery precision should be less than 5%. However, the recovery precision by atomic emission or absorption spectrometry instrumental analysis can be greater and more variable than the other recovery techniques depending on the vintage and type of instrument, and operator experience.

> Accurate quantitative results depend on the measurement of the chemical recovery of the radiochemical process for each sample.

The precision of the recovery technique should be determined and documented in order to include this statistical parameter into the total propagated uncertainty of the sample radionuclide concentration. Statistically, the chemical recovery of a technique can exceed 100%; however not to a magnitude not covered by the precision of the recovery technique.

Table 5-2 summaries the typical recovery techniques and tracers used for the 10CFR61 difficult-to-measure radionuclides.

.Table 5-2
Important Parameters Associated with 10CFR61 Analyses

Nuclide	Half Life (Years)	Radiation Emitted (keV)	Radiation Detection	MDC[a] (nCi/g)	Recovery Technique
γ Emitters	>0.1	γ (100-2000)	Ge	Cs-137-10, Others-7000	N/A
^{3}H	12.3	β (18.6)	LS	400	--
^{14}C	5,730	β (156)	LS	80	Gravimetric
^{55}Fe	2.70	X (5.89)	LS	7000	IA
^{59}Ni	75,000	X (6.92)	LS, NaI	2200	IA
^{63}Ni	100	β (65.9)	LS, NaI	35	IA
^{90}Sr	28.6	β (546)	GP, β-β	0.4	^{85}Sr, Gravimetric
^{94}Nb	20,300	β (471), γ (703, 871)	γ, GP	2	IA
^{99}Tc	2.13 E5	β (294)	GP	30	^{99m}Tc
^{129}I	1.57 E7	β (152), (39.6)	GP	0.8	Gravimetric
^{237}Np	2.14 E6	α (4771, 4788), γ (86.5, 29.4)	α Spec	1	^{239}Np
^{238}Pu	87.8	α (5436, 5499)	α Spec	1	^{236,242}Pu
^{239}Pu	24,100	α(5105, 5143, 5155)	α Spec	1	^{236,242}Pu
^{240}Pu	6,570	α (5123, 5168)	α Spec	1	--
^{241}Pu	14.4	β (20.8)	LS	35	^{236,242}Pu
^{242}Pu	3.76 E5	α (4856, 4901)	α Spec	1	^{236}Pu
^{241}Am	432	α (5443, 5485), γ(59)	α Spec	1	^{243}Am
^{243}Am	7,380	α(5234, 5279), γ(74.7)	α Spec	1	^{243}Am
^{242}Cm	0.446	α(6069, 6113)	α Spec	200	^{243}Am
^{243}Cm	28.5	α(5742, 5784), γ (228, 278)	α Spec	1	^{243}Am
^{244}Cm	18.1	α (5763, 5805)	α Spec	1	^{243}Am

[a] MDC based on 1% of the limits listed in Table 2, 10CFR61.
Codes: LS - Liquid Scintillation; NaI - Thin NAI; GP - Gas Proportional; β-β - Dual Plastic Scintillation Detectors; γ - Gamma Spectrometry; β-γ- Beta - Gamma Coincidence Spectrometry; α Spec - Alpha Spectrometry; IA - Instrumental Analysis.

Post Fusion or Leaching Isotopic Gamma Analysis

As discussed earlier, incomplete digestion of the sample or solubilization of the analyte can lead to significant under-estimation of the analyte concentration and erratic and diverse scaling factors. To ensure that there have been no significant losses and that complete sample digestion and analyte solubilization has been effected for difficult waste stream matrices, a post-digestion, isotopic gamma analysis must be taken and the quantitative results compared to the results of the pre-digestion isotopic gamma analysis. Inherent in this practice is the assumption that the recovery of the gamma-emitting radionuclides is indicative of the recovery of the difficult-to-measure nuclides.

RADIOCHEMICAL TECHNIQUES

The quality of laboratory service analyses depends on several factors, including the use of robust validated procedures and qualified staff who have met proficiency tests, maintenance of technical depth and a pro-active quality assurance program. The laboratory should only use validated radiochemical techniques for the nuclides and media of interest. Normally, the analytical method selected and the staff chosen to perform the routine analyses depends on the quality (data quality objectives) and operational performance specifications within a contract or inherent in the Lab's internal performance objectives. Highly precise and unbiased measurements means that a sophisticated and robust technique and highly experienced staff must be used.

Procedure validation involves testing a procedure with known quantities of the analytes of interest, complete with chemical and radioactive interference consistent with the expected media to be routinely analyzed. Once a procedure has been validated to meet established quality and operational performance requirements (and documented as such), the laboratory staff engaged in the routine processing of samples are qualified in the technique by hands-on training and proficiency examinations. Proficiency examinations involve processing samples spiked with known quantities of analyte (single blind spikes).

Look for the following considerations in a radiochemical technique.

When reviewing a laboratory's procedures, ensure that the following aspects have been considered in their development:

- homogeneous and representativeness of the sample;
- proper scaling between nuclides to avoid interferences;

- selecting a technique for decontamination factors as great as 10^6 for other radionuclides;
- precise rather than arty techniques that can handle a wide variation of sample chemical composition and can be performed by different analysts to the same degree of quality;
- development based on established quality performance criteria; and,
- development based on operational performance criteria such as turn-around times, optimization of batch size and minimization of radioactive and hazardous wastes.

Procedures should be evaluated to determine if chemical and radionuclide interferences would be detected, how they are removed from the quantitative measurements and how the uncertainty of the interference removal process is propagated in the total propagated uncertainty of the analyte concentration. Table 5-3 lists some known nuclide interferences for various radiochemical procedures.

Table 5-3
Radiochemical Procedures - Known Interferences[1]

NUCLIDE	INTERFERENCE
^{241}Pu	^{141}Ce, ^{144}Ce, ^{60}Co
^{99}Tc	^{110m}Ag, ^{113}Sn, ^{99}Tc
^{63}Ni	^{51}Cr
^{59}Ni	Ni x-rays
^{14}C	^{103}Ru, ^{106}Ru
TRU	Am → Pu; Pu → Am
TRU	RESOLUTION
^{90}Sr	^{140}Ba, ^{60}Co

[1] The lower the detection requirement, the greater the potential for contaminants to cause radiochemical or radiometric interferences with the analyte.

Improper sample collection, preservation and radiochemical techniques can lead to a number of inaccuracies in nuclide quantification and scaling factor ratios. Table 5-4 is a partial listing of the major procedural deficiencies and their affect on the scaling factors ratios.

Table 5-4
Impact of Improper Sampling, Sample Preparation, and Radiochemical Processing on Scaling Factors[a]

Description	Impact
< MDC Value	Scaling Factor Based On MDC Value Overestimates Disposal Ci Quantities
Improper Preservation Techniques	Significant Losses Produce Low Nuclide Results And Erratic & Very Diverse Scaling Factors
Incomplete Sample Dissolution	Inhomogeneity Between Tracer And Analyte - Produces Low Nuclide Results And Erratic & Very Diverse Scaling Factors
Sample Homogeneity	Partitioning Between Aqueous & Suspended Material & Nuclides - Erratic & Diverse Nuclide Results And Scaling Factors
Non-Tracing Or Batch Recovery Factors	Incorrect Nuclide Results And Erratic & Diverse Scaling Factors
Technique Interferences & Poor Resolution	Increased MDC Values, Positive Bias In Nuclide Results - Biased Scaling Factors
Large Uncertainty In Analytical Result	Scaling Factor Ratio Would Also Have A Large Uncertainty Consistent With The Propagation Of The Individual Nuclide Measurement Uncertainties

[a] It is assumed that the utility and laboratory have compared the calibrations of their gamma-ray spectrometry systems and have system calibrations traceable to NIST.

THIRD PARTY INDEPENDENT QUALITY CONTROL PROGRAMS

All utilities should ensure that the laboratory maintains an internal as well as an independent third party quality assurance and control program consistent with ANSI and regulatory requirements. There are several third party quality control programs available commercially or through the government, such as the commercial radioactive source manufacturers and the Nuclear Energy Institute/National Institute of Standards and Technology (NEI/NIST) Traceability program and the Nuclear Regulatory Commission/Radiological Environmental Services Laboratory confirmatory program. Also, there are a few inter-utility sample crosscheck programs. Each of these programs must maintain their traceability to NIST in accordance with ANSI guidance for traceability and monitoring laboratory activities. These programs provide quality control samples of various media, nuclides and analyte concentrations consistent with some of the nuclides and matrices for 10CFR61 waste streams. Some programs are more limited than others in terms of the diversity of media and nuclides, and sample distribution frequency. It should be noted that some of the programs may not have pre-determined quality performance criteria consistent with a utility's data quality objectives.

There are two mechanisms available to a utility to ensure that the laboratory participates in a third party quality assurance and control program. The first mechanism is for the utility to acquire the services of the organization maintaining the program and have the samples forwarded to the laboratory by either the organization or as a pass through from the utility. The second mechanism is to have the laboratory participate directly in a third party quality assurance and control program as specified within a utility/laboratory contract for services. In either case, it is recommended that the quality performance criteria for precision and bias be stipulated within the utility/laboratory contract.

6

REPRESENTATIVE SAMPLING

The unique hazard presented by radioactive material can pose a challenge to representative waste sampling. For many types of waste, concerns about representative samples can be addressed by obtaining more samples and applying statistical methods to the analytical results. The ALARA principle does not allow this "brute force" solution for higher activity radwaste to address this concern. The cost of off-site radiochemical analysis can be a significant percentage of the total radwaste cost adding a further disincentive for multiple sampling.

Off-site radiochemical laboratories have activity and radiation receipt limits on samples. These limits also pose a challenge in obtaining representative samples. In general, no sample over one R/hr can be sent to an off-site laboratory. Should a representative field sample exceed this limit a portion of the sample, or aliquot, must be obtained and sent to the laboratory. This aliquot may not be truly representative of the material being examined.

For radwaste sampling a limited number of small samples are required to comply with ALARA. The challenge is to use a technique which will provide a representative sample versus obtaining multiple samples.

Radwaste samples are obtained to provide one of two types of results. These result types are: 1) relative abundance; and, 2) absolute concentration. Relative abundance results yield the proper ratio of the nuclides in the spectrum (*e.g.*, scaling factors). Relative abundance results must be coupled with dose-to-curie calculations to determine the curie content of the waste package. Absolute nuclide concentration results need only be multiplied by the mass or volume of waste in the package to determine the curie content.

The following techniques have been accepted by regulators as a means of compliance with the intent of 10CFR61.

> Each utility is responsible for justifying that the method chosen results in a sample that is representative of the waste stream.

DRY ACTIVE WASTE

Representative samples of dry active waste (DAW) for radionuclide analysis can be obtained using surrogate or direct samples. For either case, analytical results will provide the relative abundance of the nuclides in the spectrum. Dose-to-curie calculations must be used in concert with the relative abundance to determine the activity of a DAW package.

Sample Techniques

Surrogate samples for DAW are usually used. This practice is accepted by the NRC since this waste is a small portion of the LLW source term. One technique requires the collection of several smears from various locations around the facility. Locations likely to yield contaminated smears include primary sample sinks, ventilation ducts and floor drains. Rarely will smears of actual contaminated trash alone provide successful results. The smears are stacked in a petri dish and sent to the off-site laboratory for analysis.

Utilities should contact their individual laboratories for packaging specifics and details.

> *Lessons Learned: Some RCA clean trash survey programs require a spectrum to be obtained. The use of a DAW spectrum for RCA trash may result in high reject rates. To support such "green is clean" programs, a separate spectrum for clean trash from the RCA may be beneficial. Representative surrogate samples for RCA clean trash are almost impossible to obtain. Direct sampling of RCA clean trash provides a representative spectrum. Direct sampling of trash can be accomplished by removing several items from trash bags and placing them in a small pail or drum. Successive samples from multiple bags are obtained until the drum is hand packed full. The container is sent to the off-site laboratory for ashing and analysis. This method can also be applied to DAW.*

CARTRIDGE FILTERS

Representative samples of cartridge filters may be obtained from direct samples or surrogates. Analytical results will provide the relative abundance of the nuclides in the spectrum. Dose-to-curie calculations must be used in concert with the relative abundance to determine the activity of a filter.

Sample Techniques

Low-activity filters: Obtain a sample directly. The filter media sample may be obtained by cutting, sawing or drilling of the media. This method may also be used for HEPA filters; however, their activity levels are usually so low that a DAW sample will provide an adequate spectrum.

High-activity filters: Obtaining a sample of the actual media will provide a representative relative abundance of the radionuclides. Smears of vessel internals or cartridge end- caps may not provide a representative spectrum. Smears of the filter media may not provide a representative sample, as discussed in Appendix E, since:

- Smears of wet media will tend to absorb liquid which will overestimate the activity of soluble nuclides such as cesium in the filter media.
- Smears will not remove sub-micron particulates and can underestimate the activity of ^{14}C.

A hole-saw drill bit can be used to quickly extract media samples from a high activity cartridge, see Figure 6-1. The hole-saw method enables a sample to be obtained from a filter in a shield with a side port to reduce exposure to personnel. The media sample is packaged by tapping the hole-saw bit into a petri dish. Some utilities use filter clippings as a sample.

Most cartridge filters exhibit little axial dose gradient. This should be evaluated at your site. In such cases, the sample may be taken from anywhere along the axis of the cartridge. The down-flow, inside-out vacuum filters are an exception. For these filters a sample near the bottom of the cartridge must be obtained. Figure 6-2 shows an underwater vacuum filter cask with a side port near the bottom end of the cartridge to permit sampling.

One alternative method to obtain representative samples for high activity filters is the use of an instrument sample line filter. Installation of an instrument sample filter in parallel with the filter vessel must be performed. The sample filter media must match the pore size rating and the material type of the actual cartridge filter. The flux (gpm/ft^2) through the sample filter should also match that of the actual filter cartridge.

Another alternative method is the use of installed coolant sample lines. A predetermined volume of coolant is drawn and passed through a crud filter. One disadvantage of this method is that filter papers matching the micron size of installed filters may not be available. The coolant sample may also be sent to the off-site laboratory for analysis without filtering.

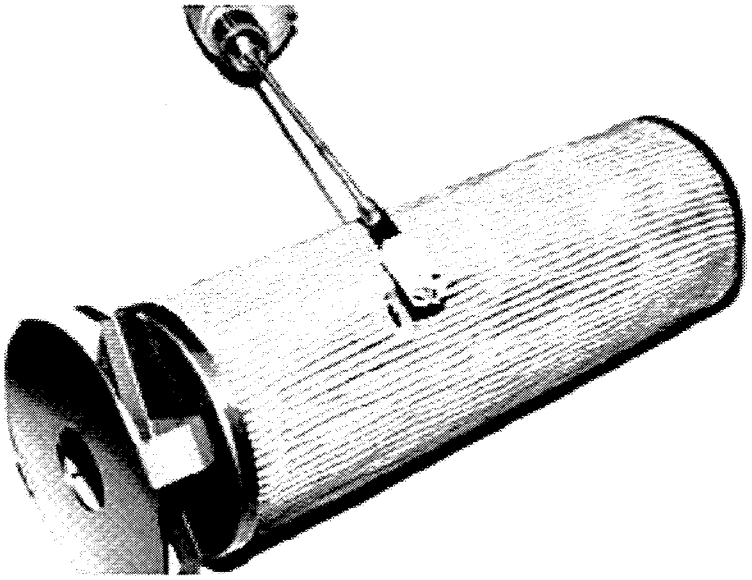

Figure 6-1 Hole Saw Drill Sample from Cartridge Filter

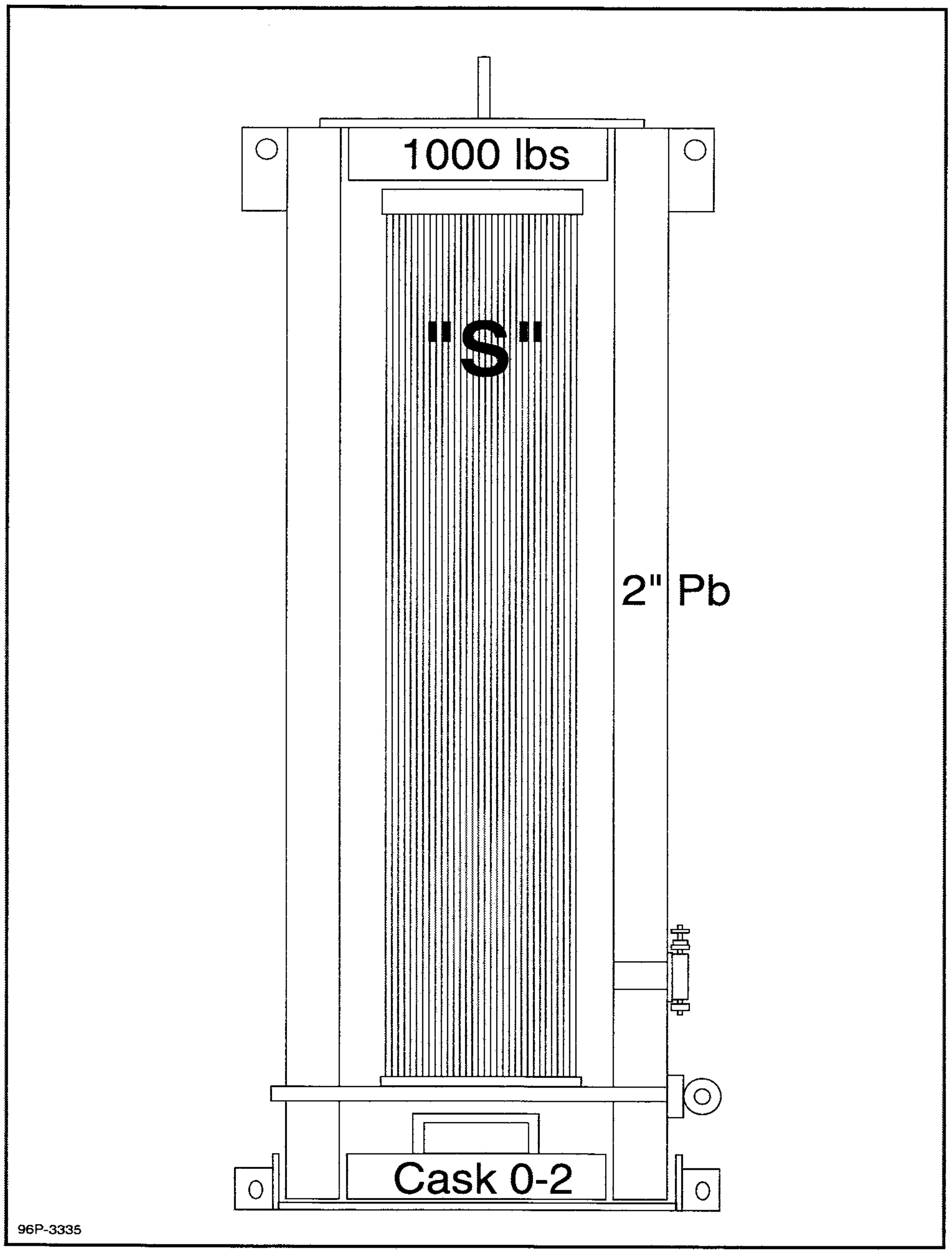

Figure 6-2 Underwater Vacuum Filter Cask with Side Port

Table 6-1 summarizes the advantages and disadvantages of the methods used to collect representative samples for cartridge filters.

Table 6-1
Comparison of Filter Sampling Alternatives

METHOD	HARDWARE	ADVANTAGES	DISADVANTAGES
Smear Samples	• Long-handled Tools • Shield bells	• Easy • ALARA (w/shield) • Inexpensive	• Surrogate Material (not filter material) • May over-report Cs and under-report ^{14}C (depending on filter pore size) • Where to smear? (handle, vessel, etc.)
Core Sampling	• Shield bell w/ side port (different sizes) • Hole saw w/ extension bit • Angle cutters (for low-activity filters)	• Actual filter material • More accurate Cs/^{14}C numbers (for smaller sub-micron filters)	• More expensive than smears • Storage space for shield bells
Instrument Sampling (parallel sample)	• Sample Tap • Filter holder • Filter disk (matches filter vessel) • Flow validation	• No shield bell • ALARA (some plants) • Allows for testing of filter options	• Representative flow? • Representative filtration? • Cost (some plants) • ALARA (some plants) • Depends on point in fuel cycle and operational transients • Extended run time to assure representative Cs
Filtered Reactor Coolant	• Existing Sample Tap	• Easy • ALARA • Geometric distribution of isotopes on filters not a factor • Inexpensive	• Surrogate material (not filter material) • Transients may affect analysis results • May have higher cesium levels than actual filters

ION EXCHANGE MEDIA

Deep bed ion exchange media may consist of cation bead resin, anion bead resin, mixtures of these resin beads or inorganic media (e.g., zeolite, carbon, glass). These media remove radionuclides by ion exchange, adsorption and filtration. The beads or granules at the top of the bed will contain more nuclides than material at the bottom of the bed. This can be verified by a vertical radiation survey or "profile" of the vessel. Particulate material will also be concentrated at the top of the bed adding to the dose gradient of the vessel.

Spent media beads or granules are inhomogeneous in radionuclide concentration in the vessel. The importance of inhomogeneity varies in proportion to the activity (radiation) level of the bed. This is observed on beds with high radiation levels that have a large vertical gradient. These conditions present a challenge to representative sampling.

Spent media is transferred from process vessels into spent media storage tanks or directly into disposable containers. The act of transferring spent media from process vessels mixes the media beads or granules to an extent. Transferring multiple beds into a storage tank or disposal container may result in discrete layers.

In general, air sparging of a media storage tank will not provide sufficient mixing to thoroughly blend the contents. Recirculation of storage tank contents may produce homogeneous spent media. In-tank mixers can produce homogeneous mixtures. However, many U.S. plant media storage tanks do not have recirculation or in-tank mixing capability.

Solidification disposal containers with internal mixers can produce homogeneous contents. Dewatering liners and HICs are not fitted with internal mixers. If inhomogeneous media or multiple transfers of dissimilar media are made into a dewatering container, the contents may be inhomogeneous.

The homogeneity (activity and isotopic distribution) of ion exchange media may not exist for the following reasons:

<u>SINGLE RESIN BED</u>

- different resin types
- beds mixed with other filter media (e.g., activated carbon)
- dose/depletion gradients within beds
- particulate distribution within beds
- channeling within beds

SPENT RESIN STORAGE TANK

- layering effect from different resin beds
- no recirculation capabilities for tank
- different resin types
- beds mixed with other filter media (e.g., activated carbon)
- dose/depletion gradients within beds
- particulate distribution within beds
- channeling within beds

These factors should be considered when obtaining representative samples for characterization of this waste stream.

Sample Techniques

Several methods are available to obtain ion exchange media samples. These include grab samples, core samples and continuous transfer in-line samples. The sample method selected to obtain a representative sample of ion exchange media may differ depending on the activity of the media.

Grab samples can be obtained directly from the disposal container or from a sample tap on a transfer line. A grab sample can provide a representative sample for absolute nuclide concentrations only if the media has been thoroughly mixed prior to obtaining the sample. This mixing could occur in the storage tank or the disposal container.

If mixing is not available, a grab sample can only provide the relative abundance of nuclides from a single bed. Relative abundance of nuclides must be used in concert with the radiation level of the bed and a dose-to-curie calculation to determine the total activity of the bed.

Core samples are obtained directly from disposal containers. A core sample will extract media from each layer in the container. A pipe with holes along its length or a "grain sampler" provide excellent core samples. The entire volume of the core sample must be analyzed to provide representative absolute nuclide concentrations.

- For low activity media, core samples are a means of providing representative samples.
- For medium and high activity media, core samples are NOT recommended.
    1. The method is not in keeping with ALARA.
    2. In addition, the entire sample may exceed laboratory radiation limits (e.g., 1 R/hr), and an aliquot of the sample is required. To insure that the

aliquot is representative, the entire sample volume must be thoroughly mixed or digested before obtaining the aliquot.

In-line samples are obtained from tank recirculation or media transfer lines. Several commercial in-line media sampling devices are available. These units are used to obtain several samples of media over the duration of a slurry transfer or tank recirculation. These samplers extract a small quantity (e.g., 5-10cc) of media from the slurry and place it in a sample bottle with each actuation and flush of the media, Figure 6-3. The units can take multiple samples over the course of a media transfer. Some plants have extended the flush lines to enable routing the media into a shielded sample bottle to improve ALARA.

If the media has already been mixed, a small quantity of waste can be obtained from the beginning, middle and end of the transfer to provide a sample representative in absolute concentrations.

If the media has not been mixed, the unit must obtain samples through the duration of the transfer in order to provide a representative sample. Since a transfer requires about 20 minutes, the representative sample volume will be 100 to 200 cc.

This sample will be representative of the batch contents. If the batch contents were layered, the sample bottle contents will be layered.

Analysis of the entire sample is required to provide absolute nuclide concentrations. This is readily achieved for low and medium activity media.

For high activity media, the entire sample must be thoroughly mixed or digested to assure that the aliquot remains representative. For details on a method to digest resin see Appendix B.

Alternatively, an aliquot of the high activity sample will provide the representative relative abundance of nuclides in the media. Some plants have utilized these relative abundance results in concert with the radiation level of the sample vial and a dose-to-curie calculation to determine the absolute nuclide concentrations for the media. This method avoids the need to mix or digest the entire sample prior to obtaining an aliquot and overcomes any inaccuracies incurred by aliquoting. It also avoids the need to obtain the "bare" dose rate of a high activity disposal container.

FILTER PRECOAT

Spent precoat filter aid may consist of powdered resin, cellulose, diatomaceous earth, carbon or mixtures of these materials. These materials are applied to filter septa in a thin coat and thus exhaust uniformly. These materials are slurried to tanks and mix well due to the small particle size. Sampling methods used for bead resin are routinely

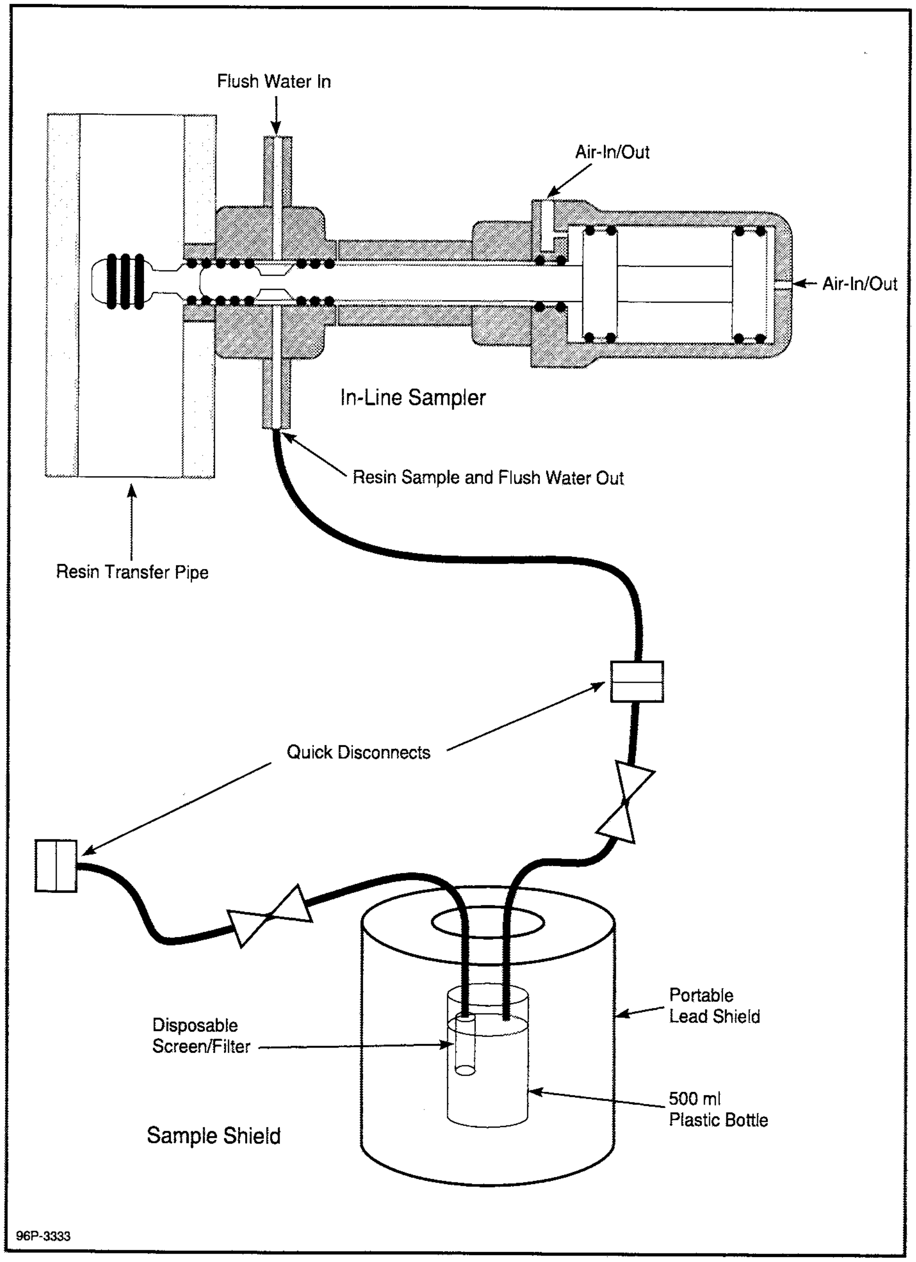

Figure 6-3 Example of In-Line Sampling

employed. In general, due to the homogeneous nature of the spent material, grab samples provide a representative sample.

