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Dete April 29,1993

To WJ Madia, General Manager, ESTD

From WJ Zielenbach, Secretary, RSC

Subject Case RSC-151, JN-1 Criticality System

Craig Jensen, acting for BCLDP, requested through the BCO RSO an RSC review of the radiation safety issues pertaining to the continued need for criticality monitors, nuclear accident dosimeters (NADs), and criticality signage in JN-1. Dr. Richard Denning and Roger Wooton, members of the former Battelle Criticality Safety Subcommittee (RSC-2), were present to add historical perspective and expert advise in this matter; Dr. Denning is also a member of the RSC. When major hot cell activities were discontinued in 1987 and the Criticality Safety Subcommittee was dissolved, two criticality monitors (out of six) were left in operation, one in the High Level Cell and one in the High Energy Cell. These monitors have been maintained since that time by the BCLDP.

Several years ago, an analysis was performed that demonstrated that there was no credible scenario whereby criticality could be achieved based on the existing inventory of fissile material in the JN-1 facility. The RSC evaluated this analysis and any possible uncertainties in any assumptions contained in this analysis. The RSC unanimously supports the conclusion that there is no credible manner in which a criticality event could be initiated in JN-1 with the amount of fissile material that is currently present. Supporting documentation is attached for the record.

Consequently, it is the opinion of the RSC that there is no radiation safety basis for the presence of the criticality monitors, dosimeters, and signage that currently remain in the JN-1 facility. This conclusion is based on the following:

1. Accountability records indicate that there is no more than approximately 50g of fissile material in the entire facility.

2. The 50g is dispersed throughout the facility in low isotopic enrichments.

3. The regulatory limit that requires a criticality accident alarm system is 450g for any combination of fissile isotopes. Therefore, the current inventory at JN-1 would have to be nine times greater than stated in the accountability records for there to be a regulatory basis for the presence of an alarm system.

4. Given the enrichment factors of the existing fissile material, many kilograms of fissile material would have to be present in an optimum configuration with suitable moderation and reflection to initiate a critical event.

The RSC further concludes that an independent verification of the accountability records is not warranted.

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Radcalc for Windows Version 2.01, Volume II: Technical Manual

J. G. McFadden Waste Management Federal Services, Inc., Northwest Operations Richland, Washington 99352 for Fluor Daniel Hanford, Inc. U.S. Department of Energy Contract DE-AC06-96RL13200

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Key Words: hydrogen generation, radioactive waste packages, decay heat, radionuclide decay, DOT classifications

Abstract: Radcalc for Windows Version 2.01 is a user-friendly menudriven Windows-compatible software program with applications in the transportation of radioactive materials. It calculates the radiolytic generation of hydrogen gas in the matrix of low- and high-level radioactive waste using NRC-accepted methodology. It computes the quantity of a radionuclide and its associated products for a given period of time. In addition, the code categorizes shipment quantities as radioactive, Type A or Type B, limited quantity, low specific activity, highway route controlled, and fissile accepted using DOT definitions and methodologies as outlined in 49 CFR Subchapter C. The code has undergone extensive testing and validation. Volume I is a User's Guide, and Volume II is the Technical Manual.

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VOL II

RADCALC FOR WINDOWS VERSION 2.01, VOLUME II: TECHNICAL MANUAL

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J. G. McFadden K. E. Hillesland J. G. Field

April 1998

Prepared for the U.S. Department of Energy by Engineering Waste Management Federal Services, Inc., Northwest Operations _____

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Version 2.01 of Radcalc for Windows was supported by the U.S. Department of Energy. Radcalc for Windows' authors would like to thank the following people for their support on this project: Vic Roetman, Tony Savino, Jess Greenborg, Jennifer Mercado, Jeff Boettger, and Margaret Andrews. In addition, we would like to acknowledge the work and support of two of the original authors of the Electric Power Research Institute report: Jim Flaherty and Paul Deltete. Acknowledgement is also due to Rich Genoni and Greg Field for their effort in the development of the Excel version of the Radcalc spreadsheet. _

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RADCALC FOR WINDOWS VERSION 2.01, VOLUME II: TECHNICAL MANUAL

J. G. McFadden K. E. Hillesland J. G. Field

ABSTRACT

Radcalc for Windows¹ is a user-friendly computer software program that has applications in the transportation and packaging of radioactive high- and low-level waste. It calculates the radiolytic production of hydrogen gas in the waste matrix of radioactive waste. In addition, Radcalc for Windows contains a robust decay algorithm that calculates the activity of a radionuclide and its associated products at a future date. Radcalc for Windows also classifies transportation quantities using the U.S. Department of Transportation definitions and methodologies outlined in Title 49, Code of Federal Regulations, Subchapter C,² as revised in the October 1, 1996, edition.

Version 2.0 of Radcalc for Windows incorporated the revisions in Title 49 of the Code of Federal Regulations found in the October 1, 1996, edition. It also incorporated changes that are designed to facilitate the input and output functions. Version 2.01 of Radcalc for Windows corrects the mixed fission product A_2 value to reflect Table 10 of 49 CFR 173.433.³ Radcalc for Windows

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¹Windows is a trademark of Microsoft Corporation.

²49 CFR Subchapter C, 1996, "Hazardous Materials Regulations, <u>Code of Federal Regulations</u>.

³49 CFR 173, 1996, "Shippers--General Requirements for Shipments and Packagings," <u>Code of Federal</u> <u>Regulations.</u>

Version 2.0 also had a minor coding error in the data output file routine, which has been corrected in Version 2.01.

Documentation for Radcalc for Windows is available in two volumes. Volume I is a user's manual, which is intended as a quick reference. It will aid the user in becoming familiar with Radcalc for Windows. Volume II is the Radcalc for Windows technical manual. The technical manual contains information on the history of Radcalc for Windows as well as the theoretical background, calculational methodology, configuration control, and verification and validation of the code.

Radcalc for Windows is sponsored by the U.S. Department of Energy. It was developed and is maintained by Engineering, Waste Management Federal Services, Inc., Northwest Operations. Copies of the software program are available by calling Engineering.

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FOREWORD

Radcalc for Windows is a software program developed and maintained by Engineering, Waste Management Federal Services, Inc., Northwest Operations, Richland, Washington, for the U.S. Department of Energy. Documentation for the software program exists in the form of *Radcalc for Windows Version 2.01*, *Volume I: User's Manual* and *Radcalc for Windows Version 2.01*, *Volume II: Technical Manual*. The software is available from Engineering by calling 509-376-0610 or 376-7111. Corrections and suggestions for Radcalc for Windows are welcomed. This page intentionally left blank.

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LIST OF TERMS

Bq/s	becquerels per second
CD	compact disk
Ci/L	curies per liter
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
EBU	Engineering Business Unit
ENDSF	 Evaluated Nuclear Data Structure File
EPRI	Electric Power Research Institute
eV	electronvolt
ENDF	Evaluated Nuclear Data File
FENDL	Fusion Energy Nuclear Data Library
g/cc	grams per cubic centimeter
HRCQ	highway route controlled quantity
ID	inside diameter
IE	Office of Inspection and Enforcement
kPa	kilopascal
LET	linear energy transfer
LLW	low-level waste
LSA	low specific activity
µCi/g	microcuries per gram
mCi/g	millicuries per gram
MeV	megaelectronvolt
MCNP	Monte Carlo N-Particle
MFP	mixed fission products
NCAW	neutralized current acid waste
NRC	U.S. Nuclear Regulatory Commission
OD	outside diameter
ORIGEN	Oak Ridge Isotope Generation and Depletion Code
010	onsite transfer cask
QA DIDC	quality assurance
RIDS	Records Inventory and Disposition Schedule
KŲ	reportable_quantity
SAKP	satety analysis report for packaging
	transuranic
WPINW	waste Management Federal Services, Inc., Northwest Operations

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1.0 INTRODUCTION

1.1 DESCRIPTION

Radcalc for Windows¹ 2.01 is a Windows-compatible software program with applications in the packaging and transportation of radioactive materials. Its primary function is to calculate the generation of hydrogen gas by radiolytic production in the waste matrix of high- and low-level radioactive waste packages. It can also be used to categorize radioactive shipments into appropriate U.S. Department of Transportation (DOT) classifications. In addition, it contains a robust decay algorithm that determines the daughter products of selected radionuclides. The various functions in Radcalc for Windows can be used separately or together.

Version 2.0 of Radcalc for Windows was issued in January 1998 and replaced Version 1.0 issued in September 1995. It offered a number of enhancements over the original code, but the primary changes reflected the DOT revisions to 49 CFR Subchapter C effective in April 1996. Version 2.01 corrects two minor errors found in 2.0. In Version 2.01 the A_2 value for mixed fission products (MFP) was changed to reflect the value given in Table 10, 49 CFR 173.433 for alpha emitters. Also the data file specific heat values were corrected. Specific heat values in the .RAD output files did not reflect the data file error and were therefore left unchanged.

The software program requires a personal computer with a 386 or higher coprocessor and Windows Version 3.1 or later. It is written in Visual C++² using the Microsoft³ Foundation Classes and Windows Library routines. It is compiled on Microsoft Visual C++ Version 1.5. It is also written in FORTRAN using the Microsoft FORTRAN 5.10 compiler. It contains libraries of radionuclide data, G values, and A_1/A_2 values for transportation classification. The software format is similar to other Windows-compatible codes.

Volume I, Rev. 0, of the documentation of Radcalc for Windows is a user's manual. The user's manual contains installation and use instructions. This volume, Volume II, Rev. 0, is the technical manual, which contains the theoretical and calculational basis of the program, the verification and validation of the program, and configuration control.

1.2 BACKGROUND

High-level and low-level radioactive wastes produce ionizing radiation, which may cause the radiolytic formation of hydrogen gas. A calculational technique for quantifying the concentration of hydrogen generated by radiolysis in sealed radioactive waste containers was developed in a

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U.S. Department of Energy (DOE) study conducted by EG&G Idaho, Inc., and the Electric Power Research Institute (EPRI) TMI-2 Technology Transfer Office. The study resulted in report GEND-041, A Calculational Technique To Predict Combustible Gas Generation in Sealed Radioactive Waste Containers (Flaherty et al., 1986). The study also resulted in a presentation to the U.S. Nuclear Regulatory Commission (NRC) which gained acceptance of the methodology for use in ensuring compliance with NRC Office of Inspection and Enforcement (IE) Information Notice No. 84-72 (NRC 1984) concerning the generation of hydrogen within packages.

NRC IE Information Notice No. 84-72, "Clarification of Conditions for Waste Shipments Subject to Hydrogen Gas Generation," applies to any package containing water and/or organic substances that could radiolytically generate combustible gases. The NRC requires the following over a period of twice the expected shipment time.

- The hydrogen gas concentration is less than 5% by volume (or equivalent limits for other inflammable gases) of the secondary container gas void.
- The secondary container and cask cavity must be inerted to ensure that oxygen is limited to 5% by volume in those portions of the package that could have hydrogen concentrations greater than 5%.

If the material being shipped classifies as low specific activity (LSA) and if it is shipped within ten days after venting, the above requirements do not apply. Notice No. 84-72 is pertinent to shipments of resins, binders, waste sludge, and wet filters. The notice requires compliance by tests or measurements of a representative package. However, in April of 1985, the NRC accepted the calculational method presented in report GEND-041 to confirm the absence of hydrogen in low-level waste (LLW) packages (Deltete 1987).

Subsequently, EPRI developed a simple computer program in a spreadsheet format utilizing the GEND-041 calculational methodology to predict hydrogen gas concentrations. The computer code was extensively benchmarked against TMI-2 (Three Mile Island) EPICOR II resin bed measurements. The benchmarking showed that the model developed predicted hydrogen gas concentrations within 20% of the measured concentrations. The computer code has since been dubbed Radcalc and is still accepted today by the NRC for Certificates of Compliance for shipping casks containing materials that may result in the radiolytic formation of hydrogen gas. Radcalc for Windows utilizes the computational methodology outlined in GEND-041 to calculate the production of hydrogen gas in the waste matrix of packages.

1.3 SCOPE

Radcalc for Windows calculates the production of hydrogen gas in the waste matrix of radioactive material containers. It contains a radionuclide library of approximately 260 radionuclides. The library is utilized by the hydrogen gas generation algorithms and can also be used to determine daughter products of selected radionuclides. The G value database (molecules of H_2 formed per 100 eV of energy absorbed) has been widely researched and includes a large number of material types.

In addition, the code categorizes packages as radioactive; Type A; Type B; limited quantity; LSA-I, -II, or -III; highway route controlled quantity (HRCQ); and fissile excepted, employing 49 CFR 173.403 definitions. It also determines reportable quantity (RQ) per 49 CFR Subchapter C and transuranic (TRU) waste per the *Hanford Site Solid Waste Acceptance Criteria* (Willis 1993). The A_1/A_2 database used by the code for these determinations is based on 49 CFR 173.435. Radcalc for Windows also calculates decay heat and pressure buildup due to hydrogen.

The Radcalc for Windows calculational methodology is outlined in Chapter 2.0. Chapter 3.0 presents the data that support the software. Chapter 4.0 gives the verification and validation documentation for the code. Configuration control is located in Chapter 5.0. Requirement specifications are summarized in Chapter 6.0, and Chapter 7.0 gives a summary.

2.0 RADCALC FOR WINDOWS METHODOLOGY

2.1 HYDROGEN GAS GENERATION

2.1.1 Theoretical Basis

Hydrogen gas generation occurs in radioactive waste materials through two main mechanisms. One of these mechanisms, the one that is addressed in Radcalc for Windows, is the release of hydrogen through radiolysis. The other mechanism is the release of hydrogen from the oxidation of metals immersed in water, and it is not included in Radcalc for Windows.

Radiolysis is the chemical change that occurs in materials as a result of ionizing radiation. The majority of radiolysis that occurs in radioactive waste materials is due to interactions initiated by alpha particles, beta particles, and photons (gamma rays and x-rays). As ionizing radiation travels through a medium, the atoms and molecules within the trajectory of the particle or photon will absorb energy and be ionized or left in an excited state. The initiating interacting radiation creates a track of excited ions and molecules. Freed electrons can, in turn, create secondary tracks or spurs branching off from the primary track. The length of the tracks and the distance traveled by the ionizing radiation will depend upon the energy and type of radiation and the chemical and physical properties of the medium. The species resulting from the interactions will depend upon the chemical structure of the absorbing medium and will, for the most part, be the same for different types of ionizing radiation. The quantity of the species formed will depend upon the amount of radiation energy absorbed within the material, the proximity of the different species formed (which will affect the chances of species recombination), and, in some part, the chemical nature of the different components.

A chemical response initiated by ionizing radiation can cause a large number of intermediary reactions. For instance, in the radiolysis of water, Spinks and Woods (1990) identifies 16 additional major reactions that follow as a result of the initial reactions caused by ionizing radiation. The The types of species produced radiolytically depend upon the chemical structure of the material through which the radiation is traveling. The major products in the radiolysis of water are:

$$H_2O = e_{aq}^{-}, OH^{-}, H_2^{-}, H_2O_2^{-}, HO_2^{-}, H_3O^{+}$$

The addition of substances such as nitrates to the water will change the chemical components produced and the quantities of these components. Some of the gases identified from the radiolysis of cation exchange resins are H_2 , CO_2 , CO, SO_2 , O_2 , and CH_4 and from the radiolysis of anion exchange resins: H_2 , CO_2 , CO_2 , CO_3 , N_2 , N_2O_3 , and NO.

In order to relate the quantity of a species produced radiolytically to the amount of energy absorbed, a term arbitrarily called the *G value* was defined (Burton 1952). The *G* value of a material is defined as the number of molecules formed or disassociated per 100 eV energy absorbed. G values are species and material specific; that is, $G(H_2)$ refers to the *G* value for the production of hydrogen gas and will differ for different absorbing media. G values are also radiation specific and depend upon the linear energy transfer (LET) of the radiation type. For instance, alpha particles have a much higher LET than beta particles and the *G* value will be correspondingly greater.

Using the G value concept, the production of hydrogen can be calculated by multiplying the total decay heat or energy produced over a specified period of time of radiation type j for radionuclide i by the fraction of decay heat or energy absorbed in the medium for that radiation type by the medium specific G value for the jth radiation type. Summing this product for all radiation types and radionuclides in the material and multiplying by the mass of the medium and a conversion factor will result in the quantity of hydrogen produced.

2.1.2 Hydrogen Gas Generation Calculations

The hydrogen gas volume is calculated as

$$H(t) = K \sum_{ij} G_j D_i(t) Z_{ij}$$
, (2-1)

where:

- i = Index for radionuclide type j = Index for radiation type. Radiation types are: $j = \alpha$ (hereafter referred to as "heavy particle"), which are decays of heavy, charged particles, such as alpha and spontaneous fission, as well as delayed neutrons, $j = \beta$ (hereafter referred to as "beta-type"), which are all electron-related radiation, such as electrons, positrons, conversion-electrons, and Auger electrons, and $j = \gamma$ (hereafter referred to as "gamma"), which includes all electromagnetic radiations, such as gamma rays, x-rays, annihilation radiation, and internal bremsstrahlung. G = Hydrogen G value for the given radiation type (molecules/100 eV) $D_i(t)$ = Total number of disintegrations for the radionuclide over time t
 - \mathcal{E}_{ij} = Total energy absorbed per disintegration K = Conversion factor from molecules to volume for an ideal gas at
 - 20 °C and standard pressure (101.325 kPa).

For known discrete gamma radiation, the total energy absorbed is calculated as follows:

$$\mathscr{E}_{i\gamma} = \sum_{k} E_{i\gamma k} A_{i\gamma k} F(\rho, E_{i\gamma k}) + \overline{E_{i\gamma}^{\prime}} ,$$

k = Index for each discrete gamma $E_{i\gamma k} = kth \text{ discrete gamma energy} \\ A_{i\gamma k} = \text{Fraction of decays exhibiting the kth gamma} \\ F(\rho, E_{i\gamma k}) = \text{Fraction of energy absorbed in the waste matrix, as discussed} \\ \hline E_{i\gamma k} = \frac{1}{2}$ $\overline{E'}_{iv}$ = Electromagnetic radiation not accounted for amongst the known discrete gammas, usually very low energy x-rays.

$$\overline{E'_{i\gamma}} = \overline{E_{i\gamma}} - \sum_{k} E_{i\gamma k} A_{i\gamma k}$$

where $\overline{E_{i\gamma}}$ is the total average gamma radiation.

For heavy particle and beta-type radiation the total energy absorbed is simply the average recoverable decay energy for the given disintegration type per disintegration. E_{ij} , E_{iyk} , and A_{iyk} data are taken from the Fusion Energy Nuclear Data Library (FENDL)/D-1.0 database discussed in Chapter 3.0.

2.2 HYDROGEN GAS GENERATION RATE

The hydrogen gas generation rate is calculated over the time that the container is sealed. It is calculated by dividing the volume of hydrogen gas produced during the seal time by the seal time duration.

2.3 TOTAL PRESSURE AND PARTIAL PRESSURE

Using van der Waals' reduced equation of state for ideal gases and assuming ideal gas conditions at constant temperature and volume, the total pressure due to the buildup of hydrogen gas is found from the original pressure times the ratio of the final total molecules to the initial total molecules. The partial pressure of the hydrogen gas is calculated from the original pressure times the ratio of the hydrogen molecules generated to the initial molecules. Pressures are calculated at the end of the user-specified seal time.

2.4 DECAY HEAT

Decay heat is calculated at the end of the user-specified decay time. The decay heat calculated by Radcalc for Windows is conservatively assumed to be equal to the total recoverable energy. The decay heat for a given radionuclide indexed by i is calculated as follows:

$$H = K \sum_{i} A_{i} E_{i} .$$

H = Decay heat rate in Watts at the start of shipment (seal time)
E_i = Total recoverable energy per disintegration of the ith nuclide in eV
A_i = Activity in curies of the ith nuclide at the beginning of shipment
K = Conversion factor of 5.93E-9 W/(Ci-eV), derived from conversion
factors listed in the student edition of the 72nd edition of the
Handbook of Chemistry and Physics (Lide 1991).

The total recoverable energy for a disintegration of each nuclide is calculated in RADCALC as follows:

$$E_i = E_{\alpha} + E_{\beta} + E_{\gamma}$$
.

 E_{α} is the average energy per decay for all heavy charged particles and delayed neutrons for the ith nuclide. E_{α} is the average recoverable energy per decay for beta-type radiation for the ith nuclide. E_{α} is the average energy per decay for all gamma radiations for the ith nuclide. E_{α} , E_{α} , E_{α} , and E_{γ} originate from the FENDL/D-1.0 database, as discussed in Chapter 3.0.

2.5 RADIOACTIVE DECAY CALCULATIONS

Given a starting inventory and a time period for decay, Radcalc for Windows will calculate the radionuclide inventory at the end of the decay time and the time-integrated disintegrations of each radionuclide during the decay time.

The decay equations can be expressed as follows:

$$\frac{dn_i}{dt} = \sum_{j \neq i} \alpha_{ij} n_j - \beta_i n_i ,$$

where:

 $\begin{array}{l} i = \mbox{Index for daughter radionuclides} \\ j = \mbox{Index for parent radionuclides} \\ n_i = \mbox{Amount of the ith radionuclide} \\ t = \mbox{Time} \\ n_j = \mbox{Amount of the jth parent} \\ \alpha_{jj} = \mbox{Decay constant for decay from radionuclide j to radionuclide i} \\ B_j = \mbox{Decay constant for depletion of radionuclide i}. \end{array}$

Appendix A, written by F. Schmittroth, shows a solution to the decay equation and demonstrates how it is implemented. The technique outlined in Appendix A was adopted in Radcalc for Windows. The amount, v_i , of the ith radionuclide produced from its immediate parent is given by

 \cdots $-\beta_{i}t \sum_{i=1}^{\infty} a^{(i)} + I$

$$v_i = \sum_j e^{-j} \sum_{r=0}^{j} c_{jr} t^{-r}$$
 (2-2)

The second sum represents a polynomial in t with constants $c_{jr}^{(1)}$. The method for calculating $c_{jr}^{(1)}$ is documented in Appendix A. The v_i are summed over all decay-chain branches to find the total amount, n_i , of radionuclide i.

The time-integrated disintegrations for n_i , designated as D_i , can be obtained from

$$D_{i} = \beta_{i} \int_{0}^{t} n_{i}(t') dt'$$

(2-3)

The result, d_j , corresponding to the single decay-chain branch represented by v_i , can be calculated by inserting equation 2-2 into 2-3 above.

$$d_{i} = \beta_{i} \sum_{j} \sum_{r=0}^{n} c_{jr}^{(i)} \int_{0}^{t} t'^{r} e^{-\beta_{j}t'} dt'$$
.

This can be rewritten as

$$d_{i} = \beta_{i} \sum_{j} \sum_{r=0}^{n} c_{jr}^{(i)} t^{r} \frac{1}{\beta_{j}} H_{r}(x_{j})$$
.

where the unitless variable $x_j = \beta_j t$. The function $H_r(x)$ is defined as

$$H_{r}(x) = \frac{1}{x^{r}} \int_{0}^{x} e^{-x'} x'^{r} dx'$$

Radcalc for Windows uses two forms to evaluate the above integral. For $x\,<\,1,$

$$H_r(x) = e^{-x} \sum_{n=0}^{\infty} \frac{r! x^n}{(r+n)!}$$

For $x \ge 1$,

$$H_{r}(x) = \frac{r!}{x^{r}} - e^{-x} \sum_{n=0}^{r} \frac{r!}{(r-n)!x^{n}} .$$

As for the v_i , the d_i are summed over all decay chain branches to find the total time-integrated disintegrations of radionuclide i, designated D_i above.

2.6 TRANSPORTATION CLASSIFICATION DETERMINATIONS

Transportation classifications are made with the source decayed to the end of the user-specified decay time. All calculations are made assuming the user input values have three significant figures. Rounding errors may occur from input values with a greater number of significant figures. Definitions, quantity calculations, and the A_1/A_2 database used in Radcalc for Windows are taken from 49 CFR Subchapter C, "Hazardous Materials Regulations." This section contains the definitions and calculations used in Radcalc for Windows.

2.6.1 Definitions

Definitions are taken from 49 CFR 173.403.

"§173.403 Definitions

For purposes of this subpart-

"A, means the maximum activity of special form Class 7 (radioactive) material permitted in a Type A package.

 A_2 means the maximum activity of Class 7 (radioactive) material, other than special form, LSA, or surface contaminated object, permitted in a Type A package. These values are either listed in § 173.435 or derived in accordance with the procedure prescribed in § 173.433 . . .

Fissile material means plutonium-238, plutonium-239, plutonium-241, uranium-233, uranium-235, or any combination of these radionuclides. The definition does not apply to unirradiated natural uranium and depleted uranium . . .

Highway route controlled quantity means a quantity within a single package which exceeds:

(1) 3,000 times the A_1 value of the radionuclides as specified in §173.435 for special form Class 7 (radioactive) material;

(2) 3,000 times the A_2 value of the radionuclides as specified in §173.435 for normal form Class 7 (radioactive) material; or

(3) 1,000 TBq (27,000 Ci), whichever is least.

Limited quantity of Class 7 (radioactive) material means a quantity of Class 7 (radioactive) material not exceeding the materials package limits specified in §173.425 and conforming with requirements specified in §173.421.

Low Specific Activity (LSA) material means Class 7 (radioactive) material with limited specific activity which satisfies the descriptions and limits set forth below. Shielding materials surrounding the LSA material may not be considered in determining the estimated average specific activity of the package contents. LSA material must be in one of three groups:

(1) *LSA-I*

(i) Ores containing only naturally occurring radionuclides (e.g., uranium, thorium) and uranium or thorium concentrates of such ores; or

(ii) Solid unirradiated natural uranium or depleted uranium or natural thorium or their solid or liquid compounds or mixtures; or

(iii) Class 7 (radioactive) material, other than fissile material, for

which the A₂ value is unlimited; or (iv) Mill tailings, contaminated earth, concrete, rubble, other debris, and activated material in which the Class 7 (radioactive) material is essentially uniformly distributed and the average specific activity does not exceed 10⁻⁶ A_2/g . (2) LSA-II (i) Water with tritium concentration up to 0.8 TBq/liter

(20.0 Ci/liter); or

(ii) Material in which the Class 7 (radioactive) material is essentially uniformly distributed and the average specific activity does not exceed $10^{-4} A_2/g$ for solids and gases and $10^{-5} A_2/g$ for liquids.

(3) ISA-III. Solids (e.g., consolidated wastes, activated materials) that meet the requirements of §173.468 and which:

(i) The Class 7 (radioactive) material is essentially uniformly distributed throughout a solid or a collection of solid objects, or is essentially uniformly distributed in a solid compact binding agent (such as concrete, bitumen, ceramic, etc.); and

(ii) The Class 7 (radioactive) material is relatively insoluble, or it is intrinsically contained in a relatively insoluble material, so that, even under loss of packaging, the loss of Class 7 (radioactive) material per package by leaching when placed in water for seven days would not exceed $0.1 A_2$; and

(iii) The average specific activity of the solid does not exceed 2 x 10^{-3} A₂/g . . .

Normal form Class 7 (radioactive) material means Class 7 (radioactive) material which has not been demonstrated to qualify as "special form Class 7 (radioactive) material."

Package means, for Class 7 (radioactive) materials, the packaging together with its radioactive contents as presented for transport . . .

Packaging means, for Class 7 (radioactive) materials, the assembly of components necessary to ensure compliance with the packaging requirements of this subpart. It may consist of one or more receptacles, absorbent materials, spacing structures, thermal insulation, radiation shielding, service equipment for filling, emptying, venting and pressure relief, and devices for cooling or absorbing mechanical shocks. The conveyance, tie-down system, and auxiliary equipment may sometimes be designated as part of the packaging . . .

Radioactive material means any material having a specific activity greater than 70 Bq per gram (0.002 microcuries per gram) . . .

Special form Class 7 (radioactive) material means Class 7 for (radioactive) material which satisfies the following conditions:

(1) It is either a single solid piece or is contained in a sealed capsule that can be opened only by destroying the capsule;

(2) The piece or capsule has at least one dimension not less than 5 millimeters (0.2 inch); and

(3) It satisfies the test requirements of §173.469. Special form encapsulations designed in accordance with the requirements of §173.389(g) in effect on June 30, 1983 (see 49 CFR Part 173, revised as of October 1, 1982), and constructed prior to July 1, 1985 and special form encapsulations designed 3

in accordance with the requirements of §173.403 in effect on March 31, 1996 (see 49 CFR Part 173, revised as of October 1, 1996), and constructed prior to April 1, 1997, may continue to be used. Any other special form encapsulation must meet the requirements of this paragraph.

Specific activity of a radionuclide means the activity of the radionuclide per unit mass of that nuclide. The specific activity of a material in which the radionuclide is essentially uniformly distributed is the activity per unit mass of the material . . .

Type A quantity means a quantity of Class 7 (radioactive) material, the aggregate radioactivity which does not exceed A_1 for special form Class 7 (radioactive) material or A_2 for normal form Class 7 (radioactive) material, where A_1 and A_2 values are given in §173.435 or are determined in accordance with §173.433.

Type B quantity means a quantity of material greater than a Type A quantity \ldots ."

2.6.2 Specific Activity

The specific activity values in Radcalc are calculated internally in the program based on the half-life and atomic weight of the radionuclide. These values are used for all radionuclides in the Radcalc library with the exception of 233 U, 235 U, and all of the plutonium isotopes; i.e., 236 Pu through 244 Pu. For these radionuclides, the specific activity values from 49 CFR 173.435 are hard-wired into Radcalc. The 49 CFR values are used for these radionuclides to ensure that the quantity of fissile material calculated by Radcalc is consistent with the DOT regulations.

2.6.3 Radioactive Material

The content of a package is classified as radioactive material if the specific activity is greater than 0.002 μ Ci/g or 70 Bq/g. Radcalc for Windows divides the total activity including daughters in the material in the units input by the user by the total mass of material to evaluate this criteria.

2.6.4 Unity Fraction/Package Type Determination

The unity fraction is used to determine Type A or Type B classification. It is calculated as follows:

Unity Fraction =
$$\sum_{i} \frac{R_i}{A}$$

where:

- i = Index for radionuclides in the material
- $R_1 =$ The activity of radionuclide i in curies A = The A₁ value for the radionuclide for material designated as "special" or the A₂ value for material designated as "normal."

The unity fraction calculated by Radcalc for Windows is equal to the unity fraction defined in 49 CFR 173.433(d).

If the unity fraction is less than or equal to 1, the payload is defined as Type A. If the unity fraction is greater than 1, the content is a Type B package. Radcalc for Windows uses 49 CFR 173.433, Table 10, A_1/A_2 values for alpha emitters for mixed fission products (MFP). If MFP include only beta or gamma emitters, this may be too restrictive.

2.6.4.1 Daughter Products in Radcalc for Windows. Radcalc for Windows uses the criteria outlined in 49 CFR 173.433(c) for identifying daughter products in transportation classification calculations. These include Type A/B, effective A₂, limited quantity, LSA, HRCQ, and RQ determination. That is: a single radioactive decay chain is considered to be a single radionuclide when no daughter radionuclide has a half-life either longer than ten days or longer than that of the parent radionuclide. Radcalc for Windows has identified 47 radionuclides that are usually shipped as by-products of a parent radionuclide. Table 2-1 shows the radionuclides that Radcalc for Windows has identified as daughter products. If the radionuclides have undergone processing and are shipped in quantities other than that produced by natural decay, then the user can input a value for the daughter. If, however, the radionuclide quantity is a product of the natural decay of a parent isotope, the user can input a 30-day or longer decay time, and Radcalc for Windows will decay all products to their equilibrium quantities.

⁴⁷ Sc	⁶⁹ Zn	90Y	91mY	⁹⁵ ™Nb	^{99™} Tc	¹⁰³ Rh	¹⁰⁶ Rh
¹⁰⁸ Ag	^{113m} In	^{115m} In	¹²⁷ Te	¹²⁹ Te	¹³² I	¹³¹ Cs	^{137m} Ba
¹⁴⁰ La	¹⁴⁴ Pr	^{144m} Pr	¹⁴⁸ Pm	¹⁶⁶ Ho	²⁰⁷ T1	²⁰⁹ Pb	²¹¹ Pb
²¹² Pb	²¹⁴ Pb	²¹⁰ Bi	²¹¹ Bi	²¹² Bi	²¹³ Bi	²¹⁴ Bi	²¹³ Po
²¹⁴ Po	²¹⁵ Po	²¹⁸ Po	²¹⁷ At	²²² Rn	²²¹ Fr	²²³ Fr	²²⁴ Ra
²²⁵ Ac	²⁸⁸ Ac	²³¹ Th	^{234m} Pa	237 _U	²³⁹ Np	²⁴² Am	

Table 2-1. Radionuclides Identified as Daughter Products in Radcalc for Windows.

NOTE: Windows is a trademark of Microsoft Corporation.

For instance, in the case of 90 Sr and 90 Y, the user can input a value for 90 Sr and input a 30-day decay time, and the program will automatically calculate the equilibrium value for 90 Y. If, however, 90 Y is to be shipped as a medical isotope, the user can input this value directly, ignoring the

"daughter" label next to the radionuclide. In most instances the user should input parent radionuclides and a 30-day (or longer) decay time and allow Radcalc for Windows to bring the radionuclides into a state of equilibrium.

2.6.5 Effective A₂

.

The effective A₂ value is calculated as:

<u>Total Activity</u> Unity Faction

where the Unity Fraction is calculated as given previously in Section 2.6.4. The effective A_2 is referred to as the " A_2 for mixture" in 49 CFR 173.433.

2.6.6 LSA Determination

The total specific activity must be less than $10^{-6} A_2/g$ for a package to be eligible for LSA-I. The limit for LSA-II is $10^{-4} A_2/g$ for solids and gases and $10^{-5} A_2/g$ for liquids. Water with tritium concentration is LSA-II if it has an activity less than 0.8 TBq/L (20.0 Ci/L). The LSA-III limit, which applies only to solids, is 0.002 A_2/g . For mixed radionuclides, the sum of the ratios of specific activities of each radionuclide to their respective limits must be less than one. It is the responsibility of the user to verify that all other conditions for LSA are met as specified in 49 CFR 173.403 and 49 CFR 173.427, including the passage of a leaching test for LSA-III.

2.6.7 Limited Quantity Determination

The following table shows the activity limits for limited quantities as used in Radcalc for Windows.

A mixture of radionuclides is considered a limited quantity when the sum of the ratios of activities of each radionuclide to its corresponding limit as indicated in Table 2-2 does not exceed one.

Nature of contents	Material package limits
Solids: Special form Other form	$10^{-3} A_1 \\ 10^{-3} A_2$
Liquidstritium* < 0.0037 TBq/L (0.1 Ci/L) 0.0037 TBq to 0.037 TBq/L (0.1 Ci to 1.0 Ci/L) > 0.037 TBq/L (1.0 Ci/L)	37 TBq (1,000 Ci) 3.7 TBq (100 Ci) 0.037 TBq (100 Ci)
Liquidsother	10 ⁻⁴ A ₂
Gases: Tritium (special or normal form) Other radionuclidesspecial form Other radionuclidesnormal form	$2 \cdot 10^{-2} A_2$ $10^{-3} A_1$ $10^{-3} A_2$

Table 2-2. Activity Limits for Limited Quantities in Radcalc for Windows 2.0.

NOTE: Windows is a trademark of Microsoft Corporation.

*Tritium is always considered to be in tritiated water form in the calculation of limited quantity when the user selects liquid for physical form in Radcalc for Windows 2.0.

2.6.8 HRCQ Determination

A package contains an HRCQ if a given activity limit is exceeded. When the source is entered in curies, the limit is 3,000 times the A_1 or A_2 value or 27,000 Ci, whichever is lower. When the source is entered in becquerels, the limit is 3,000 times the A_1 or A_2 value or 1,000 TBq, whichever is lower. For mixed radionuclides, the quantity is considered an HRCQ if the sum of the ratios of the activity of each radionuclide to its most restrictive corresponding limit exceeds one. For example, consider a package containing the following two radionuclides.

Radionuclide	Activity (Ci)	A ₂	3000*A ₂	Limit (Ci)
²²⁸ Ra	1620	1.08	3240	3240
⁴¹ Ar	14,000	16.2	48,600	27,000

For 228 Ra, the ratio of activity/limit = 1620 Ci/3240 Ci, or 0.50. For 41 Ar, the corresponding ratio = 14,000 Ci/27,000 Ci, or 0.52. Summing these ratios gives 0.50 + 0.52 = 1.02, which exceeds one. Therefore this pakage is an HRCQ package.

2.6.9 Fissile Excepted Determination

The user should reference 49 CFR 173.453 for clarification of the fissile exception regulations. The following fissile exception determinations are made by Radcalc for Windows.

- Radioactive material is specified as fissile excepted under 49 CFR 173.453(b) if the total mass of the following isotopes is less than 15 g: ²³⁸Pu, ²³⁹Pu, ²⁴¹Pu, ²³³U, and ²³⁵U.
- Radioactive material is specified as fissile excepted under 49 CFR 173.453(d) if the total mass of $^{238}{\rm Pu}$, $^{239}{\rm Pu}$, $^{241}{\rm Pu}$, $^{233}{\rm U}$, and $^{235}{\rm U}$ is not greater than 5 g/10 L.
- Radioactive material is specified as fissile excepted under 49 CFR 173.453(e) if the mass of ²³⁹Pu and ²⁴¹Pu does not exceed 20% of the total mass of plutonium, which must also be less than 1 kg. This test is only applied when there is no more than 1 kg total of plutonium and there is no ²³³U or ²³⁵U present. In Radcalc for Windows 2.0, only the following plutonium isotopes are considered: ²³⁶Pu, ²³⁷Pu, ²³⁹Pu, ²⁴⁰Pu, ²⁴¹Pu, ²⁴²Pu, and ²⁴⁴Pu.

2.6.10 TRU Waste Determination

Radcalc for Windows specifies waste as TRU if the concentration of the following are greater than 100 nCi/g of the waste matrix: 233 U, radium sources (223 Ra and 226 Ra), and all alpha-emitting radionuclides with a Z number greater than 92 and half-lives greater than 20 years. These radionuclides are shown in Table 2-3. This classification is based on the criteria given in the Hanford Site Solid Waste Acceptance Criteria (Willis 1993).

²³³ Ra	²⁴⁰ Pu	²⁴³ Am
²²⁶ Ra	²⁴² Pu	²⁴³ Cm
²³³ U	²⁴⁴ Pu	²⁴⁵ Cm
²³⁷ Np	²⁴¹ Am	²⁴⁶ Cm
²³⁸ Pu	^{242m} Am	²⁴⁹ Cf
²³⁹ Pu		

Table 2-3.	Radionuclides	Accounted	as	Transuranic.
------------	---------------	-----------	----	--------------

2.6.11 RQ Determination

The unity fraction for RQ determination is calculated in Radcalc for Windows as the sum of the ratios of activities in curies of each radionuclide to their respective RQ limits in curies per 49 CFR 172.101. The calculation is made in becquerels if the source term is entered in becquerels by the user. Daughter products listed in Section 2.6.4.1 are not included in this calculation. If the RQ unity fraction is greater than or equal to one, the package is considered to contain an RQ.

2.7 UNIT CONVERSIONS

The user may input quantities in English units (e.g., inches, feet, pounds) or in units in the Systeme International (e.g., centimeters, meters, grams). Radcalc for Windows will display the input quantities in terms of the units used when entered, as well as in terms of centimeters and grams, which are the units used internally for the calculations. All unit conversion factors and physical constants; e.g., Avogadro's number, are standard and were taken from the student edition of the 72nd edition of the Handbook of Chemistry and Physics (Lide 1991).

2.8 SOURCE OPTIONS

Several special features are offered for source terms. Inventories are multiplied or divided by 3.7×10^{10} to convert between curies and becquerels. In addition, the source term can be multiplied by a user input value to scale all entries. Daughter products are calculated from the source input, and all radionuclide quantities greater than 1×10^{-15} Ci are reported. However, the user has the option to revise the cut-off value to edit the source term information printed.

2.9 PRIMARY RADCALC CALCULATION ROUTINES

ExecuteMain (execute.cpp)

When the user selects the "calculate" button, the program calls subroutine ExecuteMain found in source file execute.cpp. ExecuteMain first performs some variable initialization, including the conversion of user input quantities to units convenient for use in the calculations. ExecuteMain then calls subroutine decayitl for the decay calculation up until seal time and calculate1 to calculate quantities relevant at the beginning of shipment. If the user has specified time of the container sealed, ExecuteMain then calls decayit2 for the decay calculation while the container is sealed and calculate2 to calculate quantities relevant at the end of the shipment. If, on the other hand, the user has specified a target percent hydrogen, ExecuteMain will call the subroutine iterate.

iterate (execute.cpp)

When the user specifies a target hydrogen percent, iterate will make a series of iterations to find the seal time required to reach or come close to the target percent specified. The volume required to reach the specified percent is first calculated. For the first try, iterate chooses one year. Extrapolation is used for each successive guess based on up to two previous results. The iteration stops without errors when (1) the time to the target volume will be less than one hour, (2) the time to the target volume will be greater than 36,500 days, (3) ten iterations have been made, or (4) the result is found for a volume between 95 and 100% of the target volume.

decayit1 (decayit.cpp)

The decay calculations for the time period before the container is sealed are conducted by decayit1. Decayit1 calls the FORTRAN subroutine decay2, supplying conversions of the data, as necessary, before and after the call. Decayit1 returns the radionuclide inventory at the start of seal time.

decayit2 (decayit.cpp)

The decay calculations for the time period during which the container is sealed are conducted by decayit2. Decayit2 calls the FORTRAN subroutine decay2, supplying conversion of the data, as necessary, before and after the call. Decayit2 returns the radionuclide inventory at the end of seal time and the total number of disintegrations for each radionuclide during seal time.

calculatel (calc.cpp)

Quantities relevant at the beginning of shipment (seal time) are calculated by calculatel. These quantities are as follows:

- Decay heat
- Unity fraction for type determination
- Type determination
- Unity fraction for LSA determination
- LSA determination
- Fissile mass for fissile excepted determination
- Fissile excepted determination
- Limited quantity determination
- HRCQ determination
- Radioactivity determination
- RQ determination
- TRU waste determination.

calculate2 (calc.cpp)

Quantities relevant at the end of shipment (end of seal time) are calculated by calculate2. These quantities are:

- Hydrogen volume generated
- Percent hydrogen in the void volume
- Partial pressure of hydrogen
- Total pressure of hydrogen and air
- Total hydrogen generation rate (total hydrogen volume divided by total seal time).

<u>DECAY2</u> (decay2.for)

This routine receives a list of initial radionuclides in terms of initial curies and converts to gram-atoms. It then calls DKSolv, which returns a list

of final radionuclides, final gram-atoms, and final disintegrations for each radionuclide. DECAY2 converts the gram-atoms into curies before returning to the calling routine (decayit1 or decayit2).

Disin (disin.for)

Calculates number of disintegrations for each radionuclide.

<u>Evaln</u> (evaln.for)

Calculates amount of each radionuclide in terms of gram-atoms.

3.0 DATABASES AND INPUT PARAMETERS USED IN RADCALC FOR WINDOWS

Radcalc for Windows uses databases for hydrogen gas and heat generation calculations and for transportation classification determinations. The following sections discuss the databases and document their origins.

3.1 DATABASES AND INPUT PARAMETERS USED FOR HYDROGEN GAS GENERATION AND HEAT DECAY

The volume of generated hydrogen gas is calculated by summing the product of decay energy, absorption fraction, and G value over all radionuclides present and all decay types. Radcalc for Windows uses radionuclide information, calculated gamma absorption fractions for selected container types, and G values to complete these calculations. Decay heat is also dependent upon radionuclide information.

3.1.1 Radionuclide Databases

Radionuclide information for hydrogen gas generation and decay heat calculations is taken from the FENDL/D-1.0 $^{\circ}$ database. Decay calculations use the Oak Ridge Isotope Generation and Depletion Code (ORIGEN)2 $^{\circ}$ database.

3.1.1.1 FENDL/D-1.0 Database. Radcalc for Windows utilizes the FENDL/D-1.0 database. The FENDL/D-1.0 database is the Evaluated Nuclear Data File (ENDF)/B-VI decay database library supplemented by experimental data from the Evaluated Nuclear Data Structure Data File (ENDSDF). See Appendix B for more information on FENDL/D-1.0. The required data was extracted from FENDL using

⁴FENDL/D Version 1, January 1992, is a decay data library for fusion (and other) applications. Summary documentation by A. B. Pashchenko. Index No. IAEA-NDS-167 in <u>Index to the IAEA-NDS-Documentation</u> <u>Series</u>.

⁵Croff, A. G., 1980, <u>A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion</u> <u>Code</u>, ORNL-5621, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

a FORTRAN code and was reformatted for use in Radcalc for Windows. The following is a list of radionuclide parameters taken from FENDL/D-1.0 and the values they are used to calculate.

Radionuclide half-lives are used in calculating specific activity, which is used to determine LSA and fissile excepted classifications.

Average heavy particle, beta-type radiation, and gamma radiation energies per disintegration are used in decay heat and hydrogen gas generation calculations.

Discrete gamma energies and abundances are used in hydrogen gas generation calculations.

3.1.1.2 ORIGEN2 Database. Radcalc for Windows uses the ORIGEN2 database for decay calculations. ORIGEN2 is managed and distributed by Oak Ridge National Laboratory and is widely accepted and used in national laboratories and the nuclear industry. The decay algorithms calculate the activity of the user-specified source and daughter products over a specified period of time and the total number of disintegrations accumulated over this same time interval for each radionuclide. Parameters relevant to these calculations include atomic mass, atomic number, and state. These parameters are used for radionuclide identification and conversions. The decay constant and the branching ratios for decay modes are also used in the decay algorithms.

3.1.1.3 FENDL/D-1.0 Verification. The data extracted from FENDL/D-1.0 and used by Radcalc for Windows were checked against the ORIGEN2 database by comparing recoverable energy. The total recoverable energy for a disintegration of each radionuclide is calculated in Radcalc for Windows as follows:

$$E_{total} = E_{\alpha} + E_{\beta} + E_{\gamma}$$

 E_{α} is the average energy per decay for all heavy particles. E_{β} is the average recoverable energy per decay for all beta-type radiation. E_{γ} is the average energy per decay for all gamma radiations. For more detailed definitions of E_{α} , E_{β} , and E_{γ} , see Section 2.1.2. E_{α} , E_{β} , and E_{γ} values originate from the FENDL/D-1.0 database.

The total recoverable energy per disintegration for each radionuclide was calculated using the values found in FENDL/D-1.0 and the results compared to those reported in the ORIGEN2 decay library. Those cases with greater than 20% difference were also checked against Browne and Firestone's *Table of Radioactive Isotopes* (Browne and Firestone 1986). In most of these cases, the FENDL numbers agreed with Browne and Firestone, and the energy values were not changed. Those that agreed with neither ORIGEN2 nor Browne and Firestone (1986) are shown in Table 3-1. The radionuclides shown in Table 3-1 rarely occur, and the values from the FENDL database were used without alteration. However, the average beta energy for ¹³³I (not shown in Table 3-1) was found to be erroneous in the FENDL database and was replaced by the average beta energy reported in Browne and Firestone (410.4 keV).

	Recoverab	le energy	Ratio of	Decay	Energy from decay type		
Radionuclide	FENDL	OR I GEN2	FENDL	concern	FENDL	Browne and Firestone	
58m _{Co}	5.23 MeV	24.7 keV	212	Y	5.21 MeV	1.83 keV	
67 _{Cu}	6.08 MeV	271 keV	22.4	Y	5.91 MeV	115 keV	
103 _{Pd}	19.5 keV	64.5 keV	0.302	β	14.5 keV	42.5 keV	
131 _{Ba}	1.06 MeV	515 keV	0.485	Y	1.02 Mev	458 keV	
139 _{Ce}	687 keV	192 keV	3.58	Y	613 keV	160 keV	
¹⁹¹ 0s	125 keV	245 keV	0.510	β ⁻ Υ	85 keV 40 keV	131 keV 74.9 keV	
199 _{Au}	890 keV	231 keV	3.85	Y	747 keV	89.4 keV	
228 _{Ra}	9.37 keV	13.0 keV	0.721	β_	8.93 keV	No data	

Table 3-1.	FENDL	Radionuc	lides	: With	Total	Recoverab	le E	nergies
Differi	ng From	n ORIGEN2	and	Browne	and	Firestone	(198	6).

NOTE: Browne, E., and R. B. Firestone, 1986, <u>Table of Radioactive Isotopes</u>, John Wiley & Sons, Inc., New York, New York.

 β^{*} = Electron related radiation (β^{*} , β^{+} , electron conversion, . . .). γ = Electromagnetic radiation (gamma, x-ray, . . .). FENDL = Fusion Energy Nuclear Data Library, used by Radcalc for Windows.*

*Windows is a trademark of Microsoft Corporation.

3.1.2 Gamma Absorption Fraction Input Parameters

Radcalc for Windows uses the total energy emitted by heavy particle and beta-type decay in calculating the volume of hydrogen produced. However, only a percent of gamma energy will be absorbed in the package and waste. The original spreadsheet (Flaherty et al. 1986) developed to calculate hydrogen gas generation uses fitted curves for four container types to calculate the absorbed gamma dose in the waste material. The four container types are a 55-gal drum, a 4x4 liner, a 5x5 liner, and a 6x6 liner. The absorbed gamma dose is a function of energy, waste density, material type, and geometry. The spreadsheet uses curve fits for each of these containers for densities ranging from 0.6 to 2.0 g/cc and energies ranging from 0.4 to 2.0 MeV. The material inside the container is assumed to be water. The data for these fits were generated using the QAD-FN computer code (Flaherty et al. 1986).

For use in Radcalc for Windows, these same four curve fits have been recalculated using the Monte Carlo N-Particle (MCNP) transport code (Breismeister 1993, Carter 1994). Along with the four original containers included in the Radcalc spreadsheet, ten additional containers have been added. The new containers are used in the DOE complex for the transportation of radioactive waste materials.

3.1.2.1 Calculation of Absorbed Gamma Dose. The original spreadsheet calculates the fraction of gamma energy absorbed in the waste material, which is assumed to be water, as a function of energy, density, and geometry. An independent fit is used for each individual container geometry. Each of these fits is a function of density and energy.
The fit is of the form:

$$F(\rho, e) = a_1 \rho e + a_2 \rho \sqrt{e} + a_3 \sqrt{\rho} e + a_4 \rho + a_5 \sqrt{\rho e} + a_6 e + a_7 \sqrt{\rho} + a_8 \sqrt{e} + a_9$$
(3-1)

where $F(\rho, e)$ is the fraction of energy absorbed, ρ is density in g/cc, and e is the emitted energy in MeV.