TANK AND SUMP SEDIMENT

Representative samples of tank or sump sediments may be obtained by a grab or core sample. Like the case for ion exchange media, the sample method depends on whether the sediment is homogeneous.

CONCENTRATES (DRYER SALT)

Representative samples of concentrated liquids may be obtained by a grab sample of a recirculated tank. Presently, few plants directly solidify concentrated liquids. This sampling technique may only be required for plants that send concentrated liquids off-site for volume reduction.

Plants that produce concentrates generally dry this waste in volume reduction equipment. These volume reduction units may or may not use a binder (e.g., paraffin, asphalt). The drying process will drive off ^{3}H and other volatile nuclides from the spectrum of the feed concentrated liquid. For systems which use asphalt as a binder a sample of the concentrated feed solution is obtained and the spectrum of the product is calculated based upon the waste loading and the removal of volatiles. Representative samples of product from volume reduction units which do not use asphalt can use different techniques.

One advantage of these volume reduction drying units for sampling is that they thoroughly mix the waste.

- For units which use paraffin, a core sample of the product drum will provide a representative sample. The core is obtained by inserting a pipe into the drum prior to hardening of the paraffin.
- If no binder is used, a grab sample from the product drum will provide a representative sample.

The analysis of these samples will yield representative absolute nuclide concentrations for the batch. The dose rates from the drums produced by drying a batch of liquid concentrates may vary. The sample results also provide the relative abundance of the nuclides in the spectrum. Dose-to-curie calculations must be used in concert with the relative abundance to determine the activity of these packages with varying dose rates.

7

USE OF LABORATORY DATA

To determine the concentration of radionuclides in a package of low level waste both on-site and off-site laboratory analysis are used. On-site analysis is in general limited to gamma emitting nuclides. Off-site analysis is used to determine non- gamma emitting or difficult-to-measure (DTM) nuclides in addition to gamma emitting nuclides. The NRC's BTP on waste classification requires that the method of curie content determination be accurate to within a factor of ten.

VALIDATION OF LABORATORY RESULTS

The utility should evaluate the validity and suitability of the laboratory's analytical results. In some cases, it is easier to evaluate the validity of the results once scaling factor ratios are calculated. The following recommendations are provided to validate analytical data:

- Laboratory to Utility Intercomparisons
    - Compare the utility's gamma isotopic concentration (sample) results (taken prior to shipping sample) to those determined by the laboratory for consistency.
    - During audits, review pre- and post-fusion (sample digestion) isotopic gamma results for nuclide losses and compare results to the utility's prior to shipping values.
- Nuclide Quantification and Detection Requirements
    - Reported minimum detectable concentration (MDC)** *(Ref. 1)* of nuclide (or nuclide category) must be less than the required detection limit (RDL) of 0.01 of Table 1 of 10CFR61

      ** MDC is used synonymous with lower limit of detection (LLD)
    - Reported MDC should be consistent with the measurement uncertainty. The relative uncertainty (1 σ) of the measurement should be ~ 30% at the MDC and should get smaller as the measured concentration increases above the MDC level; e.g. at a concentration ten times the MDC, the relative uncertainty (1 σ) should be typically between 10 and 20%.

  - The reported value should be considered real (*i.e.*, different from background and a positive activity with a 5% false detection probability) if the reported value ≥ 1.65 x its uncertainty (1 σ) of result (*Ref. 1*).
- Generic Laboratory and QA items
  - Ensure the successful implementation of the guidance presented under contract specifications.
  - Perform quality assurance audits of the laboratory's operation and internal quality control program.
  - Evaluate blind duplicate and third party quality control samples.

Data should be reviewed for consistency for a given waste stream with expected historic trends. Examples of data checks are provided below.

- Reported units and power of tens (negative versus positive) should be consistent with expected values - previous site-specific data or generic scaling factors.
- Short-lived gamma-emitting nuclides should be consistent with nuclides measured in the primary coolant for a given fuel cycle. Look for metal neutron activation products tags such as ^{110m}Ag and ^{108}Ag that are consistent with historic data. Look for unrealistic short-lived gamma emitting or activation product nuclides.
- Ratio of short-lived to long-lived gamma-emitting nuclides for a given matrix, age of matrix, waste stream and plant operation should be within expected ranges.
- Ratio of short-lived to long-lived isotopes for a given matrix, age of matrix, waste stream and plant operation should be within expected ranges - ^{89}Sr/^{90}Sr, ^{134}Cs/^{137}Cs, ^{241}Pu/^{239}Pu, ^{242}Cm/^{243,244}Cm and ^{63}Ni/^{59}Ni when measured.
- Ranking of the ratios of the long-lived isotopes, nuclides or nuclide categories should be consistent for a given waste stream; TRU/^{239}Pu, ^{55}Fe/^{60}Co/^{54}Mn, ^{90}Sr/^{137}Cs, etc.
- Magnitude of the ^{3}H concentration in the primary coolant to its concentration in other waste streams should be consistent; the ^{3}H concentration in the other waste streams should be ≤ the primary coolant value.
- Reported ^{129}I and ^{99}Tc concentration values should be consistent with the expected primary coolant values as determined by reactor physics algorithms (See Section 7).
- Ranking of TRU nuclides by concentration is typically ^{241}Pu >> ^{242}Cm > ^{238}Pu ~ ^{239}Pu ~ ^{241}Am ~ ^{244}Cm but can vary for any given sample.

SCALING FACTORS

Correlation or scaling factors are the ratio of DTM nuclides to key gamma-emitting nuclides. These factors, derived from off-site analysis, can be used with subsequent on-site gamma analysis to determine waste container curie content. Scaling factors can also be used to determine waste streams and trend waste stream spectra over time. The BTP allows the use of this technique as an accurate, cost effective and timely means of compliance.

The key gamma emitters used for scaling factor determination must be detectable by on-site analysis. Although ^{144}Ce correlates very well to transuranic nuclides, difficulty in detection in on-site analyses limits its use as a key gamma-emitting nuclide. Therefore, ^{137}Cs has been used as the key fission product gamma-emitting and ^{60}Co as the key corrosion product gamma-emitting nuclide. DTM fission products are scaled to ^{137}Cs while DTM corrosion products are scaled to ^{60}Co. These key nuclides are found together in most waste streams.

Based upon recent industry data, reliance on ^{137}Cs as a key gamma-emitting nuclide has waned *(Ref. 2)*. Transuranic nuclides and ^{99}Tc, important to waste classification, result from tramp uranium on the fuel rather than from fuel leakage. The quantity of these nuclides is not greatly affected by fuel defects. The quantity of cesium nuclides in the RCS is, however, greatly affected by fuel defects. In addition, there are waste streams in which ^{137}Cs concentrations are quite small. Cartridge filters, in general, do not remove soluble species such as ^{137}Cs. Cesium-137 cannot be used as a key nuclide for scaling when its concentration is reported as an MDC without introducing significant error *(Ref. 2)*.

The combination of these issues has led to the wide use of ^{60}Co as a key gamma nuclide for transuranics, ^{99}Tc and ^{129}I. In addition, other transuranics have been shown to correlate extremely well with ^{239}Pu. This has led to the use of ^{60}Co as a key nuclide for ^{239}Pu, and the remaining transuranics are scaled to the ^{239}Pu value. The advantage of using ^{60}Co as the primary key gamma emitter for all DTM nuclides is that new samples are not required due to small fuel defects. This avoids over reporting activity or revising waste scaling factors if ^{137}Cs had been used.

Annual and Quarterly Waste Stream Analysis

The BTP specifies that for Class B and C waste streams a confirmatory analysis should be performed annually to insure classification accuracy to within a factor of ten. Confirmatory analysis for Class A waste, like DAW, should be performed at least biennially.

As a practical definition for waste classification purposes, a waste stream is defined as any waste product or mixture of products where the DTM nuclide concentrations can be inferred by use of a single set of scaling factors.

Waste Stream Trending

For a specific waste stream, a sampling schedule should include an annual (biennual for Class A waste) off-site analysis to determine scaling factors. This may be followed at least quarterly by an on-site analysis to check the scaling factors. The ^{137}Cs-to-^{60}Co ratio is a convenient figure to compare the off-site annual spectrum to the quarterly on-site analysis. This factor represents the ratio of fission products to corrosion products and will usually indicate if there has been a significant change in the waste spectrum. In typical use, if the annual ^{137}Cs/^{60}Co ratio and the quarterly ^{137}Cs/^{60}Co ratio are within a factor of ten, the annual scaling factors are still valid. If these ratios are not within a factor of ten, an evaluation for a new off-site analysis should be made.

Use of the ^{137}Cs/^{60}Co ratio for scaling factor validity is commonly used for consistent waste streams like DAW, low activity resin and concentrates. Spent cartridge filters may also be routinely generated; however, cesium nuclides may not be present in sufficient quantities for on-site determination.

> *Lessons Learned: A change in the ^{137}Cs/^{60}Co ratio does not necessarily indicate a need to resample. The ^{137}Cs/^{60}Co ratio will increase with fuel defects, but the defect may not alter the concentration of transuranic, or ^{99}Tc. Unless ^{137}Cs was used to scale TRU or ^{99}Tc, a reanalysis is unnecessary. Alternatively, the ^{137}Cs/^{60}Co ratio could be affected by an influx of resuspended crud. Use of the ^{137}Cs/^{60}Co ratio can serve as an indicator of system changes but not the cause of the change. This practice for scaling factor checks provides no direction on what action to take unless absolute concentrations of the two nuclides are also tracked.*

Reactor Coolant Analysis for Waste Stream Trending

Analyzing and trending reactor coolant concentrations is an alternative to direct sampling to validate waste stream scaling factors. The ratio of ^{134}I to ^{60}Co in reactor coolant measures the amount of exposed fuel to activation products *(Ref. 3)*. This ratio can indicate if the waste spectrum has changed by a factor of ten. Other reactor coolant nuclide ratios (e.g., ^{134}I/^{131}I exposed fuel fraction, ^{133}I/^{131}I release index, ^{134}Cs/^{137}Cs exposed fuel age) will not indicate a change in waste spectra *(Ref. 3)*. If reactor coolant

^{134}I concentrations have remained unchanged relative to ^{60}Co, fission product scaling factors are still valid.

The advantage of reactor coolant analysis for scaling factor checks is that direct waste sampling is avoided. This can save time since reactor coolant is routinely analyzed for fuel performance anyway. In addition, worker exposure obtained by direct waste sampling can be avoided. Eliminating sampling also avoids any concerns about the representativeness of the direct waste samples.

Updating Scaling Factors

There are two methods commonly used to update waste stream scaling factors, replacement and historic scaling factor analyses. In the replacement method old scaling factors are replaced by new factors calculated from the latest off-site analysis. The advantage of replacement method is that it is simple. Replacement is a strict and rigorous rule. The disadvantage of the replacement method is that it does not make use of historic information and is heavily reliant on the accuracy of the most recent off-site analysis. Historic information can be useful in validating new scaling factors.

The other method for updating scaling factors involves combining new scaling factors with appropriate historic scaling factors. A log-mean average of the specific scaling factor is calculated from historic data to compare with the most recent analysis. The advantage of this method is that new analysis results are verified using historic data. This method can eliminate the concern that the recent sample was not representative of the waste stream. A sample result which is not within a factor (usually of ten) of the historic scaling factor data when there has been no change in fuel performance or component erosion indicates that the sample was not representative, that the analysis was inaccurate, or, less likely, that there were undetected operational changes. If inspection rejects the latter possibility, such sample data is an outlier and should not be used.

Spreadsheets can be developed or commercial software programs can be obtained to enable rapid comparison of scaling factors using these statistical methods. One of the commercial programs includes useful graphic presentations along with the statistical analyses.

INDUSTRY-WIDE (GENERIC) SCALING FACTORS

Overview

Since the onset of the regulatory requirements in 10 CFR part 61 in 1983, U.S. utilities have analyzed samples from radioactive streams in their nuclear plants. These analyses included radiochemical determinations of the absolute and relative abundance of all important radionuclides in samples taken for classification of the low-level wastes. These included analyses for gamma-ray, beta-particle and alpha-particle emitting nuclides. The Electric Power Research Institute, EPRI, assembled a data base that includes results from analyses of over 2900 samples from 23 distinct waste streams in 23 U.S. utility boiling-water (BWR) and 44 pressurized-water (PWR) reactors. One of the benefits of the data base is that it allows statistical analyses to determine any trending or constancy of scaling factors on individual plant, waste stream, or industry bases. Use of constant scaling factors, generic throughout the industry, if justified would be of value to the nuclear industry in aspects of cost savings, reduction of personnel radiation exposure, and improved accuracy in reporting of disposed radioactivity. An analysis has identified some scaling factors for hard-to-measure nuclides in LLW that are constant well within the factor of ten accuracy considered appropriate by the NRC in its Branch Technical Positions.

Statistical analyses of the EPRI data show unequivocal trending and consistency of values that would justify the use of generic scaling factors on industry-wide bases for:

- ^{55}Fe, ^{59}Ni, and ^{63}Ni, relative to ^{60}Co,
- ^{238}Pu, ^{241}Pu, ^{241}Am, ^{242}Cm , ^{243}Cm and total alpha, relative to ^{239}Pu, and
- ^{239}Pu, relative to ^{144}Ce.

Use of generic factors could significantly reduce costs for off site sample analyses, decrease reporting delays, reduce personnel radiation exposure by reducing sampling frequency, and improve accuracy in activity-content reporting. Although the use of generic scaling factors has yet to be approved for use in the U.S., their potential was already recognized by the Nuclear Regulatory Commission (NRC) in their Branch Technical Position (BTP) of 1983 where they discussed the generic potential of the ^{239}Pu - ^{144}Ce correlation. Most plants do not observe the presence of ^{144}Ce in gamma-ray analyses of their waste samples and, hence, cannot utilize the well established correlation to ^{239}Pu.

A report, *EPRI NP-5497*, *Advanced Radioactive Waste Assay Methods*, suggests a method for cerium purification and enhancement that could allow a plant to observe this activity in its wastes and thereby use the correlation, eliminating need for most TRU radiochemical analyses. This procedure is included as Appendix D.

However generic factors are derived, from the industry or individual plant data bases, they are in a practical sense, *plant-specific factors.* Their use as generic factors would require verification to assure their validity to the wastes in the plant proposing their use. Verification could usually be as simple as validating the plant historical analysis results to the industry factors and periodically verifying that the trending continues. Performing statistical comparisons for trending verification using plant historical data can provide defensible documentation. The use of computations with computer spreadsheets or off-the-shelf software programs make these analyses relatively easy. New power plants might be required to sample their waste streams over the early reactor cycles and compare results to the generic factors for verification. Actual comparisons at several stations yielded results for factors had dispersions of two in some measured factors. Additional site- or waste-stream specific factors may appear that would qualify for use by that plant.

EPRI Sample Analysis Data Base

Since the introduction of 10CFR61 in 1983, commercial nuclear power generators have implemented a program of radioactive waste classification based on sampling and radiochemical analyses and the use of site-specific scaling factors. An EPRI project (Research Project 2414-53) compiled the results from the analyses of about 2900 waste stream samples, collected from utilities, that included the period from 1982 until 1992. The results came from several commercial and DOE laboratories used by utilities to analyze their samples. The EPRI program compiled these data into two computer databases, one each for PWR and BWR data. The database now exists in several widely used commercially available database and spreadsheet formats.

The databases contain measured concentrations for 24 radionuclides that include all the nuclides important for 10CFR61 waste classification, 18 PWR and 13 BWR distinct waste streams, although not all the samples contained positive results for all of the nuclides. The database does not include the minimum-detectable limit (MDL) values given for many entries in the analysis data sheets. The PWR base contains entries for 1913 samples and the BWR base, 964 samples. Included in the data base are the plant and unit number identification, waste stream identification and sampling date. The analysis laboratory can usually be ascertained from the sample numbering identification. For PWRs, waste stream identification includes eight resin stream types, six cartridge filter stream types, dry-active waste (DAW), evaporator bottoms, charcoal filters, and reactor coolant. For BWRs, the data base includes waste streams of three cartridge filter, six resin, filter sediments, radwaste evaporator, and DAW streams, and reactor coolant.

Statistical Analysis Formalism

Data trending and statistical analyses of the data consisted primarily of calculating geometric means (log means) and dispersions for the nuclide ratios [scaling factors

(SF)], of practical importance. Studies used log-mean analyses because of the wide numerical range in the data and because the values showed a log-mean distribution. Curve-fitting regression analyses have also been used to analyze the data. Although they are statistically valid, they are not much used currently in the U.S. because the fitted curves do not appear to be through the center distribution of the measured values. Scaling factors of particular interest are those relating the difficult-to-measure nuclides to ^{60}Co and ^{137}Cs and the transuranic (TRU) isotopes to ^{239}Pu. Although inspection for and rejection of statistical outliers in the data would have improved the trending results to some respect, analyses did not exclude them in order to demonstrate trending with any value, even the *suspected "bad"* values.

Log-Mean (Geometric Average)

Equation 1 shows the analysis to obtain a log-mean of a set of data:

$$A_{SF} = e^{\frac{\sum_{i-1}^{N} \mathrm{Ln}(SF)_i}{N}} \qquad (1)$$

where,

A_{SF} is the log-mean of a scaling factor set,

$(SF)_i$ is the ith value of the scaling factor in the set for averaging, and

N is the number of values in the set.

Dispersion

The dispersion is defined as the standard deviation of the factors by which the individual values differ from the log-mean. Equation 2 shows the equation for computing the value of the dispersion in a set of values.

$$D_{SF} = e\sqrt{\frac{\sum_{i=1}^{N}[\,\mathrm{Ln}(SF)_i - \mathrm{Ln}(A_{SF})\,]^2}{N}} \qquad (2)$$

where:

> D_{SF} is the one-sigma dispersion in the scaling factor distribution of the set, and the other terms are as defined above.

This is a typical standard-deviation formalism, but done in "log space" where it results in a standard deviation from the norm of a factor rather than a value. A one-sigma dispersion in a SF for a given set of data (waste stream) implies that any new individual measurement of that SF for that stream will have a 68.3% probability of being within the factor of the dispersion value of the log-mean. Using a two-sigma (2σ) dispersion value means that any new measurement of the scaling factor would have a probability of 95.5% of being within that factor from the log mean. Hence, a large measured scaling factor distribution with a low 2σ dispersion value should justify the scaling factor's use as a generic factor.

In application to specific plants, once all waste streams are identified, searches for generic scaling factors can commence through determination of dispersion variations within historical results -- as far back in time as the data and variances will allow. These search should include any factors additional to those proposed as industry generic and listed above. Selection of allowable dispersion values is subjective although the NRC allows a accuracy (assumed 2σ dispersion) of 10. A conservative set of criteria might be that if the variation (2σ) were near 4, use of a constant factor is justified for all waste streams; if the variation (2σ) were near 5, the assumption is that different waste streams may have differing factors - calling for waste stream specific factors; a 2σ dispersion value near or above 10 indicates a usable constant factor for that correlation is not available.

Statistical Analysis Results

Analyses considered all possible scaling factors for the DTM nuclides to both ^{60}Co and ^{137}Cs, of ^{239}Pu to ^{144}Ce, and all the TRU nuclides to ^{239}Pu. From these analyses, only the DTM nuclides ^{55}Fe, ^{59}Ni, ^{63}Ni, and TRU gave promise of offering generic scaling factors. Hence, the further discussion includes only those nuclides.

Criteria for Generic Factors

A criterion for consideration of a scaling factor as generic is that the dispersion be below 10, the accuracy requirement given as guidance by the NRC in ***Low-Level Waste Licensing Branch Technical Position on Radioactive Waste Classification, 11 May 1983***. For a factor to be generic on a waste-stream specific basis, the dispersion for that stream should be less than 10. If it is to be generic on a plant or industry wide basis, the dispersion for all samples in the data base for that nuclide should be below 10. An additional criterion for a plant and industry generic factor is that the log-mean value for each stream should be consistent with the log-mean average for all samples.

Pressurized-Water Reactors. Table 7-1 summarizes the analysis results of PWR resins. The table gives the number of positive values in the database, and the log-mean average and the ***two-sigma*** dispersion values for each scaling factor and each waste stream. Table 7-2 presents a similar summary of PWR cartridge filter waste stream data analyses. Table 7-3 summarizes the analyses of the four PWR principal waste streams, resins, filter cartridges, DAW, and evaporator bottoms, and the reactor coolant samples and gives the results considering all PWR samples.

Scaling factors for resin samples, shown in Table 7-3, satisfy the criteria stated earlier for consideration as generic factors for all PWR resins, with the possible exception of ^{55}Fe - ^{60}Co. For the others, the two-sigma dispersion values are well within the factor of ten criterion and internal agreement of the log-mean averages are within a factor of 3 from the average. The data for ^{55}Fe show relatively high dispersion values and considerable differences in the log means for the different resin streams. The data for PWR filter cartridge waste streams, shown in Table 7-2, also meet the criteria for consideration as generic factors. The data for ^{55}Fe in these waste streams show considerably less scatter and lower dispersion values in waste filters, as might be expected since Fe is generally present as a colloid in reactor systems. A summary of all PWR waste streams, given in Table 7-3, indicates that all of the factors should be considered for PWR industry-wide scaling factors. The highest dispersion values in these data occur in reactor coolant, which is not a waste stream and is included only for comparison. Consistency of the values between all of the waste streams and the low values of dispersion would indicate their acceptance as generic. The ^{55}Fe scaling factor may be waste-stream generic, differentiating between filtration and ion-exchange media, whereas the others, listed in the last column of Table 7-3, should be industry-wide generic.

Boiling-Water Reactors. Table 7-4 presents a summary of statistical analyses of waste resin streams in BWRs and Table 7-5 summarizes the analyses for all BWR waste streams.

Consistency of scaling factor log-mean averages within the waste streams of BWRs and values of dispersions from the statistical analyses also satisfy the criteria for consideration as generic factors. As in the PWR data, the ^{55}Fe scaling factors have the largest internal variations and dispersion values, again most likely due to its largely colloidal form. The values listed in the last column of Table 7-5 , with the possible exception of the value for ^{55}Fe, should be acceptable industry-wide generic scaling factors. The ^{55}Fe scaling factors may be waste-stream specific generic factors, or at least distinguished between filtering and ion-exchange waste streams.

Table 7-1
Summary of PWR Resin Waste Stream Statistical Analysis Results

		Spent Resin Waste Stream					
Scaling Factor		**Boron Recovery**	**Fuel Pool**	**Rad-Waste**	**Coolant Anion**	**Coolant Cation**	**All Resin**
^{55}Fe-^{60}Co	# of values	25	15	191	2	16	249
	geom. mean	0.27	0.031	0.031	3.3	0.85	0.81
	2σ dispersion	10.0	19.0	19.0	6.8	4.0	7.8
^{59}Ni-^{60}Co	# of values	9	8	45	1	0	63
	geom. mean	0.013	0.011	0.012	0.0095		0.012
	2σ dispersion	4.6	19.0	3.2			3.1
^{63}Ni-^{60}Co	# of values	26	17	233	2	23	301
	geom. mean	0.45	0.57	0.70	1.4	0.30	0.63
	2σ dispersion	4.6	5.2	6.6	2.6	5.4	6.2
^{239}Pu-^{144}Ce	# of values	10	9	75	0	5	99
	geom. mean	0.0036	0.0053	0.0091		0.0022	0.0074
	2σ dispersion	6.2	4.0	7.0		3.8	6.5
^{238}Pu-^{239}Pu	# of values	27	20	193	2	22	264
	geom. mean	1.4	1.7	1.2	1.2	1.4	1.2
	2σ dispersion	4.4	4.6	5.0	2.6	5.2	5.0
^{241}Pu-^{239}Pu	# of values	23	17	162	2	18	222
	geom. mean	100	100	110	120	210	110
	2σ dispersion	3.8	4.6	5.4	2.6	4.2	5.1
^{241}Am-^{239}Pu	# of values	24	18	177	1	20	240
	geom. mean	0.62	0.42	0.48	0.89	0.53	0.52
	2σ dispersion	5.2	4.0	6.6		6.2	7.5
^{242}Cm-^{239}Pu	# of values	24	18	169	2	19	232
	geom. mean	0.81	0.86	1.1	3.8	2.2	1.1
	2σ dispersion	6.0	8.0	8.6	4.8	8.0	8.3
^{243}Cm-^{239}Pu	# of values	24	19	177	2	18	240
	geom. mean	0.67	0.61	0.48	2.1	0.58	0.52
	2σ dispersion	6.6	4.8	8.0	5.2	5.4	7.5
α>5yr-^{239}Pu	# of values	27	20	205	2	22	276
	geom. mean	4.1	4.0	3.6	4.9	3.8	3.6
	2σ dispersion	3.8	3.0	4.0	3.6	3.6	3.8

Table 7-2
Summary of PWR Filter Cartridge Waste Stream Statistical Analysis Results

		Spent Filter Cartridge Waste Stream						
Scaling Factor		**Boron Recovery**	**Fuel Pool**	**Rad-Waste**	**Let-Down**	**Coolant Crud**	**Seal Injection**	**All Resin**
^{55}Fe-^{60}Co	# of values	12	43	109	89	18	18	285
	geom. mean	1.1	2.6	3.0	2.7	1.15	2.1	2.51
	2σ dispersion	10.8	3.5	7.8	8.2	9.2	6.0	7.9
^{59}Ni-^{60}Co	# of values	1	14	30	36	6	10	97
	geom. mean	0.039	0.014	0.12	0.013	0.023	0.079	0.016
	2σ dispersion		3.2	3.6	4.2	4.8	9.4	5.0
^{63}Ni-^{60}Co	# of values	15	51	126	2	19	21	327
	geom. mean	0.32	0.37	0.44	1.4	0.35	0.24	0.39
	2σ dispersion	15.4	5.2	6.8	2.6	8.0	3.6	6.0
^{239}Pu-^{144}Ce	# of values	9	27	79	0	14	11	203
	geom. mean	0.0059	0.0055	0.0065		0.0036	0.0043	0.0052
	2σ dispersion	6.6	15.8	7.4		3.6	4.0	6.0
^{238}Pu-^{239}Pu	# of values	15	44	121	2	20	21	303
	geom. mean	0.9	0.84	0.48	1.2	0.95	0.96	0.81
	2σ dispersion	5.0	10.2	5.8	2.6	4.0	4.2	5.8
^{241}Pu-^{239}Pu	# of values	15	41	107	2	16	20	284
	geom. mean	96	100	83	120	97	120	90
	2σ dispersion	4.4	4.4	4.6	2.6	4.2	2.8	4.5
^{241}Am-^{239}Pu	# of values	15	43	121	1	20	21	302
	geom. mean	0.34	0.69	0.48	0.89	0.43	0.44	0.49
	2σ dispersion	8.6	5.0	5.8		4.6	3.8	5.5
^{242}Cm-^{239}Pu	# of values	15	44	120	2	19	20	300
	geom. mean	1.3	1.9	1.5	3.8	4.0	3.8	1.9
	2σ dispersion	12.2	9.0	8.2	4.8	6.6	6.0	9.4
^{243}Cm-^{239}Pu	# of values	12	42	119	2	19	21	294
	geom. mean	0.33	0.96	0.4	2.1	0.48	0.71	0.50
	2σ dispersion	8.0	6.6	7.8	5.2	6.4	8.8	7.7
α>5yr-^{239}Pu	# of values	15	47	133	2	20	20	324
	geom. mean	3.1	3.9	2.9	4.9	3.2	3.8	3.2
	2σ dispersion	4.6	3.8	3.4	3.6	3.4	3.2	3.7