For Radcalc for Windows new data were generated using a volumetric source of unity source strength (1 Bq/s) for each container. MCNP runs were made for energies varying from 0.4 to 5.0 MeV and densities ranging from 0.6 to 3.0 g/cc. A UNIX° version of Mathematica' (Wolfram Research, Inc. 1993) was used to perform least-squares fits from the MCNP data generated. The resulting coefficients for the 14 different geometries are contained in an array in the computer code. When calculating hydrogen gas generation, Radcalc for Windows calculates the gamma absorption fraction for the corresponding density and discrete gamma energy.

3.1.2.2 Container Geometry. The four original container geometries are reproduced with the same dimensions (Flaherty et al. 1986), and the 55-gal drum is reproduced with additional, more exact dimensions. The new container geometries are based on dimensions reported in their respective safety analysis reports for packaging (SARP). The model for each container has its interior volume completely filled with water, representing an upper bound in the calculation of the absorption fraction. Both English and metric units are given for each container's dimensions. The first set of numbers given represents the original units used for the container in the document referenced for dimensions. The interior and exterior dimensions and interior volume of each container are given in Table 3-2.

3.1.2.3 55-Gal Drum. Two different cylinder sizes were used in the analysis of the 55-gal drum. The first uses the same size source volume as the original QAD-FN calculations, with a diameter of 60 cm (23.62 in.) and a height of 90 cm (35.43 in.). The second model consists of a source volume with a diameter of 22.5 in. (114.4 cm) and a height of 33.25 in. (84.46 cm) (Kelly 1994). The source is encapsulated in 14 gauge steel (U.S. Standard, 0.0781 in./0.1984 cm) on top and bottom, and 18 gauge steel (U.S. Standard, 0.0500 in./0.1270 cm) on the side.

3.1.2.4 4x4 Liner. The model for the 4x4 liner is based on the dimensions used in creating the fit in the original spreadsheet. It consists only of the source volume, 120 cm (47.24 in.) in diameter and 120 cm (47.24 in.) tall.

⁶UNIX is a trademark of Novell, Inc.

⁷ Mathematica is a trademark of Wolfram Research, Inc.

Container	Geometry	Inside modeled dimensions (diameter x height)	Outside modeled dimensions	Internal volume
55-gal drum	Cylinder	22.5 in. x 33.25 in. (57.2 cm x 84.46 cm)	22.6 in. x 33.41 in. (57.40 cm x 84.86 cm)	7.65 ft ³ (2.17E5 cm ³)
4x4 liner	Cylinder	120 cm x 120 cm (47.24 in. x 47.24 in.)	NA	47.9 ft^3 (1.36E6 cm ³)
5x5 liner	Cylinder	140 cm x 140 cm (55.12 in. x 55.12 in.)	NA	76.1 ft^{3} (2.16E6 cm ³)
6x6 liner	Cylinder	180 cm x 180 cm (70.87 in. x 70.87 in.)	· NA	162 ft ³ (4.58E6 cm ³)
30-gal drum	Cylinder	18 in. x 28 in. (45.72 cm x 71.12 cm)	18.1 in. x 28.1 in. (45.97 cm x 71.37 cm)	4.12 ft ³ (1.17E5 cm ³)
85-gal drum	Cylinder	26 in. x 37.9 in. (66.04 cm x 96.27 cm)	26.13 in. x 38.03 in. (66.37 cm x 96.60 cm)	11.6 ft ³ (3.30E5 cm ³)
Doorstop sample carrier	Cylinder	4.5 in. x 5.625 in. (11.43 cm x 14.29 cm)	6.38 in. x 11.38 in. (16.21 cm x 28.91 cm)	0.0518 ft^3 (1,470 cm ³)
Ion exchange column	Cylinder	17.5 in. x 69 in. (44.45 cm x 175.3 cm)	18 in. x 69.5 in. (45.72 cm x 176.5 cm)	9.60 ft ³ (2.72E5 cm ³)
LR-56	Cylinder	140 cm x 320 cm (55.12 in. x 126.0 in.)	338 cm x 518 cm (133.1 in. x 203.9 in.)	174 ft^3 (4.93E6 cm ³)
Neutralized current acid waste	Cylinder	3.375 in. x 40.2 in. (8.573 cm x 102.1 cm)	3.75 in. x 48.88 in. (9.53 cm x 124.2 cm)	0.208 ft_{3}^{3} (5,890 cm ³)
Onsite transfer cask	Cylinder	2.375 in. x 42.75 in. (6.033 cm x 108.59 cm)	5.505 in. x 45.87 in. (13.98 cm x 116.51 cm)	0.110 ft^{3} (3,104 cm ³)
PAS-1	Cylinder	18 in. x 21.88 in. (45.72 cm x 55.58 cm)	32.5 in. x 40 in. (82.55 cm x 101.6 cm)	3.22 ft ³ (91,300 cm ³)
Sample Pig carrier	Cylinder	2.06 in. x 4.875 in. (5.23 cm x 12.38 cm)	6.71 in. x 9.275 in. (17.04 cm x 23.56 cm)	9.40E-3 ft ³ (266 cm ³)
Single pass fuel cask	Box	35.25 in. x 37.50 in. x 46.625 in. (89.54 cm x 95.25 cm x 118.4 cm)	57 in. x 59.25 in. x 69 in. (144.8 cm x 150.5 cm x 175.3 cm)	35.7 ft ³ (1.01E6 cm ³)

Table 3-2. Dimensions of Modeled Shipping Containers.

NOTE: The first dimensions shown for a container are in the original units of the referenced work.

3.1.2.5 5x5 Liner. The model for the 5x5 liner is based on the dimensions used in creating the fit in the original spreadsheet. It consists only of the source volume, 140 cm (55.12 in.) in diameter and 140 cm (55.12 in.) tall.

3.1.2.6 6x6 Liner. The model for the 6x6 liner is based on the dimensions used in creating the fit in the original spreadsheet. It consists only of the source volume, 180 cm (70.87 in.) in diameter and 180 cm (70.87 in.) tall.

3.1.2.7 30-Gal Drum. The model for the 30-gal drum is based on the DOT Specification 7A Type A Packaging (Kelly 1994). The interior of the 30-gal drum is modeled as 18 in. (45.72 cm) in diameter with a height of 28 in. (71.12 cm). The source volume is encapsulated in 18 gauge steel (U.S. Standard, 0.0500 in./0.1270 cm).

3.1.2.8 85-Gal Drum. The model for the 85-gal drum is also based on DOT 7A Type A Packaging Specification (Kelly 1994). The interior of the 85-gal drum is modeled with a diameter of 26 in. (66.04 cm) and a height of 37.9 in. (96.27 cm). The source volume is encapsulated in 16 gauge steel (U.S. Standard, 0.0625 in./0.1588 cm).

3.1.2.9 Doorstop Sample Carrier. The model for the doorstop sample carrier is based on the doorstop sample carrier SARP (WHC 1993b) and drawing H-2-32514 (WHC 1982). The doorstop interior has a diameter of 4.5 in. (11.43 cm) and a height of 5.625 in. (14.29 cm). The outside dimensions are 6.38-in. (16.21-cm) diameter and 11.38-in. (28.91-cm) height. The shielding material is steel.

3.1.2.10 Ion Exchange Column. The model for the ion exchange column is based on the Big Bertha SARP (WHC 1991). The Big Bertha is the package currently used to transport ion exchange columns on the Hanford Site, Richland, Washington. The ion exchange column is modeled as a 69.5-in.-(176.5-cm-)long (exterior) section of 18-in. schedule 10 pipe (18-in./45.72-cm outside diameter [OD]) capped on each end with a thickness equal to that of the sides (0.25 in./0.635 cm). The total interior length is 69 in. (175.3 cm).

3.1.2.11 LR-56. The model for the LR-56 is based on the *Packaging Design* Criteria for the LR-56 Cask System (WHC 1994a). The interior of the LR-56 is a cylinder 320 cm (126.0 in.) long with a diameter of 140 cm (55.12 in.). The primary containment vessel is made from 8-mm- (0.315-in.-) thick steel. There is a 20-mm (0.787-in.) air gap and a 6-mm (0.236-in.) steel secondary confinement vessel. The shielding consists of an additional 35 mm (1.38 in.) of lead and 30 mm (1.18 in.) of steel. There are also layers of wood and a final steel outer casing, which are left out of the model, as they would have negligible effect in the calculation.

3.1.2.12 Neutralized Current Acid Waste (NCAW) Cask. The model for NCAW cask is based on the SARP for the NCAW Packaging System (Khojandi 1992). The inner dimensions of the NCAW cask are a 3.375-in. (8.573-cm) diameter and a 40.2-in. (102.1-cm) height. The side shielding consists of a 3.75-in.- (9.53-cm-) OD pipe, which is 0.375 in. (0.953 cm) thick and is inside a 16-in. schedule 40 pipe (16-in./40.64-cm OD) with the annulus filled with lead. The top of the model uses a 2.68-in. (6.81-cm) steel plug and the bottom a 6-in. (15.24-cm) steel plug.

3.1.2.13 Onsite Transfer Cask (OTC). The model for the OTC is based on the SARP for the OTC (WHC 1992a). The inner dimensions of the OTC are a 2.375-in. (6.0325-cm) diameter and a 42.75-in. (108.59-cm) height. The radial shielding consists of a 0.065 in. (0.165 cm) steel shell, 1 in. (2.54 cm) of lead, and another 0.5 in. (1.27 cm) of steel. The bottom shielding consists of 0.065 in. (0.165 cm) of steel, 1.55 in. (3.94 cm) of lead, and another 0.5 in. (1.27 cm) of steel. The top is capped by 1 in. (2.54 cm) of steel.

3.1.2.14 PAS-1 Cask. The model for the PAS-1 cask is based on the *Packaging Design Criteria for PAS-1 Cask* (WHC 1993a). The interior of the PAS-1 cask has a height of 21.88 in. (55.58 cm) and a diameter of 18 in. (45.72 cm). The OD is 32.5 in. (82.55 cm) and the height is 40 in. (101.6 cm). There is 5.1 in. (12.95 cm) of lead shielding in the model. A layer of steel surrounding the interior volume is 1.5 in. (3.81 cm) thick on the top, 1 in. (2.54 cm) thick on the bottom, and 2.15 in. (5.46 cm) thick on the side. The steel on the side was calculated from the assumptions for the inside diameter (ID) and OD of the cask and the 5.1-in. (12.95-cm) lead shielding.

3.1.2.15 Sample Pig Carrier. The model for the sample Pig carrier came from the sample Pig SARP (WHC 1992b). The ID is 2.06 in. (5.23 cm), and the interior length is 4.875 in. (12.383 cm). The inner cavity is surrounded by 0.125-in. (0.318-cm) steel and 1.95-in. (4.95-cm) lead. There is also an outer steel cover 0.25 in. (0.635 cm) thick on the side and 0.125 in. (0.318 cm) thick on top and bottom.

3.1.2.16 Single Pass Fuel Cask. The model for the single pass fuel cask is based on the SARP for the cask (WHC 1993c). The single pass fuel cask is a box with outside dimensions of 57 in. (144.8 cm) x 59.25 in. (150.5 cm) x 69 in. (175.3 cm). The inner cavity is encased on the sides and bottom by a $\frac{1}{2}$ -in. (0.952-cm) steel inner shell, 10 in. (25.4 cm) of lead, and a $\frac{1}{2}$ -in. (1.27-cm) steel outer shell. The lid is made of 10 in. (25.4 cm) of lead encased in $\frac{3}{2}$ -in. (1.905-cm) steel. The outermost layer of steel was left out of the model, as it would not have a significant effect on the calculation.

3.1.2.17 Results of Gamma Absorption Fraction Analysis. The gamma absorption fraction analysis was originally presented in WHC-SD-TP-RPT-014 (WHC 1994b), and data for all figures and tables can be found in that document. The results presented in that work are discussed in the following text, and the corresponding figures and tables are available in Appendix C. Table 3-3 provides an index to the figures and tables found in Appendix C.

There are two 55-gal drum MCNP models. The first model, referred to as the simple model, is identical to the model used in the original QAD-FN work (Flaherty et al. 1986). In Appendix C, Figure C-1 shows a comparison of the data and fit generated in the original work to a parallel calculation using MCNP. The MCNP simple model gives up to a 5% higher absorption fraction for higher energies than QAD-FN, and up to 8% lower absorption fraction for lower energies. In general, however, the fits are similar enough so that the accuracy of absorbed dose calculations in Radcalc will not be altered. The data point at 1.5 g/cc, 2.0 MeV for the QAD-FN is apparently reported incorrectly in Flaherty et al. (1986).

Table 3-3. Index to	lables and drap	IS TUT EACH GE	ometry.
Container	Table of coefficients in fits	Graph of final fits	Graph comparing original and final fits
55-gal drum	Table C-1	Figure C-5	Figure C-4
4x4 liner	Table C-2	Figure C-7	Figure C-6
5x5 liner	Table C-3	Figure C-9	Figure C-8
6x6 liner	Table C-4	Figure C-11	Figure C-10
30-gal drum	Table C-5	Figure C-12	NA
85-gal drum	Table C-6	Figure C-13	NA
Doorstop sample carrier	Table C-7	Figure C-14	NA
Ion exchange column	Table C-8	Figure C-15	NA
LR-56	Table C-9	Figure C-16	NA
Neutralized current acid waste	Table C-10	Figure C-17	NA

lable 3-3. Index to lables and Graphs for Each Ge

Figure C-2 is a comparison of the MCNP simple model and the more detailed model. It shows the effects of modeling a smaller volume with shielding. The more detailed model gives absorption fractions that are up to 3% lower than the MCNP simple model, due to the reduced volume.

Table C-11

Table C-12

Table C-13

Table C-14

Figure C-18

Figure C-19

Figure C-20

Figure C-21

PAS-1 cask

Onsite transfer cask

Sample Pig carrier

Single pass fuel cask

Figure C-3 shows the difference between a curve fit to data points from densities up to 2.0 g/cc and a curve fit to data points for densities up to 3.0 g/cc. The figure shows data fits for three different energy levels. There is less than 1% difference in the range of 0.6 to 2.0 g/cc between the two curves. When the fit created from data up to 2.0 g/cc is extrapolated to 3.0 g/cc, there is a difference of as much as 4% between it and the line generated from actual MCNP data for densities between 2.0 and 3.0 g/cc. The curve fit to actual data points up to 3.0 g/cc is considered to be more accurate than the extrapolated curve and will be used in Radcalc for Windows.

Figure C-4 shows a comparison of the original spreadsheet fit and final detailed MCNP fits for three different energies. The discrepancy between the two models is greatest at low energies and low densities where the MCNP fit is lower by up to 6%.

For the 55-gal drum, fits were developed for energies between 0.4 and 2.0 MeV and energies between 2.0 and 5.0 MeV. Figure C-5 is a graph of the absorption fractions versus energy. It shows the final fit for densities

NA NA

NA NA between 0.6 and 3.0 g/cc and energies between 0.4 and 5.0 MeV. Table C-1 gives the original and final coefficients for the respective fits. The coefficients are the coefficients in the fit in the form of equation 3-3. The coefficients were developed using a least-squares fitting algorithm in Mathematica. Although there appears to be a large discrepancy in the fitting coefficients between those used in the original spreadsheet and those developed here, the fits themselves are quite good, as can be seen in Figures C-4 and C-5.

Figures C-6 through C-21 show final fits and comparisons for all other containers. Each container has a graph which shows the final fit from 0.6 to 3.0 g/cc and 0.4 to 5.0 MeV. For the containers already included in the original spreadsheet, graphs have been developed comparing the original fits to the final fits for densities between 0.6 and 2.0 g/cc and energies between 0.4 and 5.0 MeV. Table 3-3 lists the appropriate graphs and tables for each container. Tables C-1 through C-14 show the coefficients used for the fits. These coefficients are stored in an array in Radcalc for Windows and are accessed whenever absorption fractions are calculated.

3.1.3 G Value Data

The G value is defined as the number of molecules formed or disassociated in a medium per 100 eV of absorbed energy (molecules/100 eV). G values were extensively researched and a list of published conservative G values is presented in document WHC-SD-TP-RPT-014 (WHC 1994b). Radcalc for Windows uses a condensed version of that list. When maximum G values are cited for a material, the G value is used for all radiation types. When G values are missing for a radiation type, they are calculated on the basis of G beta = G gamma and G alpha = $4 \cdot G$ gamma. This is in keeping with G value information given in An Introduction to Radiation Chemistry (Spinks and Woods 1990).

The user can indicate a weight or fraction for selected materials and Radcalc for Windows will calculate a weighted-average G alpha, G beta, and G gamma. Alternatively, the user can input G values of their own choosing and combine library values and user values by indicating the fraction of each material present.

3.2 TRANSPORTATION DATA

Radcalc for Windows uses A_1 and A_2 values to determine Type A, Type B, limited quantity, low specific activity, and HRCQ classifications. The A_1 and A_2 values can be found in 49 CFR 173.435 table of A_1 and A_2 values for radionuclides as issued in the October 1, 1996, revised edition. The data used in the code was copied from an electronic version of the regulations. The user should note that the October 1, 1996, revised edition table of A_1 and A_2 values for radionuclides contains both TBq and Ci values. These values do not convert consistently. Therefore, Radcalc for Windows uses the original units provided by the user to calculate unity fractions and make classifications. All definitions used in the transportation algorithms are consistent with 49 CFR Subchapter C. Where necessary, clarification was received from the NRC.

4.0 VERIFICATION AND VALIDATION

4.1 SOFTWARE TEST PLAN: VERIFICATION CASES

Radcalc for Windows is accompanied by verification examples for benchmark problems to be run on a system following installation. The verification problems are available in the verify subdirectory of the floppy disk containing the Radcalc for Windows executable files. The test files are automatically installed in the verify subdirectory following successful completion of the Radcalc for Windows setup routine. The verification documentation of the benchmark problems is given in Appendix D. These verification problems test the various models incorporated in Radcalc for Windows and ensure the proper installation of Radcalc for Windows on a computer system. Following installation of Radcalc for Windows, running the verification examples should yield identical results as those shown in Appendix D, thus verifying the installation. See the User's Guide for more information on running the test files.

4.2 RADCALC FOR WINDOWS VALIDATION

Radcalc for Windows 2.01 modifications included the following:

- 1. The A_1 and A_2 values for MFP were changed in the RADCALC data file (DATA.BIN). The new values are the Table 10 values for alpha emitters. The MFP are now identified as "MFP alpha emitter" in the source dialog box and are listed as "MFP -a" in the RADCALC output files. The following CPP files were modified for these changes: SOURCEIO.CPP, EXTRAS.CPP, DOCWRADC.CPP.
- 2. The SOURCEIO.CPP file was modified to ensure that RADCALC reads the DATA.BIN file from the directory where the RADCALC executable file is located when the user requests a data file to be written. This corrected an error in the listing of the heat production factors in the user-requested data file.

These changes do not affect any calculational routines in RADCALC. The first change will only affect cases involving MFP, and the second change only impacts the heat production factors written to the user-requested data file. Therefore, validation of Radcalc for Windows Version 2.01 is addressed in the Version 2.0 validation documentation given in this section and in Appendix E.

Test cases were designed to exercise the various routines and models programmed into Radcalc for Windows. In Version 1.0 the test cases were divided into the four components incorporated in the code: (I) hydrogen gas generation; (II) decay calculations; (III) transportation classification determinations; and (IV) other aspects of the code, such as unit conversions and time algorithms. The tests were constructed to verify the performance of Radcalc for Windows.

Many of the algorithms developed for Version 1.0 remain unchanged in the subsequent versions. These algorithms were tested by running the cases shown in Table 4-1 in Version 2.0 and comparing them to the results given in

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Green et al. (1995), Volume II, for Version 1.0. Changes that were made in the code were validated with the cases shown in Table 4-2. Table 4-2 shows the test cases, a brief description of each case, the rad number, and the method of verification. Test cases are presented in Appendix E. Each test case includes (1) a discussion of the case and conclusion, (2) independent check results, and (3) Radcalc for Windows results.

Description	Case
Hydrogen gas generation	IA1 - IA12
Gamma absorption functions	IB1 - IB4
Use of G values	IC1 - IC5
Percent hydrogen	ID1 - ID2
Hydrogen generation rate	IE
Decay algorithms	IIA1 - IIA3
Integrated disintegrations over time	IIB1 - IIB3
Unit conversions	IVA1 - IVA3, IVA5
Target percent specified	IVB1 - IVB2
Dating routine	IVC1 - IVC3
Heat generation	IVE1
Pressure	IVF1 - IVF2

Table 4-1. Version 2.01 Cases Verified to Radcalc for Windows Version 1.0 Validation Cases.

NOTE: Windows is a trademark of Microsoft Corporation.

Table 4-2. New Radcalc for Windows Version 2.01 Validation Cases. (2 sheets)

Case		Description	Rad No.	Method of validation	
IIIA		Transportation calculations	3		
	1	Type classification for a normal form material	A1, B1	Excel spreadsheet	
	2	Type classification for a special form material	A3	Excel spreadsheet	
	3	Limited quantity classification for a special form material	LQ1, LQ3	Excel spreadsheet	
	4	Limited quantity classification for a normal form solid material	LQ4	Excel spreadsheet	
	5	Limited quantity classification for a liquid form material without tritium	LQ5	Excel spreadsheet	
	6	Limited quantity classification for a normal form liquid with tritium with a low specific activity (< 0.1 Ci/L)	LQ7, LQ10	Excel spreadsheet	

Table 4-2	New	Radcalc	for	Windows	Version	2.01
	Valida	tion Cas	es.	(2 shee	ts)	

Case		Description	Rad No.	Method of validation	
	7	Limited quantity classification for a normal form liquid with tritium with an intermediate specific activity (≥ 0.1 Ci/L and ≤ 1.0 Ci/L)	LQ8	Excel spreadsheet	
111	8	Limited quantity classification for a normal form liquid with tritium with a high specific activity (> 1.0 Ci/L)	LQ9	Excel spreadsheet	
	9	Limited quantity classification for a special form gas containing tritium	LQ19	Excel spreadsheet	
	10	Limited quantity classification for a normal form gas with no tritium	LQ28	Excel spreadsheet	
	11	Low specific activity (LSA) I (LSA-I) (≤ 10 ⁻⁶ A ₂ /g)	LSAI6, LSAI7	Excel spreadsheet	
	12	LSA-II for a normal form solid (≤ 10 ⁻⁴ A ₂ /g)	LSAII1	Excel spreadsheet	
	13	LSA-II for a normal form liquid (≤ 10 ⁻⁵ A ₂ /g)	LSAI18	Excel spreadsheet	
	14	LSA-III for a normal form solid (≤ 2.0 E-3 A ₂ /g) LSAIII1		Excel spreadsheet	
	15	HRCQ classification for a package in which activity = HRCQ1 27,000 Ci and < 3,000 A_2		Excel spreadsheet	
	16	HRCQ classification for a package in which activity < 27,000 Ci and > 3,000 A ₂	HRCQ3	Excel spreadsheet	
	17	HRCQ classification for a package requiring the most restrictive ratio selection	HRCQ21, HRCQ20	Hand calculations	
	18	Fissile excepted classification, \leq 15 g fissile material	FE1, FE3	Excel spreadsheet	
	19	Fissile excepted classification, 5 g/10 L or less (liquid)	FE4, FE6	Excel spreadsheet	
	20	Fissile excepted classification, \leq 20% ²³⁹ Pu and ²⁴¹ Pu in total Pu mass and Pu mass \leq 1,000 g	FE14, FE11, FE9	Excel spreadsheet	
	21	Reportable quantity classification		Excel spreadsheet	
	22	TRU waste classification (> 100 µCi/g)	TRU1	Excel spreadsheet	
	23	Radioactive material classification (specific activity $R1$, > 0.002 μ Ci/g)		Excel spreadsheet	
I۷		Other Radcalc features			
A		Unit conversions			
	6	Becquerels to curies	IVa4a, IVa4b	Excel spreadsheet	
	7	Ratio functions	R1	Excel spreadsheet	

HRCQ = Highway route controlled quantity. TRU = Transuranic. NOTE: Windows is a trademark of Microsoft Corporation.

*Output files archived as described in text.

5.0 CONFIGURATION CONTROL

Radcalc for Windows Version 2.01 was developed and is maintained by the Engineering Business Unit (EBU), Waste Management Federal Services, Inc., Northwest Operations (WMNW), Richland, Washington. The EBU provides technical support, software maintenance, and software updates for Radcalc for Windows. The EBU code custodian is responsible for initiating and completing all changes to Radcalc for Windows. The code custodian is also responsible for distributing new versions of Radcalc for Windows and for maintaining Radcalc for Windows files and records. Information and support for Radcalc for Windows is available by calling 509-376-0610 or 509-376-7111. Corrections and suggestions for Radcalc for Windows are welcomed.

5.1 CONFIGURATION MANAGEMENT PLAN

The following steps are recommended for software changes or identified problems.

- 1. Identify the changes that are requested or the problem in the program. Submit the change or problem to the code custodian in EBU on the form provided in Appendix F or in writing. The code custodian will keep copies of the submitted requests and a log of the requests which will include the final disposition of the request. The code custodian will also notify the requestor of the final disposition of the request. The code custodian will also notify the requestor of the final disposition of the request. The code custodian will also notify the requestor of the final disposition of the request. The code custodian will determine the extent of the change or problem and will obtain approval from the proper line of management to implement the change.
- Create a temporary version of Radcalc for Windows within a system test area, and alter this program according to the requested change.
- 3. Once the changes have been made, validate the new program by comparison with the original version and verification test cases.
- 4. Prepare documentation describing the change, and issue for approval.
- 5. Upon approval of the documentation, establish the temporary version as the official version by copying the appropriate files to new Radcalc for Windows compact disks (CD). Two CDs are stored in the WMNW records storage area, and a copy is maintained by the code custodian.
- Prepare back-ups of the new version of Radcalc for Windows per Section 5.3.

5.2 ARCHIVAL STORAGE AND RECORDS

The data and time stamp for the Radcalc for Windows Version 2.01 executable file is 03/18/98. The source and run files for Radcalc for Windows

are archived on two write-only CDs (EBU-CD-005) stored at records retention areas maintained by WMNW. A copy of the CD is also maintained by the code custodian.

Files needed to create disks with the setup utility are also archived on the CDs. The subdirectory runrad contains the files necessary for running the programming. These files are:

- radcalc.exe Radcalc.exe is the Windows executable. The Version 2.01 executable file is dated 3/18/98, 1:52 pm.
- radcalc.hlp Radcalc.hlp is the help file.
- data.bin Data.bin contains the data extracted from FENDL/D-1.0 as well as the transportation value information in binary format.
- glist.bin Glist.bin contains all G-value information in binary format.
- origen2.dk Origen2.dk is a binary formatted version of the ORIGEN2 decay database.
- verify/testn.rad Six test cases (n=1 through 6, as in test1.rad, test2.rad, etc.) are contained in the verify subdirectory to test the proper functioning of Radcalc for Windows. See Section 4.1 for further information.

Copies of the executable files can be obtained from the code custodian in EBU, WMNW, by calling 509-376-0610 or 509-376-7111.

Records for Radcalc for Windows will be located in the permanent files at WMNW.

5.3 BACKUP AND RECOVERY

Upon completion of a version of Radcalc for Windows, the code custodian will back up all files for the code and write them to two CDs for storage. The files are also contained in directory wradcalc and subdirectories source, setup, and runrad on the code custodian's computer.

5.4 DISTRIBUTION

The code custodian will maintain a record of all Radcalc for Windows users which will be filed with Radcalc for Windows documentation and program records. Upon completion of a new version of Radcalc for Windows all users will be notified of the availability of the new version.

Copies of Radcalc for Windows can be received by calling the Code Custodian at 509-376-0610 or Engineering Secretary at 509-376-7111.

6.0 REQUIREMENTS SPECIFICATION

Radcalc for Windows is written for applications in transportation and packaging. The requirements listed in this section are selected to provide guidelines appropriate to that environment. The requirements are designed to define the accepted level of performance and confirm that the code operates over the normal range of variables. The requirements are as follows:

- 1. Hydrogen gas calculation methodologies shall produce results consistent with accepted techniques as shown in Section 2.1 and 2.2.
- 2. Decay algorithms shall produce results consistent with accepted methodologies as shown in Section 2.5 and Appendix A.
- 3. Transportation quantity calculations shall produce results consistent with techniques outlined in 49 CFR Subchapter C.
- 4. Differences of a few percent or more between computed and benchmark results shall be justified.

Documented evidence that these requirements have been met is given in Chapter 4.0. This demonstrates that Radcalc for Windows produces correct results when used to analyze problems within a specific domain of applications.

7.0 SUMMARY

Radcalc for Windows is a user-friendly menu-driven Windows-compatible software with applications in the transportation of radioactive materials. It calculates the radiolytic generation of hydrogen gas in the matrix of lowlevel and high-level radioactive waste. It also calculates pressure build up due to hydrogen and the decay heat in a package at seal time. It computes the quantity of a radionuclide and its associated products for a given period of time. In addition the code categorizes shipment quantities as RQ, radioactive, Type A or Type B, limited quantity, LSA, HRCQ, and fissile excepted using DOT definitions and methodologies.

Radcalc for Windows has been extensively tested and validated. It uses NRC-accepted methodology for the calculation of the production of hydrogen gas. The radionuclide database is taken from the well-established FENDL/D 1.0 and ORIGEN2 databases. The transportation database and calculations are based directly on 49 CFR Subchapter C data and methodologies as contained in the October 1, 1996, revised edition. When used appropriately, Radcalc for Windows can be expected to give accurate and consistent results.

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APPENDIX A

A COMPUTER ALGORITHM FOR SOLVING THE BATEMAN DECAY EQUATIONS

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A COMPUTER ALGORITHM FOR SOLVING THE BATEMAN DECAY EQUATIONS

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ABSTRACT

Two features are combined to give a new algorithm for solving the Bateman decay equations. First, a hybrid representation for the solution is found that serves as the basis for a fast, general, and numerically stable method. Second, a recursive programming approach yields a convenient way to organize the calculation of complicated decay chains and conserve memory. A FORTRAN code was developed to test the method and illustrates the use of recursive methods in a nonrecursive language.

I. INTRODUCTION

The Bateman decay equations¹ arise in numerous problems in physics, chemistry, and other diverse fields. They represent a well-known example of a set of coupled first-order linear differential equations and have been studied for many years. Despite this intense scrutiny, simple solutions readily amenable to computer algorithms are often restricted to particularly simple classes of problems. This situation arises, in part, because these problems belong to the class of stiff differential equations that are especially susceptible to numerical errors, although closed-form solutions are readily obtainable.

To be specific, we consider equations of the form

$$\frac{dn_i}{dt} = \sum_{j \neq i} \alpha_{ij} n_j - \beta_i n_i$$
(1)

$$=\sum_{j}\alpha_{ij}n_{j}$$
(2)

where.

 $\alpha_{ii} = -\beta_i$.

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For nuclear transmutation and decay problems, n_i represents the amount of the i-th nuclide in a coupled system. The production rate of a daughter nuclide i by a parent j is given by the transfer or rate coefficients α_{ij} , while the destruction by either decay or transmutation is given by the constant B_i . This physical model will be followed for the purpose of discussion, although clearly the same set of equations can represent a wide variety of diverse phenomena.

An inhomogeneous, but constant, production rate is often added as a source term to these equations. However, such a term is readily included in the present set of equations by simply adding a fictitious nuclide with a unit amount, $n_k = 1$, that does not deplete $B_k = 0$. Then, a constant source S_i for any nuclide i can be generated by simply choosing a rate constant α_{ik} such that

$$\alpha_{ik} n_k = S_i . \tag{3}$$

As one might expect for such a ubiquitous set of equations, extensive literature exists for their solution. Only a few references are noted here, primarily to illustrate some of the possible approaches and their advantages and disadvantages. A typical textbook example with analytical solutions is given in Reference 2. However, these solutions are susceptible to a loss of numerical precision arising from cancellation of nearly equal terms. This problem is particularly manifest for decay chains with equal destruction constants β_i where terms involving $1/(\beta_i-\beta_i)$ are present.

A completely different approach is to numerically integrate the equations. This is a practical approach,³ but as noted, it requires special numerical techniques appropriate to stiff equations⁴ and may require significant amounts of computer time and memory. Nevertheless, good numerical packages are widely available,⁵ and this may be the option of choice in some cases. The difficulty arises in problems with widely varying system time constants. For example, in the nuclear decay problem, nuclear half-lives can easily range from fractions of a second to millions of years.

Another well-known solution is the matrix-exponential result:

$$\mathbf{n} = \mathbf{e}^{\alpha \tau} \mathbf{n}_0 \tag{4}$$

where α is the transition matrix in Eq.(2), and the exponential propagates the vector of initial concentrations n to a later time t. The matrix exponential form works especially well for smaller times where the power series implied by the exponential in Eq.(4) converges rapidly. This method forms the basis of the highly successful ORIGEN⁶ and ORIGEN2⁷ nuclear decay and transmutation codes. However, several special techniques are required to treat cases where convergence is slow. This form is also the basis for matrix-based approaches. One such example is given by Lee et al.⁸ who developed a method using matrix norms to obtain numerically dependable results. This approach taken by Lee et al. is still moderately complicated.

England takes another approach in the CINDER⁹ code by using Laplace transforms to obtain a solution. The CINDER code is well documented and provides a practical solution to a significant class of problems. It is somewhat less general than the ORIGEN algorithm and again requires attention to special cases to control numerical errors.

Other studies include recent work by Raykien and Shlyakhter¹⁰ who focus on the problem of time-dependent rate constants, an aspect not considered here and Miles¹¹ who provides an example of how equal destruction constants can be dealt with. Reference 3, a comprehensive review of nuclear decay heat, also includes a brief survey of calculational methods along with additional references.

Computer memory requirements can also pose problems. As in nuclear transmutation, the fission process can initiate hundreds of coupled chains. Thus, unless special care is taken, or the problem is restricted, memory requirements may become large.

In summary, although many practical numerical solutions are available, they often require restricting the problem to particular classes of problems, or they involve complex numerical methods problems, or they are of questionable precision. In the next section, a general algorithm is developed for solving Eq.(1), and the results are applied to a specific problem in Section III.

II. A GENERAL DECAY ALGORITHM

The algorithm given here is based on standard methods. It is fast, numerically stable, and easily implemented. As a side note, it provides a simple example of the power of recursive programming and its implementation in FORTRAN. The method may be conveniently divided into two parts. In the first part, a standard series development is summarized; it is particularly appropriate to a recursive programming approach, and it has the additional merit of having a direct physical interpretation. In the second part, the development focuses on the explicit calculation of one single step in the decay chain.

A. <u>Series Development for Nuclide Amounts</u>

In the usual way, we start by multiplying Eq.(2) with the integrating factor $e^{\beta_i t}$ to obtain

$$\frac{\mathrm{dN}_{i}}{\mathrm{dt}} = \sum_{i \neq j} \alpha_{ij} \mathrm{e}^{\delta_{ij} \mathrm{t}} \mathrm{N}_{j} , \qquad (5)$$

where

$$\delta_{ij} = \beta_i - \beta_j \tag{6}$$

and

$$N_{i} = e^{\beta_{i}t} n_{i} . \qquad (7)$$

(Note: δ_{ij} designates a difference and not the Kronecker delta function.) Integrating Eq.(5) then gives

$$N_{i} = N_{i0} + \sum_{j \neq i} S_{ij} N_{j}$$
 (8)

where the operator S_{ii} is defined by

$$S_{ij} = \alpha_{ij} \theta_{ij} \tag{9}$$

with

$$\theta_{ij}(t)f = \int_{0}^{t} e^{\delta_{ij}t'}f(t')dt'$$
(10)

Eq.(8) may then be iterated to obtain

$$N_{i} = n_{i0} + \sum_{j \neq i} S_{ij} n_{j0} + \sum_{j \neq i} \sum_{k \neq j} S_{ij} S_{jk} n_{k0} + \dots$$
(11)

Convergence of this series is not formally explored here, but based on physical grounds as well as the explicit implementation discussed later, it is not an issue. Clearly, $S_{ij}n_{jp}$ represents the contribution of the transmutation of parent j to daughter i. Repeated applications of the operator S_{ij} give successive daughters in the decay chain. Thus, the terms in Eq.(11) represent one-step decays, two-step decays, and so forth. Convergence is ensured as long as physically reasonable values are chosen for $\{B_i\}$ and $\{\alpha_{ij}\}$. Specifically, these parameters should be positive, and the sum of α_{ij} for a particular parent i should not exceed the destruction, B_i , of the parent, thus ensuring that no net nuclei are generated.

From the physical interpretation as well as from Eq.(10), all terms in Eq.(11) are positive. This ensures that, at least for this step, numerical cancellation of terms is not possible. In contrast, iteration of Eq.(2) after integration would yield negative terms because α_{ij} is negative for i=j, unlike the case for i≠j.

The simple physical picture of branching chains of daughter nuclides described by Eq.(11) lends itself readily to a recursive algorithm. One

starts with an initial amount n_{j_0} of nuclide j. Next, a daughter i is selected, and the amount $S_{i\,j}n_{j_0}$ of daughter i arising from the parent j is calculated.

Successive decays and transmutations are followed for a linear chain until some termination rule is achieved, or the final nuclide is either stable or sufficiently small. A generic implementation is shown here, where a recursive procedure TRANSMUTE is used to calculate the daughter amounts from the parent.

```
Main Program
       Global (N,)
       Set N_i = 0 (for all i)
       For i := 1 to Nlist
             T := n_{i0}
             Call TRANSMUTE(i,T)
       End For-loop
      n_i := e^{-\beta_i t} N_i
                        (for all i)
End MAIN
Procedure TRANSMUTE(j, T;)
      N_i := N_i + T_i
      If T<sub>j</sub> > eps Then
    For i := 1 to Nlist
                                         (Termination rule)
                  If (\alpha_{ii} \neq 0) Then
                        T_i := S_{ij}T_j
                        Call TRANSMUTE(i, T_i)
                  EndIf
            End For-loop
                                         (Also terminates with exhausted loop)
      EndIf
Return
```

In this illustration, $T_{\rm i}$ is the amount of daughter i arising from the amount $T_{\rm j}$ of a single immediate parent j according to the equation

$$T_i = S_{ij}T_j . (12)$$

The main program first initializes all nuclide amounts to zero. Then for each nuclide in the list of Nlist nuclides, T is set to the initial amount n_{io} , and the TRANSMUTE procedure is called to start the recursive process. The TRANSMUTE procedure first adds the parent amount T_j to the total amount N_j ; then for each daughter i, the daughter amount T_i is calculated according to Eq.(12). Finally TRANSMUTE is recursively called to continue the decay chain calculation.

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maintain only the necessary data in memory. As noted earlier, the terms at this stage are positive, so losing precision from canceled terms is not a problem. And finally, the process has a clear physical meaning. The problem is now reduced to calculating the amount of daughter i from the parent j.

B. <u>Calculation of a Single Decay Chain Step</u>

This section focuses on a single step in the decay chain calculation. To emphasize this fact, we set the daughter and parent indices, i and j, to specific values, q and p, respectively. Equation (12) then becomes

$$T_{q} = S_{qp} T_{p} . \qquad (12')$$

The daughter concentration T_q may be represented as a power series in time. As noted for the matrix-exponential solution, this works well for small values of t, but it converges slowly for larger values. An alternative representation for T_q is as a sum over the exponential functions $\{e^{\sigma_q j^t}\}$ where the sum over j includes all parents j of daughter q as well as q itself. The leading $e^{\sigma_q^t}$ implicit in $e^{\sigma_q j^t}$ occurs because we are working in the transformed system [Eq.(7)]. The disadvantage in this approach is that it is possible for near-equal terms to cancel, thus leading to large numerical errors. As an alternative approach, we use the form

$$T_{q} = \sum_{j} e^{\delta_{qj} t} \sum_{r=0}^{n} c_{jr}^{(q)} t^{r}$$
(13)

which combines the power series and exponential forms in a non-unique representation. The coefficients $\{c_{jr}^{(q)}\}$ form a two-dimensional matrix representing the concentration T_q for the daughter nuclide q. We next show how the non-uniqueness can be used to an advantage to provide a numerically accurate solution. Because of the decisive role this form plays in controlling numerical errors, it is henceforth referred to as the standard form or representation. From Eqs. (9,10 and 12,13), we find

$$T_{q} = \alpha_{qp} \int_{0}^{t} e^{\delta_{qp} t'} T_{p} dt'$$

$$= \alpha_{qp} \int_{0}^{t} e^{\delta_{qp} t'} \left[\sum_{k} e^{\delta_{pk} t'} \sum_{s=0}^{\infty} c_{ks}^{(p)} t'^{s} \right] dt' \qquad (14)$$

$$= \alpha_{qp} \sum_{k} \sum_{s=0}^{\infty} c_{ks}^{(p)} \int_{0}^{t} e^{\delta_{qk} t'} t'^{s} dt'$$

In the Appendix, it is shown that the integral in Eq.(14) is again a series of exponentials multiplied by powers of t. Two separate forms, designated as Form A and Form B, are given, depending on whether $\delta_{qk}t$ is small or large, respectively. Accordingly, the sum over k can be split into two sums, one sum consisting of terms where the Form A result is used, and the other sum consisting of terms where the Form B result is used:

$$T_{q} = T_{q}^{(A)} + T_{q}^{(B)}$$
 (15)

C. Form A Results

For small δ_{ok} t, the Form A results give

$$\int_{0}^{t} e^{\delta_{qk}t'} t'^{s} dt' = e^{\delta_{qk}t} \sum_{r=s+1}^{\infty} \frac{s!}{r!} (-\delta_{qk})^{r-s-1} t^{r} .$$
 (16)

Substituting Eq.(16) into Eq.(14) and interchanging summation limits

$$\sum_{s=0}^{\infty} \sum_{r=s+1}^{\infty} \dashrightarrow \sum_{r=1}^{\infty} \sum_{s=0}^{r-1}$$

gives

$$T_{q}^{(A)} = \alpha_{qp} \sum_{k \in A} e^{\delta_{qk} t} \sum_{r=1}^{\infty} t^{r} \left\{ \sum_{s=0}^{r-1} \left(\frac{s!}{r!} \right) (-\delta_{qk})^{r-s-1} C_{ks}^{(p)} \right\}.$$
(17)

Thus, the standard form, Eq.(13), is retained. The coefficients $\{c_{kr}^{(q)}\}$ representing the daughter concentration T_q are now obtained from the coefficients $\{c_{ks}^{(p)}\}$ representing the parent concentration T_p by comparing this result with Eq.(13). The result is

$$a_{kr}^{(q)} = \frac{1}{r!} (\alpha_{qp} t) \sum_{s=0}^{r-1} s! (-\delta_{qk} t)^{r-s-1} a_{ks}^{(p)}$$
(18)

where the dimensionless constants

$$a_{kr}^{(q)} = c_{kr}^{(q)} t^{r}$$
 (19)

are introduced.

These coefficients may be computed easily and quickly from the starting conditions

$$a_{k0}^{(q)} = 0$$
, $a_{k1}^{(q)} = (\alpha_{qp}t)a_{k0}^{(p)}$ (20)

and the recursion relation

$$a_{kr}^{(q)} = -\frac{(\delta_{qk}t)}{r} a_{k,r-1}^{(q)} + \frac{(\alpha_{qp}t)}{r} a_{k,r-1}^{(p)}$$
(21)

 $(r=2,3,\ldots), k \in A.$

Note that the recursion relation picks up coefficients from both the parent p and the daughter q. Note also that k is on both sides of Equation (18). In other words, there is no k-mixing for the Form A method; each row k in the daughter matrix $\{a_{kr}^{(q)}\}$ is generated from the same row k in the parent representation $\{a_{ks}^{(p)}\}$. For each row k, Eq.(21) is applied repeatedly, increasing the column index r until the power series in Eq.(17) converges. The number of terms required depends on the row k and the particular daughter q. This largest r-index terminates the series and is denoted by L_s for later reference.

In practice this strongly convergent power series is readily terminated by requiring the ratio of the last term in the series to the series sum to be less than some small number ϵ_r .

D. Form B Results

For $\delta_{ok}t > (s+1)/e$, the Form B results from the Appendix are used:

$$\int_{0}^{t} e^{\delta_{qk}t'} t'^{s} dt' = \left(\frac{-1}{\delta_{qk}}\right)^{s+1} s! \left[1 - e^{\delta_{qk}t} \sum_{r=0}^{s} \frac{1}{r!} \left(-\delta_{qk}t\right)^{r}\right]$$
(22)

giving

$$T_{q}^{(B)} = \alpha_{qp} \sum_{k \in B} \sum_{s=0}^{\infty} C_{ks}^{(p)} \left\{ \left(\frac{-1}{\delta_{qk}} \right)^{s+1} s! \left[e^{\delta_{qq} t} - e^{\delta_{qk} t} \sum_{r=0}^{s} \frac{1}{r!} (-\delta_{qk} t)^{r} \right] \right\}$$
(23)

where the leading 1 in brackets in Eq.(22) was rewritten as $e^{\delta qq^t}$ to demonstrate the Standard form is also preserved by Form B. Now there are two terms that generate daughter coefficients $c_{k'r}^{(i)}$ from the parent coefficients $c_{ks}^{(p)}$. (Note the use of the subscripts k' and k to index rows in the daughter and parent matrices, respectively.) The second term in brackets in Eq.(23) is the same as before except for the summation limits and sign. Thus, for this term the rows k of the parent again generate the same rows k'=k of the daughter. Specifically, for k'=k, interchanging the summation limits in Eq.(23),

$$\sum_{B=0}^{L_s} \sum_{I=0}^{B} - \cdot \sum_{S=0}^{L_s} \sum_{B=I}^{L_s} ,$$

and comparing the results with the standard form, we have (in dimensionless form)

$$a_{kr}^{(q)} = -\frac{1}{r!} (\alpha_{qp} t) \sum_{s=r}^{L_s} s! (-\delta_{qk} t)^{r-s-1} a_{ks}^{(p)}$$
(24)

 $(r = 0, 1, ..., L_{e}), k \in B$

where L_s is the upper limit of the s-sum and is established by convergence of the power series in Eq.(17). (See discussion after Eq.(21).)

Unlike the Form A case, column 1 (r=0) in the $\{a_{kr}^{(q)}\}\$ matrix is now non-zero. Also, the simplest terms in Eq.(24) are for r=L_s, and we develop the downward recursion relation

$$a_{kr}^{(q)} = -\frac{1}{\delta_{qk}t} \left[(r+1) a_{k,r+1}^{(q)} - (\alpha_{qp}t) a_{k,r}^{(p)} \right]$$
(25)

$$(r = L_{s-1}, L_{s-2}, ..., 0), k \in B$$

with the starting condition

$$\mathbf{a}_{k,L_{s}}^{(q)} = \frac{\boldsymbol{\alpha}_{qp} t}{\boldsymbol{\delta}_{qk} t} \mathbf{a}_{k,L_{s}}^{(p)} \qquad (k \in B)$$
(26)

Considering the first term in brackets in Eq.(23) completes the calculation. Again, comparing this term to the standard form in Eq.(13) yields

$$a_{q,0}^{(q)} = (\alpha_{qp}t) \sum_{k \in B} \sum_{s=0}^{L_s} \frac{s!}{(-\delta_{qk}t)^{s+1}} a_{ks}^{(p)}$$
(27)

The k=q row and r=0 column of the daughter representation thus receives contributions from all of the Form B rows of the parent. This is the only part of the calculation that mixes exponentials (i.e., adds new rows to the daughter matrix). For a particular k, the first term in the sum in Eq.(27) is $- [(\alpha_{qp}t)/(\delta_{qk}t)]a_{k0}^{(p)}$, and the simple form of the remaining terms again allows a recursive calculation of the remaining terms in the s-sum.

In summary, the set of recursive relations, Eqs.(20,21 and 25-27), provide a fast and simple algorithm. Convergence is easily controlled, and finally all operations involve only algebraic operations. The latter property both speeds the calculation and reduces the need to control the size of large numbers. The only time an exponential is calculated is at the final step. Since the standard form for T_i contains the factor $e^{\delta_{ij}t}$ (see Eq.(13)), then to obtain n_i from N_i , only multiplication by $e^{-\beta_j t}$ is needed.

III. FORTRAN IMPLEMENTATION

A FORTRAN code was written to implement and test the algorithm. Most of the code is a straightforward implementation of the above equations. Since FORTRAN is a non-recursive language, a few comments may be required to describe the recursive programming. The method used combined the recursive procedure TRANSMUTE with the calling program so entry and exit from TRANSMUTE could be explicitly programmed. For the purpose of discussion, TRANSMUTE will continue to be referred to as if it were an independent code module. Specifically, the following steps were taken and could be readily applied to other recursive applications:

- 1. A recursion index, LEVEL, is defined that keeps track of the level of nesting. Starting with one, LEVEL is incremented by one each time TRANSMUTE is "called" and decremented each time TRANSMUTE "returns" to the calling program.
- 2. Next, each local variable in TRANSMUTE that must retain its value during the recursive calls is stored in an array with the added index LEVEL. A simple variable A would become A(LEVEL), while a dimensioned variable B(I) becomes B(I,LEVEL).

- 3. In general, to "call" the recursive procedure, one must define a variable representing the return address, say with a computed GOTO statement. Then one "jumps" to the start of the recursive segment with a simple GOTO statement.
- 4. "Entry" to TRANSMUTE starts by incrementing the level index, LEVEL = LEVEL + 1.
- 5. Finally TRANSMUTE is exited by LEVEL=LEVEL-1 followed by a "jump" (i.e., GOTO) to the return address established in step 3 above.

The free use of GOTO statements may be undesirable, but it does allow explicit control of the code flow. Other necessary compromises include the exclusion of IF THEN ELSE statements, where a return simulated by a GOTO statement would otherwise jump into the scope of the IF THEN ELSE structure. On the other hand, the ability to develop the code from a recursive viewpoint can easily offset these disadvantages. In particular, it provides a very convenient and automatic means of allocating and reusing memory locations throughout the recursive tree. Obviously, using the LEVEL index limits the level of recursion to the value of the LEVEL index; nevertheless, the total number of branches is unlimited.