Table 7-3
Summary of All PWR Waste Stream Statistical Analysis Results

Scaling Factor		PWR Waste Stream					
		Resin	**Filter Cart.**	**DAW**	**Evap. Bottoms**	**Reactor Coolant**	**All Wastes**
^{55}Fe-^{60}Co	# of values	249	285	233	50	9	912
	geom. mean	0.81	2.5	2.2	0.73	1.2	1.5
	2σ dispersion	7.9	8.0	7.8	12.6	11.2	14.4
^{59}Ni-^{60}Co	# of values	63	97	41	25	23	249
	geom. mean	0.012	0.016	0.019	0.014	0.013	0.014
	2σ dispersion	3.1	5.0	15.2	3.80	5.6	5.6
^{63}Ni-^{60}Co	# of values	301	327	287	81	10	1102
	geom. mean	0.63	0.39	0.48	0.60	0.45	0.49
	2σ dispersion	6.3	6.0	3.2	9.0	10.8	7.6
^{239}Pu-^{144}Ce	# of values	99	203	105	55	33	495
	geom. mean	0.0074	0.0052	0.0054	0.0075	0.0042	0.0058
	2σ dispersion	8.5	7.9	8.2	11.4	25.2	9.3
^{238}Pu-^{239}Pu	# of values	264	303	241	91	89	988
	geom. mean	1.2	0.81	1.1	1.2	1.2	1.0
	2σ dispersion	5.0	5.8	5.0	4.6	6.0	5.6
^{241}Pu-^{239}Pu	# of values	222	284	202	65	77	850
	geom. mean	11	90	120	120	1.3	108
	2σ dispersion	5.1	4.5	5.0	7.8	5.6	5.4
^{241}Am-^{239}Pu	# of values	240	302	223	82	82	929
	geom. mean	0.49	0.49	0.64	0.74	0.61	0.55
	2σ dispersion	6.3	5.5	6.4	7.0	7.0	6.5
^{242}Cm-^{239}Pu	# of values	232	300	205	85	79	901
	geom. mean	1.1	1.9	1.32	1.36	2.0	1.5
	2σ dispersion	8.3	9.4	9.4	9.6	10.2	9.4
^{243}Cm-^{239}Pu	# of values	240	294	206	81	85	906
	geom. mean	0.52	0.50	0.53	0.54	0.69	0.54
	2σ dispersion	5.4	7.9	6.0	6.2	10.4	8.0
α>5yr-^{239}Pu	# of values	276	324	260	96	94	1050
	geom. mean	3.6	3.2	3.5	7.4	4.2	3.5
	2σ dispersion	3.6	3.7	4.0	4.0	4.4	4.1

Table 7-4
Summary of BWR Resin Waste Stream Statistical Analyses

Scaling Factor		Spent Resin Waste Stream					
		Condensate Deep-Bed	**Condensate Powdered**	**Reactor Clean-up**	**Fuel Pool**	**Rad-waste**	**All Resin**
^{55}Fe - ^{60}Co	# of values	15	40	100	4	97	256
	geom. mean	0.46	0.73	0.66	0.20	0.22	0.42
	2σ dispersion	6.4	8.20	14.6	8. 0	12.8	13.2
^{59}Ni - ^{63}Ni	# of values	4	17	19	1	21	62
	geom. mean	0.016	0.024	0.011	0.011	0.027	0.019
	2σ dispersion	4.0	11.2	7.2		7.0	7.8
^{63}Ni - ^{60}Co	# of values	30	48	132	4	112	326
	geom. mean	0.016	0.014	0.020	0.075	0.030	0.022
	2σ dispersion	7.6	7.2	6.8	3.4	3.8	7.4
^{239}Pu - ^{144}Ce	# of values	30	31	67	0	32	160
	geom. mean	0.088	0.10	0.011		0.010	0.025
	2σ dispersion	7.5	11.8	12.8		9.2	12.2
^{238}Pu - ^{239}Pu	# of values	34	31	115	0	87	267
	geom. mean	1.8	1.3	1.8		1.7	1.7
	2σ dispersion	4.4	3.8	4.2		4.0	4.2
^{241}Pu - ^{239}Pu	# of values	27	31	91	0	68	217
	geom. mean	110	99	100		99	100
	2σ dispersion	4.6	4.0	4.4		6.4	5.0
^{241}Am - ^{239}Pu	# of values	31	32	113	0	84	260
	geom. mean	0.48	0.42	1.1		1.1	0.87
	2σ dispersion	9.4	5.2	4.4		8.6	6.8
^{242}Cm - ^{239}Pu	# of values	32	34	113	0	75	254
	geom. mean	2.41	1.16	1.3		1.1	1.3
	2σ dispersion	10.0	6.2	9.0		6.5	9.4
^{243}Cm - ^{239}Pu	# of values	29	32	114	0	83	258
	geom. mean	1.07	0.43	0.85		0.97	0.83
	2σ dispersion	6.2	6.4	7.8		11.0	8.4
alpha>5yr - ^{239}Pu	# of values	36	36	127	0	96	295
	geom. mean	4.5	3.1	4.8		5.4	4.7
	2σ dispersion	3.6	3.4	3.6		4.8	4.0

Table 7-5
Summary of All BWR Waste Stream Statistical Analyses

Scaling Factor		BWR Waste Stream All Resin	Filter Sludge	Filter Cartridge	DAW	Reactor Coolant	All Samples
^{55}Fe - ^{60}Co	# of values	256	212	54	134	53	623
	geom. mean	0.42	0.88	0.87	1.2	0.33	0.63
	2σ dispersion	132.2	9.20	10.4	11.4	21.8	14.2
^{59}Ni - ^{63}Ni	# of values	62	41	13	15	15	146
	geom. mean	0.019	0.024	0.038	0.020	0.147	0.026
	2σ dispersion	7.8	24.0	8.0	41.4	35.2	18.4
^{63}Ni - ^{60}Co	# of values	30	205	51	104	44	730
	geom. mean	0.016	0.034	0.022	0.033	0.012	0.025
	2σ dispersion	736	8.6	10.2	11.0	10.6	9.0
^{239}Pu - ^{144}Ce	# of values	160	53	23	20	8	234
	geom. mean	0.025	0.0056	0.019	0.015	0.0032	0.0096
	2σ dispersion	12.2	14.4	18.6	15.2	13.4	13.2
^{238}Pu - ^{239}Pu	# of values	267	162	46	99	20	600
	geom. mean	1.7	1.7	1.6	1.5	1.2	1.6
	2σ dispersion	4.2	4.0	3.8	4.4	5.4	4.2
^{241}Pu - ^{239}Pu	# of values	217	154	37	72 110	20	500
	geom. mean	100	110	85	5.4	120	106
	2σ dispersion	5.0	5.2	4.0		5.4	5.2
^{241}Am - ^{239}Pu	# of values	260	165	47	95	25	592
	geom. mean	0.87	1.22	0.93	1.0	0.56	0.96
	2σ dispersion	6.8	7.0	7.0	5.4	7.2	7.0
^{242}Cm - ^{239}Pu	# of values	254	156	44	80	22	556
	geom. mean	1.3	1.06	0.73	0.56	2.3	1.1
	2σ dispersion	9.4	13.0	13.6	12.0	9.8	11.8
^{243}Cm - ^{239}Pu	# of values	258	165	47	91	21	582
	geom. mean	0.83	0.91	0.69	0.69	0.34	0.79
	2σ dispersion	8.4	8.2	8.0	9.2	9.8	8.8
alpha>5yr - ^{239}Pu	# of values	295	183	49	108	31	666
	geom. mean	4.7	5.3	5.1	4.5	3.3	4.8
	2σ dispersion	4.0	4.4	3.8	4.0	3.8	6.4

SITE-SPECIFIC TRENDING

Waste Stream Consolidation

Several waste streams may be associated with a single waste type generated at a power plant (Section 4). Consolidation of waste streams can be accomplished by comparing the scaling factors of each waste stream. If the set of scaling factors for DTM nuclide of one waste stream is within a factor (usually ten) of another waste stream, these waste streams may be combined into one.

In practice a single nuclide will often prevent unification of separate waste streams. For example, ^{14}C may be the only nuclide that varies by more than the factor of ten between the filters from two separate systems. For resin, ^{63}Ni may be the only nuclide that varies by more than the factor of ten between high activity and medium activity resin.

Computer spreadsheets or commercial software programs are useful to document the consolidation of waste streams. Many nuclide scaling factors need to be compared simultaneously and the magnitude of the variation in the scaling factors must be determined. Compliance with the BTP is maintained as long as the variation in the median value of a scaling factor is less than an order of magnitude (factor of ten).

In practice, laboratories will only guarantee results of two different samples from the same waste to within a factor of two. To consolidate waste streams, one must set an allowable dispersion above 2 to address these laboratory limitations. Selecting a scaling factor target dispersion of five will ensure regulatory compliance with an adequate margin of safety when consolidating waste streams.

Site Specific Constant Scaling Factors

As stated above, scaling factors for ^{55}Fe, ^{59}Ni, ^{63}Ni and transuranics have shown low variability industry wide. Among the transuranic nuclides, industry data has shown virtually no variability in the ratio of these individual nuclides to ^{239}Pu. Transuranic nuclides can therefore be correlated to the ^{239}Pu concentration. The ^{239}Pu/^{60}Co scaling factor can be used in concert with the TRU/^{239}Pu factors to derive individual transuranic nuclides.

Although generic scaling factors have yet to be approved for use in the U.S., site-specific scaling factors for these nuclides have been used. The cost of off-site analyses can be substantially reduced by deriving and using of site-specific scaling factors. The cost for off-site analysis of the corrosion products ^{55}Fe, ^{59}Ni and ^{63}Ni can be 20% of the total cost of analysis. Likewise, the cost of transuranic analysis, excluding ^{239}Pu, can be

30% of the total analysis cost. Thus, 50% of the total sample analysis cost could be saved if valid site-specific constant scaling factors were used.

Use of site-specific constant scaling factors hinges on documenting a defensible determination that constant values for the scaling factors are valid. Part of such a program should also address the conditions under which these values must be verified or changed.

Performing a statistical comparison of the above scaling factors (*i.e.*, ^{55}Fe, ^{59}Ni, and ^{63}Ni/^{60}Co and TRU/^{239}Pu) for all historic results can provide defensible documentation. The use of computer spreadsheets or commercial software programs make such comparisons relatively easy. The key to determining valid constant scaling factor values is the selection of the acceptable variability in the historic results. As stated above, the lowest "between sample" variability from off-site labs is typically a factor of two.

Actual comparisons at several stations have yielded results for the scaling factors mentioned above very close to this limit of two. This was accomplished on a waste stream specific bases. Site- specific scaling factors, like industry-wide data, is waste stream specific. Scaling factors for resin, filters and DAW will be different but, the annual values for each waste stream will vary only slightly from year to year.

Derivation of Constant Site Specific Scaling Factors

Deriving constant scaling factors can begin once all the waste streams have been identified for a station. The basic strategy is to determine the variation between all historic results for a single scaling factor. The comparison of all historic data encompasses different fuel cycles, different systems and different reactors at multi-unit sites.

Dispersion is the name of the statistical method which has been adopted to determine the variation in this data. If the dispersion is near 2, then use of a constant scaling factor is justified for all waste streams. If it is near 5, it suggests that different waste streams may have constant scaling factors but with different values. If the dispersion is near 10, it is likely that no constant scaling factor exists and analysis for this nuclide pair must continue.

As stated above constant scaling factors across all waste streams are most likely to be derived for individual transuranic nuclides (*e.g.*, ^{241}Am, ^{242}Cm, ^{243}Cm, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{241}Pu and ^{242}Pu) versus ^{239}Pu. By performing and documenting a TRU/^{239}Pu comparison, site specific TRU scaling factors can be determined. Use of these factors eliminates the need to request and pay for off-site analysis of these nuclides. This can result in a savings of 30% per sample.

The comparison of corrosion product scaling factors (e.g., ^{55}Fe, ^{59}Ni and ^{63}Ni/^{60}Co) is likely to produce at least waste stream specific constant values. Different values for resin, filters and DAW should be expected as industry data has shown. Comparing site specific results with industry data (Section 7) provides further support for the use of constant scaling factors.

Should the dispersion of the resin or filter waste stream be near 5, determination of the scaling factors for different sub- waste streams should be conducted. For filters, this may be a comparison of filters from different systems. For resin, this may be a comparison of high activity versus low activity batches. Comparing various groups within a waste stream should then be conducted to determine if a single system is the cause of the increased variability.

Results from other stations have shown that high activity and low activity resin may have different constant corrosion scaling factors. One station has determined that filters from every system except the boron recycle system have a constant set of corrosion scaling factors. The filters from the boron recycle system have a different value for ^{55}Fe/^{60}Co.

Determination and use of a constant set of waste stream specific corrosion product scaling factors eliminates the need to request and pay for analysis of these nuclides. This can result in a savings of 20% per sample.

If constant TRU and corrosion product scaling factors are used, off-site analysis results are reduced to a gamma isotopic, ^{14}C, ^{89}Sr, ^{90}Sr, ^{239}Pu, ^{99}Tc and ^{129}I. This can provide a significant reduction in annual analysis costs.

Technetium and Iodine

In general, commercial laboratory results for ^{99}Tc and ^{129}I will be LLD values. The cost of analysis for these two nuclides can be 30% of total analysis costs. EPRI has contracted with Battelle PNNL to obtain actual results using mass spectrometry analysis (MSA) for ^{99}Tc and ^{129}I from radwaste samples. PNNL is not capable of providing routine MSA analysis of these nuclides for the entire industry. The French have used a laboratory in Canada, ISOTRACE, to provide MSA services for ^{129}I.

Constant scaling factors for ^{99}Tc and ^{129}I could be developed from infrequent sample MSA results of primary resin and/or filters. These results could be used for several years with confidence because these actual results will be orders of magnitude lower than the LLD values currently obtained and used on an annual basis.

COMPUTER MODELING OF SCALING FACTORS

To date, the reported activities of the radionuclides ^{99}Tc and ^{129}I in wastes from utilities have been highly conservative, resulting in severe over-reporting of the quantities of these nuclides placed in the disposal sites. The conservatism results from very low concentrations of these nuclides in waste that are usually well below the analytical sensitivity limits of the assay laboratories unless they use expensive and time consuming mass spectrometer techniques. Since these two nuclides have critical pathways for population exposure, over-reporting can result in premature disposal site closures. More accurate reporting of these nuclide concentrations is highly desirable.

Computer models have been proposed and used in a few cases to generate more accurate scaling factors for use in plant waste management programs. The models use the production and release mechanisms for radionuclides generated in the reactor core to calculate the activities of several nuclides in the reactor coolant but particularly ^{99}Tc and ^{129}I. Empirical industry data for transport and deposition of the radionuclides on specific waste streams media in the plants allow determination of waste-stream specific scaling factors for these radionuclides.

RADSOURCE (Ref. 4), a program currently developed by EPRI, determines plant-specific scaling factors for ^{90}Sr, ^{99}Tc, ^{129}I and the TRU nuclides in waste. A commercial program, *3R-STAT (Ref. 5)* can estimate the release rate of ^{99}Tc and ^{129}I into the coolant of an operating reactor. This program yields the total production of these two nuclides over any given period and, hence, the total activity released for disposal. Gamma-ray analyses of short-lived nuclides in reactor coolant form the basis for the calculations of the concentrations of longer lived nuclides in both of the very similar models. *3R-STAT* has received NRC approval for use. Because of their calculational similarities, *RADSOURCE* presumably could also receive approval.

RADSOURCE

RADSOURCE will supplement and, with NRC acceptance, provide an alternative to the common practice of deriving some scaling factors from radiochemistry measurements on plant waste samples, especially for ^{99}Tc and ^{129}I. It is a program operating on a PC microcomputer that uses reactor coolant gamma isotopic data to compute waste stream-specific radionuclide scaling factors for the period represented by the analyzed data. *RADSOURCE* calculates a set of scaling factors for user-specified waste stream for the fuel-source radionuclides, ^{90} Sr, ^{99}Tc, ^{129}I, and TRU and for the activation-product nuclides, ^{14}C and ^{63}Ni. The program determines these latter scaling factors from the plant and industry historic databases.

Using *RADSOURCE* requires an initial setup that requires the following plant information that is then stored in the program files:

- Plant Type (*i.e.*, BWR or PWR)
- Number of Reactor Units
- Reactor Coolant Mass
- Iodine Steam Carryover Factors (BWRs only)
- Power Steam flow Rate (BWRs only)

The user must identify the generic *RADSOURCE* waste streams that will be combined into single plant-specific waste streams for packaging, shipping and disposal. The user will also enter the reactor coolant gamma spectroscopy data from routine plant measurements:

- Date
- Unit Number
- Purification or RWCU Flow Rate
- Reactor Power Level (%) (BWRs only)
- Reactor Coolant Concentrations for:
  - ^{131}I, ^{132}I, ^{133}I, ^{134}I, ^{135}I, ^{134}Cs, ^{137}Cs, and ^{60}Co.

The code uses these data to assess the reactor core conditions that dictate the release of the fission products and transuranics. It then estimates fission-product scaling factors (scaled to ^{137}Cs) for reactor coolant and corrosion product factors (scaled to ^{60}Co) from historic data. These factors, with minor adjustments to account for transport properties within the primary system, are the scaling factors for reactor coolant purification resins. *RADSOURCE* then calculates scaling factors for other waste streams through the use of waste-stream-specific factors derived from the historic plant and industry data bases.

The fuel release model contained in *RADSOURCE* is based on fundamental chemistry and physical principles and reliable accounts of release rate differences between the radionuclides used in the scaling factors. The program bases differences in removal rates for the radionuclides on purification resins partially on an empirical basis. Code validation resulted from comparisons of purification resin scaling factors produced by the code and scaling factors measured on resins. Validation used data from several plants. Scaling factors calculated using *RADSOURCE* usually exceed the accuracy achievable by measurements.

The assumptions in *RADSOURCE* are:

- All DTM nuclides except ^{129}I result only from tramp fuel or suspended corrosion products;

- The knockout (collision) mechanism is dominant for release from tramp fuel;
- ^{129}I, ^{131}I, and ^{137}Cs coolant appearance rates result from both tramp fuel and rod defect releases;
- Similarity in ^{129}I and ^{137}Cs release mechanisms are the basis for estimating ^{129}I scaling factors;
- Release rates of ^{90}Sr, ^{99}Tc and TRU are proportional to their inventories in the tramp fuel and to their concentrations in the coolant relative to that of ^{131}I;
- ^{137}Cs release rates can be accurately measured in coolant samples.

The model determines the fractional releases of radioiodine from tramp fuel and defective rods from measured concentrations of the five short-lived iodine isotopes. It partitions the releases of radiocesium and radioiodine between recoil, diffusion and knockout mechanisms on bases of empirical or theoretical relationships and coefficients for the processes.

RADSOURCE assumes a typical burnup for tramp fuel of 700 effective full-power days and uses fission product and TRU inventories as calculated by the *ORIGEN2 (Ref. 6)* burnup code. *RADSOURCE* estimates corrosion product nuclide scaling factors from well verified industry values for the nuclides ^{55}Fe and ^{63}Ni, relative to ^{60}Co.

Advantages of using RADSOURCE

- Reduce or eliminate some radiochemistry measurements made on waste samples and save on analytical costs;
- Improve accuracy of scaling factors, particularly of those for ^{99}Tc and ^{129}I;
- Identifies changing plant conditions (e.g., increased tramp fuel, rod defects, *etc.*) that could affect plant scaling factors;
- Provides scaling factors on a waste stream basis;
- Free Software for EPRI members.

Disadvantages of using RADSOURCE

- Because of the time difference between when the coolant samples are analyzed and when the wastes are generated and packaged, the utility must exercise care and judgment in tracking the proper factors for use if the scaling factors have changed over the time period;
- The program does not account for major crud burst events where large quantities of DTM and key radionuclides may be released in proportions different from normal release. This should only be a problem if several such events occur and if the quantities of radionuclides dislodged are large;

- *RADSOURCE* may not yield usable scaling factors for ^{14}C and ^{90}Sr;
- *RADSOURCE* results must still be periodically validated for accuracy to gain NRC acceptance of its use.

3R-STAT

The *3R-STAT* program provides another alternative to laboratory measurements for determining the quantities of ^{99}Tc and ^{129}I contained in nuclear plant low-level radwastes. This code assesses reactor fuel conditions through a fission-product release and transport model very nearly like that seen in *RADSOURCE*. These assessments provide release rates of ^{99}Tc and ^{129}I over a specified reactor operating period. The code assumes that, once released from the fuel, the two radionuclides will ultimately end up in the plant's low-level waste. The principal difference between *3R-STAT* and *RADSOURCE* is that *3R-STAT* computes only total activity release and not scaling factors. Hence, the accuracy of its results do not rely so heavily on accuracy of ^{137}Cs analyses in the reactor coolant or the accuracy of the historic databases in relating activities in coolant to those in plant waste streams. The code's principal application is to determine the total amounts of ^{99}Tc and ^{129}I released by the plant for waste disposal over a prescribed time.

Advantages of using 3R-STAT

- Reduce or eliminate some radiochemistry measurements made on waste samples and save analytical costs;
- Improve accuracy of reported activities in waste, particularly of those for ^{99}Tc and ^{129}I;
- Identifies changing plant conditions (e.g. increased tramp fuel, rod defects, etc.) that could affect plant scaling factors;
- The code has received NRC approval and is commercially available.

Disadvantages of using 3R-STAT

- The code does not compute scaling factors but only total releases of ^{99}Tc and ^{129}I and does not partition the nuclides into waste streams;
- Use of the code requires a mechanism, separate from manifesting, for reporting ^{99}Tc and ^{129}I to the disposal site, NRC and possibly the regional compact;
- The code treats only those two radionuclides;
- 3R-STAT results must still be periodically validated for accuracy.

8

DETERMINATION OF CURIE CONTENT

Methods used to determine curie content for use in LLW characterization and classification are summarized in this section and include:

- Measurement of Specific Nuclide Method
- Gross Radioactivity (Dose-to-Curie) Method
- Direct Assay Method

Caution: In classifying your waste, you must take into consideration the requirements of the BTP on concentration averaging referenced in Section 2.

Much of the information presented in this section on dose-to-curie and direct assay methods has been taken from the work of David James and Jene Vance under EPRI Contract RP2414-53 (Ref. 9). Some information has been modified by the Guidelines Committee for use in this document.

MEASUREMENT OF SPECIFIC NUCLIDE

The measurement of specific nuclides for waste classification requires a quantitative sample analysis of the waste. The laboratory results for each nuclide, provided in units of specific activity (e.g., μCi/cc, μCi/g), are simply multiplied by the volume or mass of the waste. This method may make use of scaling factors or it may not. For cases where scaling factors are not to be used, the off-site analysis must provide a full nuclide spectrum. If scaling factors are to be used, an on-site quantitative analysis must be used in concert with the scaling factors already calculated for that waste stream.

This method is usually used for classification of concentrated liquids which are to be solidified and low activity ion exchange media, filter media and sediments which exhibit no appreciable dose rate. The method is also used for high activity resin where significant spectrum shifts are likely due to decay or unshielded average container dose rates cannot be obtained due to the configuration of process shielding.

High activity resin is often packaged in rented shipping casks to provide sufficient shielding. The waste is packaged inside these casks with the small cask processing lid removed. Accurate disposal container average dose rates cannot be obtained when such casks are used. The number of container dose rates obtained through the processing port are limited and cannot provide an average. Dose rates obtained from the side of the casks are not useful because a significant portion of the measured dose emanates from the open port in the lid of the cask. Shielding the probe from this "sky shine" is usually not successful. These problems have led some plants to rely on the measurement of the specific nuclide method to classify high activity resin.

High activity ion exchange media is often stored for long periods to reduce radiation exposure to workers during the waste packaging process. Hence, a significant decay in activity can occur, especially for shorter-lived nuclides such as ^{58}Co. Since disposal site fees include large curie content surcharges, obtaining a waste sample during resin transfer can reduce disposal costs resulting from over-reporting of the decayed activity.

Daily rental cost for heavy shipping casks is expensive, and high cask rental costs encourage fast off-site analysis results. This situation may lead to the dilemma of requesting quick and possibly less accurate results for the very waste which contains the largest quantity of activity. This dilemma can be avoided if a waste sample can be obtained prior to filling the container in the cask or if shielded on-site storage is used.

A sample from the storage tank or from the waste as it enters the storage tank can be analyzed prior to the arrival of the shipping cask. Scaling factors can be developed from these results and saved until the waste is packaged. Upon transfer of the waste to the cask another sample is taken for on-site analysis. The gamma analysis of this sample provides a decayed spectrum. The on-site ^{137}Cs/^{60}Co ratio can be compared with the off-site analysis ratio to ensure the results agree. This allows the scaling factors from the off-site analysis to be used in concert with the on-site gamma analysis and improves the confidence that the samples taken were representative. This method reduces cask rental fees and disposal site curie surcharges. It also enables the use of scaling factors derived from analyses performed without undue stress. Further, it demonstrates that representative samples were used since two samples of the waste have been compared.

Tritium

Tritium does not scale well to any other nuclide. Tritium should be determined by measurement of this specific nuclide in reactor coolant. Alternately, ^{3}H in a waste stream can be scaled to tritium in the reactor coolant at the time of generation. Since power plants have ^{3}H analysis capability, off-site analysis of this nuclide is not required. Whenever off-site analysis of ^{3}H is considered for physical waste samples care should be taken because of the extreme mobility of ^{3}H (evaporation losses).

The moisture content of a waste stream has been used to calculate the ^{3}H content of the waste. Some typical waste stream moisture contents are as follows (Ref. 7):

- 50% Bead Resin
- 50% Powdered Resin
- 25% Cartridge Filter Media
- 0.1 to 0.001% DAW

The tritium concentration of reactor coolant multiplied by the moisture concentration of the waste stream equals the ^3H content of the waste. For PWRs, which can concentrate ^3H in reactor coolant to very high levels, consideration should be given to the use of a factor (liquid radwaste ^3H concentration versus reactor coolant), where appropriate.

DOSE-TO-CURIE METHODS

The dose-to-curie method to determine curie content is performed by calculation. This method estimates the concentrations of gamma-emitting radionuclides in a waste package based on the dose rate. The dose rate is measured at a specified distance and geometrical position from the waste package. This information, combined with the data on the relative abundance of the gamma-emitting radionuclides (determined from gamma spectrometry measurements of a sample of the waste), is used in shielding equations to calculate the radionuclide concentrations.

This dose-to-curie technique is commonly used in the nuclear industry. It is most often applied:

- where it is difficult to obtain a representative quantitative sample as in DAW and cartridge filters;
- when wastes contain very high activity and potentially high personnel exposures precludes sampling.

By measuring the dose rate from the entire waste package, this technique reduces some of the problems associated with representative sampling of a waste material for absolute concentrations. In addition, sampling for the relative abundance of radionuclides is generally easier to achieve than for absolute concentrations. This is based on the assumption that the variability of the concentration ratios of any two radionuclides in the bulk waste volume is less than that in the absolute concentrations.

General Dose-to-Curie Equations

The basic method for dose-to-curie conversion (Ref. 8) is based on the following equation:

$$C = \frac{D}{\sum_{i=1}^{n} d_i f_i}$$

Where:

C = Total Activity

D = Dose rate measured on contact or at some specified distance

f_i = Fractional abundance of the i^{th} radionuclide, and

d_i = Radionuclide specific dose-to-activity conversion factor for a given waste package configuration

The radionuclide-specific dose-to-activity conversion factors are typically determined using shielding computer codes, although hand calculations and rule-of-thumb calculations are also used. These calculations are normally performed for each individual radionuclide at a unity concentration for a range of applicable waste densities and geometry's, or for container configurations. The results may be stored in a database or may be presented using graphs or tables. It is from this data that the conversion factors can be extracted for radionuclide determinations.

Available Dose-to-Curie Methods (Ref. 9)

Several computer programs are available for estimating the curie content of waste packages using the dose-to-curie method. They include programs that are available commercially and also those that have been developed by the user (or plant). These programs include both shielding codes and waste tracking programs. Commercial and other dose-to-curie calculational programs that are used include:

Shielding Codes
ISOSHIELD
MEGASHIELD
MICROSHIELD™
QADMOD-G
WISE & SIMPLE

Waste Tracking Codes
DOSCON (PAKRAD™)
RADMAN™ (RAMSHP)
WASTETRAK™

A 1991 EPRI survey of operating plants revealed that out of the 29 plants surveyed, 16 were relying on a commercial waste tracking program and 13 were using their own program for dose-to-curie calculations. The survey observed that most of the plants used a commercial computer program to calculate the radionuclide-specific dose-to-activity conversion factors.