One final note: in the present case, using a computed GOTO statement and a return address was avoided altogether. Since TRANSMUTE is "called" only from itself and the main calling program, a "return" to the calling program is made only when the LEVEL index decrements to 1; otherwise, TRANSMUTE "returns" to itself.

IV. TESTS AND APPLICATIONS

The decay of ²³⁸Pu is an excellent example to illustrate this method. The decay chain includes a long daughter chain of 13 nuclides with half-lives that vary from 200,000 years to less than a millisecond and terminating with the stable nuclide ²⁰⁶Pb. The nuclide chain and decay data are shown in Table I. The decay data are taken from the ORIGEN2⁶ decay libraries. Two of the nuclides shown, ²¹⁸At and ²¹⁰Tl, are short-side branches that are not part of the main decay chain and for which data are not in the ORIGEN2 library. They were both assumed to be stable for this calculation. Except for these two minor branches, the chain ends with the stable nuclide ²⁰⁶Pb. Table I also includes both the total parent destruction rate {B_j} and the transition rates {a_{ij}}. For each nuclide, the sum (usually a single term) of the a_{ij} equals B_j. All values of a_{ii} not explicitly given are zero.

Table II shows the results for a decay time of 1×10^8 seconds starting with an initial amount of one gram-atom of ²³⁸Pu. The results are first compared to those from an ORIGEN2 calculation. Although the amounts may vary by 27 orders of magnitude, the two calculations agree within a few percent. For the first four nuclides in the chain, the two calculations agree with the four digits of precision output by the ORIGEN2 code. The first discrepancy (1.4 percent) is for ²²²Rn, a short-lived nuclide that follows the long-lived ²²⁶Ra. The amount of ²²²Rn calculated by ORIGEN2 can be reproduced by assuming a constant value for the amount of the parent ²²⁶Ra. Assuming instead that the amount of ²²⁶Ra is proportional to t³ as one would expect for the third member of a long-lived chain, the discrepancy is readily accounted for. The next jump in the percent difference occurs for 210 Bi, another short-lived nuclide that follows a long-lived parent.

In order to compare the present method with straightforward numerical integration of the differential equations, a solver for stiff ordinary differential equations, LSODE⁵ was obtained from the National Energy Research Supercomputing Center. Written by A. C. Hindmarsh, LSODE solves a set of differential equations of the form

$$\frac{dy}{dt} = f(t, y). \tag{28}$$

where y is a vector function of t. LSODE allows the user to directly input the Jacobian for these equations, which in our case is particularly simple, namely the constant matrix α . Error control for LSODE is provided by two parameters, either of which may be a vector quantity; a relative tolerance RTOL, and an absolute tolerance ATOL. The solver then controls the local errors in y relative to a weighted error based on

$$EWT(i) = RTOL(i) * ABS(Y(i)) + ATOL(i)$$

A method flag was set to MF=21 appropriate for stiff equations with a user-supplied Jacobian.

Because of the wide range in magnitudes for the different nuclides, the absolute tolerance ATOL was set to zero. Since EWT(i) then becomes zero for Y(i)=0, this choice required that the initial zero values for Y(i) be reset to some small number. Both this number and a common value for the relative tolerance vector RTOL were selected by trial and error. The choice of a single value for all components RTOL(i) was made to achieve input parameters that did not anticipate the solution.

The results of the LSODE calculations are compared with the previous results in Table II. Most of the calculations were made on a personal computer class machine using 32 bits for single precision real numbers. Various values of RTOL and the small non-zero initial value of Y(i) were tested. Representative results are shown. For the PC-based calculations, the LSODE results are in near perfect agreement with the results of the present method for RTOL = 0.01. (Clearly the precision of the results is much smaller than the tolerance value RTOL.) By the time RTOL is increased to 1.0, the errors in the LSODE results become excessive. The amount of required CPU time is remarkably insensitive to RTOL, so that there is little reason not to pick a small value. For this case, the current method enjoys a CPU time advantage factor of about 23/0.16=144 over direct integration.

The PC-based LSODE results used a value 1×10^{-23} for the initial value of Y(i) whenever the true initial value was zero. This value was quite constrained since any value smaller than 1×10^{-26} led to numeric overflows, and a value much larger would be comparable to the magnitude of results for some nuclides. This problem was alleviated by going to a Cray-class machine with 64 bits of precision for real numbers. Several cases were run there as well. Results are shown for the case with RTOL = 0.01. It is interesting to

note that the deviations in the results as a function of RTOL were nearly independent of the precision of the computer. The Cray CPU time was a factor of 23/0.12 = 192 better than the PC-based CPU time. A rough estimate is that about a factor of two of this increased speed arises from vectorization.

Memory utilization is a significant advantage of the present method compared to some other methods. The example given here is too small to make a direct evaluation. Nevertheless, the general behavior is clear. Since the current method always follows one decay chain branch at a time, the data storage requirements become the requirements for a single step in the chain times the maximum length of a chain. Neither the number of chains nor nuclides is important. In a method that requires the full transition matrix to be given, the memory size will grow with the problem size, even with the use of sparse matrix techniques.

As noted earlier, a good example of a problem with a large number of nuclides with limited decay chain lengths is for a fissioning nuclide. An initial fission event creates several hundred fission products. Each of these decay in relatively short chains to stable nuclides thus terminating the respective chains.

A final comment in comparing the current algebraic approach with direct numerical integration concerns the generality of the method. The direct integration method undoubtedly solves a wider class of problems; timedependent rate problems are a good example. However, the scope of the present method can be expanded. For example, it was noted earlier that inhomogeneous source terms are already included in the present scheme. And problems with time-dependent rate constants can often be treated by the simple expedient of breaking the total time interval into smaller subintervals where the rate constants can be approximated by constants.

The algebraic algorithm was found to be very robust. Not only does this ensure good solutions, but it makes it easier to obtain results without the need to iterate or otherwise "fine-tune" the input parameters. The sole parameter affecting the numerical precision of the solution for this example is the convergence parameter ϵ_r controlling the convergence of the power series in Eq.(17) and described near the end of Section II.C. A value of $\epsilon_r = 0.0001$ used here led to a maximum number of 7 terms in the power series. With the value of ϵ_r increased to 0.01, the maximum series length was only reduced by one, and the solution was nearly identical. Likewise, a stronger convergence criterion of 1 x 10⁻⁶ increased the series length by only one with a concurrent increase in the CPU time of about 30%.

Examination of intermediate results for this example shows that, as expected, cancellation is also not a problem. For example, the matrix representation for ²⁰⁶Pb at the end of the chain is representative of the more complex intermediate terms and consists of several rows of both Form A and Form B calculations. Detailed printouts show that the final result is dominated by a few terms and that it is comparable in size to the largest. That is, no significant cancellation of large terms is noted. Other validity checks include tests with both very small (1 second and zero) and large (1 x 10^{20} seconds) times. In these cases, one obtains the expected results of either no decay or complete decay to the stable end nuclide, respectively. A small amount is diverted to the two offshoot branches. A more interesting example, in the sense that other methods often do not allow it, is when feedback to a previous parent occurs. A simple case that has an analytical solution is the following set of equations:

$$\frac{\mathrm{dn}_{1}}{\mathrm{dt}} = \alpha(n_{2} - n_{1})$$

$$\frac{\mathrm{dn}_{2}}{\mathrm{dt}} = \alpha(n_{1} - n_{2})$$
(28)

which can be readily integrated to give

$$n_{1} = \frac{1}{2}n_{10}(1 + e^{-2\alpha t})$$

$$n_{2} = \frac{1}{2}n_{10}(1 - e^{-2\alpha t})$$
(29)

where the initial amounts of n_1 and n_2 are n_{10} and zero, respectively. A calculation for $\alpha t = 1$ was compared to values obtained from Eq.(29) and gave the following values precise to six decimal places:

$$n_1 = 0.5676677$$

 $n_2 = 0.4323324$.

This result was obtained within 11 (about 5 per nuclide) levels of recursion. For smaller values of αt , the process converges faster. However, for larger values of αt , convergence is slower and can become a practical limitation. This limitation is on the number of recursive levels allowed in examples with feedback loops. It is not associated with the Form A and Form B series convergence discussed in Section II.

V. SUMMARY

A new algorithm for solving the Bateman decay equations was given. Two novel aspects provide a significant alternative to previous methods. First, using simulated recursive FORTRAN programming gives an effective means of following complicated decay branches and provides a nearly automatic method of memory management. Second, the basic decay equation algorithm depends on two sets of complementary algebraic recursion relations that can be used to span the large time scales that are frequently encountered in nuclear transmutation and other fields. These relations are simple, fast, and avoid the calculation of exponentials that are time-consuming and that complicate the control of large numbers.

Additional features of the new algorithm include the ability to treat weakly coupled feedback and the lack of a multitude of different algorithms to treat special cases. Examples include short-lived nuclides that follow longlived nuclides, and situations where several nuclides have identical or nearly identical destruction rates. The utility of the method was demonstrated with both practical and analytical examples.

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	Table	I. Input	Data for ²³⁶ Pu	Decay Chain Calcul	ation.
Nuclide	Parent index,j	Daughter index,i	Half-life	ßj	α,,
²³⁶ Pu ²³⁴ U ²³⁰ Th ²²⁶ Ra ²²² Rn ²¹⁸ Po	1 2 3 4 5 6	2 3 4 5 6 7	87.74y 2.45x10 ⁵ y 7.54x10 ⁴ y 1600y 3.8235d	2.503240 E-10 8.983245 E-14 2.852458 E-13 1.372841 E-11 2.097903 E-06	2.503240 E-10 8.983245 E-14 2.852458 E-13 1.372841 E-11 2.097903 E-06
²¹⁸ At * ²¹⁴ Pb ²¹⁴ Bi	6 7 8 9 9	9 10 11	~2.5s 26.8m 19.8m	3.787690 E-03 0 4.310617 E-04 5.805253 E-04	7.574379 E-07 3.786932 E-03 0 4.310617 E-04 5.804034 E-04
²¹⁴ Po ²¹⁰ Tl * ²¹⁰ Pb ²¹⁰ Bi ²¹⁰ Po ²⁰⁶ Pb #	10 11 12 13 14 15	12 13 14 15	163.7µs 1.30m 22.3y 5.01d 138.38d stable	4.218790 E+03 0 9.850039 E-10 1.600802 E-06 5.795545 E-08 0	1.219103 E-07 4.218790 E+03 0 9.850039 E-10 1.600802 E-06 5.795545 E-08 0

* Not in ORIGEN2 library # Stable

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Index	Brecent method	ORIGEN2(a)		LSODE(b) on personal computer				LSODE(b) on Cray	
THOEX	Fresent method			(RTOL = 0.01)		(RTOL = 1.0)		(RTOL = 0.01)	
1	9.752783 E-01	9.753 E-01	0.00	9.7528 E-01	0.00	9.7528 E-01	0.00	9.7528 E-01	0.00
2	2.472157 E-02	2.472 E-02	-0.01	2.4722 E-02	0.00	2.4722 E-02	0.00	2.4722 E-02	0.00
3	1.115024 E-07	1.115 E-07	0.00	1.1150 E-07	0.00	1.1146 E-07	-0.04	1.1150 E-07	0.00
4	1.062032 E-12	1.060 E-12	-0.19	1.0620 E-12	0.00	1.1063 E-12	4.17	1.0620 E-12	0.00
5	6.851578 E-18	6.950 E-18	1.44	6.8516 E-18	0.00	7.1410 E-18	4.22	6.8516 E-18	0.00
6	3,794882 E-21	3.849 E-21	1.43	3.7949 E-21	0.00	3.9552 E-21	4.22	3.7949 E-21	0.00
7	7.161564 E-20			7.1626 E-20	0.01	8.3012 E-20	15.91	7.1616 E-20	0.00
8	3.333614 E-20	3.382 E-20	1.45	3.3336 E-20	0.00	3.4745 E-20	4.23	3.3336 E-20	0.00
9	2.475206 E-20	2.511 E-20	1.45	2.4752 E-20	0.00	2.5798 E-20	4.23	2.4752 E-20	0.00
10	3.405283 E-27	3.455 E-27	1.46	3.4053 E-27	0.00	3.5492 E-27	4.23	3.4053 E-27	0.00
15	7.517920 E-20			7.5189 E-20	0.01	8.7144 E-20	15.92	7.5179 E-20	0.00
11	3.510102 E-16	3.558 E-16	1.36	3.5101 E-16	0.00	4.0527 E-16	15.46	3.5101 E-16	0.00
12	2.106933 E-19	2.190 E-19	3.94	2.1069 E-19	0.00	2.4387 E-19	15.75	2.1069 E-19	0.00
13	3.278992 E-18	3.402 E-18	3.75	3.2786 E-18	-0.01	4.0307 E-18	22.92	3.2789 E-18	0.00
14	3.421000 E-18	3.591 E-18	4.97	3.4209 E-18	0.00	5.2885 E-18	54.59	3.4210 E-18	0.00
CPU-sec	0.16			23.18		11.87	<u> </u>	0.12	

Table II. Calculated Number of Gram-Atoms in the Decay of 1 Gram-Atom of 238 Pu to t=1x10⁸ Seconds. The variation from the present method is given in percent following each case.

a. Ref.(6) b. Ref.(5)

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APPENDIX: INTEGRAL REPRESENTATIONS

Specific forms of the integral

 $I_{n}(t,a) = \int_{0}^{t} t'^{n} e^{at'} dt \qquad (A.1)$

are central to the present development and are summarized here. The indefinite integral is readily evaluated and may be represented as an exponential function multiplied by a truncated exponential series giving:

$$I_{n}(t,a) = \frac{(-1)^{n}n!}{a^{n+1}} [e^{at}F_{n}(-at) -1]$$
(A.2)

where

$$F_n(x) = 1 + x + \frac{x^2}{2} + \dots + \frac{x^n}{n!}$$
 (A.3)

Since

$$e^{at}F_{n}(-at) = 1 - e^{at}\sum_{k=n+1}^{n} \frac{1}{k!}(-at)^{k}$$
 (A.4)

then $I_n(t,a)$ has two complementary representations:

$$I_{n}(t,a) = \left(-\frac{1}{a}\right)^{n+1} n! \sum_{k=n+1}^{n} \frac{1}{k!} (-at)^{k}$$
 Form A
= $\left(-\frac{1}{a}\right)^{n+1} n! \left[1 - e^{at} \sum_{k=0}^{n} \frac{1}{k!} (-at)^{k}\right]$ Form B (A.5)

Form A is suitable when at is small and the series converges rapidly, while Form B is suitable for larger values of at such that the two terms in brackets do not significantly cancel. Note that $(1/a)^{n+1}$ in Form A is canceled by terms in the sum so that this expression is valid for an arbitrarily small a, including zero.

The appropriate range of at for using in Form A or Form B can be made more precise by considering the following form, which is representative of that part of Eq.(A.5) in brackets where cancellation is a concern:

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$$\Delta(\mathbf{x}) = (-1)^{n+1} \left[1 - e^{-x} \sum_{k=0}^{n} \frac{1}{k!} (-x)^{k} \right].$$
 (A.6)

By the Taylor remainder theorem, we have

$$\Delta(x) = \frac{1}{(n+1)!} x^{n+1} e^{(x-x^*)} , \qquad (A.7)$$

where

 $0 \le x^* \le x .$

Then the smallest value of $\Delta(x)$ occurs for $x^* = x$. Thus, we ensure no significant cancellation in form B as long as

$$\frac{1}{(n+1)!} x^{n+1} > 1 .$$
 (A.8)

On the other hand, since the lhs is the first term in the Form A series, a value less than 1 ensures good convergence of that series. Using Stirling's approximation $n!\approx e^{-n}n^n \sqrt{2\pi n}$ and ignoring the factor $\sqrt{2\pi n}$, condition (A.8) readily becomes x > (n+1)/e. In practical calculations this condition is not critical and x > n was also found to work. The calculations reported here used the x > (n+1)/e criterion. ----

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APPENDIX F

CHANGE REPORT AND PROBLEM REPORT FORM

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Change Report and Problem Report Form.

CR/PF	Number:	Rev.:	Date:	TPCN, WO:			
1.	Software/Document Ide	ntification (Name)	1	· · · · · · · · · · · · · · · · · · ·			
2.	Prepared by:		System Name:	System Name:			
3.	CR/PR Type: [] Chang	ge Request [] Probl	em Report Requested Completion Date:				
4.	Description:			L			
5.	Justification if Change Request						
6.	Submitter's Priority:		[] High	[] Medium	[] Low		
7.	Change Authority:		[] Accept	[] Modify	[] Reject		
			Defer Until:	<u> </u>			
8.	Assigned to:	Planned Release Date:					
9.	Solution Comments: Cost/Schedule Estimate:						
10.	Software/Documents Att	ached:		<u> </u>	·		
11.	Approvals indicate CR	is Complete or PR is R	Resolved.				
	Software Developer: Date:						
<u> </u>	Cognizant Manager:			Date:			
	CR or PR Preparer:			Date:			
	Other:	<u></u>		Date:			
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APPENDIX B

FENDL/D-1.0

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FENDL/D-1.0

The decay library of the Fusion Energy Nuclear Data Library (FENDL), designated FENDL/D-1.0, was submitted by Westinghouse Hanford Company. This file started with the Evaluated Nuclear Data File (ENDF)/B-VI decay data library and was supplemented by experimental data from the Evaluated Nuclear Data Structure File (ENDSF).

The ENDF/B-VI evaluations are full evaluations consisting of both experimental and theoretical data and meet the requirements of the ENDF/B-VI formats manual. Only experimental data are considered for ENDSF. Therefore, these data often lack information for various fields. The translation from ENDSF format into ENDF/B-VI format was accomplished by Brookhaven National Laboratory program RADLIST.

The decay library has data for over 2,800 isotopes. For over 1,200 of these isotopes, rather full data sets exist, including spectra for each radiation type. Only for those isotopes near particle lines of stability does the data set only contain half-life and decay mode information.

For more information, please contact Fred Mann, Fluor Daniel Northwest, Inc., 509-376-5728.

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APPENDIX C

RESULTS OF GAMMA ABSORPTION FRACTION ANALYSIS

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Figure C-2. Comparison of the Simple and Detailed MCNP 55-Gal Drum Models for Three Energies.

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Figure C-3. 55-Gal Drum Detailed MCNP Data--Extrapolated Versus MCNP Data Fits for Three Different Energies.



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Figure C-6. Comparison of 4x4 Liner Fits (QAD-FN) and Detailed MCNP Fits for Three Energies.

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Figure C-7. Absorption Fraction Versus Energy for 4x4 Liner as a Function of Density.



Figure C-8. Comparison of 5x5 Liner Fits (QAD-FN) and Detailed MCNP Fits for Three Energies.

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Figure C-10. Comparison of 6x6 Liner Fits (QAD-FN) and Detailed MCNP Fits for Three Energies.

Absorption Fraction

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Figure C-11. Absorption Fraction Versus Energy for 6x6 Liner as a Function of Density.

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Figure C-12. Absorption Fraction Versus Energy for 30-Gal Drum as a Function of Density.



Figure C-13. Absorption Fraction Versus Energy for 85-Gal Drum as a Function of Density.







Figure C-15. Absorption Fraction Versus Energy for Ion Exchange Column as a Function of Density.



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Figure C-17. Absorption Fraction Versus Energy for Neutralized Current Acid Waste Cask as a Function of Density.

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Figure C-18. Absorption Fraction Versus Energy for Onsite Transfer Cask as a Function of Density.



Figure C-19. Absorption Fraction Versus Energy for PAS-1 Cask as a Function of Density.



Figure C-20. Absorption Fraction Versus Energy for Sample Pig Carrier as a Function of Density.



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	Original spreadsheet fit (0.6 to 2.0 g/cm ³)	Fit for future implementation (0.6 to 3.0 g/cm ³)		
	0.4 to 2.0 MeV	0.4 to 2.0 MeV	2.0 to 5.0 MeV	
a ₁	0.388	0.00101	-0.00830	
a ₂	-0.579	0.102	0.147	
a3	-0.871	0.166	-0.0297	
a ₄	-0.245	-0.391	-0.435	
a ₅	1.354	-0.560	-0.200	
a ₆	0.485	-0.153	0.0526	
a ₇	0.954	1.53	1.42	
a ₈	-0.937	0.317	-0.119	
a ₉	3.302E-6	-0.494	-0.289	

Table C-1. Coefficients in Fits for the 55-Gal Drum.

Table C-2. Coefficients in Fits for the 4x4 Liner.

	Original spreadsheet fit (0.6 to 2.0 g/cm³)	Fit for future implementation (0.6 to 3.0 g/cm ³)		
	0.4 to 2.0 MeV	0.4 to 2.0 MeV	2.0 to 5.0 MeV	
a,	-0.0430	-0.2519	0.0467	
a ₂	0.1060	0.4805	-0.1013	
a3	0.1117	0.7487	-0.1710	
a ₄	-0.5176	-0.5124	-0.2876	
a _s	-0.1340	-1.3558	0.5118	
a ₆	-0.0547	-0.4187	0.1199	
a ₇	1.3710	1.6326	0.8330	
a ₈	-0.1371	0.6262	-0.5185	
a ₉	-9.974E-07	-0.2683	0.2719	

'HNF-2549, Rev. 0

	Original spreadsheet fit (0.6 to 2.0 g/cm ³)	Fit for future implementation (0.6 to 3.0 g/cm ³)		
	0.4 to 2.0 MeV	0.4 to 2.0 MeV	2.0 to 5.0 MeV	
a ₁	-0.0977	-0.2248	0.0577	
a ₂	0.1972	0.3926	-0.1701	
a ₃	0.2759	0.6650	-0.1984	
a ₄	-0.5433	-0.4363	-0.2054	
a ₅	-0.4227	-1.1072	0.6970	
a ₆	-0.1576	-0.3517	0.1290	
a ₇	1.4278	1.4005	0.5762	
a ₈	0.0558	0.4535	-0.6044	
a ₉	-1.554E-06	-0.0773	0.4572	

Table C-3. Coefficients in Fits for the 5x5 Liner.

Table C-4. Coefficients in Fits for the 6x6 Liner.

	Original spreadsheet fit (0.6 to 2.0 g/cm ³)	Fit for future implementation (0.6 to 3.0 g/cm ³)		
	0.4 to 2.0 MeV	0.4 to 2.0 MeV	2.0 to 5.0 MeV	
a ₁	-0.3207	-0.1469	0.0707	
a ₂	0.5913	0.2187	-0.2684	
a ₃	0.7477	0.4192	-0.2212	
a ₄	-0.6337	-0.3129	-0.0598	
a ₅	-1.2520	-0.5743	0.9229	
a ₆	-0.4018	-0.1635	0.1318	
a ₇	1.5491	1.0011	0.1662	
a ₈	0.5016	0.0666	-0.6965	
a ₉	-2.947E-06	0.2549	0.7427	

	Fit for implem (0.6 to	r future entation 3.0 g/cm ³)		Fit fo implem (0.6 to	r future entation 3.0 g/cm ³)
	0.4 to 2.0 MeV	2.0 to 5.0 MeV		0.4 to 2.0 MeV	2.0 to 5.0 MeV
a ₁	0.0757	-0.0381	a ₆	-0.0337	0.0073
a ₂	-0.0134	0.2453	a ₇	1.3683	1.5199
a3	-0.0470	0.0507	a ₈	0.1607	0.0640
a ₄	-0.3161	-0.4544	a ₉	-0.4864	-0.4315
a ₅	-0.2457	-0.4911			

Table C-5. Coefficients in Fits for the 30-Gal Drum.

Table C-6. Coefficients in Fits for the 85-Gal Drum.

	Fit for implem (0.6 to	r future entation 3.0 g/cm ³)		Fit fo implem (0.6 to	Fit for future implementation 0.6 to 3.0 g/cm ³)	
	0.4 to 2.0 MeV	2.0 to 5.0 MeV		0.4 to 2.0 MeV	2.0 to 5.0 MeV	
a ₁	-0.0460	0.0029	a ₆	-0.1962	0.0688	
a ₂	0.1670	0.1039	a ₇	1.5664	1.3546	
a3	0.2751	-0.0612	a ₈	0.3380	-0.2045	
a ₄	-0.4231	-0.4307	a ₉	-0.4365	-0.1980	
a5	-0.6844	-0.0605		· · · · · · · · · · · · · · · · · · ·		

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	Fit for future implementation (0.6 to 3.0 g/cm ³)			Fit for future implementation (0.6 to 3.0 g/cm ³)	
	0.4 to 2.0 MeV	2.0 to 5.0 MeV		0.4 to 2.0 MeV	2.0 to 5.0 MeV
a ₁	0.0274	-0.0137	a ₆	0.0153	-0.0185
a ₂	-0.0545	0.0508	a ₇	0.2080	0.4095
a ₃	-0.0649	0.0493	a ₈	-0.0011	0.0983
a ₄	0.0670	0.0002	a ₉	-0.0914	-0.1644
a 5	0.0498	-0.2542			

Table C-7. Coefficients in Fits for the Doorstop Sample Carrier.

Table	C-8.	Coefficien	ts in	Fits	for	the
	Io	n Exchange	Colu	nn.		

	Fit fo implem (0.6 to	Fit for future implementation (0.6 to 3.0 g/cm ³)		Fit fo implem (0.6 to	or future mentation 3.0 g/cm ³)
	0.4 to 2.0 MeV	2.0 to 5.0 MeV		0.4 to 2.0 MeV	2.0 to 5.0 MeV
a,	0.0344	-0.0478	a ₆	-0.0672	-0.0044
a ₂	0.0526	0.2821	a ₇	1.4381	1.6473
a3	0.0323	0.0662	a ₈	0.1705	0.0819
a ₄	-0.3668	-0.5279	a ₉	-0.4383	-0.4401
a ₅	-0.3374	-0.5316			

	Fit for future implementation (0.6 to 3.0 g/cm ³)			Fit for impleme (0.6 to 3	r future entation 3.0 g/cm ³)
	0.4 to 2.0 MeV	2.0 to 5.0 MeV		0.4 to 2.0 MeV	2.0 to 5.0 MeV
a ₁	-0.1318	0.0237	a ₆	-0.2163	0.0828
a ₂	0.2193	-0.0776	a ₇	1.1119	0.5732
a3	0.4107	-0.1213	a ₈	0.2043	-0.4850
a ₄	-0.3461	-0.2369	a ₉	0.1441	0.5214
a ₅	-0.6290	0.5039			

Table C-9. Coefficients in Fits for the LR-56 Drum.

Table C-10. Coefficients in Fits for the Neutralized Current Acid Waste Drum.

	Fit for future implementation (0.6 to 3.0 g/cm ³)			Fit fo implem (0.6 to	r future entation 3.0 g/cm ³)
	0.4 to 2.0 MeV	2.0 to 5.0 MeV		0.4 to 2.0 MeV	2.0 to 5.0 MeV
a ₁	0.0418	-0.0051	a ₆	0.0458	-0.0046
a ₂	-0.0859	0.0152	a ₇	0.1171	0.2765
a ₃	-0.1131	0.0231	a ₈	-0.0697	0.0380
a ₄	0.0823	0.0333	a ₉	-0.0423	-0.0936
a ₅	0.1654	-0.1403			

	Fit for future implementation (0.6 to 3.0 g/cm ³)			Fit for future implementation (0.6 to 3.0 g/cm ³)	
	0.4 to 2.0 MeV	2.0 to 5.0 MeV		0.4 to 2.0 MeV	2.0 to 5.0 MeV
a ₁	0.0032	-0.0151	a ₆	0.0030	-0.0203
a ₂	-0.0122	0.0500	a ₇	0.1221	0.2972
a ₃	-0.0273	0.0464	a ₈	0.0091	0.0912
a ₄	0.0538	0.0024	a ₉	-0.0555	-0.1249
a ₅	0.0163	-0.2118			

Table C-11. Coefficients in Fits for the Onsite Transfer Cask Drum.

Table C-12. Coefficients in Fits for the PAS-1 Drum.

	Fit for future implementation (0.6 to 3.0 g/cm ³)			Fit fo implem (0.6 to	r future entation 3.0 g/cm ³)
	0.4 to 2.0 MeV	2.0 to 5.0 MeV		0.4 to 2.0 MeV	2.0 to 5.0 MeV
a ₁	0.0648	-0.0375	a ₆	-0.0284	-0.0206
a ₂	0.0051	0.2097	a ₇	1.1060	1.2686
a ₃	-0.0664	0.0858	a ₈	0.1916	0.1647
a ₄	-0.2162	-0.3014	a ₉	-0.4537	-0.4320
a ₅	-0.2378	-0.5672			

Table C-13. Coefficients in Fits for the Sample Pig Carrier.

	Fit for future implementation (0.6 to 3.0 g/cm ³)			Fit fo implem (0.6 to	r future entation 3.0 g/cm ³)
	0.4 to 2.0 MeV	2.0 to 5.0 MeV		0.4 to 2.0 MeV	2.0 to 5.0 MeV
a ₁	0.0030	-0.0011	a ₆	0.0036	-0.0074
a ₂	-0.0142	-0.0034	a ₇	0.0758	0.1443
a ₃	-0.0194	0.0181	a ₈	0.0069	0.0391
a ₄	0.0574	0.0508	a ₉	-0.0373	-0.0608
a ₅	0.0079	-0.0938		• • • •	<u></u>

	Fit fo implem (0.6 to	Fit for future implementation (0.6 to 3.0 g/cm ³)		Fit fo implem (0.6 to	r future entation 3.0 g/cm ³)
	0.4 to 2.0 MeV	2.0 to 5.0 MeV		0.4 to 2.0 MeV	2.0 to 5.0 MeV
a ₁	-0.0088	0.0415	a ₆	-0.1320	0.1022
a ₂	0.0062	-0.1201	a ₇	1.1491	0.7474
a3	0.1559	-0.1408	a ₈	0.0653	-0.5095
a ₄	-0.3120	-0.2337	a ₉	-0.0476	0.2967
a ₅	-0.2010	0.5028			

Table C-14. Coefficients in Fits for Single Pass Fuel Cask.

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APPENDIX D

INSTALLATION TEST CASES

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INSTALLATION TEST CASES

Six Radcalc for Windows¹ cases were developed that test proper installation of Radcalc for Windows. Below is a list of the cases and some of the features addressed by each test problem. Radcalc for Windows Input Information and Calculated Results sheets follow. Users should run the six test cases after installation of Radcalc for Windows. The test cases are located in the /verify subdirectory included with the Radcalc for Windows executable files. Identical results indicate proper installation of Radcalc for Windows. The output for two data files is also given. The data files are written for TEST2.RAD and TEST3.RAD. See Volume I, Section 2.0, for instructions on running the verification cases and saving data files.

- TEST1.RAD 30-gal drum, liquid waste. Simplest case, tests basic functionality. Decay time before sealing is entered in years, and seal time in days.
- TEST2.RAD PAS-1 Cask containing "normal" solid waste. Uses the ²⁴¹Pu decay chain (²⁴¹Pu, to ²⁴¹Am to ²³⁷Np to ²³³Pa ...) for a decay time of 200 years. This decay chain was chosen as a rigorous test of the decay calculations. Write the test case to a ".dat" file for the decayed source to check the decay algorithm. Tests unit conversions from feet and pounds to centimeters and grams. Waste void volume calculated from difference between bulk density and true density.
- TEST3.RAD Doorstop Sample Carrier containing "special" solid waste. Uses the ²⁴¹Pu decay chain for a decay time of two years. Write the test case to a ".dat" file for the decayed source to check the decay algorithm. Also tests the combination of G-values selected from the on-line database. Tests unit conversions from inches to centimeters. Tests the dating routine.
- TEST4.RAD Sample Pig Carrier (smallest container type) containing "special" solid waste. Considers only hydrogen generated from beta radiation. Tests use of "mixed waste". Tests use of Becquerels.
- TEST5.RAD NCAW cask containing "special" solid waste. Uses the iteration method for calculating to a specified target percent hydrogen. Uses ⁹⁰Y shipped as a medical isotope.
- TEST6.RAD LR-56 (largest container type) containing tritiated water. Uses the iteration method for calculating to a specified target percent hydrogen. Tests iteration technique for the case where the target percent is not reached within the maximum iteration time considered (100 years). Tests limited quantity calculation for tritiated water. Tests conversions from meters and kilograms to centimeters and grams. Tests the dating routine, including the final date calculated from the result of the iteration method.

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¹Windows is a trademark of Microsoft Corporation.

Date: 04-16-98 11:31 Radcalc for Windows 2.01 Performed By: _ Checked By: File: TEST1.RAD Source from input: Radionuclide: Curies: 1.00E+001 Cs-137 1.00E+001 Total Activity: Total Activity Minus Daughters: 1.00E+001 Waste Form: Normal Physical Form: Liquid 30 Gallon Drum Container Type: 7.00E+003 cc Package Void Volume: Waste Volume: 1.10E+005 cc Waste Mass: 1.00E+005 g Waste True Density: 1.00 g/cc Days to decay source before seal time: 365.24 days Davs container is sealed: 30.00 days G Value Material Selection: Contribution Weight G-Alpha **G-Beta** G-Gamma Name 100.00% 1.6 0.53 0.45 Water (liquid) G Values are calculated from list averaging: G Alpha G Beta G Gamma 1.6 0.53 0.45 Comments: HYDROGEN: H2 Percent Concentration: 0.543 % H2 Volume: 92.7 cc H2 Generation Rate: 0.129 cc/hour DECAY HEAT AND PRESSURE: 0.0472 Watts Heat Generated at seal time: Partial Pressure (H2): 0.553 kPa Total Pressure (H2 and Air): 102. kPa TRANSPORTATION: Note: Transportation classifications assume three significant figures. Calculations are made at user-specified decay time. Yes (1.90E+005 nCi/g) Radioactive: Effective A2 for Mixture: 13.5 Ci Type Determination: A (from unity fraction 0.72382) Limited Quantity: No No (from LSA-I unity fraction 7.2382) LSA-I Determination: *Yes (from LSA-II unity fraction 0.72382) LSA-II Determination: LSA-III Determination: Not Applicable *Per 49CFR173.403. The user must check to be sure that all other conditions for LSA are met as found in 49CFR173.403 and 49CFR173.427. HRC Quantity Determination: No 0.000 g Fissile Quantity: Fissile Excepted: NA There are no fissile radionuclides present per 49CFR173.403 TRU Waste: No (Transuranic Waste per Hanford Site WAC) Yes (from RQ unity fraction 9.7716) Reportable Quantity:

Source at start of seal time: Radionuclide: Curies: Cs-137 9.77E+000 Ba-137m 9.24E+000 (Daughter) Total Activity: 1.90E+001 Total Activity Minus Daughters: 9.77E+000

Shipping Papers and Labels:

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		Number of	Fraction of Total	Cumulative Total
	ISOTOPE	A2s	A2s	A2s
×	Cs-137 Ba-137m	7.24E-001 0.00E+000	1.00E+000 0.00E+000	1.00000

* Contains 95% of the total A2s and must be included per 49 CFR 173.433.

Date: 04-16-98 11:31 Radcalc for Windows 2.01 Performed By: __ Checked By: File: TEST2.RAD Source from input: Radionuclide: Curies: Pu-241 1.00E+000 1.00E+000 Total Activity: Total Activity Minus Daughters: 1.00E+000 Normal Waste Form: Physical Form: Solid Container Type: PAS-1 Cask Entered Values: Metric Values: 1.62 ft^3 Package Void Volume: 4.59E+004 cc 1.60 ft^3 Waste Volume: 4.53E+004 cc 9.07E+004 g 2.83E+003 cc 200. ЦЬ Waste Mass: Waste Void Volume: 0.100 ft^3 Days to decay source before seal time: 73050.00 days Days container is sealed: 73050.00 days Entered G Values: G Alpha G Beta G Gamma 1 1 Comments: HYDROGEN -18.7 % H2 Percent Concentration: 1.12E+004 cc H2 Volume: H2 Generation Rate: 0.00641 cc/hour DECAY HEAT AND PRESSURE: 0.000835 Watts Heat Generated at seal time: Partial Pressure (H2): 23.4 kPa Total Pressure (H2 and Air): 125. kPa TRANSPORTATION: Note: Transportation classifications assume three significant figures. Calculations are made at user-specified decay time. Radioactive: Yes (2.76E+002 nCi/g) Effective A2 for Mixture: 0.00542 Ci Type Determination: B (from unity fraction 4.6222) Limited Quantity: No No (from LSA-1 unity fraction 50.950) *Yes (from LSA-11 unity fraction 0.50950) *Yes (from LSA-111 unity fraction 0.025475) LSA-I Determination: LSA-II Determination: LSA-III Determination: LSA-III also requires a leach test. *Per 49CFR173.403. The user must check to be sure that all other conditions for LSA are met as found in 49CFR173.403 and 49CFR173.427. HRC Quantity Determination: No Fissile Quantity: 7.31E-007 g Fissile Excepted: Yes 15g fissile radionuclides or less per 49CFR173.453(a) 5g fissile radionuclides per 10 liters or less per 49CFR173.453(d) TRU Waste: Yes (Transuranic Waste per Hanford Site WAC) Reportable Quantity: Yes (from RQ unity fraction 2.5006)

Source	at start of seal	time:	
	Radionuclide:	Curies:	
	Pu-241	6.59E-005	
	Am-241	2.50E-002	
	Np-237	1.68E-006	
	Pa-233	1.68E-006	
	U-233	6.97E-010	
	Th-229	4.10E-012	
	Ra-225	4.10E-012	
	Ac-225	4.09E-012	(Daughter)
	Fr-221	4-09E-012	(Daughter)
	At-217	4.09E-012	(Daughter)
	Bi-213	4.09E-012	(Daughter)
	Po-213	4.00E-012	(Daughter)
	Pb-209	4.09E-012	(Daughter)
	U-237	1.62E-009	(Daughter)
			(
	Total Activity:	2.51E-002	
	Total Activity Mi	nus Daughters:	2.51E-002

Shipping Papers and Labels:

		Number	Fraction	Cumulative
		of	of Total	Total
	I SOTOPE	A2s	A2s	A2s
		•••••		•••••
*	Am-241	4.62E+000	1.00E+000	0.999880
	Np-237	3.10E-004	6.71E-005	0.999947
	Pu-241	2.44E-004	5.28E-005	1.00000
	Pa-233	6.90E-008	1.49E-008	1.00000
	U-233	2.58E-008	5.59E-009	1.00000
	Th-229	5.05E-009	1.09E-009	1.00000
	Ra-225	7.57E-012	1.64E-012	1.00000
	Ac-225	0.00E+000	0.00E+000	1.00000
	Fr-221	0.00E+000	0.00E+000	1.00000
	At-217	0.00E+000	0.00E+000	1.00000
	Bi-213	0.00E+000	0.00E+000	1.00000
	Po-213	0.00E+000	0.00E+000	1.00000
	Pb-209	0.00E+000	0.00E+000	1.00000
	U-237	0.00E+000	0.00E+000	1.00000

* Contains 95% of the total A2s and must be included per 49 CFR 173.433.

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Radcalc for Windows 2.01 Date: 04-16-98 11:37 Performed By: ____ Checked By: ____ File: TEST3.RAD Source from input: Radionuclide: Curies: Pu-241 1.00E+000 1.00E+000 Total Activity: Total Activity Minus Daughters: 1.00E+000 Waste Form: Special Solid Physical Form: Container Type: Doorstop Carrier Entered Values: Metric Values: 2.40 in^3 Package Void Volume: 39.3 cc 1.43E+003 cc 87.0 in^3 Waste Volume: Waste Mass: 3.00 Lb 1.36E+003 q Waste Void Volume: 1.00 in^3 16.4 cc Date to begin source decay: 8:00 Jul. 18, 1993 8:00 Jul. 18, 1995 8:00 Jul. 18, 1995 me: 730.00 days Date container sealed: Date container received: Days to decay source before seal time: Days container is sealed: 731.00 days G Value Material Selection: Contribution Weight G-Alpha G-Beta G-Gamma Name 30 0.4* 0.1* 30.00% 0.1 Dowex 50W 70.00% 70 0.36* 0.09* 0.09 Dowex 1 *Indicates the value is calculated from a given value G Values are calculated from list averaging: G Alpha G Beta G Gamma 0.372 0.093 0.093 Comments: HYDROGEN-H2 Percent Concentration: 14.0 % H2 Volume: 9.07 cc H2 Generation Rate: 0.000517 cc/hour DECAY HEAT AND PRESSURE: Heat Generated at seal time: 0.000131 Watts Partial Pressure (H2): 16.5 kPa Total Pressure (H2 and Air): 118. kPa TRANSPORTATION: Note: Transportation classifications assume three significant figures. Calculations are made at user-specified decay time. Radioactive: Yes (6.70E+005 nCi/g) Effective A1 for Mixture: 1.02E+003 Ci Type Determination: A (from unity fraction 0.00089738) Limited Quantity: Yes LSA-I Determination: Not Applicable LSA-II Determination: Not Applicable LSA-III Determination: Not Applicable HRC Quantity Determination: No Fissile Quantity: 0.00908 g Fissile Excepted: Yes 15g fissile radionuclides or less per 49CFR173.453(a) 5g fissile radionuclides per 10 liters or less per 49CFR173.453(d)

TRU Waste:

Yes (Transuranic Waste per Hanford Site WAC)

Reportable Quantity:

Yes (from RQ unity fraction 1.2133)

Source at start of seal time: Radionuclide: Curi

kaulonuclide:	curies:	
Pu-241	9.08E-001	
Am-241	3.05E-003	
Np-237	1.02E-009	
Pa-233	9.18E-010	
U-233	2.59E-015	
Th-229	1.14E-019	
U-237	2.23E-005	(Daughter)
Total Activity:	9.11E-001	
Total Activity Minus	Daughters:	9.11E-001

Shipping Papers and Labels:

	ISOTOPE	Number of A2s	Fraction of Total A2s	Cumulative Total A2s
	0	7 7/5.000	0 5/0 004	
~	Pu-241	3.30E+UUU	8.566-001	0.856434
*	Am-241	5.64E-001	1.44E-001	1.00000
	Np-237	1.88E-007	4.79E-008	1.00000
	Pa-233	3.78E-011	9.62E-012	1.00000
	U-233	9.58E-014	2.44E-014	1.00000
	Th-229	1.41E-016	3.59E-017	1.00000
	U-237	0.00E+000	0.00E+000	1.00000

* Contains 95% of the total A2s and must be included per 49 CFR 173.433.

Radcalc for Windows 2.01 Date: 04-16-98 11:39 Performed By: ____ Checked By: ___ File: TEST4.RAD Source from input: Radionuclide: Becquerels: Sr-90 1.00E+010 MFP - a 1.00E+010 2.00E+010 Total Activity: Total Activity Minus Daughters: 2.00E+010 Waste Form: Normal Solid Physical Form: Sample Pig Carrier Container Type: Package Void Volume: 1.00 cc 265. cc Waste Volume: 264. g Waste Mass: Waste True Density: 1.00 g/cc 30.00 days Days to decay source before seal time: Days container is sealed: 7.00 days Entered G Values: G Alpha G Beta 0 0.5 G Gamma Λ Comments: HYDROGEN: H2 Percent Concentration: 40.5 % H2 Volume: 1.36 cc H2 Generation Rate: 0.00811 cc/hour DECAY HEAT AND PRESSURE: 0.00181 Watts Heat Generated at seal time: Partial Pressure (H2): 69.0 kPa Total Pressure (H2 and Air): 170. kPa TRANSPORTATION: Note: Transportation classifications assume three significant figures. Calculations are made at user-specified decay time. Yes (1.13E+008 Bq/g) Radioactive: 4.00E+007 Bq Effective A2 for Mixture: Type Determination: B (from unity fraction 500.10) Limited Quantity: No No (from LSA-1 unity fraction 1.8943E+006) No (from LSA-II unity fraction 18943.) No (from LSA-III unity fraction 947.16) LSA-I Determination: LSA-II Determination: LSA-III Determination: HRC Quantity Determination: No Fissile Quantity: 0.000 g Fissile Excepted: NA There are no fissile radionuclides present per 49CFR173.403 TRU Waste: No (Transuranic Waste per Hanford Site WAC) Yes (from RQ unity fraction 2.6974) Reportable Quantity: Source at start of seal time: Radionuclide: Becauerels: 9.98E+009 Sr-90 Y-90 9.98E+009 (Daughter) 1.00E+010 MFP - a

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Total Activity: 3.00E+010 Total Activity Minus Daughters: 2.00E+010

Shipping Papers and Labels:

	ISOTOPE	Number of A2s	Fraction of Total A2s	Cumulative Total A2s		
*	MFP - a	5.00E+002	1.00E+000	0.999800		
	Sr-90	9.98E-002	2.00E-004	1.00000		
	Y-90	0.00E+000	0.00E+000	1.00000		

* Contains 95% of the total A2s and must be included per 49 CFR 173.433.

Radcalc for Windows 2.01 Date: 04-16-98 11:40 Performed By: _ Checked By: _ File: TEST5.RAD Source from input: Radionuclide: Curies: Y-90 1.00E+001 Medical Isotope 1.00E+001 Total Activity: Total Activity Minus Daughters: 0.00E+000 Waste Form: Normal Physical Form: Solid Container Type: NCAW 100. cc Package Void Volume: Waste Volume: 5.80E+003 cc Waste Mass: 5.70E+003 g Waste Void Volume: 100. cc Days to decay source before seal time: 0.00 days Calculate number of days sealed until 5.00% hydrogen is reached. Entered G Values: G Alpha G Beta G Gamma 1 Comments: HYDROGEN: The sealed package will contain 4.94 % hydrogen in 0.99 days. H2 Volume: 10.4 cc H2 Generation Rate: 0.438 cc/hour DECAY HEAT AND PRESSURE: Heat Generated at seal time: 0.0554 Watts Partial Pressure (H2): 5.27 kPa Total Pressure (H2 and Air): 107. kPa TRANSPORTATION: Note: Transportation classifications assume three significant figures. Calculations are made at user-specified decay time. Radioactive: Yes (1.75E+006 nCi/g) 0.000 Ci Effective A2 for Mixture: Type Determination: B (from unity fraction 1.8484) Limited Quantity: No LSA-I Determination: No (from LSA-I unity fraction 324.29) No (from LSA-II unity fraction 3.2429) LSA-II Determination: LSA-III Determination: *Yes (from LSA-III unity fraction 0.16214) LSA-III also requires a leach test. *Per 49CFR173.403. The user must check to be sure that all other conditions for LSA are met as found in 49CFR173.403 and 49CFR173.427. HRC Quantity Determination: No Fissile Quantity: 0.000 g Fissile Excepted: NA There are no fissile radionuclides present per 49CFR173.403 TRU Waste: No (Transuranic Waste per Hanford Site WAC) Reportable Quantity: Yes (from RQ unity fraction 1.0000) Source at start of seal time: Radionuclide: Curies: 1.00E+001 Medical Isotope Y-90

Total Activity: 1.00E+001 Total Activity Minus Daughters: 0.00E+000

Shipping Papers and Labels:

	ISOTOPE	Number of A2s	Fraction of Total A2s	Cumulative Total A2s	
*	Y-90	1.85E+000	1.00E+000	1.00000	

* Contains 95% of the total A2s and must be included per 49 CFR 173.433.

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Date: 04-16-98 11:40 Radcalc for Windows 2.01 Performed By: ____ Checked By: File: TEST6.RAD sussesses Input Information sussessessessessessessessessesses Source from input: Radionuclide: Curies: 1.00E+002 H-3 1.00E+002 Total Activity: Total Activity Minus Daughters: 1.00E+002 Waste Form: Normal Physical Form: Liquid Container Type: LR-56 Entered Values: Metric Values: Package Void Volume: 0.700 m^3 7.00E+005 cc Waste Volume: 4.20 m^3 4.20E+006 cc 4.20E+003 kg Waste Mass: 4.20E+006 g Waste Void Volume: 0.100 m^3 1.00E+005 cc 15:00 Jul. 24, 1995 15:00 Jul. 24, 1995 Date to begin source decay: Date container sealed: Days to decay source before seal time: 0.00 days Calculate number of days sealed until 5.00% hydrogen is reached. G Value Material Selection: Contribution Weight G-Alpha G-Beta G-Gamma Name 100.00% 1 1.6 0.53 0.45 Water (liquid) G Values are calculated from list averaging: G Alpha G Beta G Gamma 1.6 0.53 0.45 Comments: **HYDROGEN:** A hydrogen concentration of 5.00 % is not reached in 100.0 years. A hydrogen concentration of 0.311 % is reached in 100.00 years. This corresponds to date: 15:00 Jun. 29, 2095 H2 Volume+ 2.50E+003 cc H2 Generation Rate: 0.00285 cc/hour DECAY HEAT AND PRESSURE: Heat Generated at seal time: 0.00337 Watts Partial Pressure (H2): 0.316 kPa Total Pressure (H2 and Air): 102. kPa TRANSPORTATION -Note: Transportation classifications assume three significant figures. Calculations are made at user-specified decay time. Radioactive: Yes (2.38E+004 nCi/g) Effective A2 for Mixture: 1.08E+003 Ci Type Determination: A (from unity fraction 0.092593) Limited Quantity: Yes LSA-I Determination: *Yes (from LSA-I unity fraction 0.022046) *Yes (from LSA-II unity fraction 1.1905E-006) LSA-II Determination: LSA-III Determination: Not Applicable *Per 49CFR173.403. The user must check to be sure that all other conditions for LSA are met as found in 49CFR173.403 and 49CFR173.427. HRC Quantity Determination: No Fissile Quantity: 0.000 g Fissile Excepted: NA There are no fissile radionuclides present per 49CFR173.403

TRU Waste:

No (Transuranic Waste per Hanford Site WAC)

Reportable Quantity:

Yes (from RQ unity fraction 1.0000)

Source at start of seal time: Radionuclide: Curies: H-3 1.00E+002

> Total Activity: 1.00E+002 Total Activity Minus Daughters: 1.00E+002

Shipping Papers and Labels:

	ISOTOPE	Number of A2s	Fraction of Total A2s	Cumulative Total A2s
*	н-3	9.26E-002	1.00E+000	1.00000

* Contains 95% of the total A2s and must be included per 49 CFR 173.433.