Several of the available methods were selected to perform the curie estimation calculation on a hypothetical container of waste. For the hypothetical container it was assumed that it was a 6 ft x 6 ft steel liner filled with ion exchange resin at a density of 1 g/cc. It was also assumed that the dose rate was 100 mR/hr at 1 meter from the package surface at the midpoint of the container height. The following radionuclide distribution was assumed:

3.0%	^{51}Cr
0.7%	^{54}Mn
9.8%	^{58}Co
49.0%	^{60}Co
1.9%	^{65}Zn
8.6%	^{134}Cs
27.0%	^{137}Cs

Based on this hypothetical case, the total curie content of the container was estimated using each of the available methods. The results of these calculations are presented in Table 8-1.

Table 8-1
Summary of Dose-to-Curie Calculations for Selected Available Methods

Method	Total Activity (Ci)
MICROSHIELD™	1.77
WASTETRAK™	1.48
RADMAN™	1.20
WISE & SIMPLE	1.83
DOSCON (PAKRAD)	1.25
Plant A Method	0.71
Plant B Method	0.86

The results from the seven calculations are within a factor of 2 of the average. The comparison is presented as a gross check on the performance of various methods and no effort was made to resolve differences. Based upon these results, however, it is observed that the variation in dose-to-curie methods, even under uncontrolled conditions, would probably not increase misclassification of wastes. The observed variation is well within variability caused by sampling and probably within the variability of the dose rate measurement itself.

A similar comparison was conducted for DAW in a 4 ft x 4 ft x 6 ft metal box with waste density varying from 20 to 50 lb/ft^3 (Ref. 10). The box had a 1 foot dose rate of 20 mR/hr for each case and the spectrum was identical. The programs which were compared were:

- PAKRAD (DOSCON)
- MICROSHIELD
- RAMSHP 3.3.a (RADMAN)

The resulting activity for the box at the various waste densities was:

	Activity (mCi)			
Program	20 lb/ft^3	30 lb/ft^3	40 lb/ft^3	50 lb/ft^3
PAKRAD	60	70	82	100
MICROSHIELD	40	52	60	72
RAMSHP	30	40	50	60

The results of these programs are in general agreement. The programs all show the same trend as a function of waste density. The variation between the programs is 40 to 50%. The absolute differences between the program results are small when compared to the factor of 10 allowance.

Dose-to-curie models for individual cartridge filters are also used. Some programs model the cartridges as line sources (e.g., RADMAN) while others use cylindrical or annular models (e.g., PAKRAD). The line source models require that dose rates be obtained from the side of unshielded cartridges. This will result in either increased radiation exposure to workers or additional measures (such as the use of a shielded cell) to prevent additional radiation dose. The other models enable filter dose rates to be obtained from the open bottom of filter transfer casks which prevents radiation exposure to workers.

Dose-to-curie methods are based on shielding programs that have been benchmarked against measured data and are generally well established. It is noted that when radionuclide-specific dose-to-activity conversion factors are determined for a given geometry and activity content, the higher the factor the less conservative the curie estimate.

Radionuclide distributions used in the calculation must be reasonably accurate to ensure an accurate estimate. This is especially true for the key radionuclide ^{137}Cs. For ^{60}Cs, which typically dominates the distribution, it is a less significant factor. The dose rate measurement value should also represent the average of several measurements taken at the location corresponding to the dose-to-curie calculation method. For a waste container with an inhomogeneous distribution of activity concentrations, some plants use the highest measured dose rate. While this is conservative, it may significantly overestimate the actual curie content of the package.

For items or packages of lower dose rate, the distance at which the radiation level is measured is extremely important. Small errors in the distance when a 6-inch or 1-foot measurement is required can result in a significant variation in the measured dose.

> It is important that the distance at which the radiation level is measured be accurate and consistent. Use of a jig is recommended to obtain consistency and accuracy when measuring dose rates, especially at small distances.

For the dose-to-curie method, several factors can affect the accuracy of the result. These factors should be addressed as follows:

- obtain accurate average dose rates at a defined distance;
- use a representative spectrum; and
- selection of the computer model or other mathematical model.

Advantages and disadvantages of dose-to-curie methods are as follows:

Advantages

- Representative sampling for relative abundances of radionuclides is easier than representative sampling for absolute concentrations.
- The dose rate measurement averages the concentration over the volume.

Disadvantages

- Multiple dose rate measurements, at reasonably accurate locations, are required for packages with significant Inhomogeneity. This increases radiation exposure to workers for high activity packages or requires a shielded survey cell to ensure ALARA.

- The geometry of the waste package must be reasonably represented in the mathematical models in the shielding codes.
- Variations in density within the package, as well as between different packages, can have a significant impact on curie content.

Once the curie content of the gamma emitting nuclides has been determined, scaling factors must be applied to determine the curie content of the DTM nuclides. Only after these two calculations have been performed can the waste be classified.

DIRECT ASSAY METHODS

The term direct assay for waste characterization is used for integrated radiation measurement techniques made on waste packages or items. Two techniques have been developed:

- Direct gamma-ray measurements
- Direct neutron measurements

Due to the low TRU concentration in commercial LLW only direct gamma-ray measurements will be discussed in this report. EPRI report NP-5497 (Ref. 11) describes both of these techniques in detail.

Direct assay has primarily been used for activated metal from commercial power plants. Gross radioactivity (dose-to-curie) and measurement of specific nuclides (waste samples) in concert with scaling factors enable power plants to classify their waste in a cost effective manner.

Direct assay can prove useful to power plants when storing waste residuals from off-site volume reduction facilities is required. The processed waste residuals will not be in a form suitable for verification sampling. Supercompacted waste, glass and metallic waste forms must be manifested, by the plant, for final disposal shipment. These waste forms will possess considerable self shielding making common dose-to-curie techniques invalid.

Gamma-ray spectrometry of waste returned to the plant could be used to confirm that the spectrum of high energy gamma emitters is in keeping with the waste which was shipped off-site for volume reduction. Spectrometry can also identify the gamma emitters which must be manifested for final disposal.

As discussed in Section 4, under a waste process and return scenario, post processing samples, like ash, could be obtained by the volume reduction facility and sent to the plant as part of the contract. On-site gamma-ray spectrometry can aid in classifying volume reduced waste which is returned to the plant.

Note: In the direct assay process, radiation measurements must be made of the entire waste package. This can be done by scanning the entire package or surrounding it with radiation detectors. In the direct assay process there are three basic functions performed by the direct assay device:

- obtaining average gamma-ray spectrometric measurements;
- obtaining an average flux of radiation emitted from the package; and,
- using analytical calculations to convert the measured spectra and fluxes into average radionuclide concentrations.

Direct assay has been investigated primarily as a possible remedy for some of the problems associated with obtaining representative samples. Because the entire waste package (or item) is scanned it can account for the non-homogeneity in the radiological content of the package. Therefore, a more accurate determination of the curie content is achieved. With direct assay, the container is, in effect, the sample.

Direct Gamma-Ray Spectrometry

Direct Gamma-Ray Spectrometry is a quantitative method that scans the package and analyzes it for the average concentration of gamma-ray emitting activities in the waste. Several devices have been developed in the U.S., and overseas, to accomplish this. There are differences between these devices such as: the detector types; the method of scanning; and the analytical calculations used to derive the average concentrations. Three of the basic configurations are described as follows:

Gamma Spectrum Analysis with Portable Detector

This device (Ref. 11), is a high purity germanium gamma-ray detector installed in a shielded collimator. The collimator has an front aperture that allows unimpeded gamma-rays into the detector. This detector is mounted on a cart, for mobility, and can be pivoted and aimed at the waste package. A second cart holds the electronics and the microcomputer used in conjunction with the detector. The detector can be moved around the waste package or object, or the waste package can be rotated or moved up and down in front of the fixed detector. This arrangement could also be adapted for permanent installation.

The detector analyzes the incoming gamma-rays, using a multichannel pulse-height analyzer, to determine the gamma-ray spectrum. An average spectrum is determined by aiming the detector over the surface of the package.

The electronics register count-rates for each channel. Then the device calculates average concentrations for each radionuclide in the identified spectrum, using pre-determined efficiencies corresponding to the specific radionuclides; the geometry of the viewing angle; and the source volume. The detector is calibrated to known gamma sources, and the calibration is critical to the application of this technique.

Gamma Spectrum Analysis with Fixed Detector

This device developed for assaying 55 gallon drums (Ref. 12), also uses a high purity germanium detector. Here the drum is rotated in front of the detector while the detector slowly moves to scan the vertical dimensions of the drum. Similar to the Gamma Spectrum Analysis with Portable Detector, an average spectra and count-rate is determined by using a multichannel analyzer. The efficiencies used to convert from count-rate to activity concentrations (for each radionuclide) are based on calibrated measurements. In this case, a 55 gallon drum was filled with a mixture of materials of varying densities. The calibration was performed using a source of known concentration placed at different locations within the drum. The calibrated efficiencies account for inhomogeneities in the drum and for the density of the waste. The device contains a load cell to measure the weight of the drum so that the average density of the waste can be calculated.

Gross Activity Measurement with Average Gamma Spectrum

In this device (Ref. 13), a high purity germanium or a NaI detector is used to determine an average gamma-ray spectrum for the waste package. This device holds a 55 gallon drum that rotates past the detector. Only the average spectrum, not the count rate, is used from the gamma-ray spectrometry system. Plastic scintillator detectors are also used to measure the gamma-ray flux from the package. These detectors are either rotated around the package or are fixed so that the drums are rotated past them. Similar to the Gamma Spectrum Analysis with Fixed Detector (#2), a load cell is provided to determine the average density of the waste. Using the average spectrum information, the average gamma-ray flux and the density, the average concentrations are calculated by using dose-to-curie equations.

The advantages and disadvantages of direct assay methods are as follows:

Advantages

- Increased accuracy of the curie concentration determination for packaged waste by eliminating problems with sampling representativeness.
- Elimination of sampling for radionuclides measured in the direct assay.

Disadvantages

- Equipment cost and installation into a permanent facility.
- Continuing maintenance and technical support for the equipment.
- Requires measured or computed calibrations for different package sizes.

9

WASTE MANIFEST

The hazard associated with the shipment of LLW must be communicated by shipping papers which accompany the vehicle. A Bill of Lading may be required by carriers to document the goods being transported. For LLW shipped to a disposal site, a manifest must be completed. Historically, there have been two manifests for radioactive waste depending upon which disposal site was used. The Chem Nuclear manifest for Barnwell, SC is very similar to the US Ecology manifest for Richland, WA. A significant change for waste manifesting will occur with the adoption of the uniform waste manifest. Use of this manifest for the Barnwell site is required starting December 1, 1996.

Use of the uniform manifest will resolve several regulatory issues. DOT has stated that the Chem Nuclear and US Ecology manifests do not meet the requirements of 49CFR 172.203. This rule requires that additional descriptive information must appear "immediately" after the description of the hazardous material. The existing waste manifests provide this information on the continuation page. DOT has found that information on a continuation page does not constitute "immediately after" and is therefore not in compliance. Providing such information on the Bill of Lading is one way of complying with the DOT. Citation of the recently approved exemption number on each page of the Barnwell manifest is also acceptable. Another is the use of the uniform manifest.

The final rule for DOT shipping regulations issued in 1995 require the use of System International (SI) units on or before April 1, 1997. This requires that lb., Ci and mRem/hr be replaced by Kg, MBq and mSv/hr. The Barnwell site plans to adopt the use of SI units when they require use of the uniform manifest in 1996. The Barnwell site will still be using instruments with standard readings for several years. This will allow plants to continue their use of standard instruments and apply mathematical conversions to SI units in order to fill out the uniform manifest without immediately resulting in any discrepancies in disposal site receipt surveys.

> Once the disposal sites use SI instruments, SI instruments should be considered for use at power plants for radwaste surveys in order to avoid the potential for discrepancies between the manifest and the disposal site receipt surveys.

Since 10CFR20 has not adopted the use SI units for radiation protection in the U.S., separate SI instruments may have to be procured for radioactive material shipping.

Use of the uniform manifest is required by March 1, 1998 per amendments to

10CFR20 and 61. The purpose of these amendments is to:

- improve the quality and uniformity of information contained on manifests that are required to control transfers of LLW intended for disposal;
- establish a set of NRC forms, that serves as information needed to meet NRC, DOT, State and Compact information requirements;
- require LLW disposal site operators to electronically store container-specific manifest information; and
- require disposal site operators to be capable of submitting reports of stored manifest information on a computer-readable medium (e.g., magnetic disks or tapes).

Since the Barnwell site will be adopting this manifest in 1996, this regulatory requirement is met.

- The uniform manifest consists of three NRC forms: 540, 541 and 542:
- The NRC 540 form is the Shipping Paper. This form contains all the information required by DOT for radioactive waste shipments for disposal.
- The NRC 541 form provides the disposal Container and Waste Description information.
- The NRC 542 form is to be completed by waste processors to identify the waste generator.
- The 540 form must physically accompany the shipment since it serves as the DOT shipping paper.
- The other forms may accompany the shipment or they may be sent electronically if this has been agreed to by the disposal site.

Utilities have had several concerns about the uniform manifest. These concerns were addressed to the NRC via the Nuclear Energy Institute (NEI) as part of public comment. As of this date not all of these concerns have been addressed. The remaining concerns center on the specific format of the uniform manifest as issued. Some of these format issues are:

- the 540 form does not provide specific input for shipping cask information
- the 540 form has a mismatch between the entry spaces for individual radionuclides symbols (column 15) and the total activity (column 16)
- the 541 form does not support vertical listing of nuclide activity to provide a readable total container activity (column 15)

Industry was able to convince the NRC not to publish the forms as part of the rulemaking. This enables any change to the format of the forms to be conducted quickly. Adoption of the uniform manifest will require changes to the computer programs and procedures plants use to complete waste shipping manifests. Since these changes are time consuming, the industry is seeking to resolve these items before implementation.

The DOE has created a data base program to collect and manipulate this manifest data. The Low Track program is geared to Compacts which must track and report waste generation. In addition, waste generators can use the program to submit LLW information to their compact and to generate waste manifests. Low Track is designed to sort quickly and to provide reports on manifested data. The manifest generation module is basically a template. No container activity calculation, waste classification or shipment typification capability is provided by this program. The program can accept information in standard units, however, and will convert it to SI units for completing the Uniform Manifest.

NEI has set up a working group to test this DOE software. Several power plants are obtaining copies of the Low Track program for testing. Currently, a windows version of Low Track that will generate a uniform manifest is being distributed for testing. Comments are to be provided to NEI for submittal to DOE, NRC and the LLW Forum (the conference of States which requested NRC to develop the manifest).

At this time there are two options for distribution of the Low Track software. One option is for DOE to provide it free of charge to anyone who wants it. The other is to charge certain users (*i.e.*, commercial utilities and industry). No decision has been made to date on this issue.

Generation of the uniform manifest by a utility could be accomplished by several means:

- the utility could upgrade or procure existing commercial software to generate uniform manifests with SI units,
- the utility could procure an upgrade which would allow their existing software to export data to Low Track to generate the uniform manifest, or
- the utility could build specific software to generate the uniform manifest.

Since compacts may use the Low Track software, there may be an advantage for using this code. It is envisioned that LLW information could be directed to the compact electronically. Some compacts are requiring such information to be submitted on a monthly basis currently. In addition, the disposal site may allow the 541 form to be sent electronically. This would limit the shipping papers a carrier would need to the Bill of Lading and the 540 form. DOE will meet with Barnwell in 1996 to discuss this possibility.

The new DOT shipping regulations will have an impact on the uniform manifest. The order in which nuclides are listed and how many nuclides must be listed are affected by the new regulations. The nuclides must be listed in decreasing order of their hazard. Alphabetical listing is not acceptable. The method of ranking a nuclide's hazard has been specified in 49 CFR 173.433 (f). The activity of the nuclide divided by it's A_2 value is it's hazard ranking. For the 540 form all the nuclides which make up 95% of the total hazard must be listed. The alpha numeric symbols for each nuclide must be listed in entry block (15) of the 540 form. The total activity in the shipment must be listed in entry block (16) of the 540 form.

Since the 541 form requires the activity (or percentage) of each nuclide to be listed a discrepancy between the total activity of the 540 and 541 form could occur. Resolution of this item has yet to be addressed.

A discrepancy between the number of containers listed on the 540 and 541 form will occur for cask shipments of drums. For a cask shipment only one shipping container, the cask, can be listed. One is left to assume that the cask Serial Number is to be entered in block (19) of form 540 for cask shipments. For the 541 form, every drum in the cask must be listed. Resolution of this item has yet to be addressed.

In spite of these unknowns, one should insure that the uniform manifest generation method selected:

- accurately converts standard units input into SI units reported
- lists nuclides in decreasing order of their hazard per the new formula
- reports only the total shipment activity on the 540 form
- reports the same total activity on the 540 form as is reported on the 541 form

Manifest Information Requirements

These data are to be placed on new uniform manifests, NRC Uniform Manifest Forms, 540, 541, and, if necessary, 542.

General Information

1. Name, facility address, and telephone number of the licensee;
2. An explicit declaration indicating whether the shipper is acting as a waste generator, collector, processor, or some combination of these; and
3. Name, address, and phone number or the name and EPA identification number of the carrier transporting the waste.

Shipment Information

1. Date of the waste shipment;
2. Total number of packages or containers;
3. Total disposal volume and weight in the shipment;
4. Total radionuclide activity in the shipment;
5. Activity of each of the nuclides ^{3}H, ^{14}C, ^{99}Tc, and ^{129}I contained in the shipment; and
6. Total masses of ^{233}U, ^{235}U, and plutonium in special nuclear material (SNM), and total mass of uranium and thorium in source material.

Disposal Container and Waste Information

1. A unique alphanumeric identification that identifies each disposal container in the shipment;
2. Physical description of the disposal container, including the manufacturer and model of any high-integrity container;
3. Volume displaced by the container;
4. Gross weight of the container with the waste;
5. Maximum radiation level at the surface of each container for waste consigned to a disposal facility;
6. Physical and chemical description of each waste;
7. Total weight percentage of chelating agent for any waste containing more than 0.1% by weight, and the identity of the chelating agent;
8. Approximate volume of waste within each container;
9. Sorbing or solidification media, if any, and the identity of the solidification media vendor and brand name;

10. Identities and activities of individual radionuclides contained in each container, the masses of ^{233}U, ^{235}U, and plutonium in SNM and the masses of uranium and thorium in source material. For discrete waste types (*i.e.*, activated materials, contaminated materials, mechanical filters, sealed sources and devices, and wastes in solidification or stabilization media), the identities and activities of individual *significant* radionuclides associated with or contained on these waste types within each disposal container shall be reported.

> A *significant* nuclide always includes ^{3}H, ^{14}C, ^{99}Tc, and ^{129}I, and, also, any other nuclide present in a "significant concentration" defined as more than 1% of its Class A limit given in 10CFR61.55.

11. Total radioactivity within each container; and
12. For wastes consigned to a disposal facility, the classification of the waste according to 10CFR61.55. Waste not meeting the structural stability requirements of 10CFR61.56 must be identified.

Uncontainerized Waste Information

1. Approximate weight and volume of waste;
2. Physical and chemical description of the waste;
3. Total weight percentage of chelating agent for any waste containing more than 0.1% by weight, and the identity of the chelating agent;
4. For wastes consigned to a disposal facility, the classification of the waste according to 10CFR61.55. Waste not meeting the structural stability requirements of 10CFR61.56 must be identified.
5. Identities and activities of individual significant radionuclides contained in each container, the masses of ^{233}U, ^{235}U, and plutonium in SNM and the masses of uranium and thorium in source material.
6. For wastes consigned to a disposal facility, the maximum radiation levels at the surface of the waste.

Multi-Generator Disposal Container Information

This applies to containers with mixtures of waste originating from different generators and waste processors.

1. For homogeneous mixtures of waste, such as incinerator ash, provide the waste description applicable to the mixture and the volume of the waste attributed to each generator;
2. For heterogeneous mixtures of waste, such as the combined products from a large compactor, identify each generator contributing waste to the disposal container, and, for discrete waste types (*i.e.*, activated materials, contaminated materials, mechanical filters, sealed sources and devices, and wastes in solidification or stabilization media), the identities and activities of individual significant radionuclides associated with or contained on these waste types within each disposal container shall be reported.

For each generator, provide:

a. Volume of waste within the disposal container;

b. Physical and chemical description of the waste, including the solidification agent, if any;

c. Total weight percentage of chelating agent for any waste containing more than 0.1% by weight, and the identity of the chelating agent;

d. Sorbing or solidification media, if any, and the identity of the solidification media vendor and brand name if the media is claimed to meet stability requirements in 10CFR61.56; and

e. Radionuclide identities and activities of individual significant radionuclides contained in each container, the masses of ^{233}U, ^{235}U, and plutonium in SNM and the masses of uranium and thorium in source material.

The amendments also describe certification and control, tracking, and reporting requirements for shipments of waste.

10

RECORD KEEPING

The BTP on radioactive waste classification specifies that waste generators must "record on shipment manifests a description of the transferred waste, and must also carry out a quality control program to assure that classification of the waste is carried out in a proper manner." Generators must be able to defend waste characterization results. Part of any good quality control program is the documentation and maintenance of records developed as part of waste characterization process. Waste characterization data must be documented for use as described in the following sections:

- Section 7, Use of Laboratory Data, discusses record keeping requirements relative to the determination of scaling factors and trending analysis.
- Section 9, Waste Manifest, discusses record keeping requirements as they relate to the uniform waste manifest.

Most nuclear utilities have record retention policies that extend well beyond the life of the plant; therefore, retention of waste characterization results should not pose any special problems. Waste generators should, however, make sure to properly catalog results by waste stream for easy retrieval (if required).

11

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A

PERFORMANCE ASSESSMENT AND LLW CHARACTERIZATION

INTRODUCTION

The primary objective of low-level radioactive waste (LLW) characterization in utility waste management is to accurately assess the inventory of radionuclide in a waste disposal facility. For the difficult to measure (DTM) radionuclides within LLW, scaling factors are used to quantify their concentrations which dictate the radioactivity inventory within a facility. These scaling factors are known to have large variabilities, depending upon the types of waste streams and radionuclides, ranging several orders of magnitude.

The purpose of this Appendix is to investigate the importance of LLW characterization with respect to waste facility performance assessment. For this purpose, an overview of LLW performance assessment is provided and the radionuclides and waste streams of importance in performance assessment are identified. Scaling factors used in the analysis are estimated based on the shipping manifest information from the Barnwell LLW site. Sensitivities of scaling factors of each radionuclide on the estimated dose are evaluated for a test case problem.

OVERVIEW OF PERFORMANCE ASSESSMENT

The U.S. Code of Federal Regulations, Title 10, Chapter 1, Part 61 "Licensing Requirements for Land Disposal of Radioactive Waste," specifies license requirements and performance objectives for LLW disposal facilities as following:

"Concentrations of radioactive material which may be released to the general environment in groundwater, surface water, air, soil, plants or animals must not result in an annual dose exceeding an equivalent of 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public. Reasonable efforts should be made to maintain releases of radioactivity in effluents to the general environment as low as reasonably achievable."

Performance assessment is the technical analysis used to demonstrate compliance with the above stated (10CFR61.41) objectives. Performance assessment is therefore a quantitative assessment of the consequences of the events relative to the performance objectives and is done by using computer models as mathematical representation of physical phenomena that may release, transport and ultimately lead to the exposure of humans to the radionuclides contained in the waste.

Major components of performance assessment include the following:

- Characterization of LLW - Characteristics of LLW within the disposal facility such as waste volumes, radionuclide inventory (specific to waste forms and containers used for disposal).
- Source Term Model/Facility Model - Modeling of processes that control the release of radionuclides from the waste and the disposal unit. The model requires description of facility design and natural site features, degradation/failure of engineered barriers (soil covers, concrete barriers, waste containers), infiltration of water through the unsaturated soil and engineered barriers, release of radionuclides from various waste forms, and near-field transport.
- Transport Model - Modeling of transport mechanisms and pathways by which radionuclides can result in human exposures. The model includes description of transport of radionuclides from the facility through the environment via air, surface water, soil, groundwater, and biota.
- Exposure/Dose Model - Modeling of exposure and doses delivered to the public as a result of radionuclide release from the site based on exposure scenario. Exposure scenarios combine basic exposure modes - external, inhalation, and ingestion - and demographic data to postulate human activities related to exposure.

There are various computer models available for performance assessment. A representation of these models is given in Table A-1.

INVENTORY DISTRIBUTION OF 10 CFR 61 NUCLIDES IN LLW STREAMS

Performance assessment of a waste disposal facility is directed by the inventory of radionuclides stored in the facility. Table A-2 shows the distribution of radionuclide inventory in various LLW streams for the 10CFR61 radionuclides associated with the use of scaling factors. This information was obtained by using the radionuclide concentrations estimated by Dames & Moore, based on shipping manifests collected at Barnwell, SC [Dames & Moore, 1995] between 1989 and 1994, and the estimated volume of annual generation for each waste streams. Utility wastes account for the majority of inventory for all the radionuclides listed in Table A-2. The contributions from non-utility generators are separately listed in Table A-3. Within the utility wastes, dewatered resins and irradiated equipment are the two major waste streams that make up the majority of inventory for the listed radionuclides except ^{99}Tc, ^{238}Pu, ^{239}Pu and ^{241}Am.