Radcalc for Windows 2.01

Date: 04-16-98 11:31

Performed By: _____

Checked By: _____

Input File: TEST2.RAD - Source Decayed to 73050 days Isotope Data Output File: TEST.DAT SPECIFIC HEAT PROD DECAY

ISOTOPE	ACTIVITY (Bq)	ACTIVITY (Ci)	ACTIVITY (Ci/g)	QUANTITY (g)	FACTOR (Watts/Ci)	HEAT (Watts)	A2 (Ci)	# of A2s
**Pu-241 *Am-241 *Np-237 Pa-233 Th-229 Ra-225 +Ac-225 +Fr-221 +At-217 +Bi-213 +Pb-209 +U-237 TOTALS:	2.44E+006 9.25E+008 6.21E+004 2.58E+001 1.52E-001 1.51E-001 1.51E-001 1.51E-001 1.51E-001 1.51E-001 1.51E-001 1.51E-001 9.28E+008	6.59E-005 2.50E-002 1.68E-006 6.97E-010 4.10E-012 4.09E-012 4.09E-012 4.09E-012 4.09E-012 4.09E-012 4.09E-012 4.09E-012 4.09E-012 4.09E-012 2.51E-002	1.00E+002 3.43E+000 7.05E-004 2.08E+004 9.70E-003 2.13E-001 3.92E+004 5.80E+004 1.74E+008 1.61E+012 1.94E+007 1.26E+016 4.61E+006 8.16E+004	6.59E-007 7.29E-003 2.38E-003 8.08E-011 7.19E-008 1.93E-011 1.04E-016 7.05E-017 2.36E-020 2.54E-024 2.11E-019 3.18E-028 8.88E-019 1.98E-014	3.18E-005 3.34E-002 2.85E-002 2.54E-003 2.91E-002 3.07E-002 7.26E-004 3.49E-002 3.86E-002 4.27E-002 4.23E-003 5.06E-002 1.17E-003 1.95E-003	2.09E-009 8.34E-004 4.79E-008 4.26E-009 2.03E-011 1.26E-013 2.97E-015 1.43E-013 1.75E-013 1.75E-013 1.75E-013 4.79E-015 3.16E-012	2.70E-001 5.41E-003 5.41E-003 2.43E+001 2.70E-002 8.11E-004 5.41E-001 daughter daughter daughter daughter daughter daughter daughter	2.44E-004 4.62E+000 3.10E-004 6.90E-008 2.58E-008 5.05E-009 7.57E-012 daughter daughter daughter daughter daughter daughter
Total Fi	9.28E+008 ssile Radio	2.51E-002 nuclides	(excluding	daughters) 7.31E-007 g	9	0.352 004		4.022+000

+ Daughter * TRU Radionuclide ** Fissile Radionuclides *** TRU and Fissile Radionuclides

D-14

Radcalc for Windows 2.01

Date: 04-16-98 11:37

Performed By: _____

Checked By: _____

Input File: TEST3.RAD - Source Decayed to 730 days Isotope Data Output File: TEST3.DAT

ISOTOPE	ACTIVITY (Bq)	ACTIVITY (Ci)	SPECIFIC ACTIVITY (Ci/g)	QUANTITY (g)	HEAT PROD FACTOR (Watts/Ci)	DECAY HEAT (Watts)	A2 (Ci)	# of A2s
Pu-241 *Am-241 *Np-237 Pa-233 *U-233 Th-229 +U-237	3.36E+010 1.13E+008 3.77E+001 3.40E+001 9.57E-005 4.23E-009 8.24E+005	9.08E-001 3.05E-003 1.02E-009 9.18E-010 2.59E-015 1.14E-019 2.23E-005	1.00E+002 3.43E+000 7.05E-004 2.08E+004 9.70E-003 2.13E-001 8.16E+004	9.08E-003 8.90E-004 1.45E-006 4.42E-014 2.67E-013 5.38E-019 2.73E-010	3.18E-005 3.34E-002 2.85E-002 2.54E-003 2.91E-002 3.07E-002 1.95E-003	2.89E-005 1.02E-004 2.91E-011 2.33E-012 7.53E-017 3.52E-021 4.35E-008	2.70E-001 5.41E-003 5.41E-003 2.43E+001 2.70E-002 8.11E-004 daughter	3.36E+000 5.64E-001 1.88E-007 3.78E-011 9.58E-014 1.41E-016 daughter
TOTALS: Total Fi	3.37E+010 3.37E+010 ssile Radio	9.11E-001 9.11E-001 nuclides	(excluding	9.97E-003 daughters) 9.08E-003	9	1.31E-004		3.93E+000

+ Daughter * TRU Radionuclide ** Fissile Radionuclides *** TRU and Fissile Radionuclides

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APPENDIX E

VALIDATION CASES

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APPENDIX E

VALIDATION CASES

1.0 VALIDATION TO RADCALC FOR WINDOWS 1.0

Many of the algorithms contained in Radcalc for Windows Version 2.0 are identical to those contained in Radcalc for Windows Version 1.0. The algorithms that were not changed from one version to the next include those associated with the calculation of hydrogen gas volume, hydrogen gas generation rate, pressure, daughter products, unit conversions, and decay heat. The Radcalc cases that were used to test these algorithms for Version 1.0 were rerun using Version 2.0 and the output verified against the Version 1.0 output.

Validation cases also included comparisons to the original spreadsheet. Cases run include calculation comparisons to the original spreadsheet, gamma absortion fractions, use of G values, percent hydrogen, decay algorithm, disintegrations, unit conversions, target percent, dating routine, and pressure calculations. A complete listing of cases is given in Table E-1. The validation case corresponds to Version 1.0 case numbers and the file names associated with the cases have '.rad' extensions. Output was verified to Version 1.0 output. The Radcalc 2.0 output files have been written to a retrievably stored write-only compact disc (EBU-CD-001). They are contained in the subdirectory entitled 'validation\el'.

Description	Validation case	File names (.rad)
Hydrogen gas generation	IA1 - IA12	1a1 - 1a12
Gamma absorption functions	IB1 - IB4	Ib1 - Ib4
Use of G values	IC1 - IC5	Ic1 - Ic5
Percent hydrogen	ID1 - ID2	Ib2, Ib3, Ib4, Ic2, Ic3, Ic5
Hydrogen generation rate	IE	1b1, 1b2, 1b3, 1b4, 1c2, 1c3, 1c4, 1c5
Decay algorithms	IIA1 - 11A3	Ilaia, Ilaib, Ilaic, Ila2a, Ila2b, Ila2c, Ila3, Ila3a, Ila3b, Ila3c, Ila3d
Integrated disintegrations over time	11B1 - 11B3	Ia1-1c5*
Unit conversions	IVA1 - IVA3, IVA5	1b2, Ic1b, Ib4
Target percent specified	IVB1 - IVB2	IVb1, Ic5
Dating routine	IVC1 - IVC3	IVc1, Ic5
Heat generation	IVE1	1b1, 1b2, 1b3, 1b4, 1c2, 1c4, 1c5
Pressure	IVF1 - IVF2	1b1, 1b2, 1b3, 1b4, 1c2, 1c3, 1c4, 1c5

Table	E-1. Winc	Vers Iows	ion Vers	2.0 ion	File	es Va	Verified lidation	to Ca	Radcalc	for
		0.00	1013		1.0	۳a	IIUacium	ાતા	SeS.	

NOTE: Windows is a trademark of Microsoft Corporation.

Radcalc does not report the number of integrated disintegrations as a function of time. Because hydrogen generation is directly related to disintegrations, the hydrogen calculations performed for cases 1a1-1c5 were used to validate the correct functioning of this algorithm.

2.0 TRANSPORTATION CASES

Table E-2 gives a listing of the transportation validation case number, case description, output file name, and method of validation. Output files are archived on duplicate write-only compact discs labeled EBU-CD-001 in the 'validation\e2' subdirectory. Validation output files are also archived in the code custodian's computer as described in Section 5.0. These cases represent only a small number of the total cases which were run to ensure that Radcalc calculates transportation classifications correctly.

Case		Description	File name (.rad)	Method of validation
111A		Transportation calculations	3	
	1	Type classification for a normal form material	A1, B1	Excel spreadsheet
	2	Type classification for a special form material	A3	Excel spreadsheet
	3	Limited quantity classification for a special form material	LQ1, LQ3	Excel spreadsheet
	4	Limited quantity classification for a normal form solid material	LQ4	Excel spreadsheet
	5	Limited quantity classification for a liquid form material without tritium	LQ5	Excel spreadsheet
	6	Limited quantity classification for a normal form liquid with tritium with a low specific activity (< 0.1 Ci/L)	L97, L910	Excel spreadsheet
	7	Limited quantity classification for a normal form liquid with tritium with an intermediate specific activity (≥ 0.1 Ci/L and ≤ 1.0 Ci/L)	LQ8	Excel spreadsheet
	8	Limited quantity classification for a normal form liquid with tritium with a high specific activity (> 1.0 Ci/L)	LQ9	Excel spreadsheet
	9	Limited quantity classification for a special form gas containing tritium	LQ19	Excel spreadsheet
	10	Limited quantity classification for a normal form gas with no tritium	LQ28	Excel spreadsheet
	11	Low specific activity (LSA) 1 (LSA-I) ($\leq 10^{-6} A_2/g$)	LSAI6, LSAI7	Excel spreadsheet
	12	LSA-II for a normal form solid (≤ 10 ⁻⁴ A ₂ /g)	LSAII1	Excel spreadsheet
	13	LSA-II for a normal form liquid (≤ 10 ⁻⁵ A ₂ /g)	LSAI18	Excel spreadsheet
	14	LSA-III for a normal form solid (≤ 2.0 E-3 A ₂ /g)	LSAIII1	Excel spreadsheet
	15	HRCQ classification for a package in which activity = $27,000$ Ci and < $3,000$ A ₂	HRCQ1, HRCQ5	Excel spreadsheet
	16	HRCQ classification for a package in which activity $< 27,000$ Ci and $> 3,000$ A ₂	HRCQ3	Excel spreadsheet
	17	HRCQ Classification for a package requiring the most restrictive ratio selection.	HRCQ21, HRCQ20	Hand calculations
	18	Fissile excepted classification, \leq 15 g fissile material	FE1, FE3	Excel spreadsheet

Table E-2. Radcalc for Windows Version 2.0 Validation Cases. (2 sheets total)

Case		Description	File name (.rad)	Method of validation	
	19	Fissile excepted classification, 5 g/10 L or less (liquid)	FE4, FE6	Excel spreadsheet	
	20	Fissile excepted classification, $\leq 20\%$ ²³⁹ Pu and ²⁴¹ Pu in total Pu mass and Pu mass $\leq 1,000$ g	FE14, FE11, FE9	Excel spreadsheet	
	21	Reportable quantity classification	RQ1, RQ2	Excel spreadsheet	
IIIA.	22	TRU waste classification (> 100 µCi/g)	TRU1, TRU2	Excel spreadsheet	
	23	Radioactive material classification (specific activity > 0.002 µCi/g)	R1, R3	Excel spreadsheet	
IV		Other Radcalc features			
A		Unit conversions			
	6	Becquerels to curies	IVa4a, IVa4b	Excel spreadsheet	
	7	Ratio functions	R1	Excel spreadsheet	

Table E-2. Radcalc for Windows Version 2.0 Validation Cases. (2 sheets total)

HRCQ = Highway route controlled quantity.

TRU = Transuranic.

NOTE: Windows is a trademark of Microsoft Corporation.

^{*}Output files archived as described in text.

Cases IIIA1-IIIA22 compare Radcalc for Windows with calculations performed using an Excel Version 5.0 spreadsheet created specifically to evaluate transportation classifications. Cases 1 and 2 verify type classifications for both normal and special forms. Cases 3-10 verify limited quantity classifications. Cases 11-14 verify low specific activity classifications. Cases 15 and 16 verify highway route controlled quantity classifications. Cases 17-19 verify fissile excepted classifications. Reportable quantity validation is contained in Case 20, transuranic (TRU) waste classification in Case 21, and radioactive in Case 22.

A common methodology was employed to test most of the classification algorithms. First, a group of radionuclides was selected at random, and the activities of the radionuclides were chosen so that the group's combined unity fraction was exactly equal to the limiting value associated with the classification of interest. Radcalc was then run with these values to confirm that the "borderline" case was correctly categorized. Next, one of the activities was increased slightly, so as to make the combined unity fraction exceed the limiting value. Radcalc was then run using this new set of values in order to confirm that the package was once again correctly classified. Files of both the original "borderline" Radcalc output and the final "over limit" output are archived for the first example in each major classification category, but not for subsequent examples.

Case IIIA1: Type Classification for a Normal Form Material

Discussion and Conclusion

Information for a normal form solid containing five radionuclides was run on Radcalc for Windows. The activities of the five radionuclides were chosen to be exactly one fifth of their associated A_2 values, so that the combined unity fraction was exactly 1.0. The output file (designated A1) shows that the package was correctly identified as Type A.

The output file designated B1 shows the effect of slightly increasing one of the five activities used previously. With a combined unity fraction above 1.0, the package was correctly identified as Type B.

Case IIIA2: Type Classification for a Special Form Material

Discussion and Conclusion

Information for a special form material containing ten radionuclides was run on Radcalc for Windows. The activities of the ten radionuclides were chosen to be exactly one tenth of their associated A₁ values so that the combined unity fraction was exactly 1.0. The output file (designated A3) shows that the package was correctly identified as Type A. When one of the activities was increased slightly, raising the combined unity fraction above 1.0, Radcalc correctly identified the package as Type B.

Case IIIA3: Limited Quantity Classification for a Special Form Material

Discussion and Conclusion

Information for a special form material containing five radionuclides was run on Radcalc for Windows. The activities of the five radionuclides were chosen to be exactly 2.0 x 10^{-4} of their associated A₁ values so that the combined unity fraction was exactly 1.0 x 10^{-3} . The output file (designated LQ1) shows that the package was correctly identified as limited quantity.

The file designated LQ3 shows the effect of slightly increasing one of the four activities used previously. With a combined unity fraction above 1.0 \times 10⁻³, the package was correctly identified as having exceeded the limited quantity standard.

Case IIIA4: Limited Quantity Classification for a Normal Form Solid Material

Discussion and Conclusion

Information for a normal form solid containing five radionuclides was run on Radcalc for Windows. The activities of the five radionuclides were chosen to be exactly 2.0 x 10^{-4} of their associated A₂ values so that the combined unity fraction was exactly 1.0 x 10^{-3} . The output file (designated LQ4) shows that the package was correctly identified as limited quantity. When one of the activities was increased slightly, raising the combined unity fraction above 1.0×10^{-3} , Radcalc correctly identified the package as having exceeded the limited quantity limit.

Case IIIA5: Limited Quantity Classification for a Liquid Form Material Without Tritium

Discussion and Conclusion

Information for a normal form liquid containing five radionuclides was run on Radcalc for Windows. No tritium was used in this case. The activities of the five radionuclides were chosen to be exactly 2.0 x 10^{-5} of their associated A₂ values so that the combined unity fraction was exactly 1.0 x 10^{-4} . The output file (designated LQ5) shows that the package was correctly identified as limited quantity. When one of the activities was increased slightly, raising the combined unity fraction above 1.0 x 10^{-4} , Radcalc correctly identified the package as having exceeded the limited quantity limit.

Case IIIA6: Limited Quantity Classification for a Normal Form Liquid With Tritium With a Low Specific Activity (< 0.1 Ci/L)

Discussion and Conclusion

Information for a normal form liquid containing only tritium with a low specific activity was run on Radcalc for Windows. The ³H activity was chosen to be 1,000 Ci and the waste volume 10,100 L, yielding a specific activity of 0.099 Ci/L. Radcalc correctly classified the package as limited quantity as shown in the output file designated LQ7. When the activity was increased to 1,010 Ci, Radcalc correctly identified the package as having exceeded the limited quantity limit as shown in the output file designated LQ7.

Case IIIA7: Limited Quantity Classification for a Normal Form Liquid With Tritium With an Intermediate Specific Activity (≥ 0.1 Ci/L and ≤ 1.0 Ci/L)

Discussion and Conclusion

Information for a normal form liquid containing only tritium with an intermediate specific activity was run on Radcalc for Windows. The ${}^{3}\text{H}$ activity was chosen to be 100 Ci and the waste volume 500 L, yielding a specific activity of 0.2 Ci/L. Radcalc correctly classified the package as limited quantity as shown in the output file designated LQ8. When the activity was increased to 101 Ci, Radcalc correctly identified the package as having exceeded the limited quantity limit.

Case IIIA8: Limited Quantity Classification for a Normal Form Liquid With Tritium With a High Specific Activity (> 1.0 Ci/L)

Discussion and Conclusion

Information for a normal form liquid containing only tritium with a high specific activity was run on Radcalc for Windows. The 3 H activity was chosen to be 1 Ci and the package volume 0.5 L, yielding a specific activity of 2.0 Ci/L. Radcalc correctly classified the package as limited quantity as shown in the output file designated LQ9. When the activity was increased to 1.01 Ci, Radcalc correctly identified the package as having exceeded the limited quantity limit.

Case IIIA9: Limited Quantity Classification for a Special Form Gas Containing Tritium

Discussion and Conclusion

Information for a special form gas containing only tritium was run on Radcalc for Windows. The 3 H activity was chosen so that the unity fraction was equal to exactly 2.0 x 10^{-2} . Radcalc correctly identified the package as a limited quantity as shown in the output file designated LQ19. When the activity was increased, raising the unity fraction above 2.0 x 10^{-2} , Radcalc correctly identified the package as having exceeded the limited quantity limit.

NOTE: These cases were also run for normal form gas. Radcalc was also able to identify the limited quantity classification correctly.

Case IIIA10: Limited Quantity Classification for a Normal Form Gas With No Tritium

Discussion and Conclusion

Information for a normal form gas containing six radionuclides with no tritium was run on Radcalc for Windows. The activities were chosen so that the combined unity fraction was equal to exactly 1.0×10^{-3} . Radcalc correctly identified the package as a limited quantity as shown in the output file designated LQ28. When the activity was increased, raising the unity fraction above 2.0 x 10^{-2} , Radcalc correctly identified the package as having exceeded the limited quantity limit.

NOTE: These cases were also run for special form gas. Radcalc was also able to identify the limited quantity classification correctly.

Case IIIA11: Low Specific Activity (LSA) I (LSA-I) ($\leq 10^{-6} A_2/g$)

Discussion and Conclusion

Information for a normal form gas containing eleven radionuclides, was run on Radcalc for Windows. The activities of ten of the radionuclides were chosen to be exactly 10% of their associated A₂ value so that the combined A₂ unity fraction was exactly 1.0. The other radionuclide has unlimited A₂ values. The mass was chosen to be 10⁶ g, equaling the LSA-I limiting value of $10^{-6} A_2/g$, and thus the LSA-I unity fraction was equal to 1.0.

The output file (designated LSAI6) shows that the package was correctly identified as LSA-I. When one of the activities was increased slightly, raising the combined low specific activity unity fraction above 1.0, Radcalc correctly identified the package as having exceeded the LSA limit as shown in the output file designated LSAI7.

Case IIIA12: LSA-II for a Normal Form Solid ($\leq 10^{-4} A_2/g$)

Discussion and Conclusion

Information for a normal form solid containing five radionuclides was run on Radcalc for Windows. The activities of the five radionuclides were chosen to be exactly 20% of their associated A_2 value so that the combined A_2 unity fraction was exactly 1.0. The mass was chosen to be 10⁴ g, equalling the LSAII limiting values of 10⁻⁴ A_2/g for solids, and thus the LSAII unity fraction was equal to 1.0. The output file (designated LSAIII) shows that the package was correctly identified as low specific activity. When one of the activities was increased slightly, raising the combined low specific activity unity fraction above 1.0, Radcalc correctly identified the package as having exceeded the low specific activity limit.

Case IIIA13: LSA-II for a Normal Form Liquid ($\leq 10^{-5} A_2/g$)

Discussion and Conclusion

Information for a normal form liquid containing five radionuclides was run on Radcalc for Windows. The activities of the five radionuclides were chosen to be exactly 20% of their associated A_2 values so that the combined A_2 unity fraction was exactly 1.0. The mass was chosen to be 10⁵ g, equaling the LSA-II limiting value of 10⁻⁵ A_2/g for liquids, and thus the LSA-II unity fraction was equal to 1.0. The output file (designated LSAII8) shows that the package was correctly identified as LSA-II. When one of the activities was increased slightly, raising the combined LSA unity fraction above 1.0, Radcalc correctly identified the package as having exceeded the low specific activity limit. Case IIIA14: LSA-III for a Normal Form Solid ($\leq 2.0 \times 10^{-3} A_2/g$)

Discussion and Conclusion

Information for a normal form solid containing five radionuclides was run on Radcalc for Windows. The activities of the five radionuclides were chosen to be exactly 20% of their associated A_2 values so that the combined A_2 unity fraction was exactly 1.0. The mass was chosen to be 500 g, equaling the LSA-III limiting value of 2.0 x 10⁻³ A_2/g , and thus the LSA-III unity fraction was equal to 1.0. The output file (designated LSAIII) shows that the package was correctly identified as LSA-III. When one of the activities was increased slightly, raising the combined low specific activity unity fraction above 1.0, Radcalc correctly identified the package as having exceeded the LSA limit.

Case IIIA15: Highway Route Controlled Quantity Classification for a Package in Which Activity = 27,000 Ci and < 3,000 A₂

Discussion and Conclusion

Information for a Type B package having a total activity of exactly 27,000 Ci and less than 3,000 A₂ was run on Radcalc for Windows. The output file, designated HRCQ1, shows that Radcalc correctly identified the package as not requiring highway route control. When the activity was raised to above 27,000 Ci, Radcalc correctly identified the package as requiring highway route control as shown in the output file designated HRCQ5.

Case IIIA16: Highway Route Controlled Quantity Classification for a Package in Which Activity < 27,000 Ci and < 3,000 A₂

Discussion and Conclusion

Information for a Type B package having a total activity less than 27,000 Ci and less than 3,000 A₂ was run on Radcalc for Windows. The output file, designated HRCQ3, shows that Radcalc correctly identified the package as not requiring highway route control. When the case was changed so that the activity was raised to above 3,000 A₂, but still less than 27,000 Ci, Radcalc correctly identified the package as requiring highway route control.

Case IIIA17: Highway Route Controlled Quantity Classification for a Package Requiring the Most Restrictive Ratio Selection

Discussion and Conclusion

Information for a Type B package containing two radionuclides having less than 27,000 Ci and less than 3,000 A₂ values was run on Radcalc for Windows. The activities of the radionuclides were chosen so that Radcalc for Windows would be required to chose the most limiting ratio for each radionuclide. Radcalc correctly identified the package as not requiring highway route control. When one of the activities was increased, Radcalc correctly identified the package as being a highway route control quantity. The two output files for this case are HRCQ21 and HRCQ20.

Case IIIA18: Fissile Excepted Classification, \leq 15 g Fissile Material

Discussion and Conclusion

Information for a normal form solid containing the five fissile radionuclides ²³³U, ²³⁵U, ²³⁸Pu, ²³⁹Pu, and ²⁴¹Pu was run on Radcalc for Windows. The activities of the radionuclides were chosen so that the total fissile mass was equal to 15.0 g. The output file, designated FE1, shows that Radcalc correctly identified the package as fissile excepted. When one of the activities was increased slightly, raising the total fissile mass just above 15.0 g, Radcalc correctly identified the package as having exceeded the fissile excepted standard as shown in the file designated FE3.

Case IIIA19: Fissile Excepted Classification, 5 g/10 L or Less (Liquid)

Discussion and Conclusion

Information for a normal form liquid containing two fissile radionuclides was run on Radcalc for Windows. The activities of the radionuclides were chosen so that the total fissile mass was above 15.0 g. The waste volume was chosen so that the fissile mass per volume was less than 5 g/10 L of waste. The output file, designated FE4, shows that Radcalc correctly identified the package as being fissile excepted. When the fissile mass per volume was increased to above 5 g/10 L, Radcalc correctly identified the package as exceeding the fissile excepted limit as shown in the output file designated FE6.

Case IIIA20: Fissile Excepted Classification, $\leq 20\%^{239}$ Pu and ²⁴¹Pu in Total Plutonium Mass and Plutonium Mass $\leq 1,000$ g

Discussion and Conclusion

Information for a normal form liquid containing only plutonium radionuclides was run on Radcalc for Windows. The activities of the radionuclides were chosen so that the total plutonium mass was equal to 1,000 g and exactly 20% of the total plutonium mass was a combination of ²³⁹Pu and ²⁴¹Pu. The waste volume was chosen so that the fissile mass per volume was more than 5 g/10 L of waste to force Radcalc to make a fissile excepted determination using 49 CFR 173.453(e). The output file, designated FE14, shows that Radcalc correctly identified the package as being fissile excepted. When the total mass of plutonium was raised above 1,000 g, Radcalc correctly identified the package as exceeding the fissile excepted limit as shown in the output file designated FE11.

Also, when the total plutonium mass was returned to 1,000 g and the weight percent of 239 Pu and 241 Pu was raised above 20%, Radcalc correctly identified the package as exceeding the fissile excepted limit as shown in the output file designated FE9.

Discussion and Conclusion

Information for a normal form solid containing two radionuclides was run on Radcalc for Windows. The activities of the radionuclides were chosen so that sum of the Reportable Quantity fractions was equal to 1.0. The output file, designated RQ1, shows that Radcalc correctly identified the package as being a reportable quantity. When the activity of one of the radionuclides was decreased, lowering the RQ unity fraction to below 1.0, Radcalc correctly identified the package as not being a reportable quantity (RQ2).

Case IIIA22: TRU Waste Classification (> 100 μ Ci/g)

Discussion and Conclusion

Information for a normal form liquid containing a TRU radionuclide (2 > 92, half-life greater than 20 years and radium isotopes or ²³³U) was run on Radcalc for Windows. The activity of the radionuclide was chosen so the activity/total waste mass = 100 nCi/g. The output file, designated TRU1, shows that Radcalc correctly identified the package as not being TRU waste. When the activity was increased, raising the activity/total waste mass above 100 nCi/g, Radcalc correctly identified the package as exceeding the TRU waste limit as shown in file TRU2.

Case IIIA23: Radioactive Material Classification (specific activity > 0.002 μ Ci/g)

Discussion and Conclusion

Information for a Type A package containing seven radionuclides was run on Radcalc for Windows. The activities of the seven radionuclides were chosen so that their combined specific activity was exactly $0.002 \ \mu$ Ci/g. The output file for the Radcalc for Windows Input and Calculated results sheet (designated R1) shows that the package was correctly identified as not containing radioactive material. When one of the activities was increased slightly, raising the combined specific activity above $0.002 \ \mu$ Ci/g, Radcalc correctly identified the package as containing radioactive material as shown in the output file designated R3.

3.0 UNIT CONVERSIONS AND RATIOS

One of the new features that was added in Version 2.0 was the ability to convert between becquerels and curies. The code is also capable of multiplying a source term by any number to scale the source term by a desired value. The two cases in this section test the algorithms to ensure correct response of the code.

E-10

Case IVA6: Becquerels to Curies

Discussion and Conclusion

Radcalc for Windows performs calculations in terms of curies for activity. When the user enters a radionuclide inventory in terms of becquerels, Radcalc for Windows converts activities into units of curies for calculations. The resulting radionuclide inventories calculated by Radcalc for Windows at the beginning and end of seal time are then converted to becquerels in the Radcalc for Windows Input and Calculated Results sheet if the initial radionuclide inventory was specified in becquerels. The conversions from and to becquerels were checked by making two Radcalc for Windows runs. The first uses a radionuclide inventory entered in terms of becquerels. The second uses the same inventory entered in terms of curies. Results from both cases are consistent, indicating proper conversions have taken place. The two cases are designated as IVa4a and IVa4b.

Case IVA7: Ratio Functions

Ratio functions were validated by ratioing the source term in case R1 by factors of 5, 8, 13, and 16. The source dialog box values were verified using an Excel spreadsheet.

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A CALCULATIONAL TECHNIQUE TO PREDICT COMBUSTIBLE GAS GENERATION IN SEALED RADIOACTIVE WASTE CONTAINERS

James E. Flaherty Akira Fujita C. Paul Deltete Geoffrey J. Quinn

Published May 1986

EG&G Idaho, Inc. Idaho Falls, Idaho 83415

Prepared for the U.S. Department of Energy Idaho Operations Office Under DOE Contract No. DE-AC07-76ID01570

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ABSTRACT

Certain forms of nuclear waste, when subjected to ionizing radiation, produce combustible mixtures of gases. The production of these gases in sealed radioactive waste containers represents a significant safety concern for the handling, shipment and storage of waste. The U.S. Nuclear Regulatory Commission (NRC) acted on this safety concern in September 1984 by publishing an information notice requiring waste generators to demonstrate, by tests or measurements, that combustible mixtures of gases are not present in radioactive waste shipments; otherwise the waste must be vented within 10 days of shipping.

A task force, formed by the Edison Electric Institute to evaluate these NRC requirements, developed a calculational method to quantify hydrogen gas generation in sealed containers. This report presents the calculational method along with comparisons to actual measured hydrogen concentrations from EPICOR II liners, vented during their preparation for shipment. As a result of this, the NRC recently altered certain waste shipment Certificates-Of-Compliance to allow calculations, as well as tests and measurements, as acceptable means of determining combustible gas concentration. This modification was due in part to work described herein.

EXECUTIVE SUMMARY

The potential exists for the production of combustible mixtures of gasses in certain waste forms containing radioactive material (e.g., resins, binders, sludge, and wet filters). In sealed containers, radioactive waste gas production could result in pressures above atmospheric pressure. The production of combustible gases in sealed radioactive waste containers has been identified as a significant safety concern relative to the handling, shipping, and storage of radioactive waste.

At Three Mile Island (TMI-2), during preparations to ship 50 EPICOR-II liners containing ion exchange resins, it was revealed that significant concentrations of hydrogen gas were present. As a result, the U.S. Department of Energy (DOE) sponsored various research and development programs addressing gas generation and related safe management of radioactive ion exchange waste. The Nuclear Regulatory Commission (NRC) also sponsored technical studies by the Brookhaven National Laboratory (BNL) on the effects of ionizing radiation on organic ion exchange resin and other radioactive wastes. The NRC studies included a survey on resin use and radionuclide loading at operating nuclear power plants.

It was determined from the research and studies that nuclear wastes subjected to ionizing radiation produce gas through the process of radiolysis and waste degradation. The radiolytic decomposition of the residual water in the resin produces hydrogen and oxygen. In addition, the ionizing radiation acts as a catalyst for a chemical reaction between the oxygen, waste, and residual water in the container. Although hydrogen gas generation increases in an almost linear relationship with increasing curie content, pressure does not increase in a sealed container until an integrated dose is reached in which the oxygen in the air inside the container is consumed by chemical reaction. The ongoing radiolytic process continues to produce gas. Pressure will increase in a sealed container and combustible concentrations of hydrogen gas may result.

The NRC's subsequent evaluation of the hydrogen gas generation problem resulted in the issuance of new requirements (Inspection Enforcement Information Notice No. 84-72: Clarrification of Conditions for Waste Shipments Subject to Hydrogen Gas Generation) for certain certificates of compliance related to radioactive waste shipment packages. These requirements affect most radioactive waste shipments (resins, binders, waste sludge, and wet filters) from operating nuclear power plants. The NRC requirements are summarized as follows:

• For waste containers that have the potential to radiolytically generate combustible gases, plant personnel must conduct tests and take measurements to determine the factors affecting gas generation. Tests and measurements are not required if radioactivity concentration is less than that for low specific activity (LSA) material and shipped within 10 days of preparation (sealing) or venting.

Utilities (waste generators) were concerned because compliance by conducting tests and taking measurements would be difficult (if not impossible) leaving venting prior to shipment as their only alternative. The venting option is costly financially and in terms of man-rem expenditures. The Utility Nuclear Waste Management Group of the Edison Electric Institute (EEI) formed a Hydrogen Generation Task Force to study and evaluate the new NRC requirements. The task force used direct operational experience from GPU Nuclear's EPICOR II campaign and technical assistance from the EG&G Idaho/DOE Technical Integration Office at TMI. These efforts resulted in the development of a cost effective method (in man-rem and dollars) to accurately quantify hydrogen gas production in radioactive waste containers and aid utilities in compliance to NRC Information Notice No. 84-72.

Electric Power Research Institute (EPRI) demonstrated the use of this calculational method at TMI. Twenty-eight "non-leaking" EPICOR II liners were used for the demonstration. EPRI used a desk top computer with a spreadsheet program to compare the predicted hydrogen concentration from the EG&G Idaho calculational method to the hydrogen concentration actually measured when the EPICOR II liners were vented during their preparation for shipment. On the average, the predicted hydrogen concentration was within 15 percent of the measured values.

EEI 'Task Force' representatives met with NRC Transportation Branch personnel to discuss the use of calculations to determine hydrogen gas generation in radioactive waste containers. As the basis for discussion, EEI presented the calculational method that EG&G Idaho and DOE developed. EEI requested that calculations be recognized as a means of satisfying the requirements for determining hydrogen gas concentration in radioactive waste containers, as set forth in NRC Inspection Enforcement Information Notice No. 84-72. The NRC acknowledged the validity of the method. Affected certificates-of-compliance have been modified to allow calculations as an acceptable means of assessing gas generation, as well as tests and measurements set forth in NRC Inspection Enforcement Information Notice No. 84-72.

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ACRONYMS AND ABBREVIATIONS

- BNL Brookhaven National Laboratory
- DOE United States Department of Energy
- EEI Edison Electric Institute
- EPRI Electric Power Research Institute
- LSA Low Specific Activity
- NBS National Bureau of Standards
- NRC United Stated Nuclear Regulatory Commission
- SDS Submerged Demineralizer System
- TMI-2 Three Mile Island-Unit 2

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A CALCULATIONAL TECHNIQUE TO PREDICT COMBUSTIBLE GAS GENERATION IN SEALED RADIOACTIVE WASTE CONTAINERS

INTRODUCTION

1

Background

Nuclear wastes subjected to ionizing radiation produce combustible mixtures of gases in certain waste forms containing radioactive material (e.g., resins, binders, sludge, and wet filters) through the processes of radiolysis and waste degradation. In sealed containers, radioactive waste gas production could result in pressures above atmospheric pressure. The production of combustible gases in sealed radioactive waste containers has been identified as a significant safety concern relative to the handling, shipping, and storage of radioactive waste.

The radiolytic decomposition of the residual water in the resin produces hydrogen and oxygen. In addition, the ionizing radiation acts as a catalyst for a chemical reaction between the oxygen, waste, and residual water in the container. Although hydrogen gas generation increases in an almost linear relationship with increasing curie content, pressure does not increase in a sealed container until an integrated dose is reached in which all the oxygen in the air inside the container is consumed by chemical reaction. As the ongoing radiolytic process continues to produce gas, pressure will increase in a sealed container and combustible concentrations of hydrogen gas may result.

Three Mile Island Unit 2 (TMI-2) used various ion exchange media systems to process contaminated accident waste water. One system, EPICOR II, used organic ion exchange resins to process more than 500,000 gal of waste water, resulting in the generation of 72 prefilter liners with curie loadings ranging from 160 to 2200 Ci. Loadings in 22 of the liners were low enough to permit their disposal in a commercial low-level waste site, while the remaining 50 liners were stored at TMI-2. Subsequent preparations to ship these 50 liners revealed the presence of significant concentrations of hydrogen gas. As a result, the U.S. Department of Energy (DOE) sponsored various research and development programs addressing gas generation and related safe management of radioactive ion exchange waste. The Nuclear Regulatory Commission (NRC) also sponsored technical studies by the Brookhaven National Laboratory on the effects of ionizing radiation on organic ion exchange resin and other radioactive wastes. The NRC studies included a survey of operating nuclear power plants on resin use and radionuclide loading.

Parameters of Gas Generation. The DOE and NRC gas generation research efforts and actual TMI-2 on-site (EPICOR II) direct measurements and sampling revealed that certain conditions and specific relationships exist. These include:

- Ion exchange media subjected to ionizing radiation generate a variety of gases (hydrogen, oxygen, carbon dioxide, carbon monoxide, and methane) from radiolysis and resin degradation. This generation occurs at low curie loadings as well as at higher levels.
- Hydrogen gas generation rates in radioactive waste containers increase at a nearly linear rate with increasing curie content.
- Pressure in a sealed wet radioactive waste container initially decreases due to oxygen depletion, masking hydrogen production. The pressure will continue to decrease until the delivered dose is sufficient to deplete all of the gaseous oxygen.
- The time necessary to reach a given hydrogen gas concentration, in a sealed radioactive waste container, is predictable.
- United States commercial nuclear power generating plants typically ship radioactive wastes that are well below the present NRC permissible radionuclide loading limit (10 Ci/ft³); however, a few have exceeded this limit.

NRC Requirements. The NRC's subsequent evaluation of the hydrogen gas generation problem resulted in the issuance of new requirements (Inspection Enforcement Information Notice No. 84-72: Clarification of Conditions for Waste Shipments Subject to Hydrogen Gas Generation) for certain certificates of compliance related to radioactive waste shipment packages. The following paragraphs summarize these requirements, which address hydrogen gas generation and applicable safe storage and shipment periods.

For waste containers that have the potential to radiolytically generate combustible gases, plant personnel must conduct tests and take measurements to determine the factors affecting gas generation. Specifically, the following criteria must be observed during a period twice the expected shipment time:

- 1. Hydrogen gas concentration must be limited to no more than 5 percent by volume of the secondary container gas void, or
- 2. The secondary container and cask cavity must be inerted to ensure that oxygen is limited to 5 percent by volume in those portions of the waste package that could have greater than 5 percent hydrogen gas.

The above is not required if radioactivity concentration is less than that for Low Specific Activity (LSA) material and shipped within 10 days of preparation (sealing) or venting.

These NRC requirements affect most radioactive waste shipments (resins, binders, waste sludge, and wet filters) from operating nuclear power plants. Although the problem of hydrogen gas generation has been seen only in containers with greater than Type A quantities of waste (as defined in 10 CFR 61), waste generators should be aware that the same physical conditions exist in containers with Type A quantities of waste as well, even though the time to reach hazardous conditions is much longer. Waste generators were concerned because these new NRC requirements seemed to be too conservative. Compliance by conducting tests and taking measurements would be difficult (if not impossible) for many waste generators, leaving venting prior to shipment as their only alternative. The venting option is costly financially and in terms of man-rem expenditures (where the concern is to keep dose rates as-low-as-reasonably-achievable).

Hydrogen Generation - Calculation and Comparison. The Utility Nuclear Waste Management Group of the Edison Electric Institute (EEI) formed a Hydrogen Generation Task Force to study and evaluate the new NRC requirements. The task force used direct operational experience from GPU Nuclear's EPICOR II campaign and technical assistance from the EG&G Idaho/DOE Technical Integration Office at TMI. This evaluation resulted in EG&G Idaho/DOE's development of a calculational method (presented in the next section of this report) to quantify hydrogen gas generation in sealed containers. The interelation of gas production resulting from radioactive decay in the waste form, the amount of energy retained in the waste container (resulting from the decay process), and the free volume of the container are discussed, and examples given.

The Section on EPICOR Gas Measurements discusses how theoretical predictions, made with the calculational method, were checked against actual gas measurements from representative waste packages. The result of this study is discussed and clarifying appendices are presented. Appendix A gives an example of how the values from various tables and figures are used to predict combustible gas concentrations. Appendix B is a discussion of the characteristics of various waste forms that have the potential to generate combustible quantities of gas. Appendix C contains a detailed description of the computer code used to determine the absorbed dose in different container geometries with various waste forms. Appendix D provides a mathematical description of energy absorption for a spherical source and suppliments the discussions presented in Appendix C.

METHOD CALCULATION

The calculational method for predicting combustible gas generation in sealed radioactive waste containers, described in this section, is based on DOE and NRC research and General Public Utility Nuclear Corporation's (GPU Nuclear) Three Mile Island EPICOR II experience. An effort has been made to extract pertinent findings from these resources, organizing the information into a unified format. Data requirements and analytical capabilities of typical waste generators were evaluated, and the format was modified to facilitate implementation by potential users of the model. Certain parameters that require mainframe computer computation have either been tabularized or are presented in graphic form. Presented below is a complete description of the calculational method developed to quantify combustible gas generation.

Formulae and Empirical Data

Combustible gas generation in sealed radioactive waste containers involves a combination of factors that can be addressed by considering the following:

- The production of gas in the waste form resulting from radioactive decay
- Amount of energy resulting from the decay process that is retained in the waste container and absorbed by the waste
- Free volume of the container, including interstitial voids inherent in the waste form.

These three factors and their interrelations are discussed below.

Gas Production in the Waste Form Resulting from Radioactive Decay. The quantity of gas produced is a function of the amount of energy absorbed by the waste form. Gas formation is due primarily to radiolysis caused by the radioactive decay energy of nuclides present in the waste. NRC studies performed at Brookhaven National Laboratory (BNL) indicate that the amounts of combustible gases formed by radiolysis are approximately linear with dose. The relationship between gas production and energy absorption is expressed as the waste's gas generation constant or G-value, which is described as the number of radicals or molecules formed or decomposed per 100 eV absorbed. A number of G-values have been reported in the literature.¹⁻⁴ Those of concern to typical waste generators are the G-values for dewatered, or swollen resin, and solidified waste. These gas generation constants range from 0.1 to 0.6 molecules per 100 eV (See Table 1). G-Values for mixed resin bed systems can be approximated using the weight percent mix of the components.

Radioactive Decay Energy Absorbed by the Waste Form. The amount of energy absorbed by a waste form as a result of the radioactive decay process is a function of curie loading, waste properties, and container geometry. The primary decay modes for consideration by typical waste generators are beta, including other electron emissions, and gamma emission. For practical purposes, all beta and associated secondary radiation emission energies are assumed to be deposited in the waste. This is because of the relatively short travel length, or range, (as compared with gamma rays) required for the beta particle to give up its energy in the waste material, and the extremely low probability of a beta particle escaping the container. The gamma energy absorbed by the waste depends on the strength of the gamma emission, the amount of gamma ray energy absorbed by interaction (collision) with a waste particle, and the number of particles with which the gamma ray interacts. Because an interaction will either absorb or, more likely, attentuate an emission, the fraction of gamma energy absorbed depends on the number of interactions possible. Therefore, gamma energy absorption increases with increasing numbers of waste particles. For this reason, gamma absorption is a function of container waste density and geometry.

The total energy absorbed in the waste is the sum of the beta and secondary radiation decay energy and that portion of the gamma ray energy retained in the waste's matrix. In order to calculate combustible gas generation, it is necessary to determine the absorbed energy at a specific time after the waste container is loaded and sealed. This is accomplished by the use of equations that consider nuclide loading curie content; waste properties such as density, geometry, gamma ray attenuation; and absorption coefficients. The result of this exercise yields an absorbed energy dose in rads (radiation absorbed dose) for each nuclide in the waste at a time after sealing. The contributions of

Resir	Ion	ic Form (Mole	G _H cules/100 eV) ^b
DOWEX			
50WX10	Li ⁺	0.11	
DOWEX 5	$\frac{1}{H}$	0.095	
ZEO-KAR	B 215 H+	0.12	
IRN-77	н+	0.13 ±	- 0.02
IRN-78	ОН	- 0.6	
IRN-150	НО	H 0.5	
IRN-77	Na	+ 0.2	
IRN-78	CIT	0.3	
IRN-150	NaC	0.3	

Table 1. Hydrogen gas generation constants (G_H)^a

a. The hydrogen gas generation yield for a mixed bed system is the sum of the yields of the individual components. For example, a mixed bed containing 0.40 resin A and 0.60 resin B by weight with $A:G_H = x$ and $B:G_H = Y$ is

 $G_{H}(A + B) = 0.4 x + 0.6Y$

For beds of unknown composition use

anion bed (OH ⁻)	$G_{\rm H} = 0.6$
cation bed (OH ⁻)	$G_{\rm H}^{} = 0.13$
solidified resin/cement	$G_{H}^{(1)} = 0.24$
asphalt	$G_{\rm H}^{\rm H} = 0.41$

b. These values are the result of experimental analysis and appear as reported, for swollen (dewatered) resins from NUREG/CR-2830 and NUREG/CR-3383; cement values from NUREG/CR-2969, and values for asphalt from NUREG/CR-50617. The G-value is defined as the number of molecules formed or decomposed per 100 ev absorbed.

the beta, secondary, and gamma components of each nuclide are computed using the following equations:

Beta absorbed dose (rads) = $D_i^3(t)$

$$= \frac{AC_i}{\lambda_i} (E_i) (1 - e^{-\lambda} i^t)$$

Secondary radiation absorbed dose (rads)

$$= D_i^x(t) = \frac{AC_i}{\lambda_i} (E_i^x) (1 - e^{-\lambda} i^t)$$

Gamma absorbed dose (rads) = $D_1^{Y_{11}}$

$$= \frac{AC_i}{\lambda_i} (E_i^{\gamma}) (1 - e^{\lambda} i^t)$$

Combining equations yields $D_i^{J,x,\gamma}(t)$

$$= \frac{AC_i}{\lambda_i} (\overline{E}_i + E_i^x + E_i^y) (1 - e^{-\lambda}i^t)$$

where

t

 $D_i^{3,x,\gamma}(t) =$ total (beta plus secondary plus gamma) absorbed dose delivered by the ith nuclide at time t, in rads

- the decay constant of the ith nuclide in years⁻¹
- E_i = average beta energy in MeV of the ith nuclide
- E^x_i = the absorbed x-ray energy in MeV of the ith nuclide
- E_{i}^{γ} = the absorbed gamma ray energy in MeV of the ith nuclide.

The total cumulative absorbed dose for all nuclei and decay modes at time (t) is found from

$$D_{\text{Total}} = \sum_{i} D_{i}^{\beta.x.\gamma}(t)$$
.

 λ_i

A discussion of various absorption parameters is presented below.

Beta Absorbed Dose – The beta absorbed dose in the waste is calculated using the value for average beta energy (\vec{E}_i) for a nuclide. The average beta ray energy is approximately one third of the sum of the possible beta emissions multiplied by the relative abundance of the emission. For example, Iodine-131 emits two beta particles; 85 percent of the disintegrations result in a 0.608 MeV beta, while the remaining 15 percent are 0.315 MeV beta. Therefore,

$$\vec{E}_i = 1/3(0.85 \times 0.608 + 0.15 \times 0.315)$$

 $\vec{E}_i = 0.187 \text{ MeV.}^a$

These values have been tabularized and are available in standard health physics and nuclear handbooks. The beta absorbed dose is then determined by

$$D_i^{3}(t) = \frac{AC_i}{\lambda_i} (\overline{E}_i) (1 - e^{-\lambda} i^t) .$$

This is done for each beta emitting nuclide. Summing the beta dose for all nuclides at a time (t) yields the total beta absorbed dose for the waste.

Absorbed Dose Due to Secondary Radiations – Secondary radiations (E_i^x) , such as xrays and Auger electrons, result from the transition of a nuclide from an excited state to the ground state. All emissions with energies of 100 keV or less are considered to be totally absorbed by the waste form, and to participate in gas production.

Values for secondary radiations are available in standard handbooks⁵. The contribution to absorbed dose from secondary radiations for each nuclide in the waste is determined by

$$D_i^{x}(t) = \frac{AC_i}{\lambda_i} (E_i^{x}) (1 - e^{-\lambda} i^{t})$$

Summing the secondary radiation contribution for each nuclide present in the waste yields the total absorbed dose due to secondary radiations.

Gamma Absorbed Dose – Determination of the gamma absorbed dose (E^{γ}_i) is more complex than calculating the beta absorbed dose because the total energy of the gamma emission may not be deposited in the waste form. The fraction of gamma energy deposited is a function of the waste's physical characteristics and waste container geometry. The gamma energy absorbed by the waste is calculated for each nuclide present using

$$\mathbf{E}_{i}^{\gamma} = \Sigma_{j} \left[(\mathbf{n}_{ij}) (\mathbf{f}_{ij}) (\mathbf{E}_{ij}^{\gamma}) \right]$$

where

- E_{i}^{γ} = the gamma energy absorbed in MeV for the ith nuclide.
- Σ_j = the summation of the fractions of the gamma ray energies absorbed for all gamma emissions of the ith nuclide.
- n_{ij} = the abundance of the jth gamma ray per decay of the ith nuclide.
- f_{ij} = the fraction of energy, of the jth gamma ray of the ith nuclide that is absorbed in the waste.
- E_{ij}^{γ} = is the energy in MeV, of the jth gamma ray of the ith nuclide.

Example:

Cobalt 60 decays by emitting two gamma rays, 1.173 MeV and 1.332 MeV, both having an abundance of 1.0. The nuclide is present in waste with a density of 1 g/cm³ contained in a 55 gal drum.

For the 1.173 MeV gamma ray

$$n_{ij} = 1.0$$
, $f_{ij} = 0.54$, and $E_{ij}^{\gamma} = 1.173$ MeV.

a. This is an approximate value, the value used is 0.1817 MeV from Reference 5.

For the 1.332 MeV gamma ray

$$n_{ij} = 1.0, f_{ij} = 0.53, and E_{ij}^{\gamma}$$

= 1.332 MeV.

(1.0)(0.54)(1.173 MeV) + (1.0)(0.53)(1.332 MeV)

= 1.339 MeV .

Therefore, of the 2.505 MeV emitted in the gamma decay of Cobalt 60, 1.339 MeV is absorbed by the waste.

Values for n_{ij} and E^{γ}_{ij} were obtained from Reference 5. Values for f_{ij} can be obtained from Figures A-1 through A-4 in Appendix A, or computed by the methods outlined in Appendices A, B, and C.

The contribution to absorbed dose from gamma ray emission for each nuclide in the waste is

$$D_i^{\gamma}(t) = \frac{AC_i}{\lambda_i} (E_i^{\gamma}) (1 - e^{-\lambda} i^t)$$

Combining the beta, secondary, and gamma absorbed doses for each nuclide yields

$$D_{i}^{\beta,x,\gamma}(t) = \frac{AC_{i}}{\lambda_{i}} (\overline{E}_{i} + E_{i}^{x})$$
$$+ E_{i}^{\gamma} (1-e^{\lambda_{i}t}) + E_{i}^{\gamma} ($$

This is the absorbed dose in rads in the waste at time (t) due to the decay of the i^{th} nuclide. The process must be repeated for all nuclides present in the waste to arrive at the total absorbed dose at time (t) after sealing.

 $D_{\text{Total}}(t) = \sum_{i} D_{i}^{\beta,x,\gamma}(t)$.

Free Volume of the Container. The free volume of the container is the sealed internal container volume not occupied by waste.

Free Volume = Internal Container Volume

- Waste Volume

The waste itself may contain interstitial voids that can be included in the free volume. The interstitial voids can be expressed as a void fraction of the waste. The void fraction can be determined from the bulk and true densities of the waste. Values for these properties are available for organic resins from vendor supplied information.

Void Fraction =
$$\frac{\text{True Density} - \text{Bulk Density}}{\text{True Density}}$$

Void fractions for mixed systems, such as cation and anion resins, can be approximated using weight percentages of each component.

Free Volume = (Internal Container Volume

- Waste Volume) + (Waste Volume

x Void Fraction)

The free volume for solidified waste is the internal container free space over the solidified waste. Credit for internal voids can be taken if the porosity of the solidified mixture is known or can be calculated.

Gas Generation Constants, Absorbed Dose, and Allowable Hydrogen Concentration. — The three components of the combustible gas generation calculation can be assembled to predict hydrogen volume at a given time. Combining the equations developed in previous sections yields

Hydrogen volume = G-factor

x total absorbed dose x mass of waste

This equation is expressed mathematically as

$$V_{\rm H} = G_{\rm H} \times D_{\rm Total} \times M_{\rm W} \times 2.33$$

x 10^{-6} eV, cm³/rad gram molecule.

The proportionality constant, $2.33 \times 10^{-6} \text{ eV}$. cm³/rad gram molecule is determined from

$$\frac{100 \text{ Erg/gram}}{\text{rad}} \times \frac{6.25 \times 10^{11} \text{ eV}}{\text{Erg}}$$

$$\times \frac{1 \text{ gram mole}}{6.02 \times 10^{23} \text{ molecules}}$$

$$\times \frac{22.4 \text{ liters}}{\text{gram mole}} \times \frac{1000 \text{ cm}^3}{\text{liter}}$$

= $2.33 \times 10^{-6} \text{ eV}$, cm³/rad gram molecule.

The total absorbed dose is a function of time.

NOTE: The G-factor, G_H is expressed

as $\frac{\text{molecules}}{100 \text{ eV}}$

The hydrogen concentration is obtained by dividing the hydrogen volume (V_H) by the free volume of the waste container (V_F) .

PRACTICAL CONSIDERATIONS

Considering the number of variables and the complexity of the calculation, a shielding computer code (QAD-FN) was used to determine the absorbed dose in different container geometries with various waste forms. The code was modified to calculate the waste's internal absorbed dose. This was accomplished by locating the modeled detector points inside the container waste matrix.

The computer calculates the absorbed gamma dose by evaluating the dose received at up to 200 detector sub-volumes (as a result of irradiation by a maximum of 1×10^6 source sub-volumes). The detector sub-volume gamma absorbed doses are summed to yield the waste's total absorbed gamma dose. Several assumptions are made:

- The container is full to the normal fill point
- The waste is of uniform density
- Gamma absorption constants are generated using energy absorption coefficients for water (see Appendix B)
- The nuclides are uniformly distributed in the waste matrix.

A more detailed description of the computer code model and its application is provided in Appendix C. Defining many of the variables in the foregoing sections requires complex calculations and in some cases, computer analysis. Much of this work has already been done. What follows is a description of the tables, graphs, and assumptions used to categorize waste forms and a method to simplify the calculational technique into a "user friendly" format.

The attenuation and absorption characteristics of most waste materials (anion and cation resins, concrete, asphalt, and water) are similar in the energy ranges normally encountered in radioactive waste. The amount of gamma ray energy absorbed by the waste is a function of the source strength energy and waste container geometry. A study of flux to dose relationships and absorption characteristics of common waste materials was performed to determine the most appropriate means of calculating gamma absorption. A comparison of absorption profiles for these waste forms found water to be the limiting waste (See Appendix B). The gamma absorbed dose is expressed as a fraction of total gamma energy for a specific container type.

Figures A-1 through A-4 express the gamma absorbed energy fraction as a function of container geometry, waste density, and gamma ray energy. These graphs were generated using the QAD-FN shielding code mentioned earlier and described in Appendix C. Appendix A gives an example of how the values from various tables and figures, along with pertinent facility information, are used to predict combustible gas concentrations.

APPENDIX D

DEVIATION OF ANALYTICAL SOLUTION FOR ENERGY ABSORPTION AT THE CENTER OF SPHERICAL SOURCE

The energy absorption at the center of the sphere has been solved analytically without build-up factors D^{-1} . It is modified with the same notation as Equation (4) in Appendix C,

• With integral

$$E_{abs,P}^{Center} = \mu_{c} + \int_{0}^{R_{0}} exp(-\mu_{e} r) dr \qquad (D-1)$$

Solution

$$E_{abs/P}^{Center} = [\mu_{u}/\mu_{e} [1 - \exp(-\mu_{e} R_{0})]$$
 (D-2)

where, R_0 is the radius of the sphere, $E_{abs/P}^{Center}$ is the absorbed energy without buildup factor at the center of the sphere and has dimensions of MeV/s \cdot cm³. Other valuables are defined in Appendix C.

Equation (D-1) can be rewritten by combining the build-up factor as

$$E_{abs/B}^{Center} = \mu_a I_0 \int_0^{R_0} exp(-\mu, r)B(\mu, r)dr \qquad (D-3)$$

$$B(\mu_{e} r) \equiv \sum_{i=0}^{3} B_{i} \cdot (\mu_{e} r)^{i}$$
 (D-4)

where, $E_{abs/P}^{Center}$ is the absorbed energy including build-up factor, B ($\mu_e r$). Build-up factor is defined as a third order polynominal of mean free paths. $\mu_e r$, and B_i are the fitting constants.

Equation (D-3) can be solved by partial integration and becomes

$$E_{abs/B}^{Center} = \mu_a \left[0 \left[-1/\mu_c \exp(-\mu_c \tau) \right] \right]$$
$$B(\mu_c \tau) - 1/\mu_c^2 \exp(-\mu_c \tau)$$

B'(μ r) $-1/\mu_e^3 \exp(-\mu$ r)

$$B''(\mu :) - 1/\mu_e^2 \exp(-\mu - r)$$

$$\mathbf{B}'''(\mu_e \mathbf{r}) \Big] \Big|_{\mathbf{0}}^{\mathbf{R}_0} \tag{D-5}$$

where

$$B'(\mu_{e} r) = B_{1} \mu_{e} + 2B_{2} \mu_{e}^{2} r + 3B_{3} \mu_{e}^{3} r^{2}$$

$$R''(\mu_{e} r) = 2B_{2} \mu_{e}^{2} + 6B_{3} \mu_{e}^{3} r \qquad (D-6)$$

$$B'''(\mu_{e} r) = 6B_{3} \mu_{e}^{3}$$

Place Equation D-6 into Equation D-5 and evaluate at R_0 and 0, then

$$E_{abs'B}^{Center} = \mu_{a} I_{0} \left\{ 1/\mu_{e} \\ \left[B_{0} + B_{1} + 2B_{2} + 6B_{3} \right] \\ -exp(-\mu_{e} R) \\ \left[(B_{1} + B_{1} + 2B_{2} + 6B)/\mu_{e} \\ + (B_{1} + 2B_{2} + 6B_{3})R_{0} + \mu_{e} \\ (B_{2} + 3B)R_{0}^{2} + \mu_{e}^{2} B R_{0}^{3} \right] \right\}$$
(D-7.

A more generic form is

$$E_{abs/B}^{Center} = \mu_{a} I_{0} \left\{ \sum_{k=0}^{3} \frac{1}{\mu_{e}^{k}} + 1 \left[B^{(k)}(0) - \exp(-\mu_{e} R_{0}) \right] \right\}$$

$$B^{(k)}(\mu_{e} R_{0}) \right\}$$
(D-8)

where

$$B^{(K)}(\mu_e r)$$
: k-th derivative of $B(\mu_e r)$ (D-9)

REFERENCES FOR APPENDIX D

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EPICOR GAS MEASUREMENTS

An important element in the overall program to justify use of a calculational technique to predict radiolytic gas generation was the need to 'benchmark' theoretical predictions against actual gas measurements in representative waste packages. Although there has been considerable research in this area (nearly all published by the NRC), most of the published work in the United States has been directed toward laboratory scale investigations to determine the constants for gas generation as a function of deposited energy (e.g., the G-value), or to assess the effects of organic resin degradation under high integrated radiation doses. There has been little study of actual size and similarly configured waste packages representative of nuclear utility wastes.

The exception to this was the work at TMI-2 with the experience gained during shipment of the EPICOR II prefilters and Submerged Demineralizer System (SDS) zeolite liners. Although the hydrogen generation characteristics for each of these waste forms had been evaluated and published, ¹⁻⁴ the applicability of the reported hydrogen generation rates were questioned due to the higher curie loading in relation to normal power plant resin waste. The SDS liners were considered to be too highly loaded with cesium and strontium for direct use in 'benchmarking' the EG&G model. However, it was felt that the EPICOR II prefilters might offer sound operational data on hydrogen generation rates for use in validating the current calculational technique.

Consquently, it was determined that an in-depth review of actual gas measurement data from TMI-2 dewatered resin water, particularly the EPICOR II prefilters, would be undertaken with the following objectives:

- Assess the reported EPICOR II hydrogen generation data
- Determine, if possible, the hydrogen generation rate in EPICOR II prefilters by methods independent of the techniques used in Reference 6.
- Assess the linearity of long-term average hydrogen generation rates as a function of deposited curies, in order to extrapolate to curie loadings more representative of utility wastes

- Summarize the available measurements of hydrogen gas in EPICOR II prefilters for use in 'benchmarking' the EG&G calculational model
- Develop a simple computer program which could accurately predict the generation of gases in representative waste containers.

Methodology

Hydrogen gas measurement data taken during preparations to ship the initial 50 EPICOR II prefilters was obtained from the following sources:

- "Preparations to Ship EPICOR Liners"⁶
- Completed TMI-2 Operating Procedure No. 2104-4.137, "EPICOR Liner Inerting," for each prefilter shipped
- GPU Nuclear correspondence transmitting prefilter summary data to EG&G Idaho prior to shipment from TMI.

During the initial review of available data to support this current evaluation, it became clear that several characteristics of the EPICOR II liners (and the contained resin media) had the potential to significantly affect the hydrogen data gathered during the sampling evolutions at TMI. These characteristics were judged to have an even greater influence on the prediction of hydrogen evolution using the EG&G calculational model. Therefore, considerable effort was devoted to quantifying accurately each liner's characteristics in order to ensure that the correlations and conclusion developed during the evaluation were based upon the best data available.

Much of the information subsequently used in this evaluation had to be summarized to preclude violation of the proprietary agreement with EPICOR, Inc. Several of the areas requiring detailed consideration are discussed in the following paragraphs.

• The residual, or interstitial, moisture content of irradiated resin media directly influences the generation of hydrogen, primarily because the production mechanism involves the radiolysis of water. Since different types of resin media were used in the EPICOR II prefilters, it became necessary to estimate the residual moisture content from liner dewatering tests performed at TMI, manufacturer's literature, the NRC, industry studies of resin dewatering, and knowledge of the actual resin material.

- Similarly, quantification of the free volume contained within an individual EPICOR II liner became necessary. This volume would directly affect not only the hydrogen concentration buildup in the liner, but also the calculation of hydrogen volume based upon measured concentration at liner opening. Accurate resin loading data was necessary for this determination also, since the free volume surrounding a specific resin is a function of the resin's size, physical and chemical form, and shrink/swell characteristics. Again, the required level of detail for this evaluation was obtained for all 50 prefilters.
- For purposes of correlating hydrogen generation against deposited energy, the most accurate method is to calculate total integrated dose, a key element of the EG&G model. The actual curie calculations used to support liner transportation and acceptance regulations by DOE for disposal, were used in this regard to calculate the total integrated dose to the resin material.
- In calculating the total hydrogen volume present within any liner when opened, pressure and temperature corrections are required to convert the gas to volume at standard temperature and pressure (STP). This correction was deemed necessary for this study to ensure that the effects of atmospheric temperature and pressure did not influence the projection of hydrogen present in the liner. Actual average atmospheric pressure for the day each liner was opened was obtained from the National Weather Service and was used as the appropriate correction factor for pressure. Although there was a wide variation in daily temperature, it was concluded that the location of the EPICOR II liners deep within their storage modules made actual temperature variations slight. Furthermore, the overall effects on gas volume at STP from temperature variations of the '

magnitudes expected in this situation are much less than the effects from pressure variations. Therefore, no temperature correction was used when calculating hydrogen volume from opened liners.

Many of the 50 EPICOR II prefilters were suspected to be unable to completely contain the buildup of internal gases, due to small leaks in gasket materials, penetration seals, etc. Although any suspected leaks were not expected to jeopardize the containment of radioactive material. hydrogen gas would certainly be able to escape from the liner due to its small molecular size. Actual data during the opening of several liners yielded no measured hydrogen with the absence of a positive (and in several instances, a negative) internal pressure, thereby confirming the existence of liner leaks. In order to differentiate between leaking and non-leaking liners for the purpose of performing hydrogen correlation studies, an opening gas pressure of 0.2 psig was used to define a non-leaking liner.

Using this definition, a total of 28 liners were considered to be "non-leaking" and therefore suitable for detailed investigation.

- The actual dates each liner was sealed when removed from process service, and then re-opened prior to inerting for shipment, were used to accurately define the storage interval. This time interval was then used to compute individual isotope decay correction for calculation of total absorbed energy.
- The hydrogen gas generation constant (G_H) for each individual liner was determined based upon a weighted average of the resin type (e.g., cation, anion, mixed bed) multiplied by the NRC published gas constant for the specific resin type. This method is consistent with that explained in the method calculation section of this report. Since the actual EPICOR-II resin media remains proprietary information, a detailed breakdown of these calculations cannot be reported herein.

EPICOR-II Prefilter Comparison

The concentration of hydrogen gas expected in each of the 28 non-leaking EPICOR II prefilters was calculated using the hydrogen prediction methodology developed by EG&G Idaho and discussed earlier. The purpose of this effort was to compare predicted concentrations with the concentrations actually measured when these liners were vented for shipment after nearly three years of storage.

For ease of practical application, the EG&G Idaho theoretical model was implemented on a desk top computer "electronic spreadsheet" program specifically developed by EPRI for this purpose. This spreadsheet resulted in a simple to use computer program for predicting the generation of gases in the EPICOR II liners.

EPICOR II prefilter resin and liner characteristics necessary for use with the EG&G Idaho methodology were obtained using the same data and methods described previously. Additional parameters were obtained from EG&G's calculational results summarized earlier in this report. Table 2 summarizes the results of the calculations using the spreadsheet program. In addition to basic data (liner number, deposited curies, waste volume, specific activity, and years in storage) for each prefilter, a comparison of the terminal (i.e., after several years of sealed storage) hydrogen concentrations actually measured [percent H₂ (M)] and the theoretically predicted hydrogen concentrations [percent H₂ (P)] using the EG&G Idaho model is also presented. Based upon results of the previous investigation, which concluded that only 28 of the original 50 prefilters were relatively leak tight, only these 28 liners were subjected to this comparison.

On the average, the predicted hydrogen gas concentrations were within 20% of the measured values. The mean ratio of measured to predicted concentrations was 84%, with a standard deviation of 11%. The calculational model over predicted the terminal hydrogen concentration in all but one of the liners, thereby providing a consecutively safe prediction of the generation. (These averages do not consider PF-10 which was over predicted by 51%.)

					Percent	Hydrogen	
PF Number	Ci	Waste (ft ³)	Ci (ft ³)	Years Storage	Measured (M)	Predicted (P)	Percent Measured/ Predicted
10	156.2	32	4.9	2.99	2.80	5.49	51
13	986.4	32	30.8	3.30	18.90	23 75	80
14	954.1	32	29.8	3.24	20.30	23.48	86
15	961.3	32	30.0	3.27	23.20	23.80	97
17	1184.0	32	37.0	3.27	18,90	27.48	69
18	1367.0	32	42.7	2.70	26.00	26.98	96
19	1342.0	33.5	40.1	2.83	25.00	27.32	97
20	1316.2	34.5	38.2	2.59	22.40	27.22	82
21	1316.2	34.5	38.2	3.21	21.50	30.89	70
22	1154.2	34.5	33.5	3.21	22.30	28.15	79
23	895.9	34	26.3	3.01	15.50	22.29	70
24	1316.2	34	38.7	3.20	21.20	28.72	47
25	1316.2	34	38.7	2.77	22.70	26 31	86
26	1316.2	34	38.7	2.73	25.60	26.03	08
29	1316.2	34	38.7	2.87	20.50	26.05	70
30	973.1	34	28.6	3.00	17.20	28.87	60
31	1109.6	34.5	32.2	3.13	22.50	27.36	00
32	1109.6	34.5	32.2	2.75	23.90	25.15	05
33	1109.5	34	32.6	2.82	23.50	23.76	95
34	1109.6	34	32.6	2.88	19.20	23.70	99
35	1109.6	34.5	32.2	2:67	26.20	24.10	106
36	1109.5	34.5	32.2	2.71	18 60	24.03	100
37	1109.6	34	32.6	2.95	20.10	27.09	73
38	1109.6	34	32.6	2.85	21.60	27.07	/ 3
39	1109.5	34	32.6	2.77	23.90	27.05	00 00
41	1109.5	34	32.6	2.98	19.70	26.12	90
42	1109.5	34	32.6	2.64	21.40	34 20	75
43	1109.5	34	32.6	2.76	24.10	24.88	00 97
Non-Leaking	g: Pressure : Percent H	> 0.2 psig Sydrogen >	15%			Average: w/o PF-10: St Dev:	82 84
						w/o PF-10:	11

Table 2. EPICOR II prefilters

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Electric Power Research Institute (EPRI) demonstrated the use of this calculational method at TMI-2. Twenty-eight "non-leaking" EPICOR II liners were used for the demonstration. EPRI used a desk top computer with a spreadsheet program to compare the predicted hydrogen concentration from the EG&G Idaho calculational method to the hydrogen concentration actually measured when the EPICOR II liners were vented during their preparation for shipment, after nearly three years of storage on site. On the average, the predicted hydrogen concentration was within 20% of the measured values.

EEI 'Task Force' representatives met with NRC Transportation Branch personnel to discuss the use of calculations to determine hydrogen gas generation in radioactive waste containers. As the basis for discussion, EEI presented the calculational method that EG&G Idaho and DOE developed. EEI requested that calculations be recognized as a means of satisfying the requirements for determining hydrogen gas concentration in radioactive waste containers, as set forth in NRC Inspection Enforcement Information Notice No. 84-72. The NRC acknowledged the validity of the method. Affected Certificates-Of-Compliance have been modified to allow calculations as an acceptable means of assessing gas generation, as well as tests and measurements set forth in NRC Inspection Enforcement Information Notice No. 84-72.

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APPENDIX A

SAMPLE PROBLEM

APPENDIX A

SAMPLE PROBLEM

Discussion

Worksheets A-1 and A-2 can be completed from plant records, vendor information supplied to plants, and tables and graphs in this appendix. Worksheet A-1 requires information from Table A-1 for Step 8, and vendor information for Step 6. Step 5 can be determined from vendor information or plant measurements. Worksheet A-2 values are obtained from Table A-2, Figures A-1 through A-4 (whichever figure is appropriate as to container type) and plant records. Column A of Worksheet A-2 is found by dividing the number of curies of a nuclide present by the mass of the waste (Worksheet A-1, Step 7). The calculations are admittedly tedious. Therefore, the technique should be thoroughly understood before attempting use. This method is valid for both dewatered resins and solidified wastes.

The sample problem uses data from NUREG/ CR 2830 on a high level reactor cleanup resin, shipped from the Vermont Yankee Plant in 1980, for isotopic activity density values.

Figures A-1 Through A-4

Gamma Fraction Absorbed. These graphs^a provide values for Worksheet A-2, column E. Find gamma ray energy(s) of nuclides in Table A-2 and place in column C of Worksheet A-2. Use this energy (column C) to find the gamma fraction absorbed by intercepting the density line closest to the waste density^b from Step 5, Worksheet A-1. Read the fraction on the y axis and place this value in column E, Worksheet A-2. In the example problem, the waste density from Table A-2, Step 5, is 0.78 g/cm^3 —so the 0.8 g/cm^3 line is used. For example, a 1 MeV gamma would be 0.48 absorbed.

a. These graphs are valid for both resin and solidified wastes.

b. If the waste density differs from the plotted values, use the density line that is greater—but closest to the actual waste density. More exact values for the gamma fraction absorbed as a function of density can be obtained using the methods outlined in Appendix C.

Worksheet A-1

Step	Variable	Value
1.	Volume of waste container ($ft^3 \times 2.83 E + 4 cm^3/ft^3$)	$1.83 \times 10^5 \text{ cm}^3$
2.	Heights (interior) of waste container	72 cm
3.	Radius (interior) of waste container	27 cm
4.	Volume of waste ($ft^3 \times 2.83 E + 4 cm^3/ft^3$)	1.65 x 10 ⁵ cm ³
5.	Bulk density of waste (lbs/ft ³ x 1.6 E-2 g cm ³ /lb/ft ³)	0.78 g/cm ³
6.	True density (specified gravity) of waste	1.3 g/cm^3
7.	Mass of waste (Step 4 x Step 5)	<u>1.29 x 10⁵ g</u>
8.	Radiolytic hydrogen generation constant, G _H (See attached Table A-1)	0.3 ^a molecules/100 eV
9.	Waste void space (expressed as a decimal) (Step 6 - Step 5) / Step 6	0.40
10.	Free volume (Step 1 - Step 4) + (Step 9 x Step 4)	$8.4 \times 10^4 \text{ cm}^3$
11.	Maximum allowable hydrogen volume (0.05 x Step 10)	4.2 x 10 ³ cm ³
12.	Absorbed dose for Step 11 [Step 11/(Step 8 x Step 7 x 2.33 E-06)]	4.7 x 10 ⁶ rad

0.35% anion $G_H = (0.65 \times 0.13) + (0.35 \times 0.6)$ $G_H = 0.3$

A-4

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W	or	ks	he	et	A-2.
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		Column							MeV	
Nuclide	Curies	Aa (activity density)	Bb (decay constant)	C ^b (gamma energy)	Db (abundance)	E ^c (% absorption)	Fd (Eን)	G ^b (Ē _i)	H ^b (E ^x _i)	
Mn-54	0.770	5.98 10.6	0.809	0.835	1.0	0.51	0.430	0.0	0.0052	
Co-58	0.757	5.88 10.6	3.575	0.811 1.675 0.511	1.0 0.0054 0.299	0.51 0.44 0.53	0.410 0.0040 0.081	0.0290 	0.0053 — —	
Co-60	3.93	3.05 E-5	0.132	1.173 1.332	1.0 1.0	0.48 0.47	0.560 0.630	0.0958 —		
Zn-65	2.61	2.03 E-5	1.036	1.115 0.511	0.508 0.0283	0.48 0.53	0.270 0.0077	0.0020	0.0077	
Cs-134	2.29	1.78 E-5	0.336	0.475 0.567 0.605 0.797 1.039 1.168 1.365	0.0146 0.238 0.976 0.941 0.0100 0.0180 0.0304	0.53 0.52 0.52 0.51 0.49 0.48 0.47	0.0037 0.070 0.307 0.382 0.005 0.010 0.020	0.157	0.0055 	
Cs-137/Ba-137	6.24	4.84 E-5	0.023	0.662	0.8998	0.52	0.310	0.171	0.0657	

a. a = Ci/mass of Waste (Worksheet A-1 Step 7).

b. From Table A-1.

c. Figure A-1 through A-4, fraction of a gamma energy absorbed.

d. Column F = Column C x Column D x Column E.

A-S

Resin	Ionic Form	(Molecules/100 eV) ^t
DOWEX 50WX10	Li +	0.11
DOWEX 50W10	H +	0.095
ZEO-KARB 215	H +	0.12
IRN-77	н+	0.13 ± 0.02
IRN-78	OH-	0.6
IRN-150	нон	0.5
IRN-77	Na ⁺	0.2
IRN-78	Cl-	0.3
IRN-150	NaCl	0.3

Table A-1. Hydrogen gas generation constants $(G_{\mu})^a$

a. The hydrogen gas generation yield for a mixed bed system is the sum of the yields of the individual components. For example, a mixed bed containing 0.40 resin A and 0.60 resin B by weight with $A:G_H = x$ and $B:G_H = Y$ is

 $G_{H}(A + B) = 0.4 x + 0.6Y$

For beds of unknown composition use

anion bed (OH ⁻)	$G_{H} = 0.6$
cation bed (OH ⁻)	$G_{\rm H}^{\rm H} = 0.13$
solidified resin/cement	$G_{H}^{11} = 0.24$
asphalt	$G_{H}^{i} = 0.41$

b. These values are the result of experimental analysis and appear as reported, for swollen (dewatered) resins from NUREG/CR-2830 and NUREG/CR-3383; cement values from NUREG/CR-2969, and values for asphalt from NUREG/CR-50617. The G-value is defined as the number of molecules formed or decomposed per 100 ev absorbed.

-					•	
Radionuclide Constants	Half Life Years	Decay Constant	Beta Energy Average MeV	Low X-Ray Energy Average MeV	Gamma Energy MeV/Disentegrations	Intensity Abundance Fraction
	1 228 1 1	5 6 4 2 E D	0067	0.0	0.0	0.0
H-3	1.228 E + 1	3.043 E-2	.0057	0.0	0.0	0.0
C-14	5.730 E + 3	1.209 E-4	.0495	0.0	0.0	0.0
Cr-51	7.585 E-2	9.13/E+0	0.0	.0047	.3201	.0983
Mn-54	8.361 E-1	8.095 E-1	0.0	.0052	.8348	.9998
Fe-55	2.700 E + 0	2.36/ E-1	0.0	.0056	0.0	0.0
Co-57	7.417 E-1	9.344 E-1	0.0	.0232	.1230	.9720
					.6920	.0016
Co-58	1.938 E-1	3.575 E + 0	.0290	.0053	.8110	1.0010
					1.6750	.0054
					.5110	.2986
Fe-59	1.222 E-1	5.671 E + 0	.1175	0.0	.1920	.0433
					1.0992	.5850
					1.2916	.4320
Ni-59	7.500 E + 4	9.240 E-6	0.0	.0067	0.0	0.0
Co-60	5.271 E + 0	1.315 E-1	.0958	0.0	1.1732	1.0000
					1.3325	1.0000
Ni-63	1.001 E + 2	6.923 E-3	.0171	0.0	0.0	0.0
Zn-65	6.691 E-1	1.036 E + 0	.0020	.0077	1.1155	.5075
					.5110	.0283
Kr-85	1.072 E + 1	6.465 E-2	.2505	. 0.0	.5140	.0043
Sr-89	1.384 E-1	5.007 E + 0	.5830	0.0	.9091	.0002
Sr-90	2.860 E + 1	2.423 E-2	.1958	0.0	0.0	0.0
Y-90		2.423 E-2	.9348	0.0	0.0	0.0
Y-91	1.602 E-1	4.326 E + 0	.6023	0.0	1.2049	.0030
Nb-94	2.030 E + 4	3.414 E-5	.1458	.0011	.7026	1.0000
					.8711	0000.1
Zr-95	1.753 E-1	3.954 E + 0	.1161	0.0	.7242	.4370
					.7567	.5530
Nb-95	9.599 E-2	7.220 E + 0	.0433	.0009	.7658	.9981
Tc-99	2.130 E + 5	3 254 E-6	.0846	0.0	0.0	0.0
Ru-103	1.077 E-1	6432E+0	0674	0032	4970	8940
		0.432 2 . 0			6100	.0635
Rh-103m		6432 F + 0	0.0	0388	0397	0007
Ru-106	$1.008 E \pm 0$	6 874 E-1	0100	0.0	0.0	0.0
Rh-106	1.000 E PO	6 874 E-1	1 4110	0.0	5110	2060
MI-100		0.07412-1	1.4110		6710	1040
					1 2510	.1040
Cd.109	$1.270 E \pm 0$	5 455 E-1	0.0	0196	0.0	0.0
Ag.110m	1.270 E + 0	1.000 E ± 0	0.0	.0190	6170	1 2540
Ag-110III	0.927 E-1	1.000 E + 0	.000	.0010	.0170	1.3340
					1 4520	1.4115
Sm 113	2 161 1 1	2 100 5 / 0	0.0		1.4520	.4200
511-113	3.131 E-1	2.199 E+0	0.0	.0232	.4331	.0193
In-113m		2.199 E + 0	0.0	.1360	.3917	.6490
Sn-119	8.022 E-1	8.639 E-1	0.0	.0837	0.0	0.0
Sn-123	3.337 E-1	1.959 E+0	.5203	0.0	1.0886	.0060
50-124	1.648 E-1	4.205 E + 0	.3/76	.0200	.6030	.9830
					.6460	.0740
					.7240	.1580
					1.0060	.0378
		,			1.3540	.0545
					1.4750	.0381
					1.6910	.4900
					2.0910	.0573
				-		

Table A-2. Radionuclide decay properties

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Radionuclide Constants	Half Life Years	Decay Constant	Beta Energy Average MeV	Low X-Ray Energy Average MeV	Gamma Energy MeV/Disentegrations	Intensity Abundance Fraction
Sb-125	2.770 E + 0	2.502 E-1	.0865	.0254	1760	0815
					3800	0274
					4370	4020
					6170	3590
Te-125m	1.588 E-1	4.364 E+0	0.0	.1410	1093	0028
1-129	1.570 E + 7	4.414 E-8	.0409	.0380	0396	0752
I-131	2.201 E-2	3.148 E + 1	.1847	.0124	2840	0696
			•		.3640	.8140
					6370	0777
					.7230	.0180
Cs-134	2.062 E + 0	3.361 E-1	.1570	.0055	.4750	0146
					.5670	.2380
					.6050	.9760
					.7970	.9410
					1.0390	.0100
					1.1680	.0180
					1.3650	.0304
Cs-136	3.656 E-2	1.896 E + 1	.1001	.0388	.0805	.2290
					.1680	.2690
					.2740	.1270
					.3410	.4910
					.5070	.0098
					.8190	.9970
					1.0480	.7960
					1.2350	.1970
Cs-137	3.017 E + 1	2.297 E-2	.1708	0.0	0.0	0.0
Ba-137m		2.297 E-2	0.0	.0657	.6616	.8998
Ba-140	3.553 E-2	1.951 E + I	.2720	.0384	.1630	.0687
					.3050	.0450
					.4300	.0535
					.5370	.2500
La-140	4.655 E-3	1.489 E + 2	.5269	.0056	.3290	.2160
					.4840	.4840
					.8160	.3350
					.9250	.1050
					1.5960	.9550
					2.4890	.0442
Ca-141	9.028 E-2	7.676 E + 0	.1447	.0316	.1454	.4840
Ce-144	7.897 E-1	8.775 E-1	.0827	.0140	.1340	.1190
					1.4890	.0030
					2.1850	.0077
Pr-144		8.775 E-1	1.2072	.0008	.6960	.0148
					2.1860	.0097

Table A-2. (continued)



Figure A-1. Gamma absorption in a 55 gal., 7.5 ft³ drum.

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A-10

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Figure A-3. Gamma absorption in a 5 x 5 ft, 98 ft³ liner.

A-11



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ABSORBED DOSE CALCULATION

The absorbed dose is calculated for each nuclide (individually) using the information from Worksheets A-1 and A-2. The calculation is performed for several chosen times after the container is sealed. The dose to the waste from each nuclide, at a chosen time, is then summed to give a total absorbed dose at that time. For low-level waste, good choices for time intervals are t = 1, 5, 10, and 20 years. If the calculated total dose for a time interval exceeds the value of step 12, Worksheet A-1, the 5% hydrogen concentration level has been exceeded.

For each nuclide, the absorbed dose equals

$$\frac{AC_i}{\lambda} \left(\vec{E}_i + E_i^x + E_i^y \right) \left(1 - e^{-\lambda} i^t \right) .$$

In the columnar notation of Worksheet A-2

$$\frac{Aa}{\lambda_i} (g + h + f) (1-e^{-bt})$$

where the constant A

=
$$1.86 \text{ E} + 10 \frac{\text{rad gram}}{\text{MeV yr Curie}}$$

from
$$\frac{\text{rad}}{(100 \text{ erg/g})} \times \frac{1.6 \text{ E-12 erg}}{(\text{eV})}$$

 $x \frac{\text{E+6 eV}}{(\text{MeV})} \times \frac{3.7 \text{ E+10 D/sec}}{(\text{Curie})}$
 $x \frac{3.15 \text{ E+7 sec}}{(\text{yr})}$.

Solving for t = 1 year

$$Mn-54: D_{abs} = (1.86 E+10)$$

$$(5.98 E-6)(0.0052 + 0.430)$$

$$(1-exp-0.809)/0.809$$

$$= 3.32 E+4 rad$$

$$Co-58: D_{abs} = (1.86 E+10)(5.88 E-6)$$

$$[0.029 + 0.0053 +$$

$$(0.410 + 0.004 + 0.081)]$$

$$(1-exp-3.575)/3.575$$

$$= 1.57 E+4 rad$$

$$Co-60: D_{abs} = (1.86 E+10)(3.05 E-5)$$

$$(0.0958 + 0.560 + 0.630)$$

$$(1-\exp-0.1315)/0.1315$$

$$= 6.84 \text{ E} + 5 \text{ rad}$$

$$Zn-65: D_{abs} = (1.86 \text{ E} + 10)(2.03 \text{ E} - 5)$$

$$(0.0020 + 0.0077 + 0.270$$

$$+ 0.0077)$$

$$(1-\exp-1.036)/1.036$$

$$= 6.75 \text{ E} + 4 \text{ rad}$$

$$Cs-134: D_{abs} = (1.86 \text{ E} + 10)(1.78 \text{ E} - 5)(0.157$$

$$+ 0.0055 + 0.0037 + 0.070$$

$$+ 0.307 + 0.382 + 0.005$$

$$+ 0.010 + 0.020)$$

$$(1-\exp-0.336)/0.336$$

$$= 2.70 \text{ E} + 5 \text{ rad}$$

$$Cs-137/$$

$$Ba-137: D_{abs} = (1.86 \text{ E} + 10)(4.84 \text{ E} - 5)$$

$$(0.1708 + 0.0657 + 0.31)$$

$$(1-\exp-0.023)/0.023$$

$$= 4.86 \text{ E} + 5 \text{ rad}$$

The total absorbed dose at 1 year after sealing is

D (1 yr) Total =
$$3.32 E+4 + 1.57 E+4$$

+ $6.84 E+5 + 6.75 E+5 + 2.70 E+5$
+ $4.86 E+5 = 1.55 E+6 rad$.

Solving in the same manner for t = 5 years

Mn-54:	5.88 E + 4 rad
Co-58:	1.62 E + 4 rad
Co-60:	2.67 E + 6 rad
Zn-65:	1.04 E + 5 rad
Cs-134:	7.70 E + 5 rad
Cs-137/Ba-137:	2.32 E + 6 rad

The total absorbed dose 5 years after sealing is

 D_{Total} (5 yrs) 5.94 E + 6 rad .

To find the approximate time when 5% hydrogen concentration occurs, a graphical technique using Worksheet A-1 and the calculated absorbed dose is used.

Hydrogen Concentration - Graphical Solution

Hydrogen concentration is graphed versus time using the 1, 5, and 10 year doses with the values from the steps in Worksheet A-1.

- NOTE: Because the five year dose exceeds the value of Step 12 on Worksheet A-1, the 10 year dose need not be calculated.
- $H_2 = D_{Total} \times \text{Step 7} \times \text{Step 8} \times 2.33 \text{ E-6}$

x 100 percent/Step 10 .

For 1 year: percent $H_2 = (1.56 \text{ E} + 6)(1.29 \text{ E} + 5)(0.3/100)(2.33 \text{ E} - 6) (100 \text{ percent } / 8.4 \text{ E} + 4)$ = 1.67%.

For 5 years: percent $H_2 = (5.93 \text{ E}+6)(1.29 \text{ E}+5)(0.3/100) (2.33 \text{ E}-6)(100 \text{ percent})/8.4 \text{ E}4 = 6.38\%$.

These values are plotted as shown in Figure A-5. The graph predicts 3.7 years to produce 5% hydrogen. To verify, set t = 3.7 years and calculate D_{Total} . The value is found to be 4.73 E+6 rads, slightly higher than the total Step 12 value of 4.7 E+6 rads.

Calculating the percent of H₂

(4.73 E + 6)(1.29 E + 5)(0.3/100)(2.33 E - 6)

(100 percent)/8.4 E + 4 = 5.08 percent

The time to generate 5% hydrogen is approximately 4.0 years by calculation. The shipping window is then half this time, or 2 years.



Figure A-5. Hydrogen concentration values plotted against time.

Comment

The most desirable approach to solving this type of problem is through the use of computers. The information from Worksheet A-2 is placed in the computer memory along with the waste density (Step 5, Worksheet A-2), the container type, and constant A = 2.33E-6. The computer iterates the equation

Absorbed dose = $\frac{Aa}{b}$ (g + h + f) (1-e^{-bt})

for values of time (t), until the absorbed dose of Step 12, Worksheet A-1 is reached. This yields the time to produce a 5% hydrogen concentration.

APPENDIX B

WASTE MATERIAL STUDY

WASTE MATERIAL STUDY

Investigation of the characteristics of commonly encountered radioactive waste materials led to the selection of water as a generically representative substance on which calculations could be based. Calculations based on water encompass the widest spectrum of nuclear wastes generated. This choice produces gamma absorbed energy fractions that are slightly conservative (when compared to other waste materials) but more accurate than previously used techniques which have their basis in health physics considerations.

Attenuation, absorption coefficients, and build up factors are assumed to be constant for a variety of waste forms, but are varied by the changes of waste density and gamma energy.

Waste Forms

The wastes to be discussed here are those radioactive forms which have the potential to generate combustible quantities of gas. Included are ion exchange resins, filter aids, evaporator concentrates, and solidified products. Waste solidification is usually done using cement, polystyrene, polyethylene, or asphalt. Most of these materials, including solidification agents, are composed of low atomic number elements; they are expected to show similar radiological characteristics in attenuation and absorption of gamma radiation. Some important properties of these materials are listed in Table B-1. National Bureau of Standards (NBS) concrete parameters are used for cement. Portland Cement's radiological behavior will be similar to that published by NBS due to its nearly identical electron density, as shown in Table B-2. Asphalt is considered as a mixture of polystyrene, polyethylene and cellulose. Filter aid is simplified as 100 percent cellulose, and resin is represented by a mixture of polystyrene and sodium sulfate or boric acid.

Gamma radiation absorption for low atomic number elements is nearly proportional to electron density. This is because for gamma rays with an energy range of 0.2, approximately 2.0 MeV lose their energy by scattering the electrons (compton effect) in the media where they travel.

Attenuation and Absorption Coefficients, and Buildup Factors

Attenuation and absorption coefficients, and buildup factors for the majority of radioactive wastes are largely a function of the material's electron density. An evaluation of the variance of these properties was performed for water and concrete. Inspection of Table B-1 shows that waste material electron densities fall between the values reported for water and concrete (solidifying agents are combined with materials of lower electron density), so that these substances bound the waste material evaluation. Attenuation and absorption coefficients for water and concrete are listed in Table B-3; buildup factors are listed in Table B-4 and B-5. The values for attenuation and absorption coefficients for water and concrete vary by approximately 10 percent, in the energy range of interest. The actual absorption behavior for water and concrete is similar because the combination of absorption and attenuation coefficients for the individual materials (water or concrete) tends to reduce the difference between them. Energy absorption buildup factors for water and concrete also show small differences. The buildup factor variance can become significant with increasing mean free paths; however, more than 99 percent of the gamma ray's energy is absorbed within ten mean free paths, where the variance is small. Table B-6 lists buildup factor coefficients fitted for 0 to 10 mean free paths for water and concrete. Combined effects of these are illustrated by Figures B-1 and B-2 which show the absorption characteristics for concrete and water as a function of the radius of a sphere, with the source at the center. Two different source strengths, 0.4 and 1.5 MeV, were chosen to represent typical values encountered in radioactive waste. Inspection of these curves show gamma energy absorption in water to be slightly higher than absorption in concrete.

Despite the differences in absorption and attenuation coefficients, absorbed energy fractions are quite close in water and other waste materials; the water medium cases show slightly (2 to 3%) higher percents of gamma energy absorption. This overestimation is reduced when averaged over several gamma ray travel lengths.

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Components	Electron Density ^a $\left(10^{-23} \frac{\text{electrons}}{\text{gram}}\right)$	Gravimetric Density (g/cm ³)	Note
Sodium Sulfate	2.97	1.49 ^b	Concentrate
Concrete	3.01 ^c	2.35 ^c	Solidification Agent
Boric Acid	3.12	1.46	Concentrate
Cellulose	3.20	1.0	Filter Aid
Polystylene	3.24	$1.05 \sim 1.07$	Skeleton of Resin
Water	3.37	1.0	-
Polystyrene	3.44	~1.0	Solidification Agent
Asphalt	$3.20 \sim 3.44^{\mathrm{d}}$	$1.05 \sim 1.07$	Solidification Agent

Table B-1. Electron and gravimetric density of waste materials

a. Electron Densities are obtained from Table 3-11, EPRI NP-3223

- b. Ten Hydrate
- c. NBS Concrete, Nuclear Science and Engineering, 263, 56 (1975)
- d. Expected Range

The choice of water as a representative material for use in the general calculation yields results for gamma absorbed energy that are accurate and slightly conservative when considering materials other than water.

		Portland	Cement ^a	NBS Concrete ^b		
Element	Electron Density (e/g · E-23)	Composition (percent)	Electron Density (e/g · E-23)	Composition (percent)	Electron Density (e/g · E-23)	
н	5.97	_	0.00	0.56	0.03	
0	3.01	35.76	1.08	49.83	1.50	
Na	2.87	_	0.00	1.71	0.05	
Mg	2.97	1.21	0.04	0.24	0.01	
AI	2.90	4.23	0.12	4.56	0.13	
Si	3.00	10.75	0.32	31.58	0.95	
S	3.00		0.00	0.12	0.00	
К	2.93	_	0.00	1.92	0.06	
Ca	3.00	45.25	1.36	8.26	0.25	
Fe	2.80	2.80	0.08	1.22	0.03	
		100.00	2.99	100.00	3.01	

Table B-2. Comparison for elementary composition of NBS Concrete and Portland Cement element

a. CRC Handbook of Health Physics.

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b. Nuclear Science and Engineering, 263, 56 (1975).

Photon	Attenuation ^a /Absorption ^b Coefficient (cm ² /g)		
(MeV)	Water	Concrete	
0.1	0.171/0.0256	0.179/0.424	
0.15	0.151/0.0277	0.144/0.290	
0.2	0.137/0.0297	0.127/0.290	
0.3	0.119/0.0319	0.108/0.295	
0.4	0.106/0.0328	0.0963/0.298	
0.5	0.0968/0.0330	0.0877/0.300	
0.6	0.0896/0.0329	0.0810/0.0297	
0.8	0.0786/0.0321	0.0709/0.0289	
1.0	0.0707/0.0309	0.0637/0.0279	
1.5	0.0575/0.0282	0.0519/0.0254	
2.0	0.0494/0.0260	0.0448/0.0235	

Table B-3. Comparison of attenuation/absorption coefficients for concrete and water

a. Health Physics Handbook

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b. CRC Handbook of Radiation Measurement and Protection, Section A, CRC Press Inc., 1978.

Mean						Energy	r (MeV)	. <u></u>		 				· · · · · · · · · · · · · · · · · · ·
Free Path	2.000	1.500	1.000	0.800	0.600	0.500	0.400	0.300		0.200		0.150		0.100
0.0	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00		1.00		1.00		1.00
0.5	1.38	1.42	1.47	1.51	1.56	1.61	1.66	1.75		1.92 ^a		2.07 ^a		2.36
1.0	1.83	1.93	2.08	2.18	2.34	2.45	2.60	2.84		3.42 ^a		3.91		4.52
2.0	2.82	3.11	3.62	3.96	4.48	4.87	5.42	6.25		8.22		9.36		11.7
3.0	3 87	4 44	5.50	6.24	7.40	8.29	9.56	11.5		15.7		18.6		23.5
4.0	4 99	5.90	7.66	8.96	11.1	12.7	15.1	19.0		26.4		32.5	·	40.6
5.0	6 16	7 47	10.1	12.1	15.4	18.1	22.2	28.8		41.3		52.0		64.0
6.0	7.38	9.14	12.8	15.6	20.6	24.6	30.8	41.2		61.0		77.9		94.8
7.0	8.66	10.9	15.7	19.6	26.4	32.2	41.1	56.5		86.2		111		134
8.0	9.97	12.8	18.9	24.0	33.0	40.8	53.2	75.0		118		153		183
10.0	12.7	16.8	26.0	33.9	48.7	61.8	83.2	122		202		268		314
15.0	20.1	27.9	47.4	65.6	102	137	197	318		582		805		917
20.0	28.0	40 4	73.5	106	176	247	377	656	1	310	1	890	2	120
20.0	26.0	54 1	104	156	272	395	632	1180	2	580	3	840	4	260
20.0	15 2	68.8	138	213	388	582	972	1930	4	640	7	050	7	780
35.0	-J.4 51 3	84 4	175	277	525	809	1400	2950	7	890	12	100	13	100
40.0	63.6	101	214	349	683	1080	1940	4280	12	800	19	600	20	300

Table B-4. Buildup factors for water^a

a. A. B. Chilton et. al., Nuclear Science and Engineering, 97, 73, (1980).

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Mean	Energy (MeV)										
Free Path	2.000	1.500	1.000	0.800	0.600	0.500	0.400	0.300	_0.200	0.150	0.100
0.0	1.00 + 00	1.00+00	1.00+00	1.00 + 00	1.00+00	1.00 + 00	1.00+00	1.00 + 00	1.00+00	1.00 + 00	1.00 + 00
0.5	1.39 + 00	1.42 + 00	1.49 + 00	1.53 + 00	1.60 + 00	1.66 + 00	1.73 + 00	1.86 + 00	2.11 + 00	2.32 + 00	2.39+00
1.0	1.83 + 00	1.93 + 00	2.11 + 00	2.22 + 00	2.41 + 00	2.55 + 00	2.74+00	3.06 + 00	3.65 + 00	4.04 + 00	3.89+00
2.0	2.80 + 00	3.08 + 00	3.59+00	3.94 + 00	4.48+00	4.89 + 00	5.46 + 00	6.32 + 00	7.69+00	8.29+00	7.06+00
30	3.82 + 00	4.36+00	5.35+00	6.03 + 00	7.10+00	7.89+00	9.00 + 00	1.06 + 00	1.29+01	1.35 + 01	1.04 + 01
4.0	4.92 ± 00	5.77 + 00	7.35 + 00	8.48 + 00	1.03 + 01	1.16+01	1.34 + 01	1.60 ± 01	1.93 ± 01	1.96 + 01	1.41+01
5.0	6.07 + 00	7.28 ± 00	9.61 + 00	1.13 + 01	1.40 + 01	1.60 + 01	1.87 + 01	2.25 + 01	2.70 + 01	2.67 + 01	1.79+01
6.0	7.28 + 00	8.90 + 00	1.21+01	1.45 + 01	1.82+01	2.11+01	2.50 + 01	3.03 + 01	3.60 + 01	3.48 + 01	2.21 + 01
7.0	8,55+00	1.06 + 01	1.48+01	1.80 + 01	2.31+01	2.70+01	3.23 + 01	3.95 + 01	4.66 + 01	4.40 + 01	2.66 + 01
8.0	9.86 + 00	1.24 ± 01	1.78 ± 01	2.19 ± 01	2.86+01	3.37+01	4.07 + 01	5.01+01	5.88 + 01	5.45 ± 01	3.15+01
10.0	1.26 + 01	1.63 + 01	2.43 ± 01	3.07 + 01	4.13 + 01	4.96+01	6.06+01	7.62 + 01	8.85 + 01	7.91 + 01	4.21 + 01
15.0	2.00 + 01	2.71+01	4.40 + 01	5.85 + 01	8.40 + 01	1.05 + 02	1.34 + 02	1.74 + 02	1.99 + 02	1.65 + 02	7.47 + 01
20.0	2.81+01	3.93 + 01	6.79+01	9.41 + 01	1.42 + 02	1.83+02	2.43 + 02	3.26 + 02	3.72 + 02	2.92 + 02	1.16+02
25.0	3.67 + 01	5.26 + 01	9.55 + 01	1.37 + 02	2.16 + 02	2.86 + 02	3.92 + 02	5.43 + 02	6.20 + 02	4.63 + 02	1.65 + 02
30.0	4.57 + 01	6.70+01	1.26 + 02	1.86 + 02	3.05 + 02	4.14 + 02	5.83 + 02	8.34 + 02	9.58 + 02	6.86 + 02	2.22 + 02
35.0	5.50 + 01	8.21 + 01	1.60 + 02	2.42 + 02	4.08 + 02	5.67 + 02	8.20 + 02	1.21 + 03	1.40 + 03	9.63 + 02	2.86 + 02
40.0	6.46 ± 01	9.80+01	1.97 + 02	3.03 + 02	5.27 + 02	7.47 + 02	1.10+03	1.67 + 03	1.94 + 03	1.30 + 03	3.58 + 02

Table B-5. Buildup factors for concrete^a

a. C. M. Eisenhauer, G. L. Simmons, Nuclear Science and Engineering, 263, 56, (1975).

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		Water Media	Coefficient	
Energy (MeV)	2.0000 E-0	1.5000 E + 0	1.0000 E + 0	8.0000 E-1
B0	1.0000 E + 0	1.0000 E + 0	1.0000 E + 0	1.0000 E + 0
B1	7.5284 E-1	8.1045 E-1	8.6259 E-1	8.9863 E-1
B2	7.3047 E-2	1.1920 E-1	2.2233 E-1	2.9110 E-1
B3	-3.2540 E-3	-4.3743 E-3	-6.0645 E-3	-5.3580 E-3
Energy (MeV)	6.0000 E-1	4.0000 E-1	2.0000 E-1	1.0000 E-1
B0	1.0000 E + 0	1.0000 E + 0	1.0000 E + 0	1.0000 E + 0
B1	9.4167 E-1	1.0282 E + 0	1.5335 E + 0	2.0926 E + 0
B2	3.9744 E-1	5.3743 E-1	7.6967 E-1	1.2865 E + 0
B3	1.6241 E-3	1.5738 E-2	1.0831 E-1	1.6278 E-1
		Concrete Med	ia Coefficient	
Energy (MeV)	2.0000 E + 0	1.5000 E + 0	1.0000 E + 0	8.0000 E-1
B0	1.0000 E + 0	1.0000 E + 0	1.0000 E + 0	1.0000 E+0
B1	7.6801 E-1	8.1785 E-1	9.2308 E-1	9.7877 E-1
B2	6.1447 E-2	1.0697 E-1	1.8204 E-1	2.3777 E-1
B3	-2.3030 E-3	-3.7178 E-3	-4.2724 E-3	-4.0301 E-3
Energy (MeV)	6.0000 E-1	4.0000 E-1	2.0000 E-1	1.0000 E-1
B0	1.0000 E + 0	1.0000 E + 0	1.0000 E + 0	1.0000 E + 0
B1	1.0880 E + 0	1.2960 E + 0	2.0200 E + 0	2.7531 E + 0
B2	3.1374 E-1	4.3558 E-1	6.0804 E-1	1.2137 E-1
B3	-2.1601 E-3	-3.1103 E-3	5.9321 E-3	1.3481 E-3

Table B-6. Buildup factor coefficient^{a,b,c}

a. Fitted Over 0 to 10 mean free paths.

b. Buildup Factor = $B0 + B_1X + B_2X^2 + B_3X^3X$: mean free path length.

c. All constants are calculated using the least square method described in EPRI NP-3223 and fitted to the buildup factors listed in Table B-4.