Table A-1
Computer Codes for Performance Assessment

Component	Computer Codes
Percolation	VAM2D
	HELP
	CREAMS
	UNSAT-H
Source Term	BLT
	DUST
	PRESTO
	BARRIER
	IMPACTS
	PAGAN
	NEFTRAN
	GWSCREEN
Unsaturated Zone Flow/Transport	VAM2D
	UNSAT-H
	FEMWATER/FEMWASTE
	BLT
	PAGAN
	PRESTO
	PATHRAE
	GWSCREEN
Saturated Zone Flow/Transport	VAM2D
	FEMWASTE
	BLT
	PRESTO
	PAGAN
	PATHRAE
	NEFTRAN
	GWSCREEN
Surface Water	GENII
	PRESTO
	PATHRAE
Air	GENII
	CAP-88
	AIRDOS-PC
	PRESTO
Exposure and Dose	GENII
	PRESTO
	PATHRAE

Table A-2
Inventory of 10CFR61 Radionuclides in LLW Streams - Utility Waste Streams (in %: any waste stream with less than 1% contribution is not listed)

(): indicates Class	^{14}C	^{129}I	^{99}Tc	^{59}Ni	^{63}Ni	^{60}Co	^{90}Sr	^{94}Nb	^{137}Cs	^{238}Pu	^{239}Pu	^{241}Pu	^{241}Am	^{242}Cm	^{243}Cm
Resins, dewatered (AS) (chelating agent)	5.4	23.5 (1.9)	1.0	0.6 (1.8)	1.4	0.9	4.3	0.2	5.0	2.6	0.6	5.0	0.5	12.7	9.5
Resins, dewatered (AU)	3.7	2.8	0.4	-	0.4	0.1	1.2	7.3	0.8	0.4	0.2	1.4	0.3	6.2	3.3
Resins, dewatered (B)	14.0	6.9	3.0	6.4	11.8	2.2	26.0	23.9	29.7	5.4	2.0	11.4	2.4	16.3	23.8
Resins, dewatered (C)	23.5	2.0	7.7	11.6	10.0	2.3	26.7	32.2	45.3	8.9	5.4	31.2	3.2	21.0	36.6
Filter media, dewatered (AS)	0.6	0.4	0.1	0.1	0.1	-	0.4	-	0.2	0.8	0.1	1.3	0.3	4.5	2.6
Filter media, dewatered (C)	0.1	-	1.2	-	0.3	-	0.2	0.2	-	0.6	0.1	1.7	0.1	-	-
Cartridge filter, dewatered (B)	0.3	-	0.1	-	0.2	0.3	3.3	1.0	0.2	0.1	0.1	0.5	-	1.0	0.8
Cartridge filter, dewatered (C)	2.2	8.7	2.9	8.9	1.5	0.3	2.9	0.6	0.3	2.6	1.3	9.7	1.2	7.8	14.2
Cartridge filter, cement (AS)	-	-	-	-	-	-	-	-	-	26.9	-	-	-	-	-
Equipment, solid (B)	-	-	-	5.8	0.4	0.3	-	0.1	-	-	-	-	-	-	-
Equipment, solid (C)	24.0	-	3.3	63.5	73.0	91.8	0.2	31.5	-	2.2	0.1	1.9	0.6	19.8	1.9
Solid noncmbustble solid (C)	0.1	-	-	-	-	-	-	-	5.9	-	-	0.4	-	-	-
Mixed DAW, solid (AU)	3.5	23.8	3.2	0.3	0.1	0.1	0.4	-	0.4	0.9	0.5	1.7	0.4	1.9	1.6
Mixed DAW, solid (C)	0.1	0.1	1.8	-	0.1	-	0.2	-	0.2	0.3	0.5	1.1	1.0	1.2	0.3
Solidified liquid, cement (AU)	0.1	1.9	0.2	-	-	-	-	-	-	-	-	0.1	-	-	0.1
Solidified liquid, cement (C)	-	-	28.1	-	-	-	0.4	-	-	-	0.4	-	-	-	-
Incinerator ash, cement (AU)	0.1	1.9	1.6	-	-	-	-	-	-	-	-	-	-	0.1	-
Incinerator ash, solid (AU)	0.3	2.3	1.1	-	-	-	-	-	0.1	0.5	0.2	0.3	0.1	0.6	0.7
Subtotal	**78.0**	**76.2**	**55.7**	**99.0**	**99.3**	**98.3**	**66.2**	**97.0**	**88.1**	**52.2**	**11.5**	**67.7**	**10.1**	**93.1**	**95.4**
Utility %	**80.5**	**77.8**	**57.2**	**99.6**	**99.8**	**98.6**	**68.6**	**97.9**	**89.0**	**53.1**	**12.0**	**70.6**	**10.4**	**99.8**	**99.9**

Table A-3
Inventory of 10CFR61 Radionuclides in LLW Streams - NonUtility Waste Streams (in %: any waste stream with less than 1% contribution is not listed)

(Class)	^{14}C	^{129}I	^{99}Tc	^{59}Ni	^{63}Ni	^{60}Co	^{90}Sr	^{94}Nb	^{137}Cs	^{238}Pu	^{239}Pu	^{241}Pu	^{241}Am	^{242}Cm	^{243}Cm
Industry, solidified liquid, cement (AU)	2.5	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Industry, mixed DAW, solid (AU)	6.8	6.6	21.4	0.2	-	-	0.3	0.6	0.1	12.5	25.3	11.7	21.2	0.2	-
Industry, mixed DAW, solid (C)	0.3	-	-	-	-	-	-	-	-	33.0	60.7	15.7	53.3	-	-
Industry, solid noncombustible, sealed source (B)	-	-	-	-	-	1.2	0.1	-	0.1	-	-	-	-	-	-
Industry, solid noncombustible, sealed source (C)	-	-	-	-	-	-	1.7	-	0.3	-	-	-	5.3	-	-
Industry, solid noncombustible, solid (AU)	1.0	4.4	5.7	0.2	-	-	0.1	0.9	-	1.0	1.9	1.1	1.8	-	-
Industry, filter media, cement (AU)	-	5.5	0.5	-	-	-	0.1	-	-	-	-	0.2	-	-	
Industry, incinerator ash, solid (AU)	0.8	0.1	12.6	-	-	-	-	-	-	-	-	-	0.4	-	-
Government, resins cement (AS)	1.6	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Government, solid noncombustible, sealed source (C)	-	-	-	-	-	-	0.1	-	10.3	-	-	-	0.4	-	-
Government, solid noncombustible, solid (AU)	-	-	-	-	-	-	0.1	-	-	-	-	-	1.9	-	-
Government, filter media, cement (AS)	1.0	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Academic, mixed DAW, solid (AU)	1.4	0.1	-	-	-	-	-	-	-	-	-	-	0.1	-	-
Academic, solidified liquid, cement (C)	-	3.3	-	-	-	-	27.6	-	-	-	-	-	4.7	-	-
Subtotal	**15.4**	**20.0**	**40.2**	**0.4**	**-**	**1.2**	**30.1**	**1.5**	**10.8**	**46.5**	**87.9**	**28.7**	**89.1**	**0.2**	**-**
Non-Utility %	**19.5**	**22.2**	**42.8**	**0.4**	**0.2**	**1.4**	**31.4**	**2.1**	**11.0**	**46.9**	**88.0**	**29.4**	**89.6**	**0.2**	**0.1**

As shown in Table A-3, the contributions from non-utility generators to the total activity inventory are significant for ^{14}C (19.5% of total), ^{129}I (22.2%), ^{99}Tc (42.8%), ^{90}Sr (31.4%), ^{137}Cs (11.0%), ^{238}Pu (46.9%), ^{239}Pu (88.0%), ^{241}Pu (29.4%), and ^{241}Am (89.6%). Important waste streams of non-utility waste include mixed DAW (Class AU & C), solid noncombustible waste (untreated Class AU solid), cemented filter media (Class AU), and incinerator ash (Class AU), all from industry.

TEST CASE PERFORMANCE ASSESSMENT FOR 10CFR61 NUCLIDES

To identify the radionuclides of importance in performance assessment, test case analyses are made. The test case is based on a hypothetical waste disposal facility, characterized as humid permeable site. Major input parameters used are listed in Table A-4. Detailed characterization of radionuclide inventory distributions in various waste forms and containers is made based on the information collected at the Barnwell, SC, LLW site between 1989 and 1994. For the analysis of radionuclide leaching, transport, and exposure/dose modeling, a model based on PRESTO-EPA model [Hung, 1989] is used.

Table A-4
Major Inputs for the Test Case Performance Assessment

total volume of waste disposed = 11 million ft^3
facility surface area = 9.29E4 m^2
annual infiltration = 0.1172 m/yr.
Darcy velocity in the aquifer = 576.7 m/yr.
distance between the facility and the boundary well = 363.6 m
nominal depth of the disposal unit = 10.7 m
porosity of soil = 0.444
density of soil = 1.6 g/cm^3
density of waste = 1.331 g/cm^3
beginning years of cap failure = 0 year
ending years of cap failure = 300 years
hydraulic conductivity of soil = 2.2 m/yr.
number of years before waste containers begin failing = 0 year
year that containers fail completely = 300 yr.
Kd of soil & waste (ml/g):
    ^{14}C: 1; ^{3}H: 0.06; ^{99}Tc: 0.1; ^{129}I: 1; Ni: 74; ^{90}Sr: 5;
    ^{137}Cs: 280; ^{60}Co: 60; Pu: 540; Am: 940; Cm: 4000
Solubility limits (g/ml):
    ^{14}C: $3\text{x}10^{-11}$; ^{3}H: $4\text{x}10^{-9}$; ^{99}Tc: 0.115; ^{129}I: $1.39\text{x}10^{-3}$;
    ^{59}Ni: $5.9\text{x}10^{-4}$; ^{63}Ni: $6.3\text{x}10^{-4}$; ^{90}Sr: 1.0; ^{137}Cs: $1.67\text{x}10^{-4}$;
    ^{60}Co: $6\text{x}10^{-4}$; Pu: $2.42\text{x}10^{-4}$; Am: $2.41\text{x}10^{-2}$; Cm: $2.43\text{x}10^{-15}$

The model for radionuclide leaching is based on the assumption that linear isotherm exists between the solid and the liquid phase and that leaching is proportional to the reciprocal of the mean residence time of contaminant within the waste. The groundwater transport model of PRESTO-EPA is based on a semi-analytic equation, as given in the following equation [U.S. EPA, 1988].

$$C_{well}(t) = C_{source}(t-\tau)\frac{\eta \cdot \exp(-\lambda L_{uns} R_{uns} / \upsilon_{uns}) \cdot \exp(-\lambda L_{aq} R_{aq} / \upsilon_{aq})}{\upsilon_{aq}\, \varepsilon_{aq} T_{aq} [\sqrt{A_{sa}} + 2\tan(\alpha/2) L_{aq}]}$$

where,

η = long − term health effects correction factor

$C_{well}(t)$ = radionuclide concentration at the site boundary well as a function of time

$C_{source}(t)$ = radionuclide concentration at the bottom of the facility as a function of time

τ = travel time from the facility bottom to the well at site boundary

λ = decay constant of the radionuclide

L_{uns} = distance from the bottom of the facility to the nominal depth of the aquifer

$R_{uns}; R_{aq}$ ≡ retardation factor (in unsaturated region; in the aquifer) $= 1 + \frac{K_d \rho}{\varepsilon}$

υ_{uns} = vertical water velocity in the unsaturated region

L_{aq} = distance from the center of the facility to the site boundary well

υ_{aq} = velocity of groundwater in the aquifer

ε_{aq} = aquifer porosity

T_{aq} = thickness of the aquifer at the location of the well

A_{sa} = total surface area of the waste facility

α = dispersion angle of the pollutant plume in the aquifer

K_d = partition coefficient of the radionuclide

ρ = bulk density of soil

Release characteristics of radionuclides from waste packages depend on length of time that waste containers maintain their integrity. In the current analysis, time-distributed failures based log-uniform distribution are assumed for high integrity containers (HICs; between 300 and 1000 years) and carbon steel liners (5 to 50 years). No credit was given to carbon steel drums.

From the test case analyses, key radionuclides in LLW performance assessment among the 10CFR61 nuclides are found to be ^{14}C, ^{129}I, and ^{99}Tc, followed by ^{239}Pu, ^{59}Ni, and ^{94}Nb. Results of these analyses are given in Table A-5. Many of the nuclides including ^{63}Ni, ^{241}Am, ^{60}Co, ^{137}Cs, ^{238}Pu, ^{241}Pu, ^{242}Cm, and ^{243}Cm show no or negligible contribution to the projected dose. For the major nuclides in the assessment (^{14}C, ^{129}I, and ^{99}Tc), the peak dose is estimated to be in the order of 1 mrem/yr. occurring around 1,000 year after the closure of a site. Peak dose from ^{239}Pu, ^{59}Ni and ^{94}Nb is estimated to be in the order of 10^{-2} mrem/yr. with much longer time period before the observation of peak.

Table A-5
Results of Test Case Performance Assessment for 10CFR61 Nuclides

Nuclide	Peak dose (mrem/yr.)	Time to peak (yr.)	Inventory from utilities
^{14}C	3.14	1090	80.5 %
^{129}I	1.00	1090	77.8 %
^{99}Tc	0.43	1020	57.2 %
^{239}Pu	0.052	41500	12.0 %
^{59}Ni	0.025	6560	99.6 %
^{94}Nb	0.011	13000	97.9 %
^{3}H	4.4×10^{-4}	150	0.42 %
^{90}Sr	7.7×10^{-8}	750	68.6 %
^{63}Ni	0.0 ($\sim 10^{-17}$)	6560	99.8 %
^{241}Am	0.0 ($\sim 10^{-55}$)	82600	10.4 %
^{60}Co	0.0	n/a	98.6 %
^{137}Cs	0.0	n/a	89.0 %
^{238}Pu	0.0	n/a	53.1 %
^{241}Pu	0.0	n/a	70.6 %
^{242}Cm	0.0	n/a	99.9 %
^{243}Cm	0.0	n/a	99.8 %

POTENTIAL IMPACTS OF SCALING FACTOR VARIABILITIES ON PERFORMANCE ASSESSMENT

To investigate the impacts of scaling factor changes of key radionuclides on performance assessment, sensitivity calculations are made for the waste streams of importance. Radionuclides considered for this analysis include ^{14}C, ^{129}I, ^{99}Tc, ^{59}Ni, ^{239}Pu, and ^{94}Nb. Table A-6 lists scaling factors for these radionuclides estimated in each of the waste streams considered. To compare the sensitivities, these scaling factors were increased by a factor of 10.

Table A-7 shows the results of these comparisons. Each data points in the table represent the ratio of dose estimate after the increase of scaling factor by a factor of 10 to the estimated dose in the base case. The results show that the changes in scaling factors for key waste streams have a significant impact on the results of performance assessment. Important waste streams included dewatered resins (all classes), Class C solid irradiated equipment, Class A unstabilized mixed DAW, and Class C dewatered cartridge filters: Dose estimates are shown to increase by a factor of up to 3.2 for ^{14}C and ^{129}I (Class C dewatered resins and irradiated hardware for 14C; Class A stabilized dewatered resins and unstabilized mixed DAW for ^{129}I). For ^{59}Ni and ^{94}Nb, the increase in the ratio is estimated to be much higher for the Class C irradiated hardware.

IMPLICATIONS OF INVENTORY OVER-REPORTING FOR ^{129}I AND ^{99}TC

For ^{129}I and ^{99}Tc, the lower limits of detection (LLD) are often reported by the laboratory analyzing the samples when these nuclides have not been detected or measured. When the LLD, rather than actual analyzed concentration, are used in the scaling factors the concentrations of these radionuclides in the wastes are over-estimated. This over-estimation of ^{129}I and ^{99}Tc inventory was found to range by factors of 10 to 10,000 [Robertson, 1991; Vance; 1992]. Overestimation of inventory of ^{129}I and ^{99}Tc potentially have a significant implications in the licensing of any proposed LLW site.

Concentrations of ^{129}I and ^{99}Tc in LLW used in this study are not based on LLDs and reflect the recent findings of ^{129}I and ^{99}Tc inventory in LLW [EPRI, 1992; Robertson, 1991; Vance, 1992; NYLLWSC, 1995]. Results in Table A-7 show that increasing the current scaling factors for a single waste stream by a factor of 10 could increase the peak projected dose of ^{129}I and ^{99}Tc by a factor of up to 4. If the scaling factors of ^{129}I and ^{99}Tc for all waste streams are increased by a factor of 10, the projected peak dose will increase proportionally (since solubility limits of these nuclides are very high) approaching the 25 mrem/yr. of regulatory limit (the projected peak dose for ^{129}I and ^{99}Tc was estimated to be 1.0 mrem/yr. and 0.43 mrem/yr., respectively). This tells that scaling factors of ^{129}I and ^{99}Tc play important role in LLW performance assessment.

Table A-6
Estimated Scaling Factors of Nuclides of Importance Based on Barnwell Shipping Manifests

( ): indicates Class	**^{14}C**	**^{129}I**	**^{99}Tc**	**^{59}Ni**	**^{239}Pu**	**^{94}Nb**
Scaling Factor Pair	^{14}C/^{60}Co	^{129}I/^{137}Cs	^{99}Tc/^{137}Cs	^{59}Ni/^{63}Ni	^{239}Pu/^{144}Ce	^{94}Nb/^{60}Co
Resins, dewatered (AS)	3.31E-3	2.68E-4	1.08E-4	2.88E-3	1.29E-3	2.41E-6
Resins, dewatered (AU)	1.72E-2	2.07E-4	2.94E-4	6.83E-4	6.29E-3	6.76E-4
Resins, dewatered (B)	3.65E-3	1.31E-5	5.63E-5	3.67E-3	1.94E-3	1.24E-4
Resins, dewatered (C)	6.10E-3	2.49E-6	9.45E-5	7.95E-3	1.74E-2	1.66E-4
Filter media, dewatered (AS)	1.00E-2	1.54E-4	2.83E-4	5.65E-3	1.47E-3	4.19E-7
Filter media, dewatered (C)	3.25E-3	1.03E-4	1.46E-1	8.27E-4	8.91E-3	8.04E-5
Cartridge filter, dewatered (B)	7.37E-4	1.37E-6	1.76E-4	1.01E-3	6.27E-4	4.40E-5
Cartridge filter, dewatered (C)	4.47E-3	1.84E-3	5.99E-3	3.96E-2	8.04E-3	2.39E-5
Cartridge filter, cement (AS)	7.99E-5	4.52E-4	5.43E-4	-	1.17E-2	-
Equipment, solid (B)	1.61E-5	-	4.89E-4	1.12E-1	9.02E-3	2.14E-6
Equipment, solid (C)	1.52E-4	-	3.10E-1	5.94E-3	2.67E-4	3.98E-6
Solid noncmbustble solid (C)	3.85E-2	-	-	-	4.22E-4	-
Mixed DAW, solid (AU)	4.22E-2	3.78E-3	4.96E-3	1.95E-2	1.86E-2	9.32E-7
Mixed DAW, solid (C)	6.01E-3	2.45E-5	7.06E-3	1.32E-3	2.07E-2	-
Solidified liquid, cement (AU)	9.22E-3	7.54E-3	8.48E-3	5.76E-4	2.20E-1	5.38E-5
Solidified liquid, cement (C)	-	-	9.72E-1	-	-	-
Incinerator ash, cement (AU)	8.55E-2	4.03E-2	3.33E-1	1.68E-5	5.06E-2	-
Incinerator ash, solid (AU)	4.32E-2	2.08E-3	1.01E-2	6.49E-3	5.68E-2	-

Table A-7
Impacts of (Scaling Factor)x10 in Key Waste Streams on Peak Dose Estimates for Nuclides of Importance

( ): indicates Class	**^{14}C**	**^{129}I**	**^{99}Tc**	**^{59}Ni**	**^{239}Pu**	**^{94}Nb**
Scaling Factor Pair	^{14}C/^{60}Co	^{129}I/^{137}Cs	^{99}Tc/^{137}Cs	^{59}Ni/^{63}Ni	^{239}Pu/^{144}Ce	^{94}Nb/^{60}Co
Peak dose (test case)	3.14 (mrem/yr.)	1.00 (mrem/yr.)	0.43 (mrem/yr.)	0.052 (mrem/yr.)	0.025 (mrem/yr.)	0.011 (mrem/yr.)
Time to peak (yrs)	1090	1090	1020	41500	6560	13000
Resins, dewatered (AS)	1.48	3.19	1.12	1.05	1.05	1.02
Resins, dewatered (AU)	1.33	1.26	1.05	1.02	-	1.66
Resins, dewatered (B)	2.26	1.64	1.38	1.18	1.56	3.16
Resins, dewatered (C)	3.11	1.18	1.96	1.48	2.04	3.90
Filter media, dewatered (AS)	1.05	1.04	1.01	1.01	1.0	-
Filter media, dewatered (C)	1.01	-	1.16	1.01	-	1.02
Cartridge filter, dewatered (B)	1.03	-	1.01	1.01	-	1.09
Cartridge filter, dewatered (C)	1.19	1.81	1.36	1.11	1.79	1.06
Equipment, solid (B)	-	-	-	1.0	1.51	1.01
Equipment, solid (C)	3.15	-	1.41	1.0	6.67	3.84
Mixed DAW, solid (AU)	1.31	3.21	1.40	1.04	1.02	-
Mixed DAW, solid (C)	1.01	1.01	1.23	1.04	-	-
Solidified liquid, cement (AU)	1.01	1.17	1.03	-	-	-
Solidified liquid, cement (C)	-	-	1.00	1.0	-	-
Incinerator ash, cement (AU)	1.00	1.17	1.20	-	-	-
Incinerator ash, solid (AU)	1.03	1.21	1.14	1.02	-	-

Since major portion of the inventory of ^{129}I and ^{99}Tc in a LLW site is contributed by non-utility generators (about 22% and 40% of the total inventory, respectively), both ^{129}I and ^{99}Tc are expected to remain important in LLW performance assessment regardless of the contribution from the utilities.

CONCLUSIONS/DISCUSSIONS

Based on test case analyses for a hypothetical LLW disposal facility, key radionuclides in LLW performance assessment among the 10CFR61 nuclides are found to be ^{14}C, ^{129}I, ^{99}Tc. ^{59}Ni, ^{239}Pu, ^{94}Nb. The test case analysis shows that the changes in scaling factors for key waste streams could potentially impact the performance assessment significantly. Waste streams of major importance are Class C and Class B dewatered resins, Class A unstabilized mixed DAW, Class C solid irradiated equipment, and Class C dewatered cartridge filters. It is found that using accurate scaling factors for ^{129}I and ^{99}Tc could be critically important to meet the regulatory limits of LLW facility performance.

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B

MICROWAVE DIGESTION OF RESIN SAMPLES FOR 10CFR61 WASTE STREAM CHARACTERIZATION

INTRODUCTION

Accuracy of analytically determined waste stream scaling factors requires that collected samples be representative of the waste stream and that the radionuclide content of the samples be homogeneous. For many waste streams, waste stream representativeness and sample homogeneity characteristics are not a problem. However, for highly radioactive resin waste streams where only a few resin beads or milligrams of resins are analyzed, waste stream representation and sample homogeneity is difficult. For low activity resin, larger samples are available to be shipped for analysis. However, for high activity resin, the size of the sample received and analyzed by the vendor laboratory is restricted by the laboratory's radioactive materials license or by limiting radiation exposure to the laboratory staff.

The representativeness and homogeneity of high activity resin samples can be improved if the plant were to take a larger sample size, digest the resin matrix into a solution and forward a representative and homogeneous aliquant of the solution for radiochemical analysis. To facilitate this recommended practice, a resin sample preparation technique has been developed using readily available equipment in a chemistry laboratory of a nuclear power plant. The technique uses microwave digestion of gram quantities of resin in various acids under moderately elevated temperatures and pressures. Two separate protocols have been outlined for use to reflect different sample preservation or analytical requirements of the vendor radioassay. It is recommended that the plant and vendor laboratory staff review the proposed protocols for suitability prior to formal implementation.

The protocols developed do not involve the extensive use of radiotracers or the addition of a stable analyte for the determination of radionuclide losses during processing; therefore, attention must be given to ensure the integrity of the sample digestion. Special procedure testing was conducted to verify applicability of each protocol, especially for the potentially volatile radionuclides, ^{14}C, ^{99}Tc and ^{129}I. Losses in ^{14}C and ^{129}I have been identified. Both protocols generate acidic solutions containing the metals and non-volatile radionuclides and an alkaline solution containing the recovered volatile nuclides generated from the microwave digestion process. The first protocol combines the residual acidic and alkaline solutions whereas the second protocol keeps the solutions separate for transmittal to the vendor laboratory. For both protocol, excess solution can be disposed of on-site by combining the acidic and alkaline portions creating a neutral solution suitable for discharge or treatment by ion exchange systems.

These protocols have not addressed or evaluated the recovery of tritium in the resultant solutions. The reader should follow the recommendation provided in the guidelines for the analysis of tritium in waste stream samples.

MICROWAVE DIGESTION OF RESIN SAMPLES FOR 10CFR61 WASTE STREAM CHARACTERIZATION

PROTOCOL 1[1]

PRINCIPLE OF METHOD

Small quantities of spent resin are completely digested by vigorous acid treatment in conjunction with a high temperature and pressure environment created by microwaves. The volatile radionuclides are collected in sequential bubblers containing alkaline solutions of tetrabutylammonium hydroxide and sodium hydroxide. The non-volatile/gaseous radionuclides remain in the acidic media. The alkaline and acidic solutions are carefully combined and neutralized to a pH of 7. An aliquant of the solution is taken and sent to the vendor laboratory for radioassay of the gamma-emitting and difficult-to-measure nuclides.

For this protocol, the expected chemical recovery for the metals and difficult-to-measure radionuclides should be approximately 100 percent, with the exception of ^{14}C and ^{129}I. The average recoveries of ^{14}C, ^{129}I and ^{99}Tc, based on three test samples, were 0.58 " 0.05, 0.87 " 0.19 and 1.00 " 0.03, respectively. The chemical recoveries of the volatile nuclides or those elements forming gaseous byproducts are highly dependent upon the integrity of the closed off-gas collection system. Therefore, it is recommended that the user be vigilant in ensuring a leakless offgas collection system.

The protocol is applicable for the commonly found anion, cation and mixed resin types and for a resin quantity of 0.5 grams per digestion setup. The microwave hardware tested permitted the separate venting of each microwave digestion vessel (ACV). Multiple resin samples can be processed simultaneously as long as separate digestion vessels, venting and offgas collecting apparatus including the bubblers are employed.

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[1] Reference to a manufacturer by name does not mean an EPRI endorsement of the product. Any suitable equivalent may be substituted. It is highly recommended that the protocol be re-validated if it is suspected that the chemical recoveries may be affected.

Therefore, for larger quantities of the same resin sample, digest several 0.5 gram resin aliquots from the same resin sample and combine the final solutions.

LIMITATIONS

1. Samples containing contaminants such as metal or fuel fines may require further treatment for complete dissolution.
2. This digestion protocol is limited to 0.5 grams of resin per digestion.
3. The microwave digestion vessels need to be completely decontaminated prior to reuse.

REAGENTS

1. Tetrabutylammonium Hydroxide (TBAH) 60% w/V, reagent grade
2. Methanol (CH_3OH), reagent grade
3. Sodium Hydroxide (NaOH) 6M solution, reagent grade
4. Nitric Acid (HNO_3): Concentrated (16M) and 3M solutions, reagent grade
5. Deionized (DI) water, ASTM Type III or better.

RESINS

1. This technique is applicable for any spent mixed, anion or cation resin.

APPARATUS

1. Bubblers:
   Kontes Midget Impingers (MI) # 737560-0000
   Kontes Connecting Adapters (CA) # 737575-0125
   Kontes Hose Connectors (HC) # 523870 - 0125

2. Microwave/Accessories:
   CEM's MDS 2000 (900w)
   Control Cap (CC)/Cover Kits - #327030
   Microwave Transparent Valves (MTV) - #327010
   Advanced Composite Vessels (ACV) - #327010

   NOTE: Use of Microwave Transparent Valves permits the separate venting of each microwave digestion vessel (ACV). Therefore, multiple samples can be processed simultaneously as long as separate digestion vessels, valves and offgas collecting apparatus including the bubblers are employed.

3. Vacuum pump - low volume.

4. Assorted glassware and shipping container.

HAZARDS

1. Use extreme caution when handling all acids and bases. Refer to your chemistry safety manual for proper safeguards and practices.
2. Use caution when employing the microwave digestion unit. Refer to the microwave unit operating manual and your safety manual.
3. Use caution when handling radioactive materials. Refer to applicable radiation safety procedures and regulatory requirements and guidelines.

   NOTE: Follow all standard operation procedures and radiation work permits when handling radioactive materials.

PROCEDURE

1. Weigh out approximately 0.5 grams of the resin sample.
2. Add 0.5 ml of DI water to the sample.
3. Pour the slurry sample into an advanced composite vessel (ACV).
4. Add 15 ml of concentration HNO_3 to the ACV.
5. Install control caps (CC) to the ACV.
6. Place the ACV into the microwave unit.
7. With rupture discs in place, install vent tubing on the digestion vessel.
8. Install a pressure sensing line on (one of) the digestion vessel(s).
9. Start the digestion program (1.25 hours) of:

	Step 1	Step 2	Step 3	Step 4	Step 5
Power (W)	23	26	26	26	26
Time (m)	15	15	15	15	15
PSI	20	40	85	150	200

10. Prepare two impingers for each sample; one with 20 ml of tetrabutylammonium (designated as bubbler 1) and the other with 15 ml of methyl alcohol and 5 ml of NaOH (designated as bubbler 2). Attach a section of tygon tubing from the exhaust of bubbler 1 to the intake of bubbler 2. Attach a section of tygon tubing from the exit of bubbler 2 to a vacuum pump.

11. Remove the digestion vessel from the microwave unit two hours after the completion of the microwave digestion cycle..

12. Attach a section of tygon tubing between the microwave transparent valve of a digestion vessel to the intake of bubbler 1.

13. Slowly open the MTV and vent gases into the bubblers.

14. Start the vacuum pump and adjust the air flow rate to ~ 1.0 liter per minute.

15. Insert tygon tubing into the ACV via the top port and continue evacuating for 15 minutes.

16. Remove the control caps.

Note: losses in ^{14}C may occur during the next step

17. In a 50 or 100 ml pre-weighed (tared) glass vessel, slowly quantitatively combine the content of the ACV and bubblers. Rinse the ACV (with 3 M HNO_3) and bubblers (with DI water) as necessary and transfer rinses to the glass vessel.

18. Adjust the pH of the solution to 7 with pH paper using either 3M HNO_3 or 3M NaOH.

19. Weigh the combined resultant solution and glass vessel. Determine the final solution weight and calculate the ratio of the original resin weight to final solution weight.

20. Remove and transfer an appropriate aliquant of the solution for radioassay to a tared container. Determine the weight of the solution and calculate the solution's equivalent resin weight by applying the ratio of the original resin weight to final solution weight.

21. As required by the vendor laboratory, add the appropriate chemical preservatives to the sample.

22. Prepare the container for shipment to the vendor laboratory according to standard operating procedures and proper DOT regulations. Include a notation on the sample submittal or chain of custody form that refers to the equivalent resin weight.

PROTOCOL 2[1]

PRINCIPLE OF METHOD

Small quantities of spent resin are completely digested by vigorous oxidative/reductant treatment in conjunction with a high temperature and pressure environment created by microwaves. This protocol outlines separate microwave digestion processes for the two categories of the hard-to-measure nuclides, i.e., volatile nuclides and the non-volatile nuclides. Each process produces a final solution that would be analyzed by a vendor laboratory for a specific difficult-to-measure nuclide category. For the non-volatile/gaseous radionuclides (all nuclides excluding ^{3}H, ^{14}C and ^{129}I), the resin sample is combined with concentrated nitric acid and 30% hydrogen peroxide with subsequent exposure to microwaves. The volatile/gaseous radionuclides are collected in sequential bubblers containing an alkaline solution of sodium hydroxide.

For this protocol, the expected chemical recovery for the metals and difficult-to-measure radionuclides is nearly 100 percent. Refer to the table summarizing the

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[1] Reference to a manufacturer by name does not mean an EPRI endorsement of the product. Any suitable equivalent may be substituted. It is highly recommended that the protocol be re-validated if it is suspected that the chemical recoveries may be affected.

experimental test results for specific nuclide recovery data. The chemical recoveries of the volatile nuclides or those elements forming gaseous byproducts are highly dependent upon the integrity of the closed off-gas collection system. Therefore, it is recommended that the user be vigilant in ensuring a leakless offgas collection system. It should be noted that this protocol requires the addition of calibrated stable carbon and iodide carriers.