Figure B-1. Gamma absorption in spheric media.

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Figure B-2. Gamma absorption in spheric media.

APPENDIX C

COMPUTER APPLICATION

APPENDIX C

COMPUTER APPLICATION

The gamma absorbed dose was calculated for standard waste container geometries using a pointkernal computer code. The code, QAD-FN, a version of the QAD code developed by R. E. Malenfant, is in current use at the Idaho National Engineering Laboratory.^{C-1} Modifications were made to increase the number of source and detection points so that larger container geometries could be calculated. The attenuation, absorption, and buildup factors of the generic waste form discussed in Appendix B were used to define the gamma absorption characteristics of standard waste forms normally encountered in the nuclear power industry.

Basic Equations

Radiation absorption is based on an equation for point-to-point relations of radiation attenuation and absorption. For a discrete gamma ray, the energy flux, $[\phi(r)]$, at a distance (r) can be expressed as

$$\phi(r) = \frac{I_0 \exp(-\mu_e r) B(\mu_e r)}{4\pi r^2} dV_s \qquad (C-1)$$

where

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- $\phi(r) = \text{energy flux at distance r, (MeV/sec \cdot cm^3)}$
- $I_0 = \text{source intensity, (MeV/sec} \cdot \text{cm}^3)$
- μ_e = liner attenuation coefficient, (1/cm)
 - = distance from a source, (cm)

 $dV_s = source volume, (cm^3)$.

The energy absorbed at target with small volume, dV_{R} , is

$$E_{abs}(r) = \mu_a \phi(r) dV_g \qquad (C-2)$$

where

$$E_{abs}(r) = energy absorbed at distance r, (MeV/s)$$

 μ_a = liner energy absorption coefficient, (1/cm) ·

$$dV_R = Target volume, (cm^3)$$

Combining Equations (C-1) and (C-2),

$$E_{abs}(r) = \mu_{a} \frac{I_{0} \exp(-\mu_{e} r)B(\mu_{e} r)}{4\pi r^{2}}$$
$$dV_{s} dV_{R} \qquad (C-3)$$

The total amount of energy absorbed in the media can be calculated by integrating Equation (C-3) over the source and irradiated media as

$$E_{abs}^{Total} = \mu_a I_0 \int_{V_S} \int_{V_R} \frac{\exp(-\mu_e r)B(\mu_e r)}{4\pi r^2} dV_S dV_R \qquad (C-4)$$

where

 E_{abs}^{Total} is the total absorbed energy, (MeV/s), in an irradiated media.

For cylindrical geometry with homogeneous source distribution, Equation (C-4) is

$$E_{abs}^{Total} = \mu_{a} \quad l_{0} \qquad \int_{\phi_{D} Z_{D} r_{D}} \int_{\phi_{S} Z_{S} r_{S}} \frac{exp(-\mu_{e} x)B(\mu_{e} x)}{4\pi r^{2}}$$
$$r_{D} r_{S} d\phi_{S} dz_{S} dr_{S} d\phi_{D} dz_{D} dr_{D} \qquad (C-5)$$

and

$$x \equiv \sqrt{z_D^2 - z_S^2 + r_D^2 + r_S^2 - 2r_D r_S \cos(\phi_S - \phi_D)} \quad (C-6)$$

where ϕ , z, r are the coordinates of a cylinder, and subscripts D and S are detection and source points, respectively. Distance, X, is defined in Equation (C-6). A schematic of this coordinate system is shown in Figure C-1. Then, Equation (C-5) can be rewritten with integration ranges as

$$E_{aos}^{Total} = 4\pi\mu_a \quad I_0 \qquad \int_{\frac{r}{2}}^{Z} \int_{0}^{R} \left[2 \int_{0}^{r} \int_{0}^{Z} \int_{0}^{R} \frac{exp(-\mu_c x)B(\mu_c x)}{4\pi x^2} r_D r_S d\phi_S dz_S dr_S \right] dz_D dr_D . \quad (C-7)$$



Figure C-1. Source and detection points in relation to the cylindrical coordinate system.

In actual computation, the integrals are substituted by the sum of point-to-point radiation absorption

$$E_{abs}^{Total} = 2\mu_a I_0 \sum_{i=1}^{n} \left[2 \sum_{j=1}^{m} \frac{\exp(-\mu_e x_{ij})B(\mu_e x_{ij})}{4\pi x_{ij}^2} \right]$$
$$\Delta V_{S,i} \Delta V_{D,i} \qquad (C-8)$$

with

$$\Delta V_{S,j} = r_S \Delta r_S \Delta \phi_S \Delta z_S$$

$$\Delta V_{D,i} = 2\pi r_S \Delta r_D \Delta z_D$$
(C-9)

where, $V_{S,j}$ and $V_{D,i}$, are the jth source and the ith detector sub-volumes, with the source point assumed at the center of each source sub-volume. The sign Δ , with ϕ , z, and r, are subdivisions of each direction, and m and n are the number of sources and detectors, respectively.

Determination of source and detector points networks are discussed in the following sections.

Source Point Networks. Source point networks were examined by the following steps. Standard arrangements were selected for the generic calculations.

- Comparison with the analytical solution was made to verify the numerical method described in the previous section, using the analytical solution for a sphere.
- The source point networks for cylindrical configurations were developed and various arrangements examined to verify the standard setting of source point networks.

Model comparisons between analytically obtained energy absorption (see Appendix D) and numerical solutions have been done for waste densities of. $0.6 \sim 2.0$ g/cm³, and energy levels,

0.4 and 1.5 MeV. The radiological constants used are defined in Appendix B. The numerical method is principally the same as the j^{th} sum of Equation (C-8). Because of spherical symmetry, subdivisions are considered in the container radius direction with the source at the center of each subdivision. The modeled configuration and equations used are shown in Figure C-2.

Tables C-1 and C-2 show the comparisons of two methods (analytical and numerical). Table C-1 is the contribution from 0 to 10 cm in radius; Table C-2 is the contribution from 10 to 20 cm. The deviations of numerical results from analytically obtained values are decreased by increasing the number of subdivisions. Deviations are less than 1 percent for subdivisions of widths smaller than 5 cm in all cases. These results suggest that the source subdivisions surrounding detection points should be arranged such that the width is less than 5 cm.

Based on these sample calculations, absorbed energy was estimated for dewatered resin packed in 55 gal drums to determine a source point network for a cylindrical configuration. A standard source subdivision arrangement is illustrated with the relation between detection plane and source volume shown in Figure C-3. The source is assumed to be homogeneously distributed over the waste volume, and is divided into sub-volumes along three axis, Z (Height), R (Radius), and ϕ (Radial), according to the mesh generating capability of the QAD-FN code. The source region, where the detectors are included, was divided into 1-cm width subdivisions in Z and R directions, and $\pi/30$ in the radial direction. The largest width of a source subdivision is approximately 2.7 cm at the edge of a cylinder in the radial direction. To calculate the total absorbed energy, 48 detection points are placed on the detection plane in the symmetrical center plane with half-height of the waste media, according to the detection point networks shown in Figure C-4. The details of the detection point arrangement will be discussed in the next section.

The major valuables are summarized in Table C-3. The result is shown in Table C-4 with Case No. 2 (standard). Results with different source point arrangements are listed to illustrate the effects of source point network changes. The number of source subdivisions were decreased in Case 1 for every direction; the number was increased in Cases 3 and 4 for Z direction, in Case 5 for R direction, and Cases 6 and 7 for ϕ direction. These results show that the deviations of absorbed energies with various source point arrangements are usually small. Comparison of Cases 1 and 2, indicates that the source subdivisions should not be decreased from the standard. Although the relatively high deviation in Case 6 (subdivision increased in ϕ direction) is observed, further increments in this direction (Case 7) causes only a 0.17% increase from Case 6. Changes in absorbed energies are almost in equilibrium with the standard case, with deviations in the order of 1%. The standard setting of the source point arrangement discussed is used for the generic calculation, with basic source point arrangements maintained for the larger containers.

Detection Point Networks. A detection point is designed to represent a certain sub-volume of waste media (i.e., it is assumed that a detection point will show the average absorbed energy in the subvolume where it is located). The internal dose profiles for dewatered resin in a 55 gal drum are shown in Figure C-5, with the relation between detection points and representive ranges illustrated. The dose profiles of the center (z = 42 cm) and edge (z = 83 cm) are plotted against the radius (center to edge). The detection point at 7 cm on Figure C-5 is representative of the absorbed dose from 3 to 11 cm. The error associated with the choice of the representative detection point can be evaluated by calculating the upper and lower average absorbed dose as shown in Figure C-6. The upper average is defined by the midpoint dose of a range in radius or height $(X_{i-1} \text{ to } X_{i+1})$ as an average value. The lower average is the average of all applicable data points in the range of interest. When dose profiles are monotonously decreasing with a convex shape as shown in Figure C-5, the upper average will always exceed the true dose profiles but the lower average will not.

Four different lines of dose profiles are examined for two energy levels (0.4 MeV and 1.5 MeV) to estimate the possible error associated with this method.

Tables C-5 (height) and C-6 (radius) show the estimated error ranges of sample calculations. The error ranges are expressed by percent deviations. As shown in the tables, the percent deviations are small, so that the overall error is expected to be well below 1%. The detection point networks discussed are applied to generic calculations. The same approach has been used to calculate energy absorption for larger containers.



Analytical solution for sphere (with build-up factor)

$$E_{abs}^{center} = \mu_{a}I_{0} \left\{ \sum_{k=0}^{3} \frac{1}{\mu_{e}^{k+1}} \left[B^{(k)}(0) - \exp(-\mu_{e}R_{0})B^{(k)}(\mu_{e}R_{0}) \right] \right\} , (MeV/cm^{3}/s)$$

 $B^{(k)}(\mu_e r)$: k-th derivative of $B(\mu_e r)$

$$B(\mu_{e}r) = \sum_{i=0}^{3} B_{i}(\mu_{e}r)^{i}$$

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Figure C-2. Source points arrangement and energy absorption equations for spherical medium.

			Percent De	eviations from	n Analytical	Solutions	
Number of	Width of	Gamm	a Energy 0.4	(MeV)	Gamm	a Energy 1.5	(MeV)
Subdivisions (m)	Subdivisions (cm)	$\rho = 0.6^{a}$	$\rho = 1.0$	$\rho = 2.0$	$\rho = 0.6$	$\rho = 1.0$	$\rho = 2.0$
1	10	0.28	1.02	3.81	0.14	0.32	0.63
2	5	0.066	0.22	0.81	0.037	0.08	0.18
4	2.5	0.015	0.05	0.19	0.0061	0.019	0.046
5	2	0.010	0.034	0.12	0	0.011	0.032
8	1.25	0.005	0.012	0.045	0	0.004	0.013
10	1.0	0.005	0.009	0.028	0	0.004	0.008
15	0.67	0	0.003	0.012	0	0	0.004
20	0.5	0	0.003	0.007	0	0	0.002
Analytical Solution	E ^{Centerb} abs	0.198	0.326	0.597	0.163	0.262	0.478

Table C-1. Comparison of analytical and numerical method—radius of sphere = $0 \sim 10$ cm

a. Density g/cm³.

b. Defined in Figure 2.

Container Geometry. The energy absorption characteristics of several standard waste containers have been investigated. To define the standard configurations for generic calculation, the relations between internal radius and effective height are plotted in Figure C-7, the data shown as squares were obtained from an EPRI waste study^{C-2}.

All modeled containers except 55 gal drums are assumed to be right circular cylinders. Naturally, actual containers show some deviations from right circular cylinders so the magnitude of deviations are plotted against the internal radius in Figure C-8. The deviations from right circular cylinders in larger containers can vary by 50%. A discussion of these variances follows in a later section.

Radiation Absorption in Generic Waste. Calculational conditions for gamma radiation absorption in waste assumed as generic are summarized in Table C-7. The source energy range was determined by evaluating waste shipment information and 10 CFR 61 requirements. Media characteristics are presented in Appendix B.

The numbers of source and detection points taken are summarized below; detail arrangements for a 55 gallon drum are shown in Figures C-9 and C-10.

Radius/ Height (cm)	Source/ Detection Points	Note
30/90	67,500/48	55 gal drum
60/120	270,000/81	4 x 4 liner
70/140	350,000/100	5 x 5 liner
90/180	810,000/121	6 x 6 liner

The results are listed in Table C-8, and are expressed by absorption percentage with three energy levels (0.4 to 2.0 MeV), five different densities (0.6 to)

		<u></u>	Percent De	eviations from	n Analytical	Solutions	
Number of	Width of	Gamm	a Energy 0.4	(MeV)	Gamma Energy 1.5 (MeV)		
(m)	(cm)	$\rho = 0.6^{a}$	$\rho = 1.0$	$\rho = 2.0$	$\rho = 0.6$	$\rho = 1.0$	$\rho = 2.0$
1	10	0.38	0.55	3.80	0.068	0.014	1.10
2	5	0.090	0.14	0.64	0.021	0.004	0.30
4	2.5	0.021	0.033	0.15	0.007	0.004	0.10
5	2	0.011	0.022	0.090	0.007	0.004	0.043
8	1.25	0.005	0.007	0.037	0.007	0.004	0.019
10	1.0	0	0.004	0.024	0.007	0.004	0.012
15	0.67	0	0	0.010	0.007	0.004	0.007
20	0.5	0	0	0.007	_0	0.004	0.005
Analytical Solution	E ^{Centerb} /I ₀	0.189	0.274	0.292	0.146	0.212	0.280

Table C-2. Comparison of analytical and numerical method—radius of sphere = $10 \sim 20$ cm

a. Density g/cm³.

b. Defined in Figure 2.

 2.0 g/cm^3), and four kinds of containers. The absorption percentage is defined as the absorbed percentage of total emitted energy and is expressed by combining

Source intensity I_0 , (MeV/s \cdot cm³), total absorbed energy, E_{abs}^{Total} (MeV/s), and waste volume V_s (cm³), as

Absorption Percentage =
$$\frac{E_{abs}^{Total}}{I_0 V_s} \times 100$$

The actual absorbed energy can be calculated by multiplying $I_0 \times V_s$ by the Absorption Percentage. For example, the absorbed energy for 1.0 MeV gamma ray of 1μ Ci/cm³ in a 4 x 4 liner with 1.0 g/cm³ density is calculated as

$$E_{abs}^{Total} = \frac{70.3911}{100 \text{ percent}} \times 1.0[\text{MeV/d}]$$

x 3.7 x 10⁴ d/s/cm³ x V_s

where

$$V_s = \pi \tau^2 h = \pi x \ 60^2 \ x \ 120 \ [cm^3]$$

and

$$1\mu \text{Ci/cm}^3 = 3.7 \times 10^4 \frac{\text{decays}}{\text{sec cm}^3}$$
.

Then

$$E_{abs}^{Total} = 3.53 \times 10^{10} (MeV/s)^{-10}$$

or

 $E_{abs}^{Total} = 5.66 \times 10^4 (erg/s)$.

The average dose rate is obtained by using the conversion factor for rad, gram and erg, then

Average Dose Rate = $5.66 \times 10^4 (erg/s)$

x 3600 (s/h) x



Z₀: Height 84 cm, 70 subdivisions Z₁: Height 56 cm (including detection plane), $\Delta z = 1$ cm Z₂: Height 28 cm (out of detection plane, $\Delta z = 2$ cm R₀: Radium 27 cm, $\Delta r = 1$ cm, 27 subdivisions ϕ_0 : Radial π, $\Delta \phi = \pi/30$, 30 subdivisions

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Figure C-3. Source subdivisions for U.S. Department of Transportation 55 gai. Drum.

 $x \frac{1}{100}$ (rad g/erg)/1.36 x 10⁴(g)

= 1.50 (rad/h).

To apply these results to actual nuclide and waste media, data has been interpolated by the least square fitting function as

$$f(xy) = (a_1 xy) + (a_2 x\sqrt{y}) + (a_3\sqrt{xy}) + (a_4 x) + (a_5\sqrt{xy}) + (a_6 y) + (a_7\sqrt{x}) + (a_8\sqrt{y}) + (a_9)$$

where

f(xy) = absorbed portion of gamma radiation energy

x = density of waste media, g/cm^3

y = gamma radiation energy, MeV

Constants a_1 to a_9 for different containers are summarized in Table C-9. By using this function, the absorbed portion of gamma radiation energy can be calculated for various waste media and nuclides.

As an example, the absorbed energy for Co-60 is estimated below,



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Figure C-4. Detection points network for U.S. Department of Transportation 55 gal. drum.

Table C-3	. Setting o	f valuai	oles for	[,] sample	calculation
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Valuables	Characteristics	Note
Waste material	Dewatered resin	1:1 mixture of cation and anion
Density	0.78 g/cm ³	Catalog value
Container	DOT 55 gal drum dia. 54 cm height 90 cm	
Gamma ray energy Attenuation coefficient	0.4 MeV and 1.5 MeV Water ^a	
Absorption coefficient	Water ^a	
Build-up factor	Dose buildup factor ^b	

a. The coefficients used here are shown in Appendix B.

b. The same equation used in EPRI study was taken only in this calculation.

	Numb	ers of Subdivis	sions	Average Dose (rad/h)	Deviations from Standard Case	
Case Number Z	Z	R	<u>Y</u>		(%)	
1	56	18	20	9.3834	-1. <u>9</u>	
2 ^a	70	27	30	9.5073	_	
3	84	27	30	9.5070	-0.003	
4	112	27	30	9.5112	+ 0.04	
5	70	54	30	9.5054	-0.02	
6	70	27	60	9.6204	+ 1.19	
7	70	27	90	9.6370	+ 1.36	

Table C-4. Absorbed energy with various source point arrangements (dewatered resin pack in 55 gallon drum)

a. Standard case

,



Figure C-5. Internal dose profile of resin, using 1.5 MeV gamma ray.





Figure C-6. Error estimation for representative detection points.

- Media: Dewatered Resin (0.865 g/cm³)
- Energy: 1.17 MeV 100 percent abundance 1.33 MeV - 100 percent abundance
- Container: 55 gal drum.

Taking the fitting constants for a 55 gal drum, X and Y are set as 0.865 and 1.17 or 1.33. Then,

$$f(x = 0.865, y = 1.17) = 0.493$$

f(x = 0.865, y = 1.33) = 0.482.

Considering the abundance of gamma ray, then the f(xy) for ${}^{60}Co$ is,

$$f(Co-60) = \frac{1.17 \times 0.493 \times 1.0 - 1.33 \times 0.482 \times 1.0}{1.17 \times 1.0 + 1.33 \times 1.0}$$

= 0.488 .

In this case, 48.8 percent of 60 Co gamma energy is absorbed in waste media.

Consideration for Extreme Cases. As described in previous sections, generic waste forms are assumed to have

- Homogeneous media and source distributions
- Right circular cylindrical configuration for larger size containers.

Detector Coordinates		ates	1.5 M	1eV	0.4 MeV		
Number		Z	Dose (rad/h)	Deviation (%)	Dose (rad/h)	Deviation (%)	
la	0.00	83.00	7.736 E+0	0.4285	2.3942 E + 0	0.9559	
2	0.00	80.00	8.7621 E + 0	0.4915	2.9034 E + 0	0.4830	
3	0.00	76.00	9.9476 E + 0	0.1926	3.3149 E + 0	0.1956	
4	0.00	71.00	1.0845 E + 1	0.1950	3.6334 E + 0	0.2072	
5	0.00	66.00	1.1417 E + 1	0.0595	3.8301 E+0	0.0644	
6	0.00	58.00	1.1926 E + L	0.1041	4.0041 E+0	0.1095	
7	0.00	50.00	1.2163 E + 1	0.0628	4.0821 E+0	0.0617	
8	0.00	42.00	1.2233 E+1	0.0506	4.1046 E + 0	0.0458	
gb	26.00	83.00	4.0803 E+0	0.6486	1.3047 E+0	0.6492	
10	26.00	80.00	4.9329 E + 0	0.5243	1.5896 E + 0	0.5287	
11	26.00	76.00	5.4742 E + 0	0.3063	1.8087 E + 0	0.1905	
12	26.00	71.00	6.0681 E + 0	0.2568	1.9782 E + 0	0.1938	
13	26.00	66.00	6.3843 E + 0	0.0560	2.0855 E+0	0.0615	
14	26.00	58.00	6.6843 E + 1	0.1076	2.1841 E+0	0.1091	
15	26.00	50.00	6.8309 E + 0	0.0688	2.2303 E+0	0.0791	
16	26.00	42.00	6.8747 E + 0	0.0578	2.2440 E + 0	0.0531	

Table C-5. Estimate error of representing detection points (in height)

a. Center

b. Edge

Deviations from standard waste configurations are examined in this section. Figure C-11 lists devi ations from standard configurations. Height changes from a right circular cylinder and nonhomogeneous source distributions are characterized.

• Changes in Height

The height of waste media may be varied by container shape and filling level of waste. Generally, the absorption of gamma radiation energy increases with height. These increments (or decrements) are not significant.

For a 6 x 6 liner, the energy absorption deviation ranges from 1 to 3% with a 50%increase in height^a from right circular cylinder, and 2 to 8% decrease with 50% decrease in height^b. As shown in Figure C-8, height

b. Waste Configuration (90-cm height x 90-cm radius).

deviations from a right circular cylinder are bounded within 20 percent for most containers, these changes will not cause significant effect on energy absorption calculations.

Source Distribution Changes

The generic calculations assume a homogeneous source distribution over the entire waste media. There can be two types of source distributions other than homogeneous distribution.

a. One is source accumulated at the top of the waste media. It might be seen in spent cartridge type ion exchangers or filters, and is illustrated in Figure C-11 as deviation 3. When the source is accumulated in the top 50%, the energy absorption will be the same as to homogenous distribution, because the energy escaping

a. Waste Configuration (270-cm height x 90-cm radius).

Detector Coordinates		1.5 M	1.5 MeV		0.4 MeV		
Number	(cm)	Z	Dose (rad/h)	Deviation	Dose (rad/h)	Deviation (%)	
l	0.00	83.00	7.1736 E + 0	0.3639	2.3942 E + 0	1.2703	
2	7.00	83.00	7.0032 E + 0	0.2906	2.3085 E + 0	0.3057	
3	14.00	83.00	6.5013 E + 0	0.2371	2.1324 E + 0	0.2500	
4	19.00	83.00	5.8511 E + 0	0.1947	1.9594 E + 0	0.2008	
5	23.00	83.00	5.0538 E + 0	0.5179	1.6316 E + 0	0.5192	
6	26.00	83.00	4.0803 E + 0	0.5836	1.3047 E + 0	0.5726	
7	0.00	42.00	1.2233 E + 1	0.8672	4.1046 E + 0	0.8526	
8	7.00	42.00	1.1859 E + 1	0.2873	3.9745 E + 0	0.3016	
9	14.00	42.00	1.0991 E + 0	0.2443	3.6662 E + 0	0.2527	
10	19.00	42.00	9.8669 E + 0	0.2019	3.2696 E + 0	0.2527	
11	23.00	42.00	8.4488 E + 0	0.4853	2.7906 E + 0	0.4779	
12	26.00	42.00	6.8757 E + 0	0.4279	2.2440 E + 0	0.4123	

Table C-6. Estimated error of representing detection points (in radius)

Numbers 1-6: Edge

Numbers 7-12: Center





Figure C-8. Deviations from right cylinder for various waste containers.

from the top region to above is compensated by absorption in lower media. Practically, if the source region is limited to less than 50% in height at the top, the results of the generic calculation are applicable.

b. The second type of distribution considers source accumulation at the central region of the containers. This occurs when radioactive materials such as cartridge filters are placed in the center of a container and the void is filled with binders to immobilize the waste. This is shown in Figure C-11 as deviation 4. When the source has a cylindrical shape, the absorbed energy inside the source can be calculated by using the generic method.

Table C-7. Summary of calculational conditions for generic waste

Variables	Range Considered
1. Source	
Energy Level Distribution	0.4 MeV to 2.0 MeV Homogeneous
2. Media	
Density Material	0.6 to 2.0 g/cm ³ Water ^a
3. Containers	
Configuration Size	Circular cylinder 55 gal drum ^b 4 ft x 4 ft liner ^c 5 ft x 5 ft liner ^d 6 ft x 6 ft liner ^e

b. 30-cm radius x 90-cm height

c. 60-cm radius x 120-cm height

d. 70-cm radius x 140-cm height

e. 90-cm radius x 180-cm height



Figure C-9. Source points network for 55 gal. drum.

C-17



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Figure C-10. Detection point network for 55 gal. drum,

Table C-8.	Gamma	energy	absorption	percentage
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			Energy ^C		
Container ^a	Densityb	0.4	1.0	2.0	Note
30/90	6.0000E-01	45.6894	40.0910	34.3298	55 gal drum
	1.0000E + 00	60.9414	54 6901	43.0109	
	1.5000E + 00	70.6475	65.0612	49.2058	
	2.0000E + 00	75.5871	70.8238	65.9195	
60/120	6.0000E-01	63.9283	57.5887	51.1064	4 x 4 liner
	8.0000E-01	71.0530	65.3157	59.1550	
	1.0000E + 00	75.5488	70.3911	64.8325	
	1.5000E + 00	81.4383	77.4160	73.3375	
	2.0000E + 00	83.9 6 40	80.6775	77.6518	
70/140	6.0000E-01	68.1213	62.0894	55.6456	5 x 5 liner
	8.0000E-01	74.5177	69.1646	63.3853	
	1.0000E + 00	78.4773	73.7308	63.6713	
	1.5000E + 00	83.9372	79.8475	67.2470	
	2.0000E + 00	85.5449	82.2778	79.9512	
90/180	6.0000E-01	74.1570	68.6829	62.7335	6 x 6 liner
	8.0000E-01	79.3225	74.6983	69.6728	
	1.0000E + 00	82.5386	78.4175	74.1956	
	1.5000E + 00	86.3314	83.2331	80.2763	
	2.0000E + 00	87.6969	85.1385	83.1447	

a. Radius/Height, cm.

b. gram/cm³.

c. Million electron volt.

	55 godi Drum	Container Radius/Height (cm)				
Constant	30/90	60/120	70/140	90/180		
a ₁	0.388	-0.0430	-0.0977	-0.321		
a.	-0.579	0.106	0.197	0.591		
a,	-0.871	0.117	0.276	0.748		
a1	-0.245	-0.518	-0.543	-0.634		
a,	1.354	-0.134	-0.423	-1.252		
a ₆	0.485	0.0547	-0.158	-0.402		
a ₇	0.954	1.371	1.428	1.549		
ag	-0.937	-0.137	0.0557	0.502		
a _g	3.302 x 10 ⁻⁶	-9.973 x 10 ⁻⁷	-1.554 x 10 ⁻⁶	-2.947 x 10 ⁻⁶		





Figure C-11. Possible deviations from generic configuration.

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- C-1. C. W. Neilson and E. E. Hochhalter, QAD-FN, A General Purpose Shielding Design Code with Simplified Input, Exxon Nuclear Idaho Company, Inc., March 1981.
- C-2. H. R. Helmholz and J. K. Ryo, Determination of Waste Container Curie Content from Dose Rate Measurements, EPRI NP-3223, July 1983.

APPENDIX D

DEVIATION OF ANALYTICAL SOLUTION FOR ENERGY ABSORPTION AT THE CENTER OF SPHERICAL SOURCE

Building JN-1 Hot Cell Laboratory

Acceptable Knowledge Document

January 2000

TCP-98-03

UNCONTROLLED COPY

Prepared for:

Battelle Columbus Laboratories Decommissioning Project (BCLDP) Columbus, Ohio

Prepared by:

WASTREN, Inc. 1333 W. 120th Avenue, Suite 300 Westminster, Colorado 80234

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Putting Technology To Work

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PLAN APPROVAL PAGE

Prepared By: for Kevin Peters (Signed Copy on file) K. Peters Wastren, Inc.

31 3/2000 Date

This plan, *TCP-98-03, Building JN-1 Hot Cell Laboratory Acceptable Knowledge Document*, has been reviewed and approved by the following.

Approved By:

J. Eide, Site Project Manager TRU Waste Certification Program

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TCP-98-03 (psm)

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EXECUTIVE SUMMARY

In April 1943, Battelle Memorial Institute (BMI), a private research and development corporation, entered into a contract with the Manhattan Engineering District to perform atomic energy research and development activities. Since that time, a majority of the work performed has been in support of research programs for the U.S. Department of Energy (DOE) and its predecessors. BMI supported research for the nuclear power industry and performed projects for other government agencies including the U.S. Navy, U.S. Army, and U.S. Air Force. As a result of BMI's participation in this atomic energy research, portions of fifteen buildings became radioactively contaminated. These buildings and grounds are to be decontaminated and decommissioned under the Battelle Columbus Laboratories Decommissioning Project (BCLDP). The BCLDP is being overseen by the DOE-Columbus Environmental Management Project (DOE-CEMP).

Three of the affected facilities are located at the Battelle Columbus Laboratories West Jefferson North site, including the Hot Cell Laboratory (Building JN-1). This facility was constructed in 1955 primarily to support the emerging nuclear power industry. The primary mission of the facility was the examination of nuclear reactor fuels and reactor structural materials for the DOE. During the history of the Hot Cell Laboratory, ongoing defense research supporting the development of nuclear reactors was conducted primarily for the U.S. Navy. Defense-related work was also conducted for the U.S. Air Force and U.S. Army.

The purpose of this document and supporting documentation is to present the TRU waste management acceptable knowledge available for suspect transuranic (TRU) waste generated from the Hot Cell Laboratory. Acceptable knowledge refers to information that can be used for waste characterization in lieu of or in conjunction with waste sampling and analysis. The primary objective is to provide a consistent, defensible, and auditable record of information to be used during the certification of waste destined for disposal at the Waste Isolation Pilot Plant (WIPP). The materials characterized in this document consist of suspect TRU waste that may be generated based on existing documentation. These wastes will be processed and packaged for disposal at WIPP and do not represent the final waste stream.

Several hundred sources of information were reviewed during the development of this document. Relevant documentation sources were incorporated into a data management system. These references are identified in Appendix A and are grouped into four categories; published documents, unpublished data, correspondence, and discrepancy reports. These categories correspond to the "P," "U," "C," and "D" superscripts used throughout this document (e.g., P001). These superscript identifiers can be used to link the information in this document to the source. In addition, an Acceptable Knowledge Source Document Excerpts binder has been prepared in conjunction with this report. This binder contains the individual pages from the references that identify the specific information used while drafting this document.

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LIST OF ACRONYMS & ABBREVIATIONS

AK	Acceptable Knowledge
APPR	Army Package Power Reactor
BCL	Battelle Columbus Laboratories
BCLDP	Battelle Columbus Laboratories Decommissioning Project
BMI	Battelle Memorial Institute
°C	degrees centigrade
CAO	Carlsbad Area Office
CEMP	Columbus Environmental Management Project
CFR	Code of Federal Regulations
DOE	Department of Energy
EPA	Environmental Protection Agency
HEPA	High Efficiency Particulate Air
INEEL	Idaho National Engineering and Environmental Laboratory
JN-1	West Jefferson North Hot Cell Laboratory (Building JN-1)
JN-2	West Jefferson North Critical Assembly Laboratory (Building JN-2)
JN-3	West Jefferson North Research Reactor Building (Building JN-3)
JN-4	West Jefferson North Plutonium Laboratory (Building JN-4)
Mev	mega electron volts
NRC	Nuclear Regulatory Commission
PCBs	polychlorinated biphenyls
QAPD	Quality Assurance Program Document
QAPjP	Quality Assurance Project Plan
QAPP	Quality Assurance Program Plan
RAL	Radioanalytical Laboratory
RCRA	Resource Conservation and Recovery Act
TBD	to be determined
TMI	Three Mile Island
TRU	transuranic
TSCA	Toxic Substances Control Act
WAC	Waste Acceptance Criteria
WIPP	Waste Isolation Pilot Plant
WIPP-WAPP	Waste Analysis Plan for the Waste Isolation Pilot Plant, Attachment B of the
	WIPP Hazardous Waste Permit

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1.0 INTRODUCTION

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Transuranic (TRU) waste generated by the Battelle Columbus Laboratories Decommissioning Project (BCLDP) will be characterized to address the objectives of the U.S. Department of Energy (DOE) Waste Isolation Pilot Plant (WIPP) TRU Waste Characterization Program. Program requirements for TRU waste characterization are provided in the WIPP Hazardous Waste Permit, Attachment B, Waste Analysis Plan (WIPP-WAPP). Implementation of these requirements is described in the Transuranic Waste Characterization Quality Assurance Project Plan for the BCLDP TRU Waste Certification Program (QAPjP). The WIPP-WAP and QAPjP require that a consistent, defensible, and auditable record of characterization information is collected, reviewed, and managed.

The WIPP-WAP requires sites certifying TRU waste for disposal at WIPP to compile and maintain an acceptable knowledge (AK) record relating site history, mission, and waste management. In addition, waste stream specific data describing the physical and chemical composition of the waste and the originating process is required for certification. The method used to collect, review, and manage the AK documentation required by the WIPP-WAP and the BCLDP QAPjP is performed in accordance with procedure TC-AP-03.1, Collection, Review, and Management of Acceptable Knowledge Documentation. Implementation of this procedure generates three primary sources of published AK documentation:

TRU Waste Management Documentation – Summarizes the required TRU waste management information required by the WIPP-WAP, including the characterization of waste materials that represent the inputs to the BCLDP TRU waste packaging processes.

TRU Waste Process Descriptions – Summarizes the TRU waste stream information required by the WIPP-WAP for each process generating TRU waste streams. These reports are prepared prior to TRU waste processing to describe the waste inventory to be processed, process inputs, and TRU waste streams that will be certified for disposal at WIPP by the BCLDP program.

TRU Waste Confirmation Reports – Verifies AK by identifying any process deviations, resolving discrepancies, and ensuring that the Project Data Quality Objectives (DQOs) have been achieved based on the inspection and confirmation analyses of each TRU waste container.

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The primary purpose of this document is to present the required TRU waste management AK associated with the potentially TRU waste historically generated in the Hot Cell Laboratory (Building JN-1) at the West Jefferson North site, including:

Maps of the West Jefferson North site and Building JN-1

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- Summaries of the history and mission for the site and laboratory
- Descriptions of the historical laboratory operations and waste management practices
- Type and quantity of potentially TRU waste contained in JN-1

During the AK review process, potentially TRU waste materials contained in Building JN were identified and grouped into categories based primarily on the matrix of the waste (Sections 4 through 12). The waste was then characterized as it exists in the building, knowing that many of the materials (hydraulic oil, liquids, etc.) will require further processing to be eligible for disposal at WIPP. In addition, non-waste materials (i.e., light bulbs, leaded glass, and lead shielding) were characterized in advance of remediation. These waste materials represent the inputs to the BCLDP operations that will process TRU waste streams. This materials assessment and the supporting AK source documentation will be used as the basis for subsequent TRU waste stream segregation and characterization activities. This report also provides the basis for the following:

- Defense waste justification
- Spent nuclear fuel and high level waste discussions
- JN-1 Hot Cell Laboratory standard facility radionuclide distribution

The primary sources of AK were collected from historical files maintained in file cabinets in Building JN-2 at the West Jefferson site. Other sources include personal files, the Columbus Environmental Management Project (CEMP), the King Ave Records Management Office records, the DOE Chicago Operations Office Safeguards and Security Division classified files, and personal interviews.

As defined in TC-AP-03.1, the AK record consists of information from published documents, unpublished data, correspondence, and discrepancy reports which correspond to the "P", "U", "C", and "D" subscripts used throughout this document (e.g., P001).

Published Documentation – In general, published documents represent the most reliable, reviewed, and controlled sources of information. Published documentation includes, but is not limited to, controlled documents and databases, previously controlled documents and databases, procedures, reports, and studies. Published sources are referenced by the "P" superscripts in this document.

Unpublished Data – Unpublished data consists of information that has typically not received peer review and may not have been formally controlled. In many cases, this will consist of the raw data used during the development of published documentation. This AK will primarily be used to supplement and verify published information. Unpublished data includes, but is not limited to, draft documents, forms, analytical data packages, log books,

notes, and inventory lists, in addition to internal reports, studies, and databases. Unpublished sources are referenced by the "U" superscripts in this document.

Correspondence – Correspondence consists of communication records relating to specific TRU waste streams or TRU waste management. Typically, this information consists of uncontrolled records of internal and external communications. Correspondence will primarily be used to supplement and clarify the acceptable knowledge. Correspondence includes, but is not limited to, internal and external letters, memos, directives, telecommunication records, meeting minutes, and personal interview summaries. Correspondence sources are referenced by the "C" superscripts in this document.

Discrepancy Reports – This report is the documented resolution of discrepancies identified between AK sources or between AK sources and confirmation data. In general, this will consist of a letter to the AK record that will be referenced in the affected document location. Discrepancy reports are referenced by the "D" superscripts in this document.

Several hundred sources were reviewed during the development of this document. Approximately 75 sources were determined to be relevant to the scope of the AK and were incorporated into a data management system. When possible, discrepancies between sources were resolved by contacting cognizant personnel or collecting additional information. If the inconsistency was not resolved, the most conservative information was incorporated into the document. Appendix A contains the Acceptable Knowledge References listing the data sources used in compiling the AK document.

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2.0 REQUIREMENTS

This section lists the requirements, guidance documents, and procedures that dictate or direct waste characterization and certification activities associated with this the BCLDP AK program.

• TRU Waste Acceptance Criteria for the Waste Isolation Pilot Plant (WIPP WAC), WIPP/DOE-069

The WIPP WAC defines the requirements for safe handling, transportation, and disposal of TRU waste at the WIPP. The WIPP WAC requires the preparation of certification plans and associated quality assurance plans describing site-specific TRU waste certification programs for submittal to the DOE Carlsbad Area Office (CAO) for review and approval before a site can be granted authority to ship waste to WIPP.

• 40 CFR Parts 260 through 265, 268, and 270 (Implemented by Ohio Administrative Code, Rule 3745)

Title 40 of the Code of Federal Regulations (CFR) describes the federal hazardous waste regulations codified to protect the environment, including the Resource Conservation and Recovery Act (RCRA). The regulations define the requirements for making a hazardous waste determination, obtaining hazardous waste permits, and controlling the land disposal of hazardous wastes. Also included are standards for waste generators, transporters, and treatment, storage, and disposal facilities.

• Waste Isolation Pilot Plant Hazardous Waste Permit, Waste Analysis Plan (WIPP-WAP), EPA No. NM4890139088

The WIPP-WAP provides a detailed description of the quality assurance and quality control requirements for the TRU waste characterization program. The plan establishes the characterization parameters that must be addressed during the characterization of TRU waste. For each parameter, the WIPP-WAP provides requirements for waste characterization, including the use of AK, visual examination, and sampling and analysis.

• Quality Assurance Program Document (QAPD), CAO-94-1012

The QAPD identifies federal, state, and industry quality requirements applicable to the CAO quality assurance program. The QAPD also establishes the minimum requirements for CAO personnel and guidance for the development and implementation of quality assurance programs by all program participants.

• Waste Analysis at Facilities that Generate, Treat, Store, and Dispose of Hazardous Wastes; A Guidance Manual, EPA-530-R-94-024

This U.S. Environmental Protection Agency (EPA) manual provides general waste analysis guidance for facilities that generate or manage hazardous wastes. Specifically, the document promotes the use of AK characterization under certain circumstances as an alternative to or in conjunction with sampling and laboratory analysis.

• Transuranic Waste Characterization Quality Assurance Project Plan (QAPjP)

The Transuranic Waste Characterization QAPjP for the Battelle Columbus Laboratories Decommissioning Project (BCLDP) Transuranic Waste Certification Program (TCP-98-02) specifies the quality of data necessary and the characterization techniques including the use of acceptable knowledge employed by BCLDP to meet the requirements of the WIPP-WAP for TRU wastes generated at the BCL West Jefferson North facility.

• Collection, Review, And Management Of Acceptable Knowledge Documentation, TC-AP-03.1

This procedure outlines the method for collecting, reviewing, compiling, managing, and confirming AK documentation. This procedure was implemented during the preparation of this document to ensure that a consistent, defensible, and auditable record is created. This acceptable knowledge document (TCP-98-03) is the result of the implementation of TC-AP-03.1.

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3.0 WASTE GENERATING OPERATIONS

On April 16, 1943, Battelle Memorial Institute (BMI) entered into a contract with the Manhattan Engineering District to support atomic energy research and development programs. BMI was selected as one of the original participants in this program because of their metallurgical expertise.^{P026,P027} Nuclear reactor development activities, especially materials development, constituted the major portion of BMI's participation in the atomic energy program.^{P026}

During World War II, BMI played an important role in development of reactors used for plutonium production. Studies involving the extrusion and degassing of uranium were among the company's most significant contributions. During the war, the United States was largely dependent on foreign sources of uranium. BMI was responsible for the development of methods used for recovering uranium and thorium from domestic supplies of minerals containing relatively small amounts of source materials.^{P026}

The first large-scale application of nuclear power was nuclear submarine propulsion reactors. BMI was involved with this effort from its inception. When the program began, no material was available that provided adequate corrosion resistance to hot water with a low absorption cross section. BMI invented and developed Zircaloy, a zirconium corrosion-resistant alloy used in fuel elements and assemblies.^{P023} Other major accomplishments included the design and fabrication of the original reference fuel for the Nautilus program and the development of the Hot Isostatic Pressure and Picture Frame bonding technologies used to fabricate nuclear submarine cores.^{P026} Additionally, BMI pioneered the design, development, and testing of radioactive material shipping containers.^{P030}

As a result of BMI's participation in this atomic energy research for the DOE and its predecessor agencies, portions of fifteen BCL buildings and associated grounds became contaminated with varying amounts of radioactive materials.^{P038} These buildings and grounds are to be decontaminated and decommissioned under the Battelle Columbus Laboratories Decommissioning Project (BCLDP). The BCLDP is being overseen by the DOE-Columbus Environmental Management Project (DOE-CEMP).

Nine of the buildings are located at the BCL King Avenue facility in the west central portion of the city of Columbus, Ohio. The nuclear research performed in these buildings included processing and machining of enriched, natural, and depleted uranium, thorium, fuel fabrication, radiotracer studies, radiochemical analyses, and powder metallurgy studies.^{P034} Remediation of known areas of contamination in these buildings was completed in 1998.

The remaining six buildings are located at the BCL West Jefferson site located approximately 15 miles west of the King Avenue facility near the city of West Jefferson, Ohio. Research was performed at two areas at the site. Three of the buildings located at the southeast portion of the site were used for fuel element fabrication and ballistic studies. This engineering area is known as West Jefferson South; cleanup of minor contamination at this site was

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completed in 1990. The other three buildings are located in the northern portion of the site and have been known as the Nuclear Sciences Area, the Nuclear Research Center, and is currently known as West Jefferson North.^{P034}

3.1 West Jefferson North History and Mission

The Nuclear Research Center located near West Jefferson, Ohio, was constructed by BMI in 1954 and 1955 to support the on-going nuclear research programs. Currently known as the West Jefferson North facility, experiments were originally performed at the site in three main buildings: JN-1 (Hot Cell Laboratory), JN-2 (Critical Assembly Laboratory), and JN-3 (Research Reactor Building).^{P034} Building JN-4 (Plutonium Laboratory) was built in 1960 to support plutonium research and processing programs.^{P033} These buildings are identified in the aerial photograph of the West Jefferson North site provided in Figure 3-1.



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The oldest and most contaminated building at the West Jefferson North facility is the Hot Cell Laboratory (Building JN-1). Building JN-1 operations began in 1955 to support nuclear research projects such as the examination of power and research reactor fuels; post irradiation examination of fissile, control rod, source, and structural reactor materials; and examination of irradiation surveillance capsules. Building JN-1 operations are described in more detail in Section 3.2.^{P034}

Building JN-2, the Critical Assembly Laboratory, was constructed during 1954 and 1955, and began operations in 1955. In addition to critical assembly studies, the facility conducted energy conservation experiments, nuclear materials handling, and plutonium research. Active nuclear experimentation was terminated in 1970.^{P034} The Critical Assembly Laboratory was decommissioned in 1975. The decommissioned plutonium laboratory area was refurbished and currently contains the Radioanalytical Laboratory (RAL).^{U004}

The RAL is designed to conduct routine radioanalyses in support of the BCLDP, including analysis of samples for radiological constituents for the Site Environmental Report. Specifically, the RAL performs gamma spectrometry, alpha spectroscopy, gross alpha and beta counting, and various isotopic analyses on water, air, soil, sediment, sludge, smear, and waste matrices. In addition, the RAL provides technical support for analytical sampling and is responsible for obtaining, monitoring, and controlling off-site analysis of samples funded by the BCLDP.^{U015}

Building JN-3, the Research Reactor Building, was constructed during 1955 and 1956 to house the Battelle research reactor which operated from 1956 to 1974. The reactor was defueled and partially decommissioned in 1975, and Battelle's NRC R-4 license was changed to "possession only" status.^{C004} The dismantled reactor and wastes generated during decommissioning operations were shipped to DOE's Idaho National Engineering and Environmental Laboratory (INEEL). The building is currently used to store suspect TRU waste generated by decontamination and decommissioning operations in Buildings JN-1 and JN-4. Building JN-3 also houses a respirator washing room and a waste management sorting, packaging, and radiological free release area.^{P027,U004}

The Plutonium Laboratory (Building JN-4) was built in 1960 to support plutonium research and processing programs. Renovations were made to the facility in 1964 and 1967. These enlargements expanded the capabilities of the laboratory, including improved machining, mechanical testing, plutonium processing, and analytical capabilities.^{P033} With the termination of the Advanced Fuel Program in March 1977, the decision was made to decontaminate and decommission the building for unrestricted use. Decontamination of the building began in January 1978 and continued through 1982. Cleanup of contaminated soil in areas around the building is pending.^{P033}

Most of the transuranic waste generated during decontamination of Building JN-4 was shipped to either the DOE Hanford or INEEL sites for storage.^{P033} Three gloveboxes and a small

inventory of waste (291 cubic ft.) from the decontamination of Building JN-4 are stored in Building JN-3.^{D001} These wastes are not in the scope of this document.

3.2 Hot Cell Laboratory (Building JN-1)

Due to the nature of the research being conducted by BMI in the early 1950s, it was necessary to construct a remote handling facility to conduct experiments to support the development of nuclear fuels, control rods, reactor structural materials, and various other reactor components and instrumentation.^{P026} The Hot Cell Laboratory (Building JN-1) was constructed in 1955 in anticipation of the emerging commercial nuclear power industry.^{C004} Figure 3-2 provides a floor plan of the Hot Cell Laboratory that identifies the areas described in this section at the time operations ceased in the building.^{U001,U018}



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Figure 3-2. Building JN-1 Hot Cell Laboratory Floor Plan.

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3.2.1 History and Mission

Since operations began in the Hot Cell Laboratory in 1955, a variety of studies relating to the radiation performance of materials were conducted in the remote handling facilities in Building JN-1. Experiments in the Hot Cell Laboratory were largely dedicated to research supporting the DOE (including predecessor agencies) and other government agencies.^{P026} The research consisted primarily of reactor fuel studies that evaluated materials such as uranium, thorium, and plutonium alloys and compounds in pellet, dispersion, and ceramic form. Control rod material studies included rare-earth absorbers such as europium titanate dispersions in stainless steel. Structural and cladding material studies evaluated stainless steels, zirconium, Zircaloy, nickel alloys, refractory metals, and pressure vessel steels.^{U003} Table 3-1 provides a brief time line of major events in the history of Building JN-1. Table 3-2 summarizes the laboratory capabilities and the areas in Building JN-1 where these operations were conducted.

Year	Event
1954 – 1955	Hot Cell Laboratory (Building JN-1) constructed (also known as JN-1A) ^{U004}
1955	Hot Cell operations begin ^{U003,U004}
1955 - 1960	Small scale experiments on irradiated capsules ^{U003}
1964	Alpha-Gamma Cells added in the basement of Building JN-1 ^{U003,P030}
1967	Mechanical Test Cell added ^{U003,P030}
1971 – 1972	High Energy Cell and Storage/Transfer Pool added to accept full-size fuel assemblies (JN-1B) ^{D001}
1987	Operations officially end with the final off-site fuel shipment ^{C004,U018}
1989	BCLDP begins remedial decontamination and decommissioning activities to be completed by year 2003 ^{C008}

Т	able	3-1.	Building	JN-1	History.
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 Table 3-2.
 Building JN-1 Hot Cell Capabilities.^{C006}

Capability	Area(s) of Operation	
Nondestructive Examinations of Full-Size Fuel Roo	ds	
Underwater TV viewing and video tape recording of full assemblies and fuel rods	Transfer/Storage Pool	
In-cell visual examination	All Cells	
Gamma scan, gross or specific isotope	High Energy and Low Level Cells	
Profilometry	High Energy Cell	
Rod length measurement	High Energy Cell and other Cells	
Eddy current, coil, and probe techniques	High Energy Cell	
Destructive Examinations		
Fission gas sampling (qualitative and quantitative)	High Energy Cell	

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Capability	Area(s) of Operation	
	High Level Cell (before 1972)	
Rod internal void volume measurement	High Energy Cell	
	High Level Cell (before 1972)	
Rod marking and sectioning	High Energy Cell	
	High Level Cell (before 1972)	
Fuel bulk density measurements	Mechanical Test Cell	
	Low Level Cell (before 1967)	
Fuel burnup analysis	Mechanical Test Cell ^{P014} and Alpha-Gamma	
	Cells	
Neutron dosimetry	High Energy Cell	
Retained or dissolved gas analysis	Mechanical Test Cell	
Autoradiograph, alpha and beta/gamma	Alpha-Gamma and Low Level Cells	
Mechanical Testing and Metallurgical Examination	<b>IS</b>	
Tensile and bend testing	Mechanical Test Cell and Mechanical Test Room	
Burst testing	High Level and Mechanical Test Cells	
Creep testing	Low Level Subcell ^{P016} and Mechanical Test Cell ^{P014}	
Expanded mandrel testing	Mechanical Test Cell ^{P014}	
Hardness and microhardness measurements	Alpha-Gamma Cells	
Instrumented Charpy impact testing	Charpy Room and Mechanical Test Cell	
Metallography	Alpha-Gamma Cells	
	Low Level Cell (Pre-1964)	
Radiochemistry including spectrophotometry, pH,	Microprobe Counting Room	
specific resistance, titrimetry, hydrogen by inert gas-	Mechanical Test Cell	
fusion, activation analysis, and alpha, beta, and		
gamma spectrometry		
Electron microprobe analysis	Microprobe/Counting Koom	
Scanning electron microscopy and energy-dispersive X-ray analysis	Microprobe/Counting Room	

Ongoing defense-related research supporting the development of nuclear reactors was conducted primarily for the U.S. Navy. In addition, defense research and development programs were performed for other Department of Defense agencies. Section 3.2.7 describes the history of atomic energy defense research conducted in the Hot Cell Laboratory.