The tested protocol is applicable for the commonly found anion, cation and mixed resin types and for a resin quantity of 0.5 to 1.0 grams per digestion setup. The microwave unit employed did not have separate venting apparatus for each microwave digestion vessel. Therefore, multiple resin samples could not have been processed simultaneously.

LIMITATIONS

1. Samples containing contaminants such as metal or fuel fines may require further treatment for complete dissolution.
2. This digestion protocol is limited to 1.0 gram of resin per digestion.
3. The microwave digestion vessels need to be completely decontaminated prior to reuse.

REAGENTS

1. Hydrogen Peroxide (H_2O_2) - 30%, reagent grade
2. Sodium Hydroxide (NaOH) 6M solution, reagent grade
3. Nitric Acid (HNO_3): Concentrated (16M) and 3M solutions, reagent grade
4. Deionized (DI) water, ASTM Type III or better
5. Iodide carrier (NIST traceable): 12.0 mg/mL
6. Carbon carrier (NIST Traceable): 10 mg/mL. Various forms of carbon may be used, i.e., $Na_2C_2O_4$, Na_2CO_3.

RESINS

1. This technique is applicable for any spent mixed, anion or cation resin.

APPARATUS

1. Bubblers:
   - Kontes Midget Impingers (MI) # 737560-0000
   - Kontes Connecting Adapters (CA) # 737575-0125
   - Kontes Hose Connectors (HC) # 523870 - 0125
   - Tygon tubing - various lengths

2. Microwave/Accessories:
   Milestone MLS 1200 Mega
   MCR-6-E Evaporation, Concentration Rotor (ECR)
   HPR 1000/6 Digestion Rotor (DR)

   NOTE: This microwave unit does not permit the separate venting of each microwave digestion vessel. Therefore, multiple samples cannot be processed simultaneously without special equipment.

3. Vacuum system

4. Assorted glassware, 100 ml glass beakers and shipping container.

HAZARDS

1. Use extreme caution when handling all acids and bases. Refer to your chemistry safety manual for proper safeguards and practices.
2. Use caution when employing the microwave digestion unit. Refer to the microwave unit's operating manual and your safety manual.
3. Use caution when handling radioactive materials. Refer to applicable radiation safety procedures and regulatory requirements and guidelines.

   NOTE: Follow all standard operation procedures and radiation work permits when handling radioactive materials.

PROCEDURE

1. Weigh out accurately 0.5-1.0 grams of the resin sample.

2. Transfer the sample quantitatively to the microwave digestion vessel (MDV) chamber.

C-14 and I-129 Preparation:

3. Add 1 ml calibrated carbon (10 mg/ml) and iodine (12.0 mg/ml) NIST traceable stable carriers. Add 4 ml of ice-cold concentrated HNO_3 and 1 ml of ice-cold H_2O_2 to the sample in the MDV.

   CAUTION: IF ICE-COLD REAGENTS ARE NOT USED ^{14}C LOSSES MAY OCCUR WHEN THE REAGENTS ARE ADDED TO THE SAMPLE.

4. Cap the microwave digestion vessel (MDV), insert the MDV into the ECR and secure for exposure to microwaves.

5. Prepare two bubblers by adding 5 ml of 6M NaOH and 25 ml DI water to each bubbler.

6. Place the MDV/ECR unit into the microwave unit.

7. Install a vent tube from the MDV/ECR unit to the intake port of the first bubbler.

8. Install a length of tygon tubing from the exhaust port of the first bubbler to the intake port of the second bubbler. Attach another piece of tygon tubing from the exhaust port of the second bubbler to the vacuum system.

9. Turn the vacuum system on and start the digestion program of:

	Step 1	Step 2	Step 3	Step 4	Step 5	Step 6
Power (W)	250	0	720	500	400	0
Time (m)	5	2	5	5	5	5

10. Upon completion of the digestion program, continue to vent the system under vacuum for 10 minutes.

11. Transfer the alkaline solutions of the bubblers to a tared 100 ml glass beaker. Rinse the bubblers with 10 ml of DI water and add rinsate to the glass beaker.

12. Weigh the combined weight of the solution and glass beaker. Calculate the ratio of the original resin weight to the final sample solution weight.

13. Remove and transfer an appropriate aliquant of the solution for radioassay to a tared container utilized for shipment. Determine the weight of the solution and calculate the solution's equivalent resin weight by applying the ratio of the original resin weight to final sample solution weight.

14. As required by the vendor laboratory, add the appropriate chemical preservatives to the sample in the container.

15. Mark this container with the appropriate identification - this sample is to be analyzed for ^{14}C and ^{129}I. Include a notation on the sample submittal / chain of custody form that refers to the exact amount of stable carbon and iodine carriers added and the equivalent resin weight. Prepare the container for shipment to the vendor laboratory according to standard operating procedures and proper DOT regulations.

NON-VOLATILE/GASEOUS DIFFICULT-TO-MEASURE NUCLIDES:

16. Remove the MDV from the ECR and slowly open in a hood.

17. Carefully (and safely) observe the sample visually. If the sample has evaporated to dryness, add 4 ml of 16M HNO_3 and 1 ml 30 % H_2O_2 to the sample/vessel.

18. Attach the adapters for transferring the MDV to the digestion rotor. Install the MDV into the digestion rotor assembly.

19. Repeat the digestion program in step 9 above. No vacuum setup is required.

20. Vent the system for 10 minutes and allow to cool.

21. Remove the MDV and quantitatively transfer the acidic solution in the MDV to a tared 100 ml glass beaker. Rinse the MDV with 10 ml of 3M HNO_3 and add the rinsate to the glass beaker.

22. Add 10 ml of 3M HNO_3 to the glass beaker.

23. Weigh the combined weight of the solution and glass beaker. Calculate the ratio of the original resin weight to the final sample solution weight.

24. Remove and transfer an appropriate aliquant of the solution for radioassay to a tared container utilized for shipment. Determine the weight of the solution

and calculate the solution's equivalent resin weight by applying the ratio of the original resin weight to final sample solution weight.

25. As required by the vendor laboratory, add the appropriate chemical preservatives to the sample in the container.

26. Mark this container with the appropriate identification - this sample is to be analyzed for all nuclides except ^{14}C. *Note: this acidic fraction should also be analyzed for* ^{129}*I, if desired.* Include a notation on the sample submittal or chain of custody form that refers to the equivalent resin weight.

27. Prepare the container for shipment to the vendor laboratory according to standard operating procedures and proper DOT regulations.

EXPERIMENTAL TEST RESULTS
PROTOCOL 2

RADIONUCLIDE	% RECOVERY	# OF MEASUREMENTS
^{14}C*	100 ± 3	2
^{54}Mn	99 ± 3	6
^{59}Fe	99 ± 1	2
^{60}Co	98 ± 1	6
^{65}Zn	96 ± 3	6
^{99}Tc	98 ± 1	2
^{110m}Ag	91 ± 2	2
I*	57 ± 15	2
^{134}Cs	92 ± 4	6
^{137}Cs	97 ± 5	6

* ^{14}C and I recoveries refer to the alkaline fraction only. Recoveries for other nuclides listed on the table are for the acidic media.

C

ACTIVITY MEASUREMENTS IN SOLID LOW LEVEL WASTES FROM FRENCH PWRS

INTRODUCTION

LOW LEVEL WASTES PRODUCED BY EDF's PWR AND CONDITIONING METHODS

Electricité de France (EDF) is a state-owned company responsible for the generation and distribution of electricity in France. At the end of 1992 there were 53 PWR units in operation bringing the share of nuclear electricity near 75%.

EDF has applied from the beginning a rigorous policy of standardization in the design of power plants which facilitates licensing and commissioning. The concepts of liquid and solid waste treatment are almost the same inside each plant.

The generated radioactive low level wastes may be divided into the following categories:

- *maintenance wastes (called D.A.W.)* : rags, paper, vinyl sheets, gloves, bags, small metallic parts, etc. Very low activity wastes (< 200 mrem/hr on contact) are inserted in 50 gallon steel drums. Wastes with higher gamma doses are inserted in concrete containers before conditioning.
- *(small) metallic wastes* : are inserted in 50 gallon steel drums (wastes resulting from decommissioning are not addressed in this appendix).
- *filter cartridges* : the spent filter cartridges are withdrawn from the filter vessel and shipped by a lead shielded transport cask to the radwaste plant where they are deposited in a concrete container before conditioning. Cartridges with dose rates < 200 mrem/hr are dismantled and treated as D.A.W.
- *air filters* : are collected in cardboard boxes.
- *ion exchange resins* : are collected in storage tanks and stored submerged in water before conditioning.
- *evaporator concentrates* : are collected in storage tanks at 55°C before conditioning.
- *sludges* : are transferred to concrete containers in which they are conditioned.

- *oils* : are collected in flasks.

Spent resins (except resins from the steam generator blowdown system) are solidified inside concrete containers with a polymer (vinylester/styrene) mixture. The conditioning is performed with mobile treatment units.

Filter cartridges and high activity D.A.W. having dose rates > 200 mrem/hr are encapsulated with cement in concrete containers. On-site equipment is used.

Evaporator concentrates and sludges are cemented inside concrete containers with on-site equipment or with a mobile treatment unit.

In 1997-98 the low activity D.A.W., ion exchange resins (secondary reactor side), oils, evaporator concentrates and sludges will be incinerated by SOCODEI (Subsidiary of EDF) at a centralized plant located in the south of France. The ashes will be cemented. Metallic wastes will also be conditioned by fusion. The moltant products will be reused as shielding material for waste packages.

The one or two cubic meter concrete containers in which spent resins, filter cartridges and special D.A.W. are conditioned, serve for the transport of the waste and for its final disposal. All containers have a standardized, cylindrical form (height is 1.3 m, diameter is 1.1 or 1.4 m , wall thickness is between 10 and 16 cm) and are steel reinforced. Most of the containers have additional steel and lead shielding inside.

Before shipping to the repository the containers are capped with fresh concrete using on-site equipment or a mobile treatment unit.

FINAL DISPOSAL OF LOW LEVEL WASTES IN FRANCE

The "Agence Nationale pour les Déchets Radioactifs", called ANDRA, is responsible for the management of the waste disposal centers in France. ANDRA is a government agency, non-profit organization and is controlled by the french regulatory bodies. ANDRA funding comes from waste producers on an effective cost basis calculated according to volume and nature of delivered wastes.

In France, near-surface repositories are used for low level wastes. Waste producers are EDF, COGEMA (fuel reprocessor), CEA (research centers), and other companies (laboratories, hospitals, etc.). The first repository was built at LA MANCHE (near COGEMA at La Hague) and operated since 1969. It has a capacity of 500,000 cubic meters and is actually full. Site closure operations began in 1991.

Since January 1992, ANDRA opened a new repository in the southern Champagne region, called AUBE, with a capacity of 1 million cubic meters of wastes. The new facility, which is of a quite different design compared to the Manche Center, will receive the french low level wastes for the next 30-40 years (accounting a yearly total volume of 30,000 cubic meter of which about 50% is produced by EDF and 30% by COGEMA).

At the AUBE site, all waste packages (concrete containers, metallic boxes, steel drums, etc.) are placed in cubic disposal structures resembling concrete vaults. After loading, the void space is filled with concrete or gravel. The filled disposal structures are covered with a concrete cap and sealed with a polyurethane coating. Finally, a multiple layer cap (clay, bitumen, rain water runoff layers, etc.) is placed on the top of the structure.

A piping gallery network beneath the disposal structures collects any infiltrated water and is continuously monitored.

The 95 hectares disposal facility stands on a sand clay hill. Rain water pass through a single outlet, the Noues d'Amance river.

FRENCH RULES FOR FINAL DISPOSAL

Near surface storage centers in France may only accept "class A" wastes (the french waste classification is shown in the next section). EDF's process and maintenance wastes correspond to class A.

The final disposal of wastes in "raw" form (wastes which are not solidified in cement, bitumen, or polymer) is not permitted in France unless their activity content is very low. Wastes converted to dry form must also be solidified. This rule is particularly stringent for most waste producers. The reason for ANDRA's specification is to prevent escape of the raw waste in case of voids in the waste package.

France has one of the most stringent quality requirements for "stabilizing" solidified wastes or concrete packages. A "stabilized" material must possess "durable" (>300 years) characteristics for resisting irradiation, cracking, leaching and temperature effects. ANDRA has required characterization tests include leaching tests, tritium diffusion tests on concrete containers, thermal cycling tests, irradiation tests, fire tests and different small scale tests (compressive strength, shrinkage, gas permeability, etc.). Leaching, diffusion and irradiation tests are time consuming (> 1 year).

Quality requirements are less stringent for solidified wastes having an activity content below 1 curie/ton (Ci/t) (*37 GBq/t*). For these wastes no "stabilized" form is required.

ANDRA's disposal agreements are delivered individually per waste type and waste conditioning process. They are definitive.

WASTE CLASSIFICATION - DIFFERENCES BETWEEN FRANCE AND USA

The differences between 10CFR61.55 and the french regulations in waste classification are summarized in the tables below :

"CLASS A" WASTES

10CFR61.55	**FRANCE**
No waste form requirement	All wastes must be solidified in a matrix (cement, polymer, bitumen)
	"Stabilized" (300 yrs) waste form or package if total activity > 1 Ci/t or if specified isotopes exceed defined concentration levels. A "stabilized" form possess "durable" (>300 years) characteristics for resisting to irradiation, cracking, temperature effects and leaching.
No site design requirement	Site design requirements depending upon the activity level

Max. concentrations for short lived β and β-γ isotopes : ^{60}Co < 40 Ci/m^3 ^{137}Cs < 1 Ci/m^3 ^{90}Sr < 0.04 Ci/m^3 ^{3}H < 700 Ci/m^3	Max. concentrations for "stabilized" short lived β and β-γ isotopes : ^{60}Co $< 1{,}400$ Ci/t * ^{137}Cs < 8.9 Ci/t ^{90}Sr < 2.5 Ci/t ^{3}H < 5.4 Ci/t + limits for other isotopes The limits are lower for non "stabilized" forms.
Max. concentrations for long lived β and β-γ isotopes : ^{129}I < 0.008 Ci/m^3 ^{14}C < 0.8 Ci/m^3 ^{99}Tc < 0.3 Ci/m^3 ^{63}Ni < 3.5 Ci/m^3	Max. concentrations for "stabilized" long lived β and β-γ isotopes : ^{129}I < 0.001 Ci/t* ^{14}C < 5.4 Ci/t ^{99}Tc < 0.027 Ci/t ^{63}Ni < 320 Ci/t * Conversion ton in m^3 : densities are between 0.6 and 2.2 for EDF wastes (the mean value is 1.7 - waste package included) **Lower limits for non "stabilized" forms. Other max. concentrations for ^{59}Ni, ^{93}Zr, ^{93}Mo, ^{94}Nb, ^{107}Pd, ^{135}Cs, ^{151}Sm (stabilized and non-stabilized)**
Max. concentrations for α isotopes : ^{241}Pu < 0.35 Ci/t ^{241}Cm < 2 Ci/t Other TRU < 0.01 Ci/t	Max. concentrations for α isotopes : ^{241}Pu: no concentration limit is specified, but the total disposed activity on the repository is restricted ^{241}Cm : no limit (see above) Other TRU : no limit (see above) **Total α activity is limited to 0.1 Ci/t for "stabilized" wastes and to 0.005 Ci/t for non stabilized wastes. The total disposed activity of ^{239}Pu, ^{241}Am and ^{237}Np is restricted on the repository**

"CLASS B" WASTES

10CFR61.55	FRANCE
Waste or package in stabilized form (for 300 years) No max. concentration limits except for: ^{63}Ni (3.5 Ci/m^3) ^{90}Sr (150 Ci/m^3) ^{137}Cs (44 Ci/m^3)	The french "type B" wastes are solidified or encapsulated low level and medium level wastes containing > 0.1 Ci/t of alpha activity. Generated principally at the fuel reprocessing plant of La Hague Surface disposal is not permitted

"CLASS C" WASTES

10CFR61.55	FRANCE
Stabilized form as for class B with site design requirements. Max. concentration limits 10 x higher than class A waste (no limits for ^{3}H and ^{60}Co)	The french "type C" wastes are high level and alpha wastes solidified in glass blocks (originating from fuel reprocessing)

COMPARISON OF THE ALLOWABLE ISOTOPE CONCENTRATIONS

The waste classification is different between France and USA.

The french PWR low level wastes are closest to the 10CFR61 "Class C" wastes. These wastes are in a stabilized form and are protected in the final repository by special structures (site design requirements). However, the maximum allowable concentrations for long-lived isotopes in these wastes are higher than the french ones (principally in the case of ^{129}I and ^{99}Tc). The differences may be explained by the way in which protective measures are taken in France and in USA for reducing exposures to an inadvertent intruder.

The tables hereafter show how exposure conditions are developed in both countries.

ALLOWABLE DOSE LIMITS

RADIATION PROTECTION	USA	FRANCE (Site of AUBE)
EXPOSURE LIMITS (Objectives)	25 mrem/yr. whole body 500 mrem/yr. for an inadvertent intruder	100 mrem/yr. whole body for normal situations * 100 mrem/yr. for an inadvertent intruder This presumes an accidental situation on the repository with barrier losses

* Tendency is actually 10 mrem/yr. in France (0.1 mSv/yr.)

USED EXPOSURE SCENARIOS AND PATH WAYS

Isotope	USA	FRANCE (accidental condition with barrier losses)	FRANCE MAX. DOSES AT
^{60}Co	Agriculture - Direct gamma	Repository cap collapse	Before closure
^{137}Cs	Agriculture - Food uptake	Residence at 330 yr.	330 yr.
^{3}H	Agriculture - Food uptake	Cap collapse	Before closure
^{90}Sr	Agriculture - Food uptake	Water well and food uptake	359 yr.
^{129}I	Agricult. - Food/inhalation	Water well and food uptake	364 yr.
^{14}C	Agriculture - Food uptake	Water well and food uptake	1,300 yr.
^{63}Ni	Agriculture - Food uptake	Water well and food uptake	340 yr.
TRU	Construction - Inhalation	Road construction (Inhalation)	After closure

The "accidental conditions" in France presume barrier losses :

- complete repository cap loss with maximum water infiltration,
- all wastes and packages are completely destructed and in form of concrete stuff,
- maximum leaching of the wastes (the repository drain system has disappeared)

Pathway models are not "normalized" in France or in USA. The repository Designer proposes "reasonable" ways which humans could come in contact with the waste or which the radionuclides in the disposal facility could be transported to humans.

EDF'S POLICY IN WASTE CHARACTERIZATION

"Waste Characterization" means all techniques which lead to the determination of the activity content in solid wastes.

CASE OF THE "EASY-TO-MEASURE" β - γ EMITTERS

In order to reduce radiation exposures EDF's policy is to avoid sampling on solid wastes as much as possible . For most of the wastes (see table below) the activity content is evaluated by "dose-to-curie" methods.

Dose-to-curie methods are calculational techniques. The dose rate is measured at a specified distance and geometrical position from the waste package (concrete container or steel drum). The gamma doses are combined with the data on the relative abundance of the gamma-emitting isotopes (determined from a gamma spectrum). EDF uses a "theoretical" gamma spectrum called "standard spectrum". The standard spectrum is established on the basis of the radioactive sources. For example, the standard spectrum of spent filters takes into account the activity of the major gamma-isotopes in the filtrated water. Conservative hypotheses are used for the retention of radionuclides on filters and on resins. The calculated γ spectrum is assumed "stable" as a function of time and applies to all reactor sites. Adaptation of the spectrum is only made in cases where important activity variations in the reactor water are observed.

TYPE OF WASTE	Sampling	Evaluation by gamma spectrometry	Evaluation by γ dose measurement on the container
Spent resins (primary side)	No	Yes, on the metering tank for resins of the mobile conditioning unit	Yes, on the metering tank of the mobile unit for control of the "consistency" of the gamma spectrum
Spent filter cartridges (> 200 mrem/hr on contact)	No	No	Yes, on the surface of the concrete container. Dose-to-curie method using a "standard" γ spectrum
D.A.W.	No	No	Yes, on the surface of the 50 gallon steel drum. Dose-to-curie method using a "standard" γ spectrum
Spent resins (secondary side)	No	No	Same as for D.A.W.
Evaporator concentrates (Radwaste facility)	Yes	Yes	No

CASE OF THE "DIFFICULT-TO-MEASURE" SHORT-LIVED β EMITTERS (PERIOD < 31 YR.)

The concerned isotopes are :

ISOTOPE	HALF LIFE	RADIATION CHARACTERISTICS
^{90}Sr	28 yr.	Pure β emitter
^{3}H	12 yr.	Pure β emitter (weak β)

The tritium activity is evaluated on the basis of the titriated water source. No sampling and tritium measurement are performed by EDF on solid wastes :

TYPE OF WASTE	TRITIUM EVALUATION METHOD	RESULT
Spent resins (primary side)	Max. tritium concentration in the residual water of the dewatered resins (interstitial and occluded water)	Negligible concentrations in spent resins
Spent filter cartridges (> 200 mrem/hr on contact)	Max. tritium concentration in the residual water of the dewatered spent filters (around 5 liters)	Negligible
D.A.W	No residual water (dry wastes)	Negligible
Evaporator concentrates (Radwaste facility)	Mean volume of tritiated water treated on the evaporator	< 1 GBq/m^3 (0.027 Ci/m^3). Evaporator concentrates have normally < 1 Ci/m^3 of total β-γ activity

For ^{90}Sr the evaluation techniques are much more complicated. For a long time, scaling factors have been used with total α and ^{60}Co activity measured in reactor water. This scaling is very unreliable while it depends on reactor transients, fuel conditions and reactor materials. Scaling of ^{90}Sr with ^{137}Cs has also proved to be very unreliable (see subsection on scaling factors).

EDF is actually investigating other evaluation techniques for ^{90}Sr. These techniques are based upon :

- scaling measurements with ^{140}Ba. This isotope is a short lived γ emitter (13 days) but is chemically close to ^{90}Sr and is easy to measure.
- measurements of ^{90}Sr/^{140}Ba in reactor water in order to verify the influence of fuel cycles, reactor transients (e.g., spiking effects during crud burst) and reactor operation (normal operation, cold shutdown, etc.). Comparison of the radiochemical behavior of ^{90}Sr and ^{140}Ba in the reactor water, on the primary filter and the primary demineralizer of the CVCS (Chemical Volume and

Control System). This is performed for normal reactor operations and for cold shutdown transients which generate crud burst.

- retention characteristics of ^{90}Sr/^{140}Ba on filters and demineralizers. Measurement campaigns are performed on a small-scale filter and demineralizer, branched on a reactor water sampling line, which should simulate the CVCS (the PWR's Reactor Water Purification System).

In addition, a calculational technique is in development based upon the french PROFIP code (from the C.E.A.) The (improved) code calculates the reactor water activity of various isotopes originating from the fuel leakage, tramp fuel or from contaminated reactor materials.

The objective is to obtain a computerized method for determining the ^{90}Sr content in spent filters and in resins on the basis of chemically-close γ isotopes measured in reactor water and to predict the influence of :

- crud burst transients (cold shutdown),
- "memories"effects (from ^{90}Sr disposals during former fuel cycles),
- fuel conditions (dammages, type of fuel),
- reactor materials (primary circuit).

This evaluation technique will be ready in 1998.

CASE OF THE ALPHA ACTIVITY

Although the total alpha activity shipped by EDF to the repository is low compared to other waste producers (COGEMA, etc.), it is necessary to guarantee its "negligible" content with evaluation techniques.

As for ^{90}Sr, EDF is developing a calculational technique based on an improved PROFIP code. No scaling factors are used in this case. The technique compares with the RADSOURCE code in which reference is taken from the spectrum of the iodines, measured in the reactor water, and from other significant isotopes (^{137}Cs, ^{134}Cs, etc.). The PROFIP technique will be qualified by alpha measurements on power plants with the use of the above described "CVCS simulator" branched on the reactor water sampling line.

As for ^{90}Sr, the computerized evaluation technique will be ready in 1998.

CASE OF THE "DIFFICULT-TO-MEASURE" LONG-LIVED ISOTOPES

The for ANDRA concerned long-lived isotopes are :

ISOTOPE	**HALF LIFE**	**RADIATION CHARACTERISTICS**	**PRINCIPAL SOURCE**
^{129}I	1.6 x 10^7 yr	Very weak β emitter	Fuel /tramp uranium
^{14}C	5,730 yr	Weak β emitter	Activation of ^{17}O in RW
^{99}Tc	2.1 x 10^5 yr	Weak β emitter	Tramp uranium
^{94}Nb	2.1 x 10^4 yr	β - γ emitter	Tramp uranium
^{63}Ni	123 yr	Very weak β emitter	Activation of Ni
^{59}Ni	8 x 10^4 yr	EC with very low X-rays	Activation of Ni
^{93}Mo	3,500 yr	EC with very low X-rays	Activation of Mo
^{93}Zr	9.5 x 10^5 yr	Very weak β emitter	Fuel+ activation of Zr
^{151}Sm	90 yr	Very weak β and γ emitter	Fuel/tramp uranium
^{135}Cs	2.9 x 10^6 yr	Very weak β emitter	Fuel
^{107}Pd	7 x 10^6 yr	Very weak β emitter	Fuel/tramp uranium
^{239}Pu	2.4 x 10^4 yr	α emitter	Tramp uranium
^{247}Am	470 yr	α emitter + ^{237}Np X-rays	Tramp uranium
^{238}U	4.5 x 10^9 yr	α emitter	Reprocessing plant

EDF's policy is to reduce the number of long-lived isotopes for which the presence in solid wastes has to be verified. Following strategies are used :

a) Step 1 :

It is shown which of the above listed long-lived isotopes contribute significantly to public radiation exposure on the final repository after 300 years and which do not. A rough evaluation has been made by EDF on the exposure doses in mrem/year (inadvertent intruder scenario) which each long-lived isotope may bring. The applied evaluation method is very conservative in regard to used scaling factors and other hypotheses. An example is given in the next chapter. If dose rates are below 0.1

mrem/yr for a long-lived isotope, EDF considers that it is not worthwhile to continue investigations on the radionuclide.

b) Step 2 :

For the exposure dose "significant" long-lived isotopes, scaling factors are established which correlate the long-lived radionuclides with ^{60}Co or ^{137}Cs.

The scaling factors are evaluated by measurement campaigns on solid wastes (principally CVCS resins and filters) originating from different reactor sites. After obtaining a reasonable quantity of scaling factor values these campaigns will cease.

In parallel, PROFIP calculations are made which verify all influences on scaling factors (reactor operation, materials, fuel cycle, etc.). As for ^{90}Sr, the calculations are qualified on small-scale CVCS simulators.

CLASSIFICATION OF LONG-LIVED ISOTOPES IN FUNCTION OF THEIR IMPORTANCE

As shown before, EDF's policy is to classify the long-lived isotopes in function of their importance in public radiation exposure (> 300 years - inadvertent intruder scenario).

The dose calculations are made by using the following techniques :

a) EDF's total ^{60}Co and ^{137}Cs content in stored wastes is calculated. It is asssumed that 50 power stations have shipped wastes during 35 years to a repository site (site of AUBE). Cobalt and cesium activities are calculated from the standard gamma spectrums of the wastes (see above) or from direct measurements. In the case of filters and sludges, the activities of the normally soluble ^{137}Cs istope have been intentionally over-estimated. An example is shown below:

WASTE TYPE	**TOTAL ^{60}Co ACTIVITY (35 YEARS)**	**TOTAL ^{137}Cs ACTIVITY (35 YEARS)**
All spent filters	64,750 Ci	6,475 Ci
All spent resins (except secondary side)	266,000 Ci	154,000 Ci
D.A.W. (< 200 mrem/hr)	1,750 Ci	175 Ci
D.A.W. (> 200 mrem/hr)	26,250 Ci	625 Ci
Evaporator concentrates	875 Ci	875 Ci
Sludges	175 Ci	175 Ci
TOTAL VALUE	359,800 Ci (13,300 TBq)	164,300 Ci (6,100 TBq)

b) the activities of the long-lived (^{129}I, ^{14}C, ^{99}Tc, ^{63}Ni, ^{239}Pu, ^{241}Am) isotopes are calculated by correlation with the above ^{60}Co or ^{137}Cs content. In this case conservative USA scaling factors, reported by EPRI reports, are used but with "conservative" envelopes. For the long-lived isotopes for which no scaling factors exist (^{94}Nb, ^{107}Pd, ^{93}Mo, etc.) the activities are deducted, either by data from reactor water activities calculated by the french code PROFIP (fission products) and PACTOL (activation products), or by taking their relative abundance in the fuel rods. Conservative hypotheses are made on the retention of these istopes in the filters and in the resins of the demineralizers.

c) the conversion of activity from radiation exposure is performed with the same calculational models which are used in the Safety Analysis Report (S.A.R.) of the AUBE site using the same pathways, leaching and migration characteristics, etc. In fact, total long-lived isotope activities (all waste producers) and associated doses have been reported by ANDRA in the S.A.R., so a simple extrapolation was possible.

Results are given below :

ISOTOPE	**TOTAL ACTIVITY (all wastes) 35 YEARS OF PRODUCTION**	**ENVIRONMENTAL MAX. DOSE ON REPOSITORY AND SCENARIO (1 mrem/yr = 0.01 mSv/yr)**	**IMORTANT ISOTOPE TO EDF ?**
^{129}I	5 Ci (0.2 TBq)	210 mrem/yr (well at 364 yr)	YES
^{14}C	3,287 Ci (122 TBq)	20 mrem/yr (well at 1,300 yr)	YES
^{99}Tc	47 Ci (2 TBq)	4 mrem/yr (well at 365 yr)	YES
^{94}Nb	35 Ci (1 TBq)	3 mrem/yr (residence at 330 yr)	YES
^{63}Ni	197,226 Ci (7,297 TBq)	2 mrem/yr (well at 340 yr)	YES
^{239}Pu	24 Ci (0.9 TBq)	0.4 mrem/yr (road at 330 yr)	LESS
^{241}Am	15 Ci (0.6 TBq)	0.2 mrem/yr (road at 330 yr)	LESS
Other	< 1 Ci (<0.04 TBq)	10E-2 to 10E-5 mrem/yr	NO

Considering the above listed exposure rates, EDF's concluded to make an evaluation effort primarily for **^{129}I and** ^{14}C, and secondarily for **^{99}Tc, ^{94}Nb, and ^{63}Ni.**

PROCEDURES USED FOR CORRELATION FACTOR MEASUREMENT

As stated above, the for EDF "important" isotopes are the shortlived ^{90}Sr (28 yr) and the long-lived ^{129}I, ^{14}C, ^{99}Tc, ^{94}Nb and ^{63}Ni. In addition, the short-lived "difficult-to-measure" ^{55}Fe isotope (2.6 yr) has been measured.

The correlation factors are evaluated by direct measurement on sampled, solid wastes originating from different PWR power plants. The wastes are filter cartridges from purification systems (CVCS, fuel pool cleaning, and radwaste system) and ion exchange resins from the intermediate storage tanks (resins originating from different cleaning systems - partially mixed).

CRITERIA ON MINIMAL DETECTABLE CONCENTRATIONS

The selected "minimal detectable concentration" values (MDC) for each of the long-lived istopes are related to the level of radiation exposure each radionuclide contributes to the repository site after 300 years (site of AUBE).

This criterium has been settled up by EDF, ANDRA and the other french waste producers . The objective is to bring the total dose (all isotopes) below 10 mrem/yr (0.1

mSv/yr) in the intruder scenario. In this case it has been conservatively assumed that each of the long-lived isotopes do not exceed 1/10th of the total dose (1 mrem/yr).

The conversion of the 1 mrem/yr limit into activity concentrations is again done by extrapolation of the S.A.R. reported doses and total activity values (see point c of the chapter above).

The obtained MDC values are shown hereafter and are compared to 10CFR61 :

ISOTOPE	CORRELATION	MDC (OBJECTIVE) in nCi/g	COMPARISON MDC 10CFR61 in nCi/g
^{129}I	^{137}Cs	0.0002	0.8
^{99}Tc	^{137}Cs	0.005	90
^{14}C	^{60}Co	0.1	80
^{63}Ni	^{60}Co	70	35
^{59}Ni	^{60}Co	14	2200
^{94}Nb	^{60}Co	0.004	2

USED MEASUREMENT AND SAMPLING TECHNIQUES

As shown inthe above table, the required MDC values in France, except for ^{63}Ni, are much lower of those of the 10CFR61, particularly for ^{129}I, ^{99}Tc, and ^{14}C-14. These very low MDCs lead to the use of sophisticated (and costly) measurement methods.

The development of sampling and measurement techniques and the measurements themselves are performed by the C.E.A. (Commissariat d'Energie Atomique). C.E.A. is an EDF Contractor which operates mostly in the field of specialized laboratory analyses. The CEA/EDF program is also partially financed by the CCE (European Atomic Energy Community) in the frame of a more international program.

The table below summarizes the C.E.A. applied measurement techniques and the achieved MDC values.

ISOTOPE	**MDC OBJECTIVE nCi/g**	**MDC OBTAINED nCi/g**	**Summary of CEA measurement technique**
^{129}I	0.0002	0.0001 10E-6	Pyrohydrolysis at 900°C- acid conversion - ICPMS* (for filters) CCl_4/KI extraction + MSA * *(other wastes)
^{99}Tc	0.005	0.01	Column elution - extraction - ICPMS or Pyrohydrolysis 900°C - acid conversion - ICPMS
^{14}C	0.1	0.01	Pyrohydrolysis - CO_2 conversion-carbosorb - liquid scintillation
^{63}Ni	70	0.01	Dimethylglyoxime method - liq. scintillation
^{59}Ni	14	0.01	Dimethylglyoxime method - liq. scintillation
^{94}Nb	0.004	0.01	Fusion - hydroxylamine extraction - Ge/Li count. or Dissolution - TBP extract. - Ge/Li counting.

* ICPMS = Inductively Coupled Plasma Atomic Emission Spectrometry
* * M.S.A. = Mass Acceleration Spectrometry

In most cases "detection capabilities" are tested on samples by using a radioactive tracer.

The filter cartridges are sampled by cutting and extracting a piece of the filter membrane on the top, in the middle, and on the bottom (3 samples per cartridge - 1 to 3 gram per sample). The ^{60}Co activities (used for correlation) vary from top to bottom. However, it could be seen that these differences do not affect the scaling factors.

The resins are sampled in the intermediate storage tanks. About 10 cm^3 per tank is taken and 0.3 to 1 gram is analyzed.

The applied dissolution technique is a digestion with sulfuric acid and hydrogen peroxyde (in a microwaver digester). The volatile radionuclides (^{129}I, ^{99}Tc) are recovered and off-gases (^{14}C in CO_2) are trapped in NaOH. The chemical yields are detected by using a radioactive tracer.

SCALING FACTOR AVERAGING

EDF applies a linear regression mode on the related logarithms of the DTM (difficult-to-measure) isotope concentrations and the logarithms of the ETM (easy-to-measure) isotope concentrations :

log (Conc. DTM) = B. log(Conc. ETM) + log.K or

(Conc.DTM) = K.(Conc.ETM)exp b

K = intercept   B = slope

The "correlation degree" is tested with the Fisher-Snedecor method resulting in a R^2 factor. Good correlation is obtained if R^2 is between 0.7 and 0.9. For facility reasons, EDF assumes an 0% measurement error in the linear logarithmic regression mode. If linear logarithmic regression was not possible, "robust" regression is applied (arithmetic mean values).

In the case of linear logarithmic regression, the scaling factor is called "**Kreg**". In the case of robust regression it is called "**Kmean**".

Kreg and Kmean factors may be different for a same sample. For example the ^{59}Ni/^{60}Co scaling factors measured on 11 different filter cartridges (3 samples per cartridge) give a Kreg of 5.3×10^{-4} ($R^2 = 0.85$ = good correlation) and a Kmean of 7.0×10^{-4}.

EDF'S SCALING FACTOR RESULTS - COMPARISON WITH USA

The following tables give a summary of the EDF's scaling factor results for spent filter cartridges and spent resins. These results were obtained during the measurement campaigns of 1991-96. The french scaling factors are discussed and compared to the USA averages reported in the Characterization Guidelines.

SCALING FACTORS FOR FILTERS

The total number of analyzed samples is 298.

ISOTOPE	**CORREL.**	**TYPE OF MEAN VALUE**	**R² in case of Kreg**	**EDF Scaling Factor**	**Sum of measur. errors (mean)**	**Comparison with obtained USA S.F (all filters) * * * ( ) = Root Mean Square Dispersion**
^{129}I *	^{137}Cs	Kmean		1.0 E-6	30%	2.1 E-4 (3.82)
^{99}Tc * *	^{137}Cs	Kreg	0.64	4.2 E-4	55%	3.5 E-3 (12.86)
^{14}C	^{60}Co	Kreg	0.61	1.1 E-2	20%	6.5 E-3 (8.49)
^{90}Sr	^{60}Co	Kmean		2.6 E-2	25%	3.6 E-4 (7.47)
^{55}Fe	^{60}Co	Kreg	0.90	2.1	20%	2.5 E-0 (3.95)
^{63}Ni	^{60}Co	Kreg	0.81	2.3 E-1	15%	3.9 E-1 (3.00)
^{59}Ni	^{60}Co	Kreg	0.85	5.3 E-4	50%	-
^{94}Nb	^{60}Co	Kreg	0.70	1.3 E-4	40%	1.6 E-4 (3.30)

* Number of valid measurement points is only 23
* * Number of valid measurement points is only 5
* * * Mean values are log. geometric means (are reported in the Guidelines).

EDF concludes that the activation products ^{63}Ni, ^{59}Ni, ^{55}Fe and ^{94}Nb may be used as "generic scaling" factors (S.F. which are applicable for all french PWR's). The $^{63/59}$Ni and ^{55}Fe input in the reactor water is the same as for the nickel-cobalt-iron input originating from corrosion or erosion of reactor materials (stainless steel and inconnel). The chemical behaviors of the reactor water generated insoluble Ni/Fe/Co isotopes are close and scaling with cobalt seems to be reliable. Niobium-94 is a corrosion/erosion product of inconnel and acts in the same way.

Iodine-129 is considered as a more difficult candidate for "generic" scaling factor. The first difficulty is to obtain a sufficient number of measurement results. The number of "valid" measurement points is limited. Following reasons explain this restriction :

- the M.A.S. (Mass Acceleration Spectrometry) is proved to be the only "reliable" measurement technique for determining low ^{129}I concentrations. This technique is only applied in specialized laboratories. EDF was obliged to send samples outside France in this case. This procedure was very expensive and has therefore been limited.

- Many filter cartridges did not show a "measurable" ^{137}Cs content (^{137}Cs is mostly in soluble form).

The measurement of ^{129}I scaling factors will be continued in the next years for obtaining more measurement points.

The second problem with ^{129}I concerns the scaling itself. We are not quite sure if iodine and cesium adsorption on filtrable particles or on the filter membrane is occuring all the time in the same way.

Carbon-14 is also considered a bad candidate. It is expected that the correlation between ^{14}C and ^{60}Co is subjected to the reactor materials, the reactor transients (bringing activity spiking effects) and the variable chemical conditions of the reactor water (temperature, pH, etc.). Carbon-14 is principally generated by activation of the reactor water dissolved oxygen (neutron/alpha reaction of ^{17}0 which has an 0.038 % abundance in natural oxygen). An EDF subcontracted theoretical study shows that the chemical forms of ^{14}C may be variable in function of the reactor water temperature. At 300°C and at pH = 7.1, ^{14}C exists in the form of gaseous CH_4 and soluble methanol ($CH_3 0H$). At cold shutdown and after aeration of the primary circuit (pH = 9.6), ^{14}C is in the form of methanol, bicarbonates (HCO_3-) and carbonates (CO_3-).

The poorest scaling is between ^{90}Sr and ^{60}Co (high variability). This is due to the strong differences in chemical behavior between ^{90}Sr and ^{60}Co especially during reactor transients. As shown in the chapter before, scaling with ^{140}Ba and the use of a computerized calculational method may be a help.

For ^{99}Tc the measurement of scaling factors will be continued by EDF for obtaining more measurement points.

It is observed that the french scaling factors for ^{129}I, ^{99}Tc and ^{90}Sr are much lower than the USA factors :

SCALING FACTOR	Number of measurement points FRANCE	Number of measurement points USA	DIFFERENCES WITH USA
^{129}I/^{137}Cs	23	22	100 x lower
^{99}Tc/^{137}Cs	5	53	10 x lower
^{90}Sr/^{60}Co	40	256	100 x lower

The ^{129}I difference may be explained by the different measurement techniques which have been used (M.A.S. is used in all cases at E.D.F.). The above described influences of ^{129}I and ^{137}Cs on filters are also an explanation. For ^{99}Tc, the number of EDF measurement points does not allow for a comparison with USA results. The ^{90}Sr differences may come from the reactor materials used but, in both cases, scaling is assumed to be very difficult between ^{60}Co and ^{90}Sr.

SCALING FACTORS FOR RESINS

The total number of analyzed samples is 126.

ISOTOPE	CORREL.	TYPE OF MEAN VALUE	R² in case of Kreg	EDF Scaling Factor	Sum of measur. errors (mean)	Comparison with obtained USA S.F (all filters) * * * ( ) = Root Mean Square Dispersion
^{129}I *	^{137}Cs	Kmean		1.0 E-6	30%	1.8 E-4 (14.55)
^{99}Tc * *	^{137}Cs					5.3 E-4 (9.40)
^{14}C	^{60}Co	Kmean		1.8 E-2	20%	1.9 E-2 (7.28)
^{90}Sr	^{60}Co	Kreg	0.63	2.3 E-3	35%	2.0 E-3 (7.86)
^{55}Fe	^{60}Co	Kreg	0.84	1.4 E-1	30%	8.1 E-1 (3.93)
^{63}Ni	^{60}Co	Kreg	0.84	1.4	20%	6.3 E-1 (3.09)
^{59}Ni	^{60}Co	Kreg	0.93	1.1 E-3	35%	
^{94}Nb	^{60}Co	Kreg	0.70	1.2 E-4	50%	3.2 E-4 (10.62)

* Number of valid measurement points is only 6

* * Number of valid measurement points is not sufficient

* * * Mean values are log. geometric means (are reported in the Guidelines).

As for the filters and for the same reasons, the ^{63}Ni, ^{59}Ni, ^{55}Fe and ^{94}Nb isotopes are plausible candidates for "generic" scaling factors.

A statement on ^{129}I is not possible. To few measurement results exist. Measurement of scaling factors will continue by EDF on ^{129}I (also for ^{99}Tc).

The ^{14}C isotope is a difficult candidate for the same reasons as outlined for the filters.

It is observed that the french scaling factor for ^{129}I is about 100 times lower compared to the USA. As for the resins the difference may be explained by the different measurement techniques and the variations in behavior. The number of measurement results is also to different to permit a comparison. EDF brings 6 results and USA 21 results.

CONCLUSION

Differences exist between France and USA in waste classification and in allowable isotope concentrations for long-lived, "difficult-to-measure" radionuclides. The differences in concentration limits may be explained by the isotope pathways which have been considered in the scenarios for evaluating the environmental radiation exposures risks.

For preventing over-estimation of public exposure risks, EDF recommends a calculation of their radiological impact on the repository to obtain a picture of the impact isotopes have on the evaluation.

This exercise has shown EDF that only ^{129}I, ^{14}C, ^{99}Tc, ^{63}Ni and ^{94}Nb are to be considered as "significant" in radiological impact on the french AUBE disposal site (after 300 years). These isotopes may bring environmental doses up to 1 mrem/yr (0.01 mSv/yr) in "inadvertent intruder" scenarios. The greatest doses result from the presence of ^{129}I and ^{14}C. Alpha and ^{90}Sr are less important.

EDF's policy is to prevent the necessity of direct measurements on solid waste samples to reduce radiation exposures to the plant operators. Except for evaporator concentrates, gamma isotopes are measured without any waste sampling. This is done by direct gamma spectrometry on waste containers or by "dose-to-curie" methods.

EDF's evaluation of "difficult-to-measure" isotope concentrations is based upon :

- the use of "generic" scaling factors which correlate "difficult-to-measure" istopes with ^{60}Co, ^{137}Cs or other gamma emitters (for cases where scaling is possible),
- and, in parallel, on calculational techniques in which the french codes PROFIP (fission products) and PACTOL (activation products) are applied.

These techniques will be later computerized (objective is 1998). In this program, the PROFIP and PACTOL codes are submitted to qualification tests by using small filter/demineralizer equipment, branched on the reactor water sampling line, which simulate the CVCS (PWR's Reactor Water Purication System).

Since 1991, measurement campaigns on solid wastes have started on french power plants for determining scaling factors with ^{60}Co and ^{137}Cs. The sampling and measurement work is subcontracted to the CEA (Commissarait d'Energie Atomique). Till 1996, about 298 filter samples and 126 resin samples have been analyzed.

It seems that ^{63}Ni, ^{59}Ni, ^{55}Fe and ^{94}Nb may be used as "generic" scaling factor for ^{60}Co (scaling factor which applies on all PWR's).

Carbon-14 and ^{90}Sr have proven to be more difficult candidates for generic scaling. Calculational techniques must be used in parallel. Strontium-90 may be scaled with ^{140}Ba (gamma emitter).

D

RADIOCHEMICAL NUCLIDE ENHANCEMENT TECHNIQUES

This Appendix material is taken from EPRI NP-5497, Advanced Radioactive Waste Assay Methods, November 1987

SUMMARY OF NUCLIDE ENHANCEMENT TECHNIQUES

Two problems often faced by plant personnel in analyzing radwaste samples for the gamma-ray emitting activities prior to shipment are:

1. High specific activities in waste that prevent taking samples for gama-ray analyses that permit an acceptable level of assurance of representativeness for the bulk waste, and
2. High relative activities of other gamma-emitting nuclides that mask the presence of ^{144}Ce that would be the more accurate scaling nuclide for TRU activities in the waste.

The first of these is common for spent primary bead resins where the activity from only a few beads may saturate the analytical counting capability of the plant laboratory spectrometer. If there were a simple chemical extraction technique that could reproducably remove the activities into a solution, an aliquant could be taken that could be more easily analyzed by gama-ray spectrometry. The method should be one that requires easily handled chemicals and simple apparatus and should be capable of being done quickly.

The second is common to all wastes where the ^{60}Co and ^{137}Cs activities usually dominate the gamma-ray spectrum and mask the region at 133 KeV where the stronger ^{144}Ce gama ray would be observed. If another aliquot, taken from the extraction discussed above could have the ^{144}Ce quantitatively, reliably and easily removed, an analysis of the sample for the cerium activity would permit the use of the more accepted and reliable scaling factors for TRU activities.

A method has been developed and tested to do both the extraction of activities from a sample of most radwaste materials found in plants and the separation of the ^{144}Ce activity.

CHEMICAL EXTRACTION OF ACTIVITY FROM WASTES

The technique often used to prepare the waste samples for chemical analyses is a combination of acid dissolution and alkaline fusion. The procedure, designed to dissolve even the most insoluble transuranic elements, is beyond the desires of most plant chemical laboratories for on-site analyses. Hence, the ,search was for a simpler method that would extract the gama-ray emitting activities, including ^{144}Ce, from the waste. The technique that gave the more satisfactory and reproducible results is a leach-dissolution of the material in concentrated nitric acid. Samples of filter cartridges resins and evaporator bottoms were leached and the leachate gamma-ray-emitting nuclide concentrations compared with those of the original concentrations. The extraction efficient was above 90% for the samples tested. Dissolution was for a period of one hour while the beaker was agitated in a sonic bath. Detailed procedures, included with the procedure for cerium enhancement, are given below.

ENHANCEMENT OF CERIUM ACTIVITIES

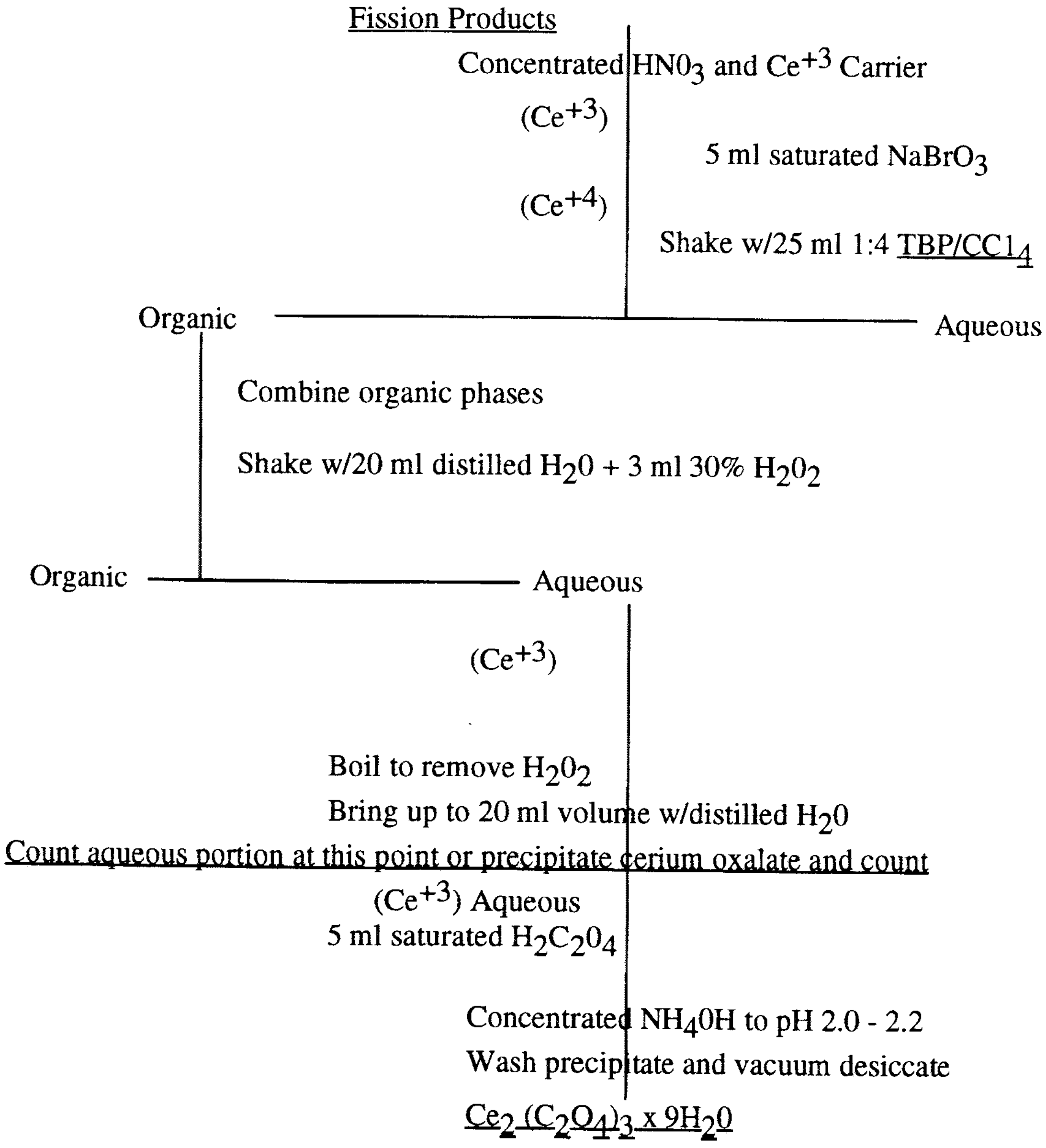
The method developed to enhance cerium activities in waste uses a sample of the nitric-acid leached solution together with an oxidation-reduction cycle and liquid-liquid extraction step. Sodium bromate oxidized the cerium to the +4 valence state where it is extracted into the organic phase of 20% tri-butyl-n-phosphate carbon tetrachloride mixture. It can be reduced with hydrogen peroxide and extracted into a water phase from which it can be precipitated as the oxalate for gamma-ray analysis. Appendix B lists detailed procedures for the radiochemical technique.

Table D-1 lists five typical samples analyzed by the method, the activity found in the sample, and the chemical yield determined from the recovery of the ^{139}Ce activity used as a tracer. Except for the single sludge sample, the yields are good. Cartridge filter samples gave the most difficulty in obtaining yields above 80% but showed consistent values. The method could be easily used by plant radiochemical staff.

RADIOCHEMICAL PROCEDURES FOR CERIUM ACTIVITY ENHANCEMENT

The radiochemical separation of cerium is based on the oxidation-reduction and liquid-liquid extraction cycle using a 20% tri-butyl phosphate-carbon tetrachloride mixture.

FLOW DIAGRAM



Fission Products

Concentrated $HN0_3$ and Ce^{+3} Carrier

(Ce^{+3})

5 ml saturated $NaBrO_3$

(Ce^{+4})

Shake w/25 ml 1:4 TBP/$CC1_4$

Organic

Aqueous

Combine organic phases

Shake w/20 ml distilled H_20 + 3 ml 30% H_20_2

Organic

Aqueous

(Ce^{+3})

Boil to remove H_20_2

Bring up to 20 ml volume w/distilled H_20

Count aqueous portion at this point or precipitate cerium oxalate and count

(Ce^{+3}) Aqueous

5 ml saturated $H_2C_20_4$

Concentrated NH_40H to pH 2.0 - 2.2

Wash precipitate and vacuum desiccate

$Ce_2 (C_2O_4)_3$ x $9H_20$

Table D-1
Recovery Efficiency of ^{139}Ce Tracer in Elution and Cerium Enhancement Tests

Sample Number	^{139}Ce Tracer Recovery	Activity (μCi/gram)
Cartridge Filter	74%	0.402
Filter Solids	34%	0.070
Evaporator Bottoms	87%	0.0013
Bead Resin	99%	14.6
Bead Resin	84%	6.51

Necessary Apparatus

- 250 ml separatory funnels
- 1 - 500 ml separatory funnel
- 250 ml beakers
- 4 - fritted glass crucibles for standardization vacuum desiccator sonic bath
- magnetic stirring plate
- magnetic stirring bars

Reagents

- $HN0_3$: concentrated
- NH_4OH: concentrated
- H_20_2 : 30%
- CCl_4: reagent grade
- HC1: 6N
- Na_2CO_3 :1M
- Tri-n-butyl phosphate, 1:4 mixture TBP in CCl_4 (See Note 1)
- $H_2C_20_4$: Saturated
- Methyl Alcohol (Methanol)
- $NaBrO_3$: saturated

- Ce^{+3} carrier: 8.1 mg/ml: Dissolve 25 grams of $Ce(NO_3)_3$ x $6H_20$ in 1 liter of distilled water.

Note 1: Pretreatment of tri-n-butyl Phosphate. TBP must be pretreated with 1M Na_2CO_3 in order to remove any phosphoric acid that may be present in the TBP.

Place 250 ml TBP in a 500 ml separatory funnel. Add 100 ml of 1 M Na_2CO_3 and shake for`10 minutes. Let solution settle until upper TBP phase has separated from the lower aqueous phase. (The TBP will appear cloudy). After draining off aqueous phase, place TBP in a glass bottle, add 150 ml of CCl_4 and mix. Let 1:4 mixture settle until uniform and clear.

Radiochemical Procedure for Cerium

1. Weigh samples (maximum size of 5 grams) to be analyzed and transfer to a 250 ml beaker with concentrated nitric acid, bring to 25 ml volume in $HN0_3$.
2. Add 1 ml of Ce^{+3} carrier, followed by 5 ml of saturated $NaBrO_3$ to oxidize the Ce^{+3} to Ce^{+4}.
3. Place beakers in sonic bath (cover with a watch glass) for 1 hour, then transfer to stirring plate and let stir overnight to ensure sufficient time for the acid leach.
4. Transfer the sample to a 250 ml separatory funnel and shake for 10 minutes with 25 ml of a 1:4 $TBP:CC1_4$ mixture. The Ce^{+4} goes into the organic phase. Separate and shake the aqueous phase with a second 25 ml portion of $TBP:CC1_4$. This may need to be done a third and fourth time if any yellow or orange color is left in the aqueous phase.
5. Combine the organic phases in a beaker. Place colorless aqueous phase in acid radwaste.
6. Pour organic phases back into a clean separatory funnel and add 20 ml distilled water and 3 ml of 30% H_20_2 to each. (This will reduce the Ce^{+4} to Ce^{+3}). Shake for 10 minutes to extract the Ce^{+3} into the aqueous phase. Repeat this at least 3 more times with fresh aliquots of water and H_20_2.
7. Combine the aqueous phases in a beaker and boil the sample to remove excess H_20_2.
8. When cool, transfer to a 20 ml scintillation vial and count or precipitate oxalate and count. (See Note 2).

Oxalate Precipitation

9. Add 10 ml distilled H_20, 5 ml saturated oxalic acid and adjust to pH 2.0-2.2 with concentrated NH_40H to precipitate $Ce_2(C_20_4)_3$ x $9H_20$- Mount the precipitate on a glass fiber filter (Whatman 934-AH) and wash successively with distilled water and methanol before vacuum desiccating for 30 minutes.