When operations officially ceased in 1987, the Hot Cell Laboratory housed three large hot cells, ten small Alpha-Gamma Cells, a Mechanical Test Cell, and numerous support areas including a large transfer and storage pool, chemical test room, counting room, machine shop, shear testing room, and waste storage rooms.^{C004,U002}

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Figure 3-3. General Flow of Materials Through Building JN-1.

The BCLDP began remedial decontamination and decommissioning activities in 1989. The objective of the BCLDP is to decontaminate the West Jefferson buildings and associated grounds. According to the DOE-Ohio Strategic Plan, decontamination of the West Jefferson buildings is to be completed by 2003, conditioned on availability of requisite funding.

#### 3.2.2 JN-1 Process Operations

The Hot Cell Laboratory primarily supported experiments on small scale irradiation capsules during the first five years of operations and over the years developed the capacity to examine complete reactor fuel assemblies.^{U003} Figure 3-3 illustrates the general flow of test materials through Building JN-1 as described in Sections 3.2.2.1 through 3.2.2.7.

Figure 3-4 provides a process flow diagram for the Hot Cell Laboratory. The diagram identifies laboratory subprocesses, chemicals, and materials introduced during process

Building JN-1 Hot Cell Laboratory Combustibles (Section 4.0) Materials to be Examined fuel assemblies, bundles, rods, cladding, irradiated Glass capsules, and other (Section 5.0) materials of construction Mechanical Testing Nondestructive Examination Process Chemicals acetic, boric, bromic, chromic, Tensile & Bend TV Viewing & Video Taping Heterogeneous Burst Debris (Section 6.0) Gamma Scan citric, hydrochloric, hydrofluoric, Creep lactic, nitric, oxalic, & plorle acids Profilometry Expanded Mandrei Rod Length Measurement Inorganic Solids butyl, ethyl, & methyl alcohols Hardness & Microhardness Weiahing (Section 7.0) Charpy Impact Horizontal & Vertical Bow aluminum (powder), cadmium, Eddy Current Leaded Rubber/ lead (shielding), mercury, Destructive Examination potassium, rhodium, ruthenium, Plastic (Section 8.0) Fission Gas Sampling and sodium metals Specimen/Sample Prep Rod Void Volume Marking & Sectioning Liquids acetone Fuel Bulk Density Cutting (Section 9.0) ammonia Fuel Burnup Analysis ammonium nitrate Grinding · Retained Gas Analysis ammonium thiocyanate Polishina Metal · Dissolved Gas Analysis barium sulfate Washing (Section 10.0) benzene Autoradiography carbon tetrachloride 1,1,1-trichloroethane **Organic Solids** Metallurgical Examination cupric chloride (Section 11.0) dioctylphthalate Metallography ethylene glycol Radiochemistry Solidified Inorganic glyceragealethant • pH Waste (Section 12.0) alvcerine hydrogen peroxide Titrimitry, Specific Resistance isopropyl ether Electron Microprobe kerosene SEM & Energy Dispersive X-ray methyl ethly ketone potassium dichromate potassium ferricyanate potassium permanganate silver nitrate sodium hydroxide trichloroethylene Miscellaneous Process Materials wood jumber, pallets, & tool parts glass windows, labware, & apparatus ion exchange resins Decontamination Products light bulbs Radiacwash cartridge water filters Shaklee Basic I roughing and HEPA filters electronic equipment and batteries Shaklee Basic H activated charcoal (carbon) metal, glass, & plastic cans, bottles, & bags Zep Old Smoky cioth & paper wipes, rags, towels, & vacbags Radsorb, Drierite, and Floor Dry absorbants Spray Nine leaded gloves and aprons metal tools, wire, pipe, and parts lead-acid batteries hydraulic and silicon-based oils protective clothing, respirators, & cartridges

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operations, and potential TRU waste materials contained in the cells. The waste materials have been segregated into waste groups that are further described and characterized in sections 4.0 through 12.0. Specific chemical usage is summarized in Table 3-4. As the materials in the waste groups are packaged, Acceptable Knowledge process Descriptions will be prepared describing the specific generating processes and TRU waste streams that will be certified for disposal at WIPP.

### 3.2.2.1 High Energy Cell, Transfer/Storage Pool and Supporting Areas

In 1971 and 1972 the High Bay was constructed to house the High Energy Cell, the Transfer and Storage Pool, and support areas.^{D002} The High Energy Cell and pool were specifically designed to accommodate the receipt, storage, transfer, and examination of entire fuel assemblies which was not feasible before this time.^{D001,P003,P005} Prior to the construction of this cell, shipments of radioactive materials were received at the loading dock and introduced directly into the cells.^{P032}

Shipping casks containing fuel assemblies and fuel rod bundles were received at the west door of the High Bay. A 50-ton crane was used to transfer the casks to the Washdown Room where road film and dirt were removed using water and soaps including Radiacwash, Spray Nine, and other non-hazardous soaps and detergents.^{C001,P025} After cleaning, the casks were backfilled with water. The displaced air and steam generated during the backfill process was vented by metal tubing to the High Energy Cell. The casks were then lowered into the Transfer and Storage Pool to be opened.^{P003,P002}

The dimensions of the pool are 20 feet by 20 feet and 49 feet deep. A motorized, moveable personnel bridge running north and south across the pool was used to facilitate work in the pool. A one-ton pool bridge crane is located on a monorail attached to the bridge. The pool contained deionized water filtered by 12 ion exchange columns in the Pump Room. The Pump Room also contained a tank of deionized water to replenish water lost due to evaporation.^{P002} Draining of the Transfer and Storage Pool water began in November of 1985. Evaporation of the pool water was completed in September 1997.^{C014, D004}

After the cask lid was removed under water, the fuel assemblies or bundle of rods were lifted out of the cask using the pool bridge crane and placed in storage racks in the pool. The cask was lifted out of the pool, decontaminated with deionized water and soap, reassembled, and placed back on the bed of the truck that delivered the cask. The fuel rod bundles or fuel assemblies were moved to the access lid in the floor of the High Energy Cell through a transfer canal in the west wall of the pool. The assemblies and bundles were lifted into the cell by a 5-ton crane and the access lid closed. ^{P002, P003} Materials and equipment could also be introduced into the cell directly through the Washdown Room by moving an 18-ton steel door, through two 6-inch drop-in ports, or through a 9x12-foot door in the celling of the cell.

Once in the High Energy Cell, several nondestructive examinations were performed on the assemblies, bundles, and rods. Fuel assemblies would be weighed, dimensionally measured,

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temperature measured, photographed, and video taped. Holes were then cut into the nozzle/cap of the assembly and the rods (encased by the cladding) would be removed. Each rod would be photographed, weighed, measured, and tested. The nondestructive examination included eddy current, profilometry, horizontal and vertical bow, and gamma scan. Fission gases were collected and analyzed from a hole drilled in the cladding. These examinations accounted for approximately 90 percent of the work performed in the cell. Other activities included studies and characterization of Three Mile Island (TMI) resins used to decontaminate water, effects of cobalt-60 on instrumentation, and fuel rod compaction.^{C001,P003,P005} Fuel rods were cut into 4-foot lengths using a tubing cutter or an abrasive wheel (non-fuel containing materials only), placed in transfer casks, and transferred to the High Level Cell for destructive testing.^{C004,C006,P003,P005}

#### 3.2.2.2 Controlled Access Area

The primary purpose of the Controlled Access Area was to support operations conducted in the High Level, Low Level, Mechanical Test, and Alpha-Gamma Cells, and the Charpy Room. Equipment, specimens, and other materials were moved in and out of the cells through a number of ports and doors accessible in the Controlled Access Area. A crane was used to move heavy equipment into the High Level Cell through the main cell door. In addition to material transfer and special project support capabilities, the area was used for manipulator arm service, drum compaction, and equipment and specimen decontamination.^{P012}

A repair bench in the area was used to repair and service manipulators. The repair bench was accessible by four glove holes through a window for shielding. The manipulators fit through five other holes in the window and rested on a device that controlled their movement to accommodate repair. The contaminated portion of the arm was generally repaired inside the Controlled Access Area. When possible, repairs were performed through the gloves in the window to reduce personnel exposure.^{P012} A portable 55-gallon drum compactor was also operated in a plastic enclosure located next to the Mechanical Test Cell door. The purpose of the enclosure was to contain airborne contamination which was vented to the Controlled Access Area.

The Sabotage Program (1981 to 1983) was conducted in the Controlled Access Area and represents one of the last major projects performed in the Hot Cell Laboratory. The purpose of the program was to determine the effects of a terrorist attack on nuclear fuel shipping casks. The program involved shooting a shaped charge at a small model cask containing fuel. The solids and gases generated by the collision were collected and analyzed. The apparatus is bolted on top of a small empty pool which was used for storage of small casks prior to 1972.^{P012}

During operations, the Controlled Access Area was in constant use and was repeatedly contaminated by numerous programs. Even though the area has been cleaned many times, the floors and other surfaces remain contaminated with the radionuclides from these programs.^{P012}

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#### 3.2.2.3 High Level and Low Level Cells

The High Level Cell and the Low Level Cell were the original cells constructed in Building JN-1 in 1955. These cells consist of two main cells located above grade level with two subterranean cells (subcells) located directly below. The cells are situated back-to-back and form a block that is 24 feet by 24 feet. The Low Level and High Level Cells are separated by a shielding wall and were designed to provide shielding for 10,000 curies and 10 million curies of a 1-Mev gamma emitter, respectively.^{P030} Entrance into each cell was gained through hydraulically operated steel doors in the Controlled Access Area. Materials were introduced through numerous access ports. Primary access into each cell was through a small access port in the hydraulic cell doors.^{U002,P015} Transfer trays were used to transfer materials between the High Level, Low Level, and Mechanical Test Cells.

Nondestructive examination, destructive testing, and material preparation were conducted in the High Level and Low Level Cells before the addition of the High Energy and Mechanical Test Cells. They were the only cells available for remote manipulation of nuclear reactor materials before the addition of the Alpha-Gamma Cells in 1964.^{C006,P032} Operations performed in the High Level and Low Level Cells included rod marking, sectioning, defueling, visual examination, and dimensional measuring, in addition to gamma scan, tensile, fission gas, rod void volume, fuel bulk density, autoradiography, and burst testing analyses.^{C006,P017,P018} The High Level Cell is highly contaminated due to destructive cutting and grinding of fuel materials and the cobalt-60 work performed in this cell.^{P018}

In 1988 and 1989 some waste materials remaining in the High Level Cell were compacted into 105 metal cans and most were transferred to the Low Level Cell where they were gamma scanned.^{P017,P018}

Directly beneath the Low Level and High Level Cells are subterranean cells that were designed to support operations in the main cells and conduct operations that did not require remote manipulations.^{P032} Creep testing was performed in the subcell below the Low Level Cell. The creep equipment, wiring, and supplies are still in the subcell. The subcell below the High Level Cell was not used for project work and is currently used for storage of used contaminated HEPA filters.^{P016}

#### 3.2.2.4 Mechanical Test Cell

The Mechanical Test Cell was constructed in 1967 to increase the mechanical properties testing capabilities of spent fuel, irradiated cladding, and structural materials.^{U002,U003} A variety of examinations were conducted in the cell including tensile, creep, vacuum fusion, burst, radial burnup, expanded mandrel, and density testing. In addition to spent fuel, other materials studied in the cell included Zircaloy, stainless steel, nickel alloys, refractory materials, and pressure vessel steels.^{P030}

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Specimens prepared in the High Level and Low Level Cells were placed in aluminum vials and other containers and then transferred into the Mechanical Test Cell through the tray in the Low Level Cell. Burst and tensile testing were performed on fuel cladding. In addition to the destructive mechanical testing performed in the cell, density testing of spent fuel was conducted using elemental mercury metal. The mercury used for this testing became contaminated with metal dust and fragments from the fuel specimens. Additionally, the cell was contaminated by mercury that could be observed on the floor and work surfaces.^{P014}

#### 3.2.2.5 Alpha-Gamma Cells

The Alpha-Gamma Cells consist of 10 small interconnected cells located in the basement under the shipping and receiving dock. The original nine cells (Cells 1-9) were constructed in 1964 to support metallography testing of fuel rod specimens. The room adjacent to Cell 1 was constructed as an evaporator room for low level radioactive liquids. Cell 10 was added in this area in the early 1970s.^{C012} Each of the cells have a window and manipulators on the front and roll out on wheels for access. Figure 3-4 illustrates the general floor plan and lists the operations performed in the Alpha-Gamma Cells.^{P001,P005,P024}

Cells 1, 2, 3, and 10 were dedicated to support metallography operations. In the Low Level Cell, 1-3/4 inch diameter specimens (Metmounts) of fuel rods were prepared and transported to Cell 3 (through Cell 4). Cell 3 is considered the most contaminated cell, and the two grinders in the cell are the most contaminated equipment in the Alpha-Gamma Cells. The Metmounts were ground to the desired thickness in Cell 3; polished, washed (alcohol and water), and acid etched in Cell 2; and hardness tested and photographed using a Metallograph in Cell 1. In the early 1970s the equipment in Cell 1 was abandoned and operations were transferred to Cell 10 where a new Metallograph was installed.^{C012,P001}

The primary purpose of Cell 4 was to transport materials to and from the Controlled Access Area. Metmounts were transferred through a connecting chute in the Controlled Access Area. Pneuatic transfer tubes from the Low Level Cell were seldom used because of operational problems with this system.^{C001,P017} A Macro Camera was used to photograph items in the cell. Cell 4 was not modified from its original design and was never used for project work.

Cells 5 and 6 were designed and constructed to prepare californium-252 sources. Californium wire used to produce the sources was introduced into Cell 5 through a port in the floor of the back dock. The wire was cut, weighed, and put into small containers. The containers were sealed by welding on the lids and then leak tested and washed. These sources were dropped through another port in the back dock into Cell 6 where they were placed and sealed (welded) in a larger container that was also leak tested and washed. Cell 6 should be the least contaminated cell since only sealed sources were introduced into this area.^{P001} Cells 5 and 6 were not used until the mid-1970s when the californium source program was initiated, and have not been used for any other projects.^{C001} Wastes generated in Cells 5 and 6 will not be TRU contaminated due to the short half-life of californium-252 (< 3 years) and low specific activity of curium-248 (daughter product).

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Figure 3-5. Alpha-Gamma Cells Floor Plan.

Cell 7 was used primarily for dissolution and solidification of burnup fuel specimens. Fuel clad samples from burnup analysis in the High Level Cell were transferred into Cell 7 through a port in the floor of the back dock. The fuel was dissolved from the cladding, and the cladding was returned to the High Level Cell or Low Level Cell to be weighed. The fuel was dissolved with nitric acid, diluted, mixed with cement, and allowed to solidify in foam cups (referred to as *slugs*).^{C012,P001} Fumes from the solutions escaped the box through a faulty pipe connection and contaminated the rear area of Cells 7 and 8. The rear window in Cell 8 is severely etched from these fumes.^{P001} Additional information on slugs is presented in Section 12.0.

In Cell 8, unclad fuel samples were tested for thermal conductivity. The samples were brought into the cell through pass-through ports from Cell 4. The test instrument consisted of a two-foot diameter plate with a small electric furnace at the center. The sample was placed in the furnace and heated. The sample was exposed to a laser beam which was directed to a detector for analysis.^{P001}

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X-ray diffraction testing of Metmounts was performed in Cell 9. The samples were brought into the cell through pass-through ports from Cell 4 and analyzed (crystalline structure) with an energy dispersive x-ray analysis unit and goniometer (instrument used to measure crytalline structure).^{P001}

#### 3.2.2.6 Charpy Room

Shear testing of irradiated nuclear reactor structural materials was performed in the Charpy Room. Encapsulated specimens were received in shipping casks that were opened in the Controlled Access Area. The casks were opened and the capsules containing the specimens were transferred to the High Level Cell for opening, separating, and accounting. The specimens were placed in metal cans, cleaned in an ultrasonic bath in the Controlled Access Area, then transferred to the Charpy Room for analysis.^{P019}

Irradiated specimens were tested using an automated Charpy Impact Machine. Temperature-controlled oil and alcohol baths were used to heat and cool specimens, respectively. Although Charpy testing was the primary operation performed in this area, reconstituting broken specimens was also performed in this room. Milling machines operated by manipulators were used for this activity.^{P019}

Storage holes (74) constructed in the floor of the Charpy Room were used to store stacked cans of specimens. These holes were originally part of the Controlled Access Area and were used to store specimens from the High Level and Low Level Cells. These 4-, 6-, and 8- inch diameter storage holes are currently empty but are expected to contain high levels of radioactive contamination.^{P019}

#### 3.2.2.7 Supporting Areas

In addition to the activities involving irradiated and highly radioactive materials, examination and analysis of nonradioactive and low activity materials were performed in several areas in Building JN-1. The Chemistry Laboratory, Counting Room, Microprobe Room, and Machine Shop were located to the east of the main office area and south of the hot cells. Wastes generated from the cleanup of these areas were not mixed with wastes generated during hot cell operations and are not included in the suspect TRU waste inventory.^{C007,P009}

#### 3.2.3 Radionuclides

This section describes the methods used to identify and quantify the isotopes present in radioactive waste generated in Building JN-1 which support the TRU waste determination. Since gamma rays emitted by radionuclides can be readily detected and quantified by common measurement techniques, the modeling method described in this section will chiefly use emitted gamma rays as a means of modeling the quantity of isotopes present in a standard waste stream. Because isotopes other than gamma emitters are known to be present from the operating history

of the facility, the method combines laboratory measurements of the isotopic distribution with a computer-generated distribution so all isotopes are accounted for.^{P040}

The measured isotopic distribution is based on laboratory analysis (alpha and gamma spectroscopy) of smear samples taken from the Controlled Access Area.^{P040} As described in Section 3.2.2.2, contamination in the Controlled Access Area should be a good approximation of the contamination as a whole in Building JN-1 because this area was in constant use and was repeatedly contaminated by projects being conducted in all the hot cells.^{P012,C004} Using the measured distribution as a base, the remaining isotopes are scaled according to the distribution generated by a computer code (ORIGEN2) that models the production and decay of fission and activation products of commercial nuclear power plants, since the majority of activities in Building JN-1 is attributable to nuclear power reactor research.^{C004,P040}

A given quantity of the combined distribution is then used as the radiation source with the MicroShield computer shielding code to generate external gamma ray dose rates for various package and form weights. These dose rates are used to generate dose rate versus weight conversion equations for each package and form. These equations are then used to calculate the activity content, in millicuries, for individual packages and waste forms.^{P040}

The first step in developing packaging dose rate-to-package activity relationships is the determination of a representative source term for a given waste stream. In this case, a specific source composition was determined for the West Jefferson North Facility. The source term utilized in these dose-to-curie content models is based on gamma and alpha spectroscopic analyses of smear samples taken from the Controlled Access Area in 1996 and on ORIGEN2 code output for low enrichment fuel representative of that examined at the facility. Since a majority of the research performed in this area supported commercial reactor materials, the code output utilized in this characterization is based on a standard commercial nuclear fuel composition with 3.2 percent enriched uranium-35, burnup of 33,000 MegaWatt-Days per Metric Ton Uranium, and a decay time of 15 years (as of May 1998).

Because cesium-137 is the dominant isotope in the analysis of smear samples and in the ORIGEN2 output, it is used as the basis for the standard mix specification. Ratios for all isotopes not present in the smear samples are based on the ORIGEN2 calculations and then normalized to cesium-137. The source activities are further normalized to a total activity of 1 millicurie. The isotopic distribution in Table 3-3 represents a total activity of 1 millicurie distributed among the isotopes in proportion to their presence as indicated by the ORIGEN2 code calculations and the sample analysis. This distribution is considered to be the Building JN-1 standard isotope mix.^{P040} This mix should be a good approximation that will be evaluated periodically to incorporate future sampling and analysis and account for radioisotope decay. Comparison of assay results from analysis of low level waste boxes verified that the isotopic ratios were within an order of magnitude of the calculated isotopic mix.^{C026}

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Isotone	Distribution (%)	Isotope	Distribution (%)
Cesium-137	2.68E-01	Plutonium-239	6.33E-04
Berium-137m	2.54E-01	Tellurium-125m	3.31E-04
Darium-157m Plutonium-241	2.24E-01	Cadmium-113m	1.00E-04
Stroptium-90	1.06E-01	Rhodium-106	6.55E-05
Vttrium.90	1.06E-01	Ruthenium-106	6.55E-05
Promethium_147	9.42E-03	Americium-243	6.10E-05
Curium 244	6.66E-03	Neptunium-239	6.10E-05
Diutonium 238	6 33E-03	Curium-243	5.24E-05
Amoriaium 241	4 99E-03	Technetium-99	4.79E-05
Europium 154	3 87E-03	Americium-242m	2.43E-05
Europium 155	2.87E-03	Americium-242	2.42E-05
Coholt 60	1.99E-03	Curium-242	2.00E-05
Coolum 134	1 40E-03	Uranium-234	4.54E-06
Antimony 125	1.36E-03	Uranium-238	1.17E-06
Anumony-125 Distorium 240	1.08E-03	Uranium-236	9.45E-07
Samarium-151	1.07E-03	Uranium-235	6.46E-09

 Table 3-3.
 Isotopic Distribution for Building JN-1 Waste.

## 3.2.4 Chemical Use

Chemicals historically used in Building JN-1, and how and/or where the chemicals were used, are presented in Table 3-4. The information in the table was created during review of the source documentation describing operations in the hot cells. In addition, a list of chemicals was prepared using Material Safety Data Sheets compiled based on a chemical inventory taken of the building. Cognizant personnel reviewed the list to determine the chemicals used in the hot cells that may have contaminated the suspect TRU waste inventory characterized in Sections 4.0 through 12.0.

Chemical	Use
Acetic acid	Photograph printing. ^{C001}
Acetone	Decontamination of shipping casks. ^{C001}
-	Cleaning and degreasing. Cell and equipment decontamination. ^{P024,P032}
Acid (unspecified)	Metmount etching. ^{P001}
Alcohol (unspecified)	Dissolving fuel specimens. ²⁰⁰ Metmount washing. ²⁰⁰¹
	Charpy specimen cooling. ^{P019}
	Cleaning.
Aluminum powder	Metmount polisning.
Ammonia	Unknown.
Ammonium nitrate	Low Level and Alpha-Gamma Cells.

Table 3-4.Historical Chemical Usage.

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Chemical	Use	
Ammonium thiocyanate	Unknown. ^{C001}	
Barium sulfate	Alpha-Gamma Cells. ^{C011}	
Benzene	Laboratory operations. ^{C006}	
Boric acid	Alpha-Gamma Cells. ^{C011}	
Bromine or bromic acid	Alpha-Gamma Cells. ^{C011}	
Butyl alcohol	Low Level and High Level Cells. ^{C001}	
Cadmium	Dosimeter capsules. ^{C001}	
	Shielding.	
Carbon tetrachloride	Immersion density, Mechanical Test Cell. Cool, Coll	
	Laboratory operations. ^{C006}	
	Low Level, High Level, and Alpha-Gamma Cells. ^{C011}	
	Cell and equipment decontamination. ^{P024,P032}	
Chlorothene (1,1,1-trichloroethane)	Cleaning and degreasing. COLL, POL9	
Chromic acid	Glass cleaner. ^{C001}	
Citric acid	Ultrasonic cleaning. ^{C001}	
Cupric chloride	Alpha-Gamma Cells. ^{C011}	
Diamond paste	Metmount polishing. ^{C001}	
Dioctylphthalate	Filter testing. ^{C001}	
Ethanol	Decontamination of shipping casks. ^{C001}	
	Cleaning and degreasing. ^{C001}	
Ethylene glycol	Metmount polishing. ^{C001}	
Freon	Refrigerant. ^{P006}	
Glycerigia	Metal Etchant. ^{C001}	
Glycerin	Unknown. ^{C001}	
Hydraulic oil (Unocal Unax Rx 32) ^{P025}	In system for raising/lowering Cell doors. ^{P015}	
Hydrochloric acid	Low Level, High Level, Mechanical Test, and Alpha-Gamma Cells. ^{C001,C011}	
Hydrofluoric acid	Alpha-Gamma Cells. ^{C001}	
Hydrogen peroxide (30% or less)	Alpha-Gamma Cells. ^{C001}	
Isopropyl ether	Low Level and High Level Cells. ^{C001}	
Kerosene	Metmount polishing. ^{C001}	
	Cell and equipment decontamination. ^{P024,P032}	
Lactic acid	Alpha-Gamma Cells. ^{C001}	
Mercury	Immersion density in Mechanical Test Cell. ^{C001,C011}	
	In inventory in the Controlled Access Area, High Level, Low Level, and High Energy Cells. ^{C011}	
Methanol	High Level Cell. ^{C001,C011}	
Methyl ethyl ketone	Low Level, High Level, and Alpha-Gamma Cells. ^{C011}	
Nitric acid	Low Level, High Level, Mechanical Test, and Alpha-Gamma Cells. ^{C011}	
Oxalic acid	Alpha-Gamma Cells. ^{C011}	

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Chemical	Use	
Picric acid	Unknown. ^{C001}	
Potassium	Low Level and High Level Cells. ^{C001}	
Potassium dichromate	Low Level and Alpha-Gamma Cells. ^{C011}	
Potassium ferricyanate	Unknown. ^{C001}	
Potassium permanganate	Alpha-Gamma Cells. ^{C014}	
	Metal etching. ^{C014}	
Radiacwash ^{P025}	Decontamination of shipping casks. ^{C001}	
	Ultrasonic cleaning. ^{C001}	
	Cell and equipment decontamination. ^{P024,P032}	
Rhodium	Alpha-Gamma Cells. ^{C011}	
Ruthenium	Alpha-Gamma Cells. ^{C011}	
Shaklee Basic I and Basic H ^{P025}	Decontamination of shipping casks. ^{C001}	
Silicon-based oil	Hot oil baths in Charpy Room. ^{P006}	
Silver nitrate	Ultrasonic cleaning. ^{C001}	
	Low Level and High Level Cells. ^{C011}	
Sodium	Low Level and High Level Cells. ^{C001}	
Sodium hydroxide	Low Level, High Level, Mechanical Test, and Alpha-Gamma Cells. ^{C001,C011}	
Spray Nine ^{P025}	Decontamination of shipping casks. ^{C001,P003}	
	Cleaning of surfaces in Charpy Room. ^{P019}	
Sulfuric acid	Low Level, High Level, and Alpha-Gamma Cells. ^{C011}	
Trichloroethylene	Cleaning and degreasing. ^{C001,C011}	
UCAR phenolic resins ^{P025}	Metmount material in Low Level and Alpha-Gamma Cells. ^{C001,C011}	
Water	Metmount washing. ^{C001,P001}	
	Cell and equipment decontamination. ^{P024,P032}	
Zep Old Smoky ^{P025}	Cleaning of surfaces in all areas. ^{P019}	

#### 3.2.5 Waste Characterization

During the review of acceptable knowledge, waste materials were identified and assigned matrix parameter categories. Similar waste materials are combined to create the "waste groups." Table 3-5 summarizes the waste assessed for the development of this document including the matrix parameter category, waste description, and EPA Hazardous Waste Numbers. Matrix parameter categories and EPA Hazardous Waste Numbers are assigned in accordance with TC-AP-03.1, Collection, Review, and Management of Acceptable Knowledge Documentation. The characterization for each stream is discussed in Sections 4.0 through 12.0.

To be determined (TBD) was used for the wastes in Table 3-5 when existing acceptable knowledge was insufficient to assign EPA Hazardous Waste Numbers. The appropriate numbers

will be assigned utilizing additional analysis or evaluation of these streams as they are processed. This supplemental information will be incorporated into the acceptable knowledge record as it is developed per procedure TC-AP-03.1.

Matrix Parameter Category	Waste Description	EPA Hazardous Waste Numbers	Section Reference
S5319	Plastic/Rubber	None	4.0
S5390	Combustibles	D005, D007, D008, D009,	
		D011, F001, F002, and F005	_
S5122a	Glass	None	5.0
S5122b	Leaded Glass	D005, D008	
S5122c	Light Bulbs	D008, D009	
S5410	Composite Filter Debris	TBD	6.0
S5460	Electronic Equipment	D008, D011	
S5490	Fixtures and Equipment	D009	· <u>····································</u>
S3114	Absorbed Oil	TBD	7.0
S3118	Activated Carbon	None	
S3119a	Inorganic Particulate Materials	TBD	
	(Spent)		
S3119b	Inorganic Particulate Materials	None	
	(Unused)		
S5121	Concrete	None	
S5123	Ceramic/Brick	D005, D007, D008, D009,	
		D011, F001, F002, and F005	
S5129	Insulation	None	
S5311	Leaded Rubber	D008	8.0
X7410	Lead Acid Batteries	D008	
L1110	Acidic Wastewaters	D002	9.0
X7100	Elemental Mercury	D009	
L2120	Aqueous/Nonhalogenated Organic	TBD	
	Liquids		
L2220a	Nonhalogenated Organic Liquids	None	
L2220b	Nonhalogenated Organic Liquids	D001, F003	
L9000	Unknown Liquids	TBD	
S5111	Metal Debris	None	10.0
S5112	Metal Debris with Lead	D008	
X7211	Elemental Lead	D008	
X7220	Elemental Cadmium	D006	
X7510	Bulk Reactive Metals	D003	
X7530	Pyrophoric Metals	D001	
S3211	Organic Resin	None	11.0
S3150	Slugs	TBD	12.0

Table 3-5.Waste Characterization.

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#### 3.2.6 Waste Management

With the exception of gloveboxes and a small inventory of containers generated during the decontamination of the Plutonium Laboratory (Building JN-4), all suspect TRU waste stored at the site was generated in Building JN-1. These materials consist of process-related waste, equipment, furniture, and other items remaining in the cells, and the equipment and waste materials that have been removed from the cells and stored in various locations in Buildings JN-1 and JN-3.

Due to the radiological research nature of operations in the Hot Cell Laboratory, care was taken to minimize cross contamination of test materials; however, there was extremely limited consideration for waste segregation. Based on the research performed to date and verified by personnel who worked in the area, most of the wastes generated by different programs were mixed together and would be indistinguishable. Additionally, due to the numerous projects conducted in these areas and the movement of materials between the cells, equipment and material still contained in the cells could be cross-contaminated with radionuclides from almost any project. Historically, the only waste segregation performed was the separation of compactible waste for volume reduction purposes.^{C004,P012,P027} During historical operations there were no waste material transfers between buildings at the site, and therefore Building JN-1 wastes would not be mixed with wastes from other buildings (JN-2, JN-3, or JN-4).^{C001}

Containers (drums and hoppers) of waste in storage at the site were removed from the cells during campaigns to clean out the cells. This waste was not mixed with low level wastes from the supporting areas in Building JN-1. For this reason, all containers of suspect TRU waste will have originated from hot cell operations or clean-up operations and will not be contaminated with compounds used exclusively outside of the cells.

TRU waste is contained in the hot cells and stored in other locations in Building JN-1, including the Low Level Subcell, High Level Subcell, Waste Storage Area, Equipment Storage Room, Pump Room, Washdown Room, High Energy Cell Mezzanine, and Waste Storage Shed. In addition, hoppers and drums of Building JN-1 waste are stored in the empty pools in Building JN-3.^{U007,P004,P007,P008} It is estimated that after this waste has been segregated, decontaminated, and packaged, approximately 2,500 cubic feet of predominantly remote-handled TRU waste will remain to be certified for disposal at WIPP.^{C008}

#### 3.2.7 Defense Operations

A variety of defense-related research was conducted in the Hot Cell Laboratory. Based on the review of available documentation, most of the defense research was conducted for the U.S. Navy (both directly and through DOE's Naval Reactors Program), although defense research and development were also performed for the U.S. Air Force and U.S. Army. Based on review of shipping logbooks/forms, project questionnaires, correspondence, research reports, and personnel interviews, defense work was conducted in Building JN-1 starting no later than 1958 and continuing until at least 1984.^{C001,C002,C004,C005,C009,C025,U005,U010,U014,U016,P028,P029}

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The first large-scale application of nuclear power was submarine propulsion. BMI was heavily involved in the Naval Reactor Program from its inception. BMI invented and developed Zircaloy, a zirconium corrosion-resistant alloy used in fuel elements and assemblies. Other major accomplishments included the design and fabrication of the original reference fuel for the Nautilus program and the development of the Hot Isostatic Pressure and Picture Frame bonding technologies used to fabricate nuclear submarine cores.^{P036}

Defense research and development work performed for the U.S. Air Force included support of the Aircraft Nuclear Propulsion Program (1947-1961). In addition, reactor studies were conducted involving air-cooled, molten salt-fueled, and supercritical water-cooled reactors. Very extensive work was performed in the area of producing activated forms of oxide ceramics having a density approaching theoretical.^{P036}

BMI contributed to the development of the Army Package Power Reactor (APPR) program which began in 1954 and continued to the late 1960s. The mission of this program was to develop a portable and transportable reactor for use at the South Pole.^{C001} Because the APPR was a water-cooled reactor, naval reactor technology could be utilized. BMI had considerable experience with the fuel selected for this reactor, and also contributed to the refinement of the processes used to fabricate fuel clad elements.^{P036} In addition to U.S. Army reactor research, the Hot Cell Laboratories were used in 1981 and 1982 for research on the destruction of chemical agents using high level gamma fields. The research involved irradiating capsules of chemical agents using cobalt-60 sources.^{C002} Chemical agent capsules were not introduced into the cell; irradiation occurred in the closed beam-type port.^{C014}

Based on guidance from CAO, a TRU waste is eligible for disposal at WIPP if it has been generated in whole or part by one of the *atomic energy defense activities* listed in section 10101(3) of the Nuclear Waste Policy Act of 1982. By definition, this includes any activity performed in carrying out any of the following functions:^{P041}

- A) Naval reactors development
- B) Weapons activities, including defense inertial confinement fusion
- C) Verification and control technology
- D) Defense nuclear materials production
- E) Defense nuclear waste and materials by-products management
- F) Defense nuclear materials security and safeguards and security investigations
- G) Defense research and development

Based on the review of acceptable knowledge, TRU wastes generated in the hot cell areas of Building JN-1 clearly meet the definition of defense waste in the areas of naval reactors development (A) and defense research and development (G). Even though a majority of the work performed in the hot cells was not done in support of defense programs, no attempt was

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made to segregate the defense-related wastes (see Section 3.2.6). Since segregation of this waste is no longer feasible, by definition this waste is eligible for disposal at the WIPP facility.

## 3.2.8 Spent Nuclear Fuel and High Level Waste

There are several criteria for determining if a material is spent nuclear fuel or high level waste. A material is spent nuclear fuel if <u>all</u> of the following are true:

- 1) The material is fuel withdrawn from a nuclear reactor after irradiation.
- 2) The constituent elements have not been separated by reprocessing.
- 3) Test specimens of developmental reactor fuels were not irradiated solely for research and development purposes.

The material is high level waste if any of the following are true:

- 1) The waste is from first cycle separation.
- 2) The material is test specimens of fissionable material not irradiated solely for research and development purposes.
- 3) The material is derived from fuel withdrawn from a nuclear reactor after irradiation where the constituent elements have been separated by reprocessing.

As previously described, Building JN-1 was used for research and development testing of spent fuel withdrawn from nuclear reactors after irradiation. The types of testing did not involve separation of constituent elements from the fuel. The spent fuel inventory does not include specimens of developmental reactor fuels, and therefore is considered spent nuclear fuel.^{U008} Spent nuclear fuel is not eligible for disposal at the WIPP site and will be segregated from other wastes.^{P041}

The radioactive waste inventory generated in Building JN-1 consists of materials such as wipes, glass, and metal contaminated with spent nuclear fuel (see Sections 4.0 through 12.0). These materials were not withdrawn from a nuclear reactor and were not generated from first cycle separation. Therefore, the waste is not high level waste and eligible for disposal at WIPP.^{P041}

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## 4.0 COMBUSTIBLES

This waste group consists of combustible wastes generated by the research and development activities conducted in Building JN-1. A description of the combustibles wastes and associated matrix parameter categories are provided in Section 4.1. The RCRA characterization of the combustibles wastes is presented in Section 4.2.

## 4.1 Waste Description

*Matrix Parameter Category S5319, Plastic/Rubber:* This waste consists of plastic or rubber debris material including polyethylene, polyvinyl chloride, nylon, styrofoam, Tygon, Plexiglas, and neoprene. Waste items may include sheeting, rope, tape, empty bottles and bags, booties, hose/tubing, lids from spray cans, cups, packing material, respirators, gloves, boots, rain suits, o-rings, electrical cords, safety glasses, and vials.^{P004,P006,U009} Plexiglas panels may also be included.^{P001,P006}

*Matrix Parameter Category S5390, Combustibles:* This waste consists of paper, cloth, and wood debris materials, including canvas, leather, and other porous materials. Waste items may include wipes, rags, towels, tissues, blotter paper, mop heads, empty vacuum bags, paper filters, cardboard, sand paper, smear paper, masking tape, notebook paper, documents, and protective clothing (e.g., gloves, booties, suits, safety shoes/boots), lumber, plywood, pallets, and tool handles.

## 4.2 Waste Characterization

Combustibles wastes are characterized based on knowledge of the material, knowledge of the processes generating the waste, and visual examination. This section provides a RCRA hazardous waste determination for combustible wastes. EPA Hazardous Waste Numbers applicable to the combustible waste group are presented by matrix parameter category in Table 4-1. These conclusions are supported by the evaluation in Sections 4.2.1 and 4.2.2.

Matrix Parameter Category	Waste Description	EPA Hazardous Waste Numbers
S5319 -	Plastic/Rubber	None
S5390	Combustibles	D005, D007, D008, D009, D011,
		F001, F002, and F005

Table 4-1.	Combustibles	Waste	Characterization.
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#### 4.2.1 Characteristic Waste

The materials in this waste group may exhibit a characteristic of hazardous waste as defined in 40 CFR 261, Subpart C, as a toxic waste (metals only). Based on the acceptable

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knowledge documentation reviewed, the materials do not exhibit the characteristics of ignitability (40 CFR 261.21), corrosivity (40 CFR 261.22), reactivity (40 CFR 261.23), or toxicity for organics (40 CFR 261.24).

*Ignitability*: The materials in this waste group do not meet the definition of ignitability as defined in 40 CFR 261.21. The materials are not liquid, and visual examination is performed to ensure free liquids are not added to containers during packaging. In addition, absorbents are added to wastes having the potential of generating free liquids (e.g., wet wipes). The materials are not capable of causing fire through friction or absorption of moisture. Cellulose-based materials such as wipes are visually examined to verify the absence of visible nitrocellulose contamination. The materials in this waste group are therefore not ignitable wastes (D001).

*Corrosivity*: The materials in this waste group do not meet the definition of corrosivity as defined in 40 CFR 261.22. The materials are not liquid, and visual examination is performed to ensure free liquids are not added to containers during packaging. In addition, absorbents are added to wastes having the potential of generating free liquids (e.g., wet wipes). The materials in this waste group are therefore not corrosive wastes (D002).

*Reactivity*: The materials in this waste group do not meet the definition of reactivity as defined in 40 CFR 261.23. The materials are stable and will not undergo violent chemical change. The materials will not react violently with water, form potentially explosive mixtures with water, or generate toxic gases, vapors, or fumes when mixed with water. The materials do not contain cyanides or sulfides, and are not capable of detonation or explosive reaction. The materials in this waste group are therefore not reactive wastes (D003).

*Toxicity*: The materials in this waste group meet the definition of toxicity as defined in 40 CFR 261.24. The toxicity characteristic contaminants fall into one of two categories: metals and organics. Organic compounds include halogenated- and nonhalogenated-solvents, pesticides, herbicides, and other toxic compounds. This waste group does not exhibit the characteristic of toxicity for organics. This waste group may exhibit the characteristic of toxicity for metals including barium, chromium, lead, mercury, and silver metals.

Barium sulfate, chromic acid, potassium dichromate, mercury, and silver nitrate were used in various processes in Building JN-1 (see Table 3-4). Lead shielding materials were also used (see Section 8.0). The combustibles waste (S5390) is potentially contaminated with these materials. Therefore, waste S5390 is assigned EPA Hazardous Waste Numbers D005, D007, D008, D009, and D011 since a representative sample of this waste cannot be obtained for verification purposes.

Benzene, carbon tetrachloride, methyl ethyl ketone, and trichloroethylene were used in Building JN-1. These compounds were typically used as solvents. Therefore, waste S5390 is regulated as a listed hazardous waste and not a characteristic waste since these compounds are specifically addressed in the treatment standards for listed hazardous waste.
Plastic/rubber waste (S5319) is visually examined prior to or during packaging to ensure no residue is present, or this waste meets the definition of empty container (40 CFR 261.7). Therefore, waste S5319 does not exhibit the characteristic of toxicity (D004-D043).

#### 4.2.2 Listed Hazardous Waste

The material in this waste group is a listed hazardous waste because it was mixed with spent solvents listed in 40 CFR 261, Subpart D. Based on the acceptable knowledge documentation reviewed, the material is not, or was not mixed with, a hazardous waste from specific sources (40 CFR 261.32), or a discarded commercial chemical product, an off-specification species, a container residue, or a spill residue thereof (40 CFR 261.33).

Carbon tetrachloride, 1,1,1-trichloroethane, trichloroethylene, benzene, and methyl ethyl ketone were used in laboratory operations and for cleaning/degreasing (see Table 3-4). The combustibles waste (S5390) is potentially contaminated with these spent solvents. Therefore, waste S5390 is assigned EPA Hazardous Waste Numbers F001, F002, and F005.

Plastic/rubber waste (S5319) is visually examined prior to or during packaging to ensure no residue is present, or this waste meets the definition of empty container (40 CFR 261.7). Any contact with listed solvents is considered incidental. Therefore, waste S5319 is not a listed hazardous waste.

This waste may be contaminated with spent acetone and methanol used for cleaning and degreasing. However, this waste is not assigned F003 because it does not exhibit the characteristic of ignitability (F003-listed solvents are listed solely for ignitability).

F004-listed solvents were not used in the areas or processes generating combustible wastes. Therefore, this waste group is not an F004-listed hazardous waste.

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#### 5.0 GLASS

This waste group consists of glass wastes generated by the research and development activities conducted in Building JN-1. A description of the glass wastes and associated matrix parameter categories are provided in Section 5.1. The RCRA characterization of the glass wastes is presented in Section 5.2.

# 5.1 Waste Description

*Matrix Parameter Category S5122a, Glass:* This waste consists of glass debris, including laboratory glassware, windows, and various glass apparatus.^{P001,P006}

Matrix Parameter Category S5122b, Leaded Glass: This waste consists of leaded glass windows.^{P001}

Matrix Parameter Category S5122c, Light Bulbs: This waste includes various light bulbs such as fluorescent, incandescent, and mercury vapor light bulbs.

# 5.2 Waste Characterization

Glass wastes are characterized based on knowledge of the material, knowledge of the processes generating the waste, and visual examination. This section provides a RCRA hazardous waste determination for glass wastes. EPA Hazardous Waste Numbers applicable to the glass waste group are presented by matrix parameter category in Table 5-1. These conclusions are supported by the evaluation in Sections 5.2.1 and 5.2.2.

Matrix Parameter Category	Waste Description	EPA Hazardous Waste Numbers
	Glass	None
S5122b	Leaded Glass	D005, D008
S51220	Light Bulbs	D008, D009

Table 5-	1 (	Glass	Waste	Characte	rization	n
		CIIGOS.		~11mm ~~~~~		-

#### 5.2.1 Characteristic Waste

The materials in this waste group may exhibit a characteristic of hazardous waste as defined in 40 CFR 261, Subpart C, as a toxic waste (metals only). Based on the acceptable knowledge documentation reviewed, the materials do not exhibit the characteristics of ignitability (40 CFR 261.21), corrosivity (40 CFR 261.22), reactivity (40 CFR 261.23), or toxicity for organics (40 CFR 261.24).

*Ignitability*: The materials in this waste group do not meet the definition of ignitability as defined in 40 CFR 261.21. The materials are not liquid, and visual examination is performed to

ensure free liquids are not added to containers during packaging. The materials in this waste group are therefore not ignitable wastes (D001).

*Corrosivity*: The materials in this waste group do not meet the definition of corrosivity as defined in 40 CFR 261.22. The materials are not liquid, and visual examination is performed to ensure free liquids are not added to containers during packaging. The materials in this waste group are therefore not corrosive wastes (D002).

**Reactivity:** The materials in this waste group do not meet the definition of reactivity as defined in 40 CFR 261.23. The materials are stable and will not undergo violent chemical change. The materials will not react violently with water, form potentially explosive mixtures with water, or generate toxic gases, vapors, or fumes when mixed with water. The materials do not contain cyanides or sulfides, and are not capable of detonation or explosive reaction. The materials in this waste group are therefore not reactive wastes (D003).

*Toxicity*: The materials in this waste group meet the definition of toxicity as defined in 40 CFR 261.24. The toxicity characteristic contaminants fall into one of two categories: metals and organics. Organic compounds include halogenated- and nonhalogenated-solvents, pesticides, herbicides, and other toxic compounds. This waste group does not exhibit the characteristic of toxicity for organics. This waste group may exhibit the characteristic of toxicity for barium, lead, and mercury metals.

Glass wastes (S5122a) are visually examined prior to or during packaging to ensure no residue is present, or these wastes meet the definition of empty container (40 CFR 261.7). Therefore, waste S5122a does not exhibit the characteristic of toxicity (D004-D043).

Analytical data for leaded glass windows indicate lead and barium leach above the toxicity characteristic level.^{C013,U021} Therefore, waste S5122b is assigned EPA Hazardous Waste Numbers D005 and D008.

Fluorescent and mercury vapor light bulbs may exhibit the characteristic of toxicity for mercury, and incandescent light bulbs may exhibit the characteristic of toxicity for lead (solder). Therefore, waste S5122c is assigned EPA Hazardous Waste Numbers D008 and D009.

#### 5.2.2 Listed Hazardous Waste

The material in this waste group is not, or was not mixed with, a waste listed in 40 CFR 261, Subpart D as a hazardous waste from non-specific sources (40 CFR 261.31), as a hazardous waste from specific sources (40 CFR 261.32), or as a discarded commercial chemical product, an off-specification species, a container residue, or a spill residue thereof (40 CFR 261.33).

Glass wastes are visually examined prior to or during packaging to ensure no residue is present, or these wastes meet the definition of empty container (40 CFR 261.7). Any contact with listed solvents is considered incidental. Therefore, this waste group is not a listed hazardous waste.

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# 6.0 HETEROGENEOUS DEBRIS

This waste group consists of heterogeneous debris wastes generated by the research and development activities conducted in Building JN-1. A description of the heterogeneous debris wastes and associated matrix parameter categories are provided in Section 6.1. The RCRA characterization of the heterogeneous debris wastes is presented in Section 6.2.

#### 6.1 Waste Description

*Matrix Parameter Category S5410, Composite Filter Debris:* This waste consists of filters constructed of more than one material type (e.g., metal, inorganic nonmetal, and organic materials). Filters include air intake and exhaust filters such as roughing filters (prefilters) and HEPA filters, as well as Tri-Nuc cartridge filters used to filter the transfer and storage pool water.

*Matrix Parameter Category S5460, Electronic Equipment:* This waste consists of electronic equipment constructed of more than one material type (e.g., metal, inorganic nonmetal, and organic materials). Waste items may include electric panels (ultrasonic cleaners).^{P001}

*Matrix Parameter Category S5490, Fixtures and Equipment:* This waste consists of debris materials that are a combination of both organic and inorganic compositions. Waste items may include contaminated fixtures and equipment such as vacuum cleaners and thermostats.^{P006,P011}

### 6.2 Waste Characterization

Heterogeneous debris wastes are characterized based on knowledge of the material, knowledge of the processes generating the waste, and visual examination. This section provides a RCRA hazardous waste determination for heterogeneous debris wastes. EPA Hazardous Waste Numbers applicable to the heterogeneous debris waste group are presented by matrix parameter category in Table 6-1. These conclusions are supported by the evaluation in Sections 6.2.1 and 6.2.2.

Matrix Parameter Category	Waste Description	EPA Hazardous Waste Numbers
	Composite Filter Debris	TBD
S5460	Electronic Equipment	D008, D011
S5490	Fixtures and Equipment	D009

Table 6-1.	Heterogeneous	Debris	Waste	Characterization.
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#### 6.2.1 Characteristic Waste

The materials in this waste group may exhibit a characteristic of hazardous waste as defined in 40 CFR 261, Subpart C, as a toxic waste (metals only). Based on the acceptable knowledge documentation reviewed, the materials do not exhibit the characteristics of ignitability (40 CFR 261.21), corrosivity (40 CFR 261.22), or reactivity (40 CFR 261.23).

**Ignitability**: The materials in this waste group do not meet the definition of ignitability as defined in 40 CFR 261.21. The materials are not liquid, and visual examination is performed to ensure free liquids are not added to containers during packaging. The materials are not capable of causing fire through friction or absorption of moisture. Air exhaust filters are visually examined to verify the absence of visible nitrocellulose contamination. The materials in this waste group are therefore not ignitable wastes (D001).

*Corrosivity*: The materials in this waste group do not meet the definition of corrosivity as defined in 40 CFR 261.22. The materials are not liquid, and visual examination is performed to ensure free liquids are not added to containers during packaging. The materials in this waste group are therefore not corrosive wastes (D002).

**Reactivity:** The materials in this waste group do not meet the definition of reactivity as defined in 40 CFR 261.23. The materials are stable and will not undergo violent chemical change. The materials will not react violently with water, form potentially explosive mixtures with water, or generate toxic gases, vapors, or fumes when mixed with water. The materials do not contain cyanides or sulfides, and are not capable of detonation or explosive reaction. The materials in this waste group are therefore not reactive wastes (D003).

**Toxicity**: The materials in this waste group meet the definition of toxicity as defined in 40 CFR 261.24. The toxicity characteristic contaminants fall into one of two categories: metals and organics. Organic compounds include halogenated- and nonhalogenated-solvents, pesticides, herbicides, and other toxic compounds. This waste group does not exhibit the characteristic of toxicity for organics. This waste group may exhibit the characteristic of toxicity for lead, mercury, and silver metals.

Air intake filters (S5410) have not come in contact with toxicity characteristic compounds, and therefore are not toxic wastes (D004-D043).

Air exhaust filters (S5410) may be contaminated with dust and particulate material from cutting and grinding fuel specimens. Additional research is necessary to determine if these filters exhibit the characteristic of toxicity due to the metal impurities in fuel. If air exhaust filters are determined to be hazardous waste, they will be segregated from air intake filters.

Electronic equipment (S5460) may contain lead or silver above the toxicity characteristic level due to the presence of solder on circuit boards. Therefore, waste S5460 is assigned EPA Hazardous Waste Numbers D008 and D011.

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Contaminated fixtures and equipment (S5490) such as vacuum cleaners and thermostats may exhibit the characteristic of toxicity for mercury.^{P006,P011} Therefore, waste S5490 is assigned EPA Hazardous Waste Number D009.

Benzene, carbon tetrachloride, methyl ethyl ketone, and trichloroethylene were used in Building JN-1 (see Table 3-4). However, air exhaust filters (S5410) are designed to entrap dust and particulate material and will not retain toxicity characteristic levels of these compounds. In addition, filters are visually examined to verify the absence of visible contamination. Therefore, waste S5410 does not exhibit the characteristic of toxicity for organics (D018-D043).

#### 6.2.2 Listed Hazardous Waste

The material in this waste group is not, or was not mixed with, a waste listed in 40 CFR 261, Subpart D as a hazardous waste from non-specific sources (40 CFR 261.31), as a hazardous waste from specific sources (40 CFR 261.32), or as a discarded commercial chemical product, an off-specification species, a container residue, or a spill residue thereof (40 CFR 261.33).

Air exhaust filters (S5410) may have come in contact with solvent vapors. However, vapors do not meet the definition of a solid waste. Filters are visually examined prior to or during packaging to ensure no visible contamination is present. Therefore, waste S5410 is not a listed hazardous waste.

Electronic equipment (S5460) and other contaminated fixtures and equipment (S5490) are visually examined prior to or during packaging to ensure no residue is present. Any contact with listed solvents is considered incidental. Therefore, wastes S5460 and S5490 are not listed hazardous wastes.

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## 7.0 INORGANIC SOLIDS

This waste group consists of inorganic solids wastes generated by the research and development activities conducted in Building JN-1. A description of the inorganic solids wastes and associated matrix parameter categories are provided in Section 7.1. The RCRA characterization of the inorganic solids wastes is presented in Section 7.2.

### 7.1 Waste Description

*Matrix Parameter Category S3114, Absorbed Oil:* This waste consists of oils mixed with an inorganic absorbent material such as Drierite or Floor Dry.^{P006,P025}

*Matrix Parameter Category S3118, Activated Carbon:* This waste consists of unused charcoal contained in the filter bed of the fan unit in the Mezzanine above the High Energy Cell.^{P003,P006}

*Matrix Parameter Category S3119a, Inorganic Particulate Materials (spent):* This waste may include floor sweepings and vacuum cleaner contents. U009,P001,P004,P006

*Matrix Parameter Category S3119b, Inorganic Particulate Materials (unused):* This waste may include soda ash and cement that has not been used.^{U009,P001,P004,P006}

*Matrix Parameter Category S5121, Concrete:* This waste consists of concrete debris such as concrete blocks used for radiation shielding (does not include construction materials).^{P001,P004,P006}

*Matrix Parameter Category S5123, Ceramic/Brick:* This waste may include ceramic refractory material (firebrick).^{U009}

*Matrix Parameter Category S5129, Insulation:* This waste consists of pipe insulation.^{U009}

## 7.2 Waste Characterization

Inorganic solids wastes are characterized based on knowledge of the material, knowledge of the processes generating the waste, and visual examination. This section provides a RCRA hazardous and Toxic Substances Control Act (TSCA) waste determination for inorganic solids wastes. EPA Hazardous Waste Numbers applicable to the inorganic solids waste group are presented by matrix parameter category in Table 7-1. These conclusions are supported by the evaluation in Sections 7.2.1, 7.2.2, and 7.2.3.

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Matrix Parameter Category	Waste Description	EPA Hazardous Waste Numbers	
S3114	Absorbed Oil	TBD	
S3118	Activated Carbon	None	
S3119a	Inorganic Particulate Materials (spent)	TBD	
S3119b	Inorganic Particulate Materials (unused)	None	
S5121	Concrete	None	
S5123	Ceramic/Brick	D005, D007, D008, D009,	
		D011, F001, F002, and F005	
S5129	Insulation	None	

Table 7-1.	Inorganic	Solids	Waste	Characterization.
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#### 7.2.1 Characteristic Waste

The materials in this waste group may exhibit a characteristic of hazardous waste as defined in 40 CFR 261, Subpart C, as a toxic waste (metals only). Based on the acceptable knowledge documentation reviewed, the materials do not exhibit the characteristics of ignitability (40 CFR 261.21), corrosivity (40 CFR 261.22), or reactivity (40 CFR 261.23).

*Ignitability*: The materials in this waste group do not meet the definition of ignitability as defined in 40 CFR 261.21. The materials are not liquid, and visual examination is performed to ensure free liquids are not added to containers during packaging. In addition, absorbents are added to wastes having the potential of generating free liquids. The materials in this waste group are therefore not ignitable wastes (D001).

*Corrosivity*: The materials in this waste group do not meet the definition of corrosivity as defined in 40 CFR 261.22. The materials are not liquid, and visual examination is performed to ensure free liquids are not added to containers during packaging. In addition, absorbents are added to wastes having the potential of generating free liquids. The materials in this waste group are therefore not corrosive wastes (D002).

**Reactivity:** The materials in this waste group do not meet the definition of reactivity as defined in 40 CFR 261.23. The materials are stable and will not undergo violent chemical change. The materials will not react violently with water, form potentially explosive mixtures with water, or generate toxic gases, vapors, or fumes when mixed with water. The materials do not contain cyanides or sulfides, and are not capable of detonation or explosive reaction. The materials in this waste group are therefore not reactive wastes (D003).

*Toxicity*: The materials in this waste group meet the definition of toxicity as defined in 40 CFR 261.24. The toxicity characteristic contaminants fall into one of two categories: metals and organics. Organic compounds include halogenated- and nonhalogenated-solvents, pesticides, herbicides, and other toxic compounds. This group does not exhibit the characteristic of toxicity for organics. This waste group may exhibit the characteristic of toxicity for barium, chromium, lead, mercury, and silver metals.

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Absorbed oil (S3114) and spent inorganic particulate materials (S3119a) may be contaminated with toxic metals or organics depending on how the oil was used. Additional research is necessary to determine if these wastes exhibit the characteristic of toxicity (D004-D043). A representative sample of wastes S3114 and S3119a will be obtained for verification purposes.

Activated carbon (S3118) is unused charcoal contained in a filter bed of a fan unit. Inorganic particulate materials (S3119b) are also unused. Since these materials were not used, they will not exhibit the characteristic of toxicity (D004-D043). A representative sample of wastes S3118 and S3119b will be obtained for verification purposes.

Concrete blocks (S5121) and pipe insulation (S5129) are visually examined prior to or during packaging to ensure no residue is present. Therefore, wastes S5121 and S5129 do not exhibit the characteristic of toxicity (D004-D043).

Firebrick (S5123) was removed from an incinerator in Building JN-1. The incinerator was used to burn combustible materials generated in the building. It is assumed the firebrick is contaminated with the same constituents as the combustibles (see Section 4.0). Therefore, waste S5123 is assigned EPA Hazardous Waste Numbers D005, D007, D008, D009, and D011 since a representative sample of this waste cannot be obtained for verification purposes.

#### 7.2.2 Listed Hazardous Waste

The material in this waste group is a listed hazardous waste because it was mixed with, or derived from the treatment of, spent solvents listed in 40 CFR 261, Subpart D. Based on the acceptable knowledge documentation reviewed, the material is not, or was not mixed with, a hazardous waste from specific sources (40 CFR 261.32), or a discarded commercial chemical product, an off-specification species, a container residue, or a spill residue thereof (40 CFR 261.33).

Activated carbon (S3118) is unused charcoal contained in a filter bed of a fan unit. Inorganic particulate materials (S3119b) are also unused. Since these materials were not used, they are not listed hazardous wastes.

Concrete blocks (S5121) and pipe insulation (S5129) are visually examined prior to or during packaging to ensure no residue is present. Any contact with listed solvents is considered incidental. Therefore, wastes S5121 and S5129 are not listed hazardous wastes.

Firebrick (S5123) was removed from an incinerator in Building JN-1. The incinerator was used to burn combustible materials generated in the building. It is assumed the firebrick is contaminated with the same constituents as the combustibles (see Section 4.0). Therefore, waste S5123 is assigned EPA Hazardous Waste Numbers F001, F002, and F005.

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# 7.2.3 TSCA Waste Determination

The material in this waste group is not TSCA regulated waste as defined in 40 CFR 761. The source of the oil which was absorbed (S3114) is believed to be from the hydraulic cylinders used to open and close the doors to the High Level Cell and Low Level Cell.^{P006} This hydraulic oil does not contain polychlorinated biphenyls (PCBs).^{C006} Therefore, waste S3114 is not a TSCA regulated waste. A representative sample of this waste will be obtained for verification purposes.

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# 8.0 LEADED RUBBER/PLASTIC

This waste group consists of leaded rubber/plastic wastes generated by the research and development activities conducted in Building JN-1. A description of the leaded rubber/plastic wastes and associated matrix parameter categories are provided in Section 8.1. The RCRA characterization of the leaded rubber/plastic wastes is presented in Section 8.2.

### 8.1 Waste Description

Matrix Parameter Category S5311, Leaded Rubber: This waste consists of leaded gloves and aprons.

*Matrix Parameter Category X7410, Lead Acid Batteries:* This waste consists of lead acid battery casings which have been drained of their electrolyte.

# 8.2 Waste Characterization

Leaded rubber/plastic wastes are characterized based on knowledge of the material, knowledge of the processes generating the waste, and visual examination. This section provides a RCRA hazardous waste determination for leaded rubber/plastic wastes. EPA Hazardous Waste Numbers applicable to the leaded rubber/plastic waste group are presented by matrix parameter category in Table 8-1. These conclusions are supported by the evaluation in Sections 8.2.1 and 8.2.2.

Table 8-1.	Leaded	Rubber/Plastic	Waste	Characterization.
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Matrix Parameter Category	Waste Description	EPA Hazardous Waste Numbers
S5311	Leaded Rubber	D008
X7410	Lead Acid Batteries	D008

#### 8.2.1 Characteristic Waste

The materials in this waste group may exhibit a characteristic of hazardous waste as defined in 40 CFR 261, Subpart C, as a toxic waste (metals only). Based on the acceptable knowledge documentation reviewed, the materials do not exhibit the characteristics of ignitability (40 CFR 261.21), corrosivity (40 CFR 261.22), reactivity (40 CFR 261.23), or toxicity for organics (40 CFR 261.24).

**Ignitability**: The materials in this waste group do not meet the definition of ignitability as defined in 40 CFR 261.21. The materials are not liquid, and visual examination is performed to ensure free liquids are not added to containers during packaging. The materials in this waste group are therefore not ignitable wastes (D001).

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*Corrosivity*: The materials in this waste group do not meet the definition of corrosivity as defined in 40 CFR 261.22. The materials are not liquid, and visual examination is performed to ensure free liquids are not added to containers during packaging. The materials in this waste group are therefore not corrosive wastes (D002).

**Reactivity:** The materials in this waste group do not meet the definition of reactivity as defined in 40 CFR 261.23. The materials are stable and will not undergo violent chemical change. The materials will not react violently with water, form potentially explosive mixtures with water, or generate toxic gases, vapors, or fumes when mixed with water. The materials do not contain cyanides or sulfides, and are not capable of detonation or explosive reaction. The materials in this waste group are therefore not reactive wastes (D003).

*Toxicity*: The materials in this waste group meet the definition of toxicity as defined in 40 CFR 261.24. The toxicity characteristic contaminants fall into one of two categories: metals and organics. Organic compounds include halogenated- and nonhalogenated-solvents, pesticides, herbicides, and other toxic compounds. This waste group does not exhibit the characteristic of toxicity for organics. This waste group may exhibit the characteristic of toxicity for lead metal.

Leaded rubber (S5311) and lead acid batteries (X7410) contain lead above the toxicity characteristic level. These wastes are visually examined prior to or during packaging to ensure no residue is present. Therefore, wastes S5311 and X7410 are assigned EPA Hazardous Waste Number D008.

#### 8.2.2 Listed Hazardous Waste

The material in this waste group is not, or was not mixed with, a waste listed in 40 CFR 261, Subpart D as a hazardous waste from non-specific sources (40 CFR 261.31), as a hazardous waste from specific sources (40 CFR 261.32), or as a discarded commercial chemical product, an off-specification species, a container residue, or a spill residue thereof (40 CFR 261.33).

Leaded rubber (S5311) and lead acid batteries (X7410) are visually examined prior to or during packaging to ensure no residue is present. Any contact with listed solvents is considered incidental. Therefore, wastes S5311 and X7410 are not listed hazardous wastes.

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### 9.0 LIQUIDS

This waste group consists of liquid wastes generated by the research and development activities conducted in Building JN-1. A description of the liquid wastes and associated matrix parameter categories are provided in Section 9.1. The RCRA characterization of the liquid wastes is presented in Section 9.2.

### 9.1 Waste Description

#### 9.1.1 Inorganic Liquids

*Matrix Parameter Category L1110, Acidic Wastewaters:* This waste consists of various acids and acid solutions including nitric acid.^{P001,P006}

*Matrix Parameter Category X7100, Elemental Mercury:* This waste consists of liquid elemental mercury (mercury contaminated equipment is not included in this waste).^{P006}

#### 9.1.2 Organic Liquids

*Matrix Parameter Category L2120, Aqueous/Nonhalogenated Organic Liquids:* This waste consists of hydraulic oil (Unocal Unax RX 32), wash water, and a sludge of sand and mixed fission products (dust, small fragments) located on the floor of the hydraulic door room beneath the Controlled Access Area.^{C006,P006,P015} Small items such as tools may also be present in this waste.^{P006} Over the years, the oil leaked from hydraulic cylinders used to open and close the doors to the High Level Cell and Low Level Cell. The floor of the Controlled Access Area has been washed many times with water and detergent, and some of the wash water drained into the opening between the door and Controlled Access Area floor.^{P015,P037}

*Matrix Parameter Category L2220a, Nonhalogenated Organic Liquids:* This waste consists of nonhalogenated organic liquids with a flash point greater than or equal to 60°C, such as glycols and oils.

*Matrix Parameter Category L2220b, Nonhalogenated Organic Liquids:* This waste consists of nonhalogenated organic liquids with a flash point less than 60°C, such as alcohols.

#### 9.1.3 Unknown Liquids

*Matrix Parameter Category L9000, Unknown Liquids:* This waste consists of unmarked bottles of liquids contained inside the various cells.^{P001}

# 9.2 Waste Characterization

Liquid wastes are characterized based on knowledge of the material, knowledge of the processes generating the waste, and visual examination. This section provides a RCRA hazardous and TSCA waste determination for liquid wastes. EPA Hazardous Waste Numbers applicable to the liquids waste group are presented by matrix parameter category in Table 9-1. These conclusions are supported by the evaluation in Sections 9.2.1, 9.2.2, 9.2.3.

Matrix Parameter Category	Waste Description	EPA Hazardous Waste Numbers
T 1110	Acidic Wastewaters	D002
¥7100	Elemental Mercury	D009
T 2120	A queous/Nonhalogenated Organic Liquids	TBD
L2120	Nonhalogenated Organic Liquids	None
L2220a	Nonhalogenated Organic Liquids	D001, F003
L22200	Indmatogenated Organic Diquids	TBD
L9000	Unknown Liquids	

Table 9-1.	Liquid	Waste Chara	cterization.
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#### 9.2.1 Characteristic Waste

The materials in this waste group may exhibit a characteristic of hazardous waste as defined in 40 CFR 261, Subpart C, as an ignitable waste (40 CFR 261.21), a corrosive waste (40 CFR 261.22), and a toxic waste (metals only). Based on the acceptable knowledge documentation reviewed, the materials do not exhibit the characteristic of reactivity (40 CFR 261.23).

**Ignitability:** Nonhalogenated organic liquids (L2220b) meet the definition of ignitability as defined in 40 CFR 261.21(a)(1), and therefore are assigned EPA Hazardous Waste Number D001.

*Corrosivity*: Acidic wastewaters (L1110) meet the definition of corrosivity as defined in 40 CFR 261.22, and therefore are assigned EPA Hazardous Waste Number D002.

**Reactivity:** The materials in this waste group do not meet the definition of reactivity as defined in 40 CFR 261.23. The materials are stable and will not undergo violent chemical change. The materials will not react violently with water, form potentially explosive mixtures with water, or generate toxic gases, vapors, or fumes when mixed with water. The materials do not contain cyanides or sulfides, and are not capable of detonation or explosive reaction. The materials in this waste group are therefore not reactive wastes (D003).

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**Toxicity**: The materials in this waste group meet the definition of toxicity as defined in 40 CFR 261.24. The toxicity characteristic contaminants fall into one of two categories: metals and organics. Organic compounds include halogenated- and nonhalogenated-solvents, pesticides, herbicides, and other toxic compounds. This waste group does not exhibit the characteristic of toxicity for organics. This waste group may exhibit the characteristic of toxicity for mercury metal.

Elemental mercury (X7100) waste exhibits the characteristic of toxicity for mercury and is therefore assigned EPA Hazardous Waste Number D009.

Aqueous/nonhalogenated organic liquids (L2120) may be contaminated with toxic metals or organics depending on the contamination from the Controlled Access Area washed into the hydraulic door room below. Additional research is necessary to determine if these wastes exhibit the characteristic of toxicity (D004-D043). A representative sample of the treated waste will be obtained for verification purposes.

Nonhalogenated organic liquids (L2220a) do not exhibit the characteristic of toxicity (D004-D043). A representative sample of the treated waste will be obtained for verification purposes.

Unknown liquids (L9000), acidic wastewaters (L1110), and nonhalogenated organic liquids (L2220b) may exhibit the characteristic of toxicity and will be characterized on a caseby-case basis. A representative sample of the treated waste will be obtained for verification purposes.

#### 9.2.2 Listed Hazardous Waste

The material in this waste group is a listed hazardous waste because it is a spent solvent or was mixed with spent solvents listed in 40 CFR 261, Subpart D. Based on the acceptable knowledge documentation reviewed, the material is not, or was not mixed with, a hazardous waste from specific sources (40 CFR 261.32), or a discarded commercial chemical product, an off-specification species, a container residue, or a spill residue thereof (40 CFR 261.33).

Nonhalogenated organic liquids (L2220a) were not mixed with spent solvents. Therefore, waste L2220a is not a listed hazardous waste.

Nonhalogenated organic liquids (L2220b) may include spent solvents such as methanol. Therefore, waste L2220b is an F003-listed hazardous waste.

Unknown liquids (L9000) may be listed spent solvents or mixed with listed spent solvents and will be characterized on a case-by-case basis.

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#### 9.2.3 TSCA Waste Determination

The material in this waste group is not TSCA regulated waste as defined in 40 CFR 761. The source of the oil (L2120) is from the hydraulic cylinders used to open and close the doors to the High Level Cell and Low Level Cell.^{P006} This hydraulic oil does not contain PCBs.^{C006} Waste L2220a does not contain PCBs. Therefore, wastes L2120 and L2220a are not TSCA regulated wastes. A representative sample of these wastes will be obtained for verification purposes, as applicable.

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### 10.0 METAL

This waste group consists of metal wastes generated by the research and development activities conducted in Building JN-1. A description of the metal waste and associated matrix parameter categories are provided in Section 10.1. The RCRA characterization of the metal waste is presented in Section 10.2.

### 10.1 Waste Description

*Matrix Parameter Category S5111, Metal Debris:* This waste consists of metal debris including stainless steel, aluminum, iron, copper, and zirconium (Zircaloy).^{P023} Waste items may include cable, wire, planchets, signs, respirator filters, valves, piping, strapping, tools, foil, sheeting, fixtures, ballasts, equipment (e.g., pumps, motors, etc.), hardware (e.g., nuts, bolts, brackets, etc.), empty cans, specimen vials, fuel rod cladding, Metmounts, and empty tackle boxes.^{U009,P001,P003,P004,P006,P008} Aerosol cans which have been punctured and emptied of their contents are also included.^{P001}

*Matrix Parameter Category S5112, Metal Debris with Lead:* This waste consists of metal items containing lead as part of the matrix. Waste items may include lead-lined tubing.^{P006}

*Matrix Parameter Category X7211, Elemental Lead:* This waste consists of pure lead materials used primarily as radiation shielding. Waste items may include lead wool, shot, bricks, containers (e.g., pigs, casks), sheeting, and disks.^{P001,P003,P006}

*Matrix Parameter Category X7220, Elemental Cadmium:* This waste consists of cadmium wire.^{P019} The cadmium was possibly used to make covers for dosimeter capsules to increase the efficiency of dosimetry measurements.^{C001}

*Matrix Parameter Category X7510, Bulk Reactive Metals:* This waste includes metals which may react violently with water, such as sodium and potassium.^{C001}

*Matrix Parameter Category X7530, Pyrophoric Metals:* This waste includes metals which may spontaneously ignite in air, such as aluminum powder.^{C001,P001}

# 10.2 Waste Characterization

Metal wastes are characterized based on knowledge of the material, knowledge of the processes generating the waste, and visual examination. This section provides a RCRA hazardous waste determination for metal wastes. EPA Hazardous Waste Numbers applicable to the metal waste group are presented by matrix parameter category in Table 10-1. These conclusions are supported by the evaluation in Sections 10.2.1 and 10.2.2.

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Matrix Parameter Category	Waste Description	EPA Hazardous Waste Numbers
S5111	Metal Debris	None
S5112	Metal Debris with Lead	D008
X7211	Elemental Lead	D008
X7220	Elemental Cadmium	D006
X7510	Bulk Reactive Metals	D003
X7530	Pyrophoric Metals	_D001

Table 10-1. Metal Waste Characterization.

#### 10.2.1 Characteristic Waste

The materials in this waste group may exhibit a characteristic of hazardous waste as defined in 40 CFR 261, Subpart C, as an ignitable waste (40 CFR 261.21), reactive waste (40 CFR 261.23), and a toxic waste (metals only). Based on the acceptable knowledge documentation reviewed, the materials do not exhibit the characteristics of corrosivity (40 CFR 261.22), or toxicity for organics (40 CFR 261.24).

*Ignitability*: The materials in this waste group meet the definition of ignitability as defined in 40 DFR 261.21(a)(2). The materials are liquid, but visual examination is performed to ensure free liquids are not added to containers during packaging. Pyrophoric metals (X7530) will spontaneously ignite in air and therefore are assigned EPA Hazardous Waste Number D001.

*Corrosivity*: The materials in this waste group do not meet the definition of corrosivity as defined in 40 CFR 261.22. The materials are not liquid, and visual examination is performed to ensure free liquids are not added to containers during packaging. The materials in this waste group are therefore not corrosive wastes (D002).

**Reactivity:** The materials in this waste group do not meet the definition of reactivity as defined in 40 CFR 261.23(a)(2). The materials are stable and will not undergo violent chemical change. The materials will not form potentially explosive mixtures with water, or generate toxic gases, vapors, or fumes when mixed with water. The materials do not contain cyanides or sulfides, and are not capable of detonation or explosive reaction. Bulk reactive metals (X7510) will react violently with water and are therefore assigned EPA Hazardous Waste Number D003.

*Toxicity*: The materials in this waste group meet the definition of toxicity as defined in 40 CFR 261.24. The toxicity characteristic contaminants fall into one of two categories: metals and organics. Organic compounds include halogenated- and nonhalogenated-solvents, pesticides, herbicides, and other toxic compounds. This waste group does not exhibit the characteristic of toxicity for organics. This waste group may exhibit the characteristic of toxicity for cadmium and lead metals.

Metal debris waste (S5111) is visually examined prior to or during packaging to ensure no residue is present, or these wastes meet the definition of empty container (40 CFR 261.7).

Visual examination will also verify the absence of toxicity characteristic metals (e.g., lead bricks). Therefore, waste S5111 does not exhibit the characteristic of toxicity (D004-D043) as long as all surfaces of the metal can be examined to verify the absence of surface contamination.

Metal debris with lead (S5112) and elemental lead (X7211) exhibit the characteristic of toxicity for lead. Therefore, wastes S5112 and X7211 are assigned EPA Hazardous Waste Number D008. The waste stream will not exhibit the characteristic of toxicity for any other constituent as long as all surfaces of the lead can be examined to verify the absence of surface contamination.

Elemental cadmium (X7220) exhibits the characteristic of toxicity for cadmium. Therefore, waste X7220 is assigned EPA Hazardous Waste Number D006.

#### 10.2.2 Listed Hazardous Waste

The material in this waste group is not, or was not mixed with, a waste listed in 40 CFR 261, Subpart D as a hazardous waste from non-specific sources (40 CFR 261.31), as a hazardous waste from specific sources (40 CFR 261.32), or as a discarded commercial chemical product, an off-specification species, a container residue, or a spill residue thereof (40 CFR 261.33).

Metal wastes are visually examined prior to or during packaging to ensure no residue is present, or these wastes meet the definition of empty container (40 CFR 261.7). Any contact with listed solvents is considered incidental. Therefore, this waste group is not a listed hazardous waste.

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### 11.0 ORGANIC SOLIDS

This waste group consists of organic solids wastes generated by the research and development activities conducted in Building JN-1. A description of the organic solids wastes and associated matrix parameter categories are provided in Section 11.1. The RCRA characterization of the organic solids wastes is presented in Section 11.2.

#### 11.1 Waste Description

*Matrix Parameter Category S3211, Organic Resin:* This waste consists of ion-exchange resin contained in bags which was used for deionizing the transfer and storage pool water. ^{U009,P002,P004} CM-2 Regenerated Mixed Bed Resin was used. ^{P025} The containers of resin likely have free liquids. ^{C001} Also included is TMI resin which was sampled, analyzed, and packaged; no other testing was conducted.

# 11.2 Waste Characterization

Organic solids wastes are characterized based on knowledge of the material, knowledge of the processes generating the waste, and visual examination. This section provides a RCRA hazardous waste determination for organic solids wastes. EPA Hazardous Waste Numbers applicable to the organic solids waste group are presented by matrix parameter category in Table 11-1. These conclusions are supported by the evaluation in Sections 11.2.1 and 11.2.2.

Table 11-1. Organic Solids Waste Characterization.

Matrix Parameter Category	Waste Description	EPA Hazardous Waste Numbers
S3211	Organic Resin	None

#### 11.2.1 Characteristic Waste

Based on the acceptable knowledge documentation reviewed, the materials do not exhibit the characteristics of ignitability (40 CFR 261.21), corrosivity (40 CFR 261.22), reactivity (40 CFR 261.23), or toxicity (40 CFR 261.24).

**Ignitability**: The materials in this waste group do not meet the definition of ignitability as defined in 40 CFR 261.21. The materials are not liquid, and visual examination is performed to ensure free liquids are not present or are not added to containers during packaging. In addition, absorbents are added to wastes containing free liquids or having the potential of generating free liquids. The materials in this waste group are therefore not ignitable wastes (D001).

*Corrosivity*: The materials in this waste group do not meet the definition of corrosivity as defined in 40 CFR 261.22. The materials are not liquid, and visual examination is performed to ensure free liquids are not added to containers during packaging. In addition, absorbents are

added to wastes having the potential of generating free liquids. The materials in this waste group are therefore not corrosive wastes (D002).

**Reactivity:** The materials in this waste group do not meet the definition of reactivity as defined in 40 CFR 261.23. The materials are stable and will not undergo violent chemical change. The materials will not react violently with water, form potentially explosive mixtures with water, or generate toxic gases, vapors, or fumes when mixed with water. The materials do not contain cyanides or sulfides, and are not capable of detonation or explosive reaction. The materials in this waste group are therefore not reactive wastes (D003).

*Toxicity*: The materials in this waste group do not meet the definition of toxicity as defined in 40 CFR 261.24. No source for toxic compounds has been identified for organic resin (S3211), and therefore does not exhibit the characteristic of toxicity (D004-D043). A representative sample of waste S3211 will be obtained for verification purposes.

#### 11.2.2 Listed Hazardous Waste

The material in this waste group is not, or was not mixed with, a waste listed in 40 CFR 261, Subpart D as a hazardous waste from non-specific sources (40 CFR 261.31), as a hazardous waste from specific sources (40 CFR 261.32), or as a discarded commercial chemical product, an off-specification species, a container residue, or a spill residue thereof (40 CFR 261.33).

Organic resin (S3211) did not come in contact with listed solvents. Therefore, this waste group is not a listed hazardous waste.

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# **12.0 SOLIDIFIED INORGANIC WASTE**

This waste group consists of solidified inorganic wastes generated by the research and development activities conducted in Building JN-1. A description of the solidified inorganic waste and associated matrix parameter categories are provided in Section 12.1. The RCRA characterization of the solidified inorganic waste is presented in Section 12.2.

#### 12.1 Waste Description

*Matrix Parameter Category S3150, Slugs:* Slugs were produced from dissolving fuel specimens in an acid solution which was then diluted several times and mixed with cement and water and allowed to solidify in foam cups.^{P001} The slugs will contain limited amounts of radionuclides from fuel because of this dilution.^{C006}

### 12.2 Waste Characterization

Solidified inorganic wastes are characterized based on knowledge of the material, knowledge of the processes generating the waste, and visual examination. This section provides a RCRA hazardous waste determination for solidified inorganic wastes. EPA Hazardous Waste Numbers applicable to the solidified inorganic waste group are presented by matrix parameter category in Table 12-1. These conclusions are supported by the evaluation in Sections 12.2.1 and 12.2.2.

Table 12-1. Solidified Inorganic Waste Characterization.

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Matrix Parameter Category		Waste Description	EPA Hazardous Waste Numbers
S3150	Slugs		TBD

#### 12.2.1 Characteristic Waste

Based on the acceptable knowledge documentation reviewed, the materials do not exhibit the characteristics of ignitability (40 CFR 261.21), corrosivity (40 CFR 261.22), or reactivity (40 CFR 261.23). Additional research is necessary to determine if the wastes exhibit the characteristic of toxicity (40 CFR 261.24).

**Ignitability**: The materials in this waste group do not meet the definition of ignitability as defined in 40 CFR 261.21. The materials are not liquid, and visual examination is performed to ensure free liquids are not added to containers during packaging. The materials in this waste group are therefore not ignitable wastes (D001).

*Corrosivity*: The materials in this waste group do not meet the definition of corrosivity as defined in 40 CFR 261.22. The materials are not liquid, and visual examination is performed to

ensure free liquids are not added to containers during packaging. The materials in this waste group are therefore not corrosive wastes (D002).

**Reactivity:** The materials in this waste group do not meet the definition of reactivity as defined in 40 CFR 261.23. The materials are stable and will not undergo violent chemical change. The materials will not react violently with water, form potentially explosive mixtures with water, or generate toxic gases, vapors, or fumes when mixed with water. The materials do not contain cyanides or sulfides, and are not capable of detonation or explosive reaction. The materials in this waste group are therefore not reactive wastes (D003).

*Toxicity*: The materials in this waste group may meet the definition of toxicity as defined in 40 CFR 261.24. The toxicity characteristic contaminants fall into one of two categories: metals and organics. Organic compounds include halogenated- and nonhalogenated-solvents, pesticides, herbicides, and other toxic compounds. The toxicity characteristic contaminants in this waste group are to be determined.

Slugs (S3150) may exhibit the characteristic of toxicity depending on the leachability of metal fuel impurities in this waste. Additional research is necessary to determine if these wastes exhibit the characteristic of toxicity (D004-D011). A representative sample of waste S3150 will be obtained for verification purposes.

#### 12.2.2 Listed Hazardous Waste

The material in this waste group is not, or was not mixed with, a waste listed in 40 CFR 261, Subpart D as a hazardous waste from non-specific sources (40 CFR 261.31), as a hazardous waste from specific sources (40 CFR 261.32), or as a discarded commercial chemical product, an off-specification species, a container residue, or a spill residue thereof (40 CFR 261.33).

Listed solvents were not used in the process that generated slugs. Therefore, this waste group is not a listed hazardous waste.

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# ACCEPTABLE KNOWLEDGE INVENTORY

Appendix A to Building JN-1 Hot Cell Laboratory Acceptable Knowledge Document

Ref	Title / Author	Summary	Date
C001	Group Interview Record of Eugene Sands, Larry Stickel, Harley Toy, Max Berchtold, Mike Failey, and George Kirsch, BCL, conducted by Kevin Peters/Jeff Harrison. WASTREN, Inc.	Interview summary of group interview with Eugene Sands, Larry Stickel, Harley Toy, Max Berchtold, Mike Failey, and George Kirsch. Includes notes taken from a very general discussion of operations in JN-1 including chemical use, flow of materials, general operations by area, radionuclides, waste management, and defense-related projects. The information collected was general and used to focus subsequent AK research.	1998. May 13.
C002	Miscellaneous Internal Correspondence for the U.S. Army Project (Project Number G8109). Battelle Columbus Laboratory.	The correspondence includes descriptions of a program conducted for the Army to examine the destruction/immobilization of toxic substances using intense gamma irradiation from Co-60 source in JN-1. Demonstrates that defense related materials from the Army was introduced into JN-1 in 1970.	November 1992 – January 1983.
C003	Internal Correspondence from Michael P. Failey to Louis B. Myers. "Characterization of the JN-1 Hot Cell Waste Drums." Battelle Columbus Laboratory.	The correspondence provides isotopic characterization data for 15 drums of Hot Cell waste materials assayed by a gamma-ray scanning system in 1985 and 1986. Isotopes include only gamma-ray emitting radionuclides and does not quantify any transuranics (Pu, Am).	1997. May 1.
C004	Interview Record of Harley Toy and George Kirsch, BCL, conducted by Kevin Peters. WASTREN, Inc.	Interview summary of interview with Harley Toy and George Kirsch. Includes summary of discussions of waste management, shut down of JN-1, operation history, chemical use and defense related work. Includes identification of defense projects from a list of projects (U014) and documentation linking Shippingport work to naval research in JN-1. In the interview, Harley and George verifies that highly enriched uranium (greater than 93%) could only be attributed to naval reactor fuel.	1998. July 13.
C005	Interview Record of Scott Kitts, BCL, conducted by Kevin Peters. WASTREN, Inc.	Interview summary of interview with Scott Kitts. Documents that Mr. Kitts identified N-Reactor fuel cladding from a dual purpose Hanford reactor in the Low Level cell. This verifies defense related materials were being tested in JN-1 during the early 1980s.	1998. July 16.
C006	Interview Record of Max Berchtold, BCL, conducted by Kevin Peters. WASTREN, Inc.	Interview summary of interview with Max Berchtold. Documents which areas operations identified in the 1970s brochure (P031) were performed, in addition of how materials were introduced into the cells. The composition of the hydraulic oil used to lift the LLC and HLC doors (no PCBs) was discussed (MSDS attached) verifies carbon tetrachloride and benzene used in early operations.	1998. July 15.
C007	Interview Record of Scott Kitts, BCL, conducted by Kevin Peters. WASTREN, Inc.	Interview summary of interview with Scott Kitts. Documents that all suspect JN-1 TRU waste in inventory (hoppers and drums) stored at the site was generated in the cells and not mixed with wastes or chemicals from the area supporting the cells.	1998. July 22.

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Ref No.	Title / Author	Summary	Date
C008	Memorandum from Thomas A. Baillieul to Mike Brown, DOE/CAO. "Certification Strategy for Transuranic Waste Generated from the Battelle Columbus Laboratories Decommissioning Project." DOE Ohio Field Office.	This memorandum summarizes the certification strategy for TRU waste generated during the Battelle Columbus Laboratories Decommissioning Project (BCLDP). The memo summarizes BCLDP and provides an estimate of RH and CH waste to be generated by the program. Attachment includes the TRU Waste Certification Program Integrated Schedule (including assumptions and commitments), Certification Strategy, and a memo requesting a small quantity site certification audit.	1998. June 25.
C009	Miscellaneous BMI correspondence. Various Authors.	Miscellaneous correspondence documenting the initial communications associated with naval reactor research (February 1956) and descriptions of navy research conducted in 1966 - 1968.	February 1956 – October 1968.
		Documents: (1) External correspondence from R. W. Dayton to W. H. Wilson, 2/9/56; (2) Memorandum from R. F. Dickerson to R. W. Dayton, 8/27/56; (3) Memorandum from J. B. Brown to J. W. Ray, 3/22/66; (4) Memorandum from J. B. Brown to J. W. Ray, 9/15/66; (5) Memorandum from J. W. Ray to R. W. Dayton, 10/27/67; (6) Memorandum from J. W. Ray to R. W. Dayton, 10/27/67; (7) Memorandum from D. C. Minton, Jr. to S. J. Paprocki, 10/29/68.	
C010	Miscellaneous DOE and Battelle Columbus Correspondence. Various Authors.	Miscellaneous correspondence relating to discussions of the amount of government research performed by Battelle laboratories, in addition to identifying the responsibility for the cost and management of the D&D operations. Several of the correspondence include attachments estimating the amount of work performed by the DOE, industry, and other government agencies (DOD, Army, Navy, Air Force, NRC). Limited information on where research was performed.	November 1985 – December 1995.
		Documents: (1) External correspondence from David A. Zorich to Pete Greenwalt, 12/27/95; (2) External Correspondence from J. O. Neff to William Daily, 11/6/91; (3) External correspondence from Jerome R. Bahlmann to Martin A. Langsam, 6/22/87; (4) Memorandum from James W. Vaughan, Jr. to Troy Wade, 8/12/87; (5) DOE Memorandum of Understanding under Contract No. W- 7405-Eng-92-M.	
C011	JN-1 Chemical Use Lists. Battelle Columbus Laboratories.	This list of chemicals was compiled by reviewing files containing MSDSs for chemicals found in JN-1. The list of chemicals was reviewed by Max Berchtold and Eugene Sands to identify those chemicals used in hot cells.	1998. July 24.
C012	Interview Record of George Kirsch, BCL, conducted by Kevin Peters. WASTREN, Inc.	Interview identifies the transfer of metallography operations from alpha/gamma Cell 1 to Cell 10, and identifies the acid used to dissolve burnout fuel in the alpha/gamma cells.	1998. July 28.
C013	Internal correspondence from D. L. Kidd to A. A. Church. "Status of TCLP Analysis on Leaded Gloves and Leaded Glovebox Windows." EG&G Rocky Flats, Inc.	Analytical data for leachable metals in leaded gloves and leaded glass windows.	1991. March 13.

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# ACCEPTABLE KNOWLEDGE INVENTORY Appendix A to Building JN-1 Hot Cell Laboratory Acceptable Knowledge Document

Ref	Title / Author	Summary	Date
C014	Group Interview Record of Eugene Sands and George Kirsch, BCL, conducted by Kevin Peters. WASTREN, Inc.	Interview summary of group interview with Eugene Sands and George Kirsch. This interview was conducted to answer specific questions raised during the first review of the AK document. Interview verified that cyanide will be contained in negligible amounts in the waste; that the Transfer/Storage Pool evaporator was completed in 1989; and that Army chemical agents will not be present in the TRU waste streams.	1998. October 22.
C025	Letter to James H. Eide, BMI, authored by Kevin Peters, WASTREN, Inc.	Letter summarizing historical research performed at West Jefferson during the 1970s and 1980s for Hanford. Based on the review of existing AK (C005, P035, P043, P044, U006), defense research includes N-reactor and LOCI programs performed in JN-1.	1999 October 30
C026	Internal Correspondence from Jim Sarge to James H. Eide	Review of Benchmark Environmental assay data for waste boxes comparing BCLDP JN Standard and Pool waste stream models with ORIGEN2 output.	1999. April 30.
D001	Telephone Interview Record of Scott Kitts and George Kirsch, conducted by Kevin Peters. "Discussions about JN-4 Waste Stored in JN-3." WASTREN, Inc.	This interview verifies that there is a small inventory of plutonium contaminated waste from D&D of the Plutonium Laboratory JN-4. Previously it had been assumed that the only waste remaining was gloveboxes that would not be disposed of at WIPP. Includes miscellaneous inventory information sent by George Kirsch.	1998. July 22.
D002	Interview Record of George Kirsch, BCL, conducted by Kevin Peters. WASTREN, Inc.	Interview summary with George Kirsch discussing the actual date that the HEC and transfer pool were constructed and operational in JN-1. The AK record identifies 1972 and 1975 as the start date.	1998. July 28.
D004	Telephone Interview Record of Cidney Voth, BCL, conducted by Kevin Peters WASTREN, Inc.	This interview was conducted to describe the operations involved with the draining of the JN-1 Transfer/Storage Pool Water and to resolve a desrepancy relating to the date when this activity was completed. In an interview with Gene Sands and George Kirsch (C014) the interviewer was informed that the evaporation of the pool was completed in 1989. Based on review of subsequent AK documentation it was determined that this activity was not completed until the mid 1990s. Cidney Voth was interviewed and documentation provided that the operation was performed between 1995 and 1997.	1999 April 27
P001	Alpha Gamma Cells JN-1A. Decontamination and Decommissioning Operation. Louis B. Myers, Eugene Sands, Paul A. Tomlin, and William E. Bruce.	This report describes general operations in the 10 Alpha Gamma Cells in the basement of JN-1A, including a description of the cells construction, cell access, and equipment and waste contained in each cell. Operations include metal specimen (Metmounts) grinding, washing, polishing, metallography analysis, production of californium sources, preparation of fuel samples for disposal, thermal conductivity testing, and x-ray diffraction. Attachments include miscellaneous photos, drawings, and narrative (unknown source), in addition to hand-written inventory lists, 1996 update, and a supplement to Battelle's procedure manual for the cells (1964). Limited information relating to specific projects and dates.	1994. August.

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# ACCEPTABLE KNOWLEDGE INVENTORY

Ref No.	Title / Author	Summary	Date
P002	Fuel Storage Pool, Pump Room and Washdown Room JN-1B. Decontamination and Decommissioning Operations. Louis B. Myers, Max B. Berchtold, and James L. Stickel.	This report describes general operations and configuration of the fuel storage pool, pump room, and washdown room in JN-1B. Operations include fuel storage (assemblies, strongbacks, rod bundles, rod holders and tools), deionization of pool water, and washing of casks. Attachments include 1996 update, miscellaneous drawings/photos, and health physics survey reports and data (including isotopics).	1995. January.
P003	High Energy Cell, Mezzanine, and Top of HEC JN-1B. Decontamination and Decommissioning Operations. Louis B. Myers, Max B. Berchtold, Paul D. Faust, and Paul A. Tomlin.	This report describes general operations and configuration of the mezzanine, high bay, High Energy Cell (HEC), and area above the HEC in the JN-1B. Operations described include receipt and transfer of fuel assemblies into the HEC and transfer of cutup rods to the High Level Cell (HLC). Nondestructive examination, including weighing, dimension measuring, visual examination, photography, and gamma scan. Attachments include a 1996 update, miscellaneous drawings/photos, and health physics survey reports and data (including isotopics). Specific projects are not identified or described.	1994. December.
P004	Waste Storage Shed JN-1A. Decontamination and Decommissioning Operations. Louis B. Myers, and Max B. Berchtold.	This report describes the contents of the Waste Storage Shed (WSS) behind Building JN-1A. The inventory includes numerous drums and hoppers of waste from the HEC, HLC, and Alpha/Gamma Cell; resin bags from the pool deionizing tanks; and other miscellaneous contaminated items. Attachments include drawings of the WSS, waste location maps, and inventory lists.	1995. June.
P005	Hot Cell Purposes and Activities. Decontamination and Decommissioning Operations. Battelle Columbus Laboratories.	Very brief and general report (2 pages) describing operations in the High Energy Cell, High Level Cell, Low Level Cell, Mechanical Test Cell, Charpy Room, and Alpha Gamma Cells.	1997. September 22.
P006	Contents of the West Jefferson North Hot Cells and Storage Areas. Louis B. Myers, Max B. Berchtold, and Eugene H. Sands.	This report describes the equipment, wastes, supplies, and other contaminated materials contained in the High Energy Cell, High Level Cell, Low Level Cell, Hydraulic Room, Pump Room, Mechanical Test Cell, Charpy Room, and Alpha Gamma Cells.	1995. May.
P007	Reactor Pool, Thermal Column, and Contractor Pool in JN-3. Decontamination and Decommissioning Operations. Louis B. Myers, and James L. Stickel.	This report provides a physical and functional description of the concrete structure that houses the remains of the retired research reactor including Reactor Pool, the Thermal Column, and the Contractor (GE) Pool. Describes waste stored in the area. Attachments include photos of the reactor. Specific projects and dates are not identified.	1995. September.
P008	West Jefferson North Hopper Location and Contents. Decontamination and Decommissioning Operations. Louis B. Myers, and Max B. Berchtold.	This report describes the contents of 24 waste hoppers stored in JN-3, JN-1B (High Bay), and the Waste Storage Shed. Attachments include logbook entries of specific materials placed into certain hoppers.	1995. June.
P009	Chemistry Laboratory, Counting Room and Microprobe Room. Decontamination and Decommissioning Operations. Louis B. Myers, Max B. Berchtold, Paul A. Tomlin, and Michael P. Failey.	This report describes the operations and configuration of the Chemistry Laboratory, Counting Room, and Microprobe Room. Operations described include x-ray diffraction, gamma spec, alpha spec, gross alpha/beta, and isotopic analysis. Attachments include a 1996 update and photographs of these areas. Specific projects and dates are not identified.	1994. December.

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# ACCEPTABLE KNOWLEDGE INVENTORY Appendix A to Building JN-1 Hot Cell Laboratory Acceptable Knowledge Document

Ref No.	Title / Author	Summary	Date
P010	Evaporator Room JN-1A. Decontamination and Decommissioning Operations. Louis B. Myers, Max B. Berchtold, and Paul A. Tomlin.	This report describes the purpose and configuration of the Evaporator Room adjacent to the Charpy Room in JN-1A. Operations include an evaporator tank that produced sludge evaporated from liquids from the Radiological Analytical Laboratory, Hot Sump, hot drain sump (behind the Machine Shop), and drains from the CAA and OBD. Attachments include a 1996 update, health physics survey reports and data, and a certificate of analysis for TCLP of the sludge (1993).	1994. November.
P011	Controlled Access Area Storage Rooms JN-1A. Decontamination and Decommissioning Operations. Louis B. Myers, Max B. Berchtold, Paul D. Faust, and Paul A. Tomlin.	This report describes the contents of the two storage rooms located to the east of the Controlled Access Area. Attachments include a health physics survey report and data, and a hand-written report of the contents of the rooms.	1994. October.
P012	Controlled Access Area JN-1A. Decontamination and Decommissioning Operations. Louis B. Myers, Max B. Berchtold, Paul D. Faust, and Paul A. Tomlin. West Jefferson North historical files.	This report describes the Controlled Access Area (CAA) that provided support to the High Level Cell, Low Level Cell, and Mechanical Test Cell. Operations included staging and transferring materials between cells, service and repair manipulators, and performing special projects. The Sabotage Program performed in this area (1981-1983) is also described. Attachments include a 1996 update and health physics survey reports and data.	1994. September.
P013	Mezzanines JN-1A. Decontamination and Decommissioning Operations. Louis B. Myers, and Max B. Berchtold.	This report describes the mezzanine areas above the High Level Cell, Low Level Cell, and the Mechanical Test Cell. These areas contain the ventilation and filtering systems for these cells. Attachments include a 1996 update.	1994. September.
P014	Mechanical Test Cell JN-1A. Decontamination and Decommissioning Operations. Louis B. Myers, Carl A. Redd, Sr., and Max B. Berchtold.	This report describes the purpose and background of the Mechanical Test Cell (MTC) used primarily for tensile testing. Other analyses performed in this cell include creep, vacuum fusion, burst, radial burnup, density, and expanding mandrel testing. Attachments include a 1996 update, photographs, hand-written calculations of isotope content (Co-60 and Sb-125), and waste can volume.	1994. July.
P015	High Level Cell and Low Level Cell Hydraulic Doors and Hydraulic Door Room JN-1A. Decontamination and Decommissioning Operations. Louis B. Myers, and Max B. Berchtold.	This report describes the configuration and operation of the High Level and Low Level hydraulic doors, in addition the hydraulic fluid and contamination on the floor of the Hydraulic Room below the doors. Attachments include drawings and diagrams of the area and a health physics survey report and data.	1994. September.
P016	Subcells of the High Level and Low Level Cells in JN-1A. Decontamination and Decommissioning Operations. Louis B. Myers, Max B. Berchtold, and Paul A. Tomlin.	This report describes the subcells below the High Level and Low Level Cells. The subcells are used for storage of used HEPA filters and were used historically for creep testing. Attachments include a 1996 update, drawings and diagrams of the area and a health physics survey report and data, in addition to a drawing showing hydraulic oil seeping into the area from the Hydraulic Room.	1994. November.

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# ACCEPTABLE KNOWLEDGE INVENTORY

Ref No.	Title / Author	Summary	Date
P017	Low Level Cell JN-1A. Decontamination and Decommissioning Operations. Louis B. Myers, Max B. Berchtold, Thomas A. Beddick, Paul D. Faust, and Paul A. Tomlin.	This report describes the operations and configuration of the Low Level Cell (LLC) in JN-1A. Projects included fuel cutting, grinding, tensile tests of cobalt samples, and gamma scanning of waste containers filled in the HLC. Attachments include a 1996 update, logbook pages