10. As soon as a constant weight is achieved, count the sample for ^{144}Ce.

<u>Note 2</u>. *The cerium sample may be counted in either liquid form or mounted as an oxalate without significant variation in the* ^{144}*Ce values. Both methods (counting as a liquid and solid) are included so as to utilize different efficiencies for the gamma detector you will be using.* ^{141}*Ce has a 145 KeV gamma and* ^{144}*Ce has a 133 KeV gamma and its short lived daughter* ^{144}*Pr has gamma rays of 696, 1489 and 2185 KeV. These gammas are detected by gamma ray spectrometry using Ge(Li) or HPGe detectors.*

E

REPRESENTATIVE SAMPLING OF CARTRIDGE FILTERS: CORING OR CLIPPING VS. SMEARING

A common practice for obtaining samples of filter cartridges is to smear the filter housing, end cap or the filter, itself. An alternate technique used by some utilities is to extract a coring or clipping from the filter media. Some radwaste managers suspect that smears of cartridge end-caps, filter media or housing may not provide a representative sample since:

- Smears of wet media will tend to absorb liquid which will overestimate the activity of soluble nuclides such as cesium in the filter media.
- Smears may not pick up smaller particles and can underestimate the activity of activities associated with these particles (*e.g.*, ^{14}C).

Use of sub-micron filter cartridges in some plants can exacerbate the non-representative neture of smears used as surrogate samples for this media.

EPRI sponsored a set of measurements to investigate the magnitude of these effects in an operating PWR. The study obtained samples from two filter cartridges used in the plant; a 1-μ and a 0.2-μ filter cartridge. Samples included were:

1-μ Filter

- smear of filter housing interior
- smear of filter end cap
- smear of filter side
- coring from filter

0.2-μ Filter

- smear of filter end cap
- smear of filter side
- coring from filter

These samples were radiochemically analyzed for the 10CFR61 nuclides, both gamma emitting and DTM, and the scaling factors (to ^{60}Co) compared to examine for nuclide biasing in any of the samples.

Analysis of the 1-μ filter samples showed no observable bias in the resulting scaling factors between the different samplings. However, the results from the 0.2-μ filter samples showed significant differences between the coring and the smear samples.

Table E-1 presents the analytical results for the 1-μ sample. No entry for a nuclide results in the Table indicates that the activity was below a minimum detectable limit. Table E-2 shows the scaling factors of the important nuclides, related to ^{60}Co, for those samples. Figure E-1 shows the data in Table E-2 in graphical form for easier comparison of the results. Except for the very small concentration of ^{58}Co in the end-cap smear, the results from the four samples are very comparable, indicating that any of the samples is representative of the activity on the 1-μ filter.

Table E-3 presents the radionuclide analysis results for the 0.2-μ filter cartridge samples. Table E-4 and Figure E-2 give the scaling factors for these samples. For this filter cartridge sampling there are some substantive differences in the scaling factors derived from the different sampling techniques. The large differences occur for ^{14}C and the colloids ^{55}Fe, ^{54}Mn and ^{58}Co. Smears of the filter end cap would substantially underestimate the ^{14}C content whereas smears of the filter would significantly underestimate the corrosion-product colloids.

Table E-1

Summary of Radiochemical Results for the Analysis of the 1-μ Filter Cartridge Samples

NUCLIDE	Side Filter Smear μCi/Sample	End Cap Smear μCi/Sample	Inside Vessel Smear μCi/Sample	Filter Core μCi/Sample
^{14}C	6.0E-03	2.3E-02	1.7E-02	1.2E+00
^{55}Fe	5.4E-01	4.7E+00	2.9E+00	1.3E+02
^{63}Ni	4.9E-02	5.7E-01	3.8E-01	1.1E+01
^{51}Cr	2.2E-03	-	2.1E-02	2.4E-01
^{54}Mn	8.5E-03	7.8E-02	7.6E-02	9.0E-01
^{57}Co	4.4E-04	3.0E-03	4.3E-03	5.7E-02
Co-58	3.2E-02	2.9E-02	4.4E-01	4.0E+00

Table E-1
Summary of Radiochemical Results for the Analysis of the 1-μ Filter Cartridge Samples

^{59}Fe	6.8E-04	-	5.5E-03	1.7E-01
^{60}Co	1.6E-01	1.7E+00	1.1E+00	2.5E+01
^{65}Zn	6.5E-04	4.0E-03	-	-
^{95}Zr	1.4E-03	-	2.5E-02	1.3E-01
^{95}Nb	2.5E-03	1.4E-03	6.6E-02	2.5E-01
^{103}Ru	-	-	-	-
^{106}Ru	-	-	-	-
^{108m}Ag	-	-	-	-
^{110m}Ag	2.9E-04	1.3E-03	2.0E-03	1.1E-01
^{124}Sb	-	-	-	-
^{125}Sb	2.1E-03	2.0E-02	1.7E-02	3.9E-01
^{134}Cs	-	-	-	-
^{137}Cs	-	-	-	-
^{140}Ba/La	-	-	-	-
^{141}Ce	-	-	-	-
^{144}Ce	-	-	-	-
^{113}Sn	2.4E-04	7.2E-04	3.4E-03	4.1E-02
^{238}Pu	3.1E-06	1.8E-05	7.2E-06	2.5E-04
^{239}Pu	1.4E-06	2.1E-05	8.0E-06	2.1E-04

Table E-2
Scaling Factors (to ^{60}Co) Resulting From Analyses of the 1-µ Filter Cartridge Samples

Nuclide	Side Filter Smear	End Cap Smear	Inside Vessel Smear	Filter Core
^{14}C	0.038	0.014	0.016	0.051
^{55}Fe	3.42	2.77	2.74	5.45
^{63}Ni	0.31	0.33	0.36	0.46
^{54}Mn	0.054	0.046	0.072	0.037
^{58}Co	0.20	0.02	0.41	0.16
TRU	2.8E-05	2.3E-05	1.4E-05	1.9E-05

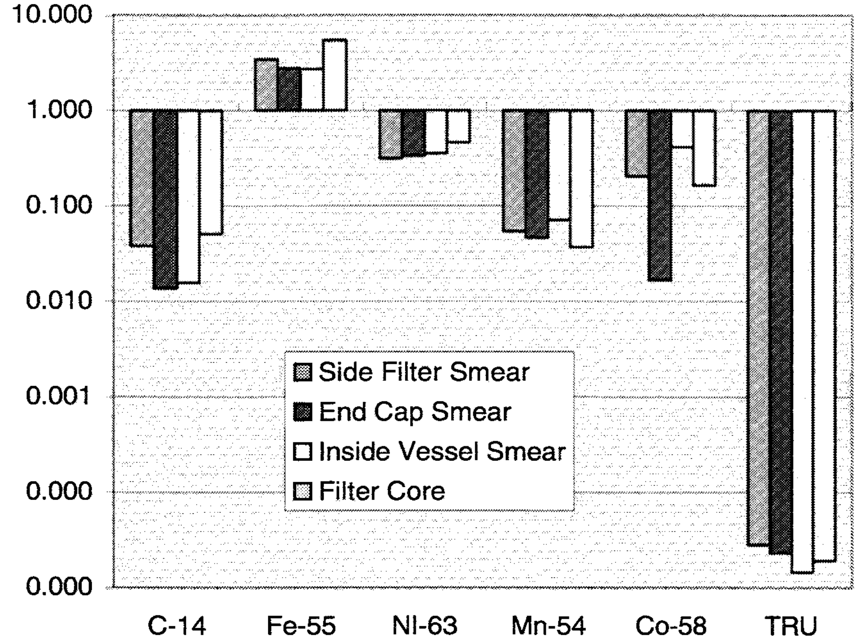


Figure E-1 Comparison of Scaling Factors (to ^{60}Co) from the Analysis of the 1-µ Filter Cartridge Samples

Table E-3
Summary of Radiochemical Results
for the Analysis of the 0.2-μ Filter Cartridge Samples

NUCLIDE	Filter Coring μCi/Sample	Filter Smear μCi/Sample	End Cap Smear μCi/Sample
^{14}C	1.83E-01	4.28E-03	4.80E-05
^{55}Fe	1.92E+01	1.81E-01	4.58E-01
^{63}Ni	2.05E+00	4.04E-02	6.41E-02
^{51}Cr	1.11E+01	1.25E-02	1.22E-01
^{54}Mn	1.05E+00	5.24E-03	7.27E-02
^{57}Co	7.05E-02	2.05E-04	2.42E-03
Co-58	2.59E+01	5.63E-02	8.82E-01
^{59}Fe	1.07E+00	1.36E-03	2.70E-02
^{60}Co	1.72E+00	9.13E-02	5.40E-02
^{65}Zn	1.25E-02	-	-
^{95}Zr	2.86E+00	2.91E-03	5.90E-02
^{95}Nb	3.99E+00	4.25E-03	8.56E-02
^{103}Ru	1.64E-01	-	-
^{106}Ru	-	-	-
^{108m}Ag	-	-	-
^{110m}Ag	6.10E-03	-	-
^{124}Sb	3.51E-02		8.10E-04
^{125}Sb	1.90E-01	3.32E-03	5.10E-03
^{134}Cs	-	-	-
^{137}Cs	3.80E-03	3.30E-04	1.05E-03
^{140}Ba/La	-	-	-
^{141}Ce	5.45E-02	-	-
^{144}Ce	2.21E-01	-	3.55E-03
^{113}Sn	2.97E-01	5.50E-04	7.47E-03
^{238}Pu	1.52E-04	6.20E-06	1.37E-06
^{239}Pu	1.33E-04	1.14E-05	1.79E-06

Table E-4
Scaling Factors (to ^{60}Co) Resulting from Analyses of the 0.2-μ Filter Cartridge Samples

Nuclide	Filter Coring	Filter Smear	End Cap Smear
^{14}C	0.11	0.047	0.0009
^{55}Fe	11.1	2.0	8.5
^{63}Ni	1.19	0.44	1.19
^{54}Mn	0.61	0.06	1.35
^{58}Co	15.0	0.62	16.3
^{137}Cs	2.2E-03	3.6E-03	1.9E-02
TRU	1.7E-04	1.9E-04	5.9E-05

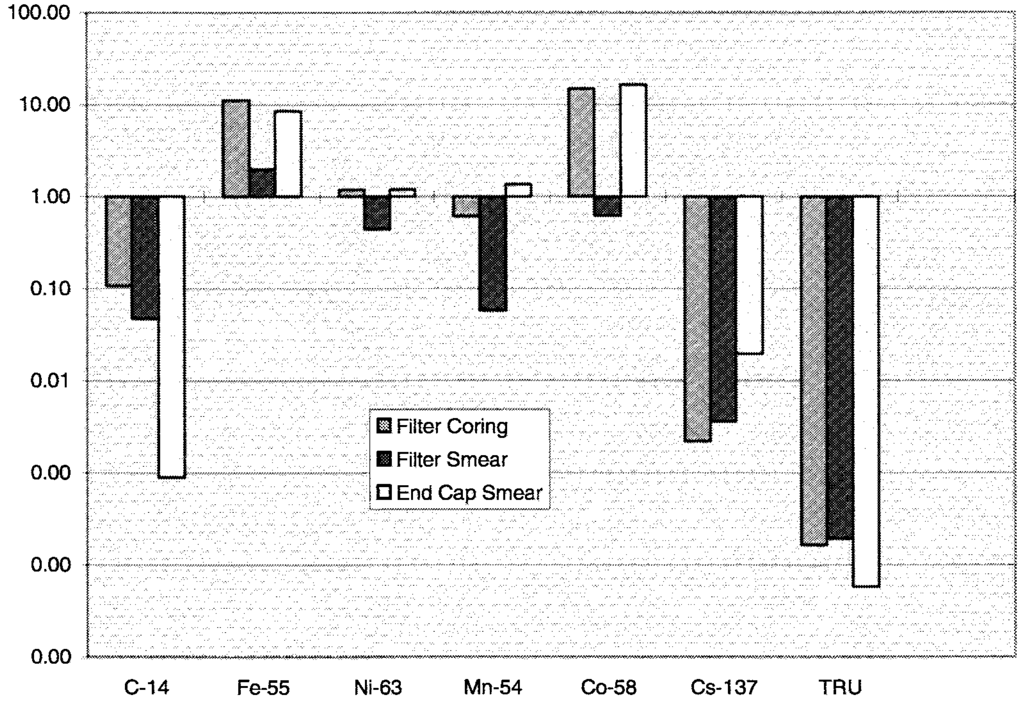


Figure E-2 **Comparison of Scaling Factors (to ^{60}Co) from the Analyses of the 0.2-μ Filter Cartridge Samples**

F

GLOSSARY

10 CFR XXX	Code of Federal Regulations, Title 10, Part XXX.
49 CFR XXX	Code of Federal Regulations, Title 49, Part XXX
Absorbent	A material that takes in another material, as a sponge absorbs water.
Absorption	The action of an absorbent.
Absorption, self	See self absorption.
Activation	The process of inducing radioactivity by irradiation. Activation occurs primarily in neutron radiation fields.
Activation DRP	A discrete radioactive particle formed by activation. The redominated activation product in activation DRPs in ^{60}Co.
Activation product	A radionuclide formed by activation. ^{60}Co is often the predominated activation producer.
Adsorbent	A solid material that holds a contaminant on its surface, as activated carbon holds iodine.
AEA	Atomic Energy Act of 1954, as amended.
Agreement state	A state that has assumed, by virtue of its agreement with the NRC, certain regulatory and licensing responsibilities of the NRC.
ALARA	As low as is reasonably achievable. As low as is reasonably achievable taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety and other social and socioeconomic considerations, and in relation to the utilization of atomic energy in the public interest. (This definition is from pre-1989 versions of 10CFR20).

Aliquant	One of several sub-samples of unequal size or quantity.
Aliquot	One of several sub-samples of equal size or quantity.
Alpha emitter	A radionuclide that emits alpha radiation.
Alpha radiation	Radiation consisting of alpha particles, helium nuclei without associated electrons. The mass of an alpha particle is that of 4 protons.
Analyte	Generic term for substance of interest, e.g. CO_2, ^{137}Cs, total U, etc.
Anion	A negatively charged ion.
Background Radiation	The radiation that is independent of the sample being measured. It is also used to mean natural background radiation.
Barnwell	The state-licensed low level waste disposal site at Barnwell, South Carolina operated by Chem-Nuclear Systems, Inc..
Beta emitter	A radionuclide that emits beta-radiation.
Beta Radiation	Radiation consisting of charged particles emitted from the nucleus of an atom during radioactive decay. The mass of a beta particle is 1/1837 that of a proton.
Binder	The solidification media used in processing waste for stabilization and/or disposal (e.g., cement, asphalt).
Boric Acid	The additive used for "chemical shim" reactivity control in PWRs and used in the emergency shutdown systems of BWRs.
BRE	Boron recovery evaporator (PWRs only).
Breakthrough	(Of on exchange resin) the first indication of the failure of a resin column to continue to remove a contaminant that it previously removed.
By-Product	A secondary waste or material from decontamination or other process.
Byproduct Material	Any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation

	incident to the process of producing, or utilizing special nuclear material.
Cation	A positively charged ion.
CF	Cubic feet. Also, abbreviated ft^3.
CFR	Code of Federal Regulations.
Channeling	(Of resin columns) A condition resulting from improper packing of a bed in which the solution being treated follows relatively particle-free path down through the bed.
Characterize	Usually, to determine radiological characteristics, sometimes including chemical and/or physical characteristic (chemical form and/or particle size).
Charcoal	A term that is often used for activated carbon adsorbent in either liquid and gaseous cleanup systems.
Chelating Agent	An organic compound that forms a chemically stable ring with a metal ion, promoting its solubility. Chelating agents are sometimes used in decontamination solutions. There disposal is limited because that may promote the migration of radioactive ions in waste.
Class A Waste	A specific class of low activity waste defined in 10CFR61.
Class B Waste	A specific class of low activity waste defined in 10CFR61. Class B waste contains more activity that Class A waste.
Class C Waste	A specific class of low activity waste defined in 10CFR61. Class C waste contains more activity than Class B waste.
Clean Waste Verification	Verification that waste contains no radioactive contamination and therefore can be disposed of without restrictions because of radioactivity.
Coincidence Counting	Counting that used two stacked detectors. Only counts that occur within the same time interval (due to the sample particle or same nuclear decay) are recorded. The same equipment may also be used in the opposing mode to record events that occur in only one of the two detectors. Technically, this is anti-coincidence counting.

Colloid	Very small, non-settling suspended particles.
Compact	Two or more states authorized as a compact to develop LLW disposal facilities, from which generators in other states cold be denied access.
Condensate Polishing System	The secondary cleanup system in a PWR.
Container	Any portable device in which a material is stored, transported, treated, or disposed, or otherwise handled.
Contamination	Contamination means material where it is not wanted. Usually used to mean radioactive contamination; radioactive material where it is not wanted.
CPM	Counts per minute. The number of radiation events detected by an instrument in one minute.
CPS	Condensate Polishing System. The secondary system in a PWR.
cps	Counts per second. The number of radiation events detect by an instrument in one second.
Critical Radionuclide	The radionuclide that will give the most biologically significant dose to the human body under the postulated release conditions.
Daughter Radionuclide	Decay product. A radionuclide resulting from the radioactive disintegration of a radionuclide, formed either directly or as the result of successive transformations in a radioactive series. A decay product may be either radioactive or stable.
DAW	Dry Active Waste.
DB	Deep bed. Deep bed ion exchange columns are large vessels containing bead (not powdered) resin. Activated carbon or other absorbents may also be present in deep bed systems.
Decay Rate	The rate which a radioisotope decays. Decay rate is usually given in disintegrations per minute.

Demineralizer	An ion exchange vessel used to prepare very low conductivity (demineralized) water. However, in radwaste, the term is often used to mean any ion exchange vessel used for water cleanup.
Dewatering	The removal of water, usually by gravity or pumping (as distinguished from drying). As the term applies to the processing of spent resin, there are two similar but distinctively different meanings: (1) When solidifying spent resin, the resin is dewatered to obtain the desired resin:waterbinder ratio. It is not intended that all free standing water be removed from the resin before addition of the binder. This dewatering effort is a subcomponent of the utility's solidification process control program approved by the NRC. (2) When dewatering resin for disposal without the intent of solidifying, the dewatering pump and system continues to draw water from the resin until the free-standing water is less than 0.5 percent. This dewatering effort is a subcomponent of the utility's dewatering process control program approved by the NRC.
Difficult-to-Measure Radionuclides	Radionuclides that are not abundant and do not release easily detected gamma radiation. Difficult-to-measure radionuclides include ^{90}Sr, ^{14}C, and ^{55}Fe.
Discrete Radioactive Particle	A highly radioactive small particle. Also called hot particle or fuel flea.
Dose	A general term denoting the quantity of radiation or energy absorbed. For special purposes, it must be qualified (Radiological Health Handbook, US Department of Health Education and Welfare, 1970).
Dose Rate	A term denoting the rate at which radiation dose is absorbed. See dose.
DOT	U.S. Department of Transportation.
dpm	Disintegrations per minute. The number of nuclear events occurring in the sample in one minute.

dps	Disinregrations per second. The number of nuclear events occurring in the sample in one second.
Dry Active Waste	Dry, solid, low level radioactive waste.
DTM	See difficult-to-measure radionuclides.
EEI	Edison Electric Institute.
Efficiency	(of a counter) A measure of the probability that a count will be recorded when radiation strikes a detector. The counts per unit time recorded divided by the disintegrations in the same time.
EPRI	Electric Power Research Institute.
Fission Product	A nuclide that is formed by nuclear fission. Fission products of concern are usually the radioactive one such as ^{137}Cs and the plutonium isotopes. Ordinarily, fission products are confined within the fuel and are only of concern if fuel has been breached.
Fixed Contamination	Contamination that is not easily removed without disturbing the base material. Contamination that is not transferred to the sample when a wipe or smear sample is taken.
ft^3	Cubic feet. Also abbreviated cf.
Fuel Cladding Failure	Breaches in the cladding of fuel of any size that release dissolved or particulate radioactive contamination into the primary system.
Fuel Failure	See fuel cladding failure.
Gamma Emitter	A radionuclide that emits gamma radiation from its nucleus.
Gamma Radiation	High energy, short wavelength electromagnetic radiation emitted from the nucleus. Gamma radiation is more penetrating than either alpha or beta radiation. Gamma rays are similar to X-ray, except in their formation. They are normally more energetic.
GC	Gas chromatograph or gas chromatography.
GeLi	Lithium drifted germanium. Sometimes abbreviated Ge(Li). A common and useful radiation detector. A GeLi detector

	coupled with the right electronics, will resolve the energy of the radiation striking it so that isotopes can be identified.
G-M detector	Geiger-Mueller detector.
Generator	Generator of low-level waste.
Greater-Than-Class-C Waste	A specific class of low activity waste, defined by 10CFR61, which contains more activity than Class C Waste.
Half-Life	The time required for one half of a quantity of a pure radionuclide to decay.
HEPA filter	High efficiency particulate air filter. HEPA filters, by definition, must retain at least 99.97% of particulate material 0.3 microns or larger in diameter.
ISOSHLD	The name of a shielding code.
Isotopes	Nuclides having the same number of protons in their nuclei, and hence the same atomic number, but differing in the number of neutrons, and therefore the mass number. Almost identical chemical properties exist between isotopes of the same element.
Isotopic Composition	The content of a sample by radioisotope. Stable (non-radioactive) isotopes are usually not considered.
IX	Ion exchange.
Live Time Counter	A counter that automatically corrects for dead time in sample counting.
LLD	See lower limit of detection.
LLW	Commercial low-level radioactive waste. Those wastes containing source, special nuclear, or byproduct material that are acceptable for disposal in a land disposal facility. This definition has the same meaning as that in the Low-Level Waste Policy Act, that is radioactive waste not classified as high-level radioactive waste, transuranic waste, spent nuclear field, or byproduct material as defined in section 11e.(2) of AEA (uranium or thorium tailings and waste).
LLWPAA	Low Level Waste Policy Amendments Act.

Long-Lived Radionuclide	A relative term, indicating the period of time a radionuclide will be detectable in a significant amount. Radionuclides with a half-life of less than 60 days are rarely considered long-lived. Those with a half-life of 100 years or more almost always are.
Lower Limit of Detection	Usually the least amount of material that can be detected with 95% confidence.
LRW	Liquid radwaste.
LWR	Light water reactor, any water cooled reactor except a heavy water cooled reactor.
Manifest	Documents that must accompany shipments of LLW, hazardous waste, and mixed waste. Refer also to 10 CFR 20.311 and 40 CFR 262, appendix.
MDA	See minimum detectable activity.
MDC	Minimum detectable concentration as defined by L.A. Currie. Limits for Qualitative Detection and Quantitative Determination. Anal. Chem. 40(3): 586-693: 1968. This terminology is synonymous with the frequently used term "Lower Limit of Detection" (LLD).
MICROSHIELD	The name of a shielding code.
Minimum Detectable Activity	The smallest concentration or quantity of a radioactive material in a sample that yields a net count above system background with 95% confidence.
Mixed waste	Commercial low-level radioactive waste (LLW) that is also regulated as hazardous waste under EPA's hazardous waste program. Mixed waste is listed as hazardous waste in 40 CFR 261, Subpart D, contains hazardous substances that are listed as hazardous waste in 40 CFR 261, Subpart D, or exhibits any hazardous characteristic defined in 40 CFR 261, Subpart C.
Multi-Channel Analyzer	An electronic counting system that is able to discriminate between pulses of different voltages. Multi-channel analyzers are widely used for gamma spectroscopy for radionuclide identification.

NaI	Sodium iodide, usually means a thallium drifted sodium iodide detector.
NaITI	Thallium drifted sodium iodide detector. Sometimes abbreviated NaI(TI).
Natural Background Radiation	The radiation in the natural environment, including cosmic rays and radiation from naturally-occurring radioactive elements, both inside and outside the human body.
NCRP	National Council on Radiation Protection.
New Waste Sream	A term used to refer to a waste stream that has not been characterized according to 10CRF61.
Non-Compatible DAW	DAW consisting of pipe, valves and reactor components that cannot be reduced in volume by the equipment available. What is considered compactible from one plant to the next will vary depending on the compaction equipment available.
NRC	U.S. Nuclear Regulatory Commission.
Origen-2	A computer code used to determine the radionuclide inventory from fission and activation during reactor operation. It has been used to determine the radionuclide inventory of activated components.
Packaged Volume	The volume of waste as it is packaged ready for disposal. This includes the volume of the burial containers.
Parent Radionuclide	A radionuclide that decays to form a radioactive daughter radionuclide. A single radionuclide may be both a parent and daughter in a decay chain.
Pathway	The route by which a radionuclide travels to give a radiation dose. Radionuclides in airborne or liquid effluents may give a population does through a good chain pathway.
Predominate Radionuclide	An isotope which, because of its concentration, half-life, or chemical characteristics, appears to be the most prevalent constituent of a sample or waste.

Pressured Water Reactor	A light water reactor that transfer heat to a secondary (uncontaminated) stream/water system which powers the turbine.
PWR	Pressured water reactor.
Radioassay	Analysis for radionuclides.
Radiochemical Analysis	Analysis using the chemical properties of the radionuclide of interest to separate it from the bulk of the material before radiological counting. Radiochemical procedures generally have much lower MDAs than direct radiological counting.
Radioiodine	A radioactive isotope of iodine. The most common ones are ^{129}I and ^{131}I.
RADMAN	The name of a shielding code that is geared to the needs of the radwaste professional.
Radwaste	Radioactive waste.
Radwaste Desk Reference	Publication by EPRI intended to provide quick reference to t typical problems and issues encountered in the management of LLW at nuclear power plants.
Reg Guide	See Regulatory Guide.
Regulatory Guide	One of a series of documents published by the NRC which identify one acceptable method of performing some required task or function.
Removable Contamination	Contamination that can be removed by hand wiping or other nondestructive measures. It is also called smearable contamination. It is usually quantified as the number of disintegrations per 100 square centimeters.
Scaling Factor	Known ratios of radionuclides in reactor waste streams which are used to calculate the quantity of some radionuclides which are known to present but are not directly measured in each waste package. Activation radionuclides such as ^{59}Fe and ^{63}Ni are usually scaled to ^{60}Co. Fission radionuclides such as the plutonium isotopes are usually scaled to ^{137}Cs.
Segregation	The process of separating waste according to radiological considerations. This may involve separating bags of trash by

	dose rate (e.g., 0-5, 5-50, and >50 mrad/hr), separating trash containing discrete radioactive particles from the more routine radioactive trash, or similar segregation systems.
Self Absorption	Absorption of radiation by the material in which the radioactive atoms are located: in particular, the absorption of radiation within a sample being assayed.
Short-Lived Radionuclide	A relative term indicating the period of time a radionuclide will be detectable in a significant amount. Radionuclides with a half ife of less than 30 to 60 days may be considered short-lived for radwaste management purposes.
SI(units).	The term for an international system of standard radiation units. See conversion tables for specific units.
SIMPLE	The name of a shielding code.
SNM	See special nuclear material.
Sock Filter	A slang term for a cloth bag filter used in liquid processing systems.
Source material	Uranium or thorium, or any combinations thereof, in any physical or chemical form OR ores which contain by weight one-twentieth of one percent (0.05%) or more of (1) Uranium, (2) thorium or (3) any combination thereof. Source material does not include special nuclear material.
Special nuclear Material (SMN)	Plutonium, ^{233}U, uranium enriched in the isotope 233 or the isotope 235, and any other material which the NRC, pursuant to the provisions of section 51 or AEA, determines to be special nuclear material, but does not include source material OR any material artificially enriched by any of the foregoing but does not include source material.
Specific Activity	The total activity of a radionuclide (or all radionuclides) per unit weight of a compound, element, radionuclide or waste.
Transuranic	Material that contains more than 100 nano-Curies (nCi) per gram of trans-uranium isotopes.
Treatment	Any method, technique, or process, including neutralization, designed to change the physical, chemical, or biological character of composition of the waste so as to neutralize such

	waste, or so as to recover energy or material resources from the waste, or so as to render such waste non-hazardous, or less hazardous; safer to transport, store, or dispose of or amenable for recovery, amenable for storage, or reduce in volume.
TRU	See transuranic.
Uniform Hazardous Waste Manifest	The form required by EPA in 40 CFR 262.20 through 262.23 to accompany all shipments of hazardous waste to assure that waste is properly accounted for "from cradle to grave."
Uniform Low-Level Radioactive Waste Manifest	The form and information required by 10 CFR 20.311 to accompany all shipments of LLW ultimately intended for disposal.
WISE	Waste Inventory and Shielding Estimator - a shielding and curie estimation code.

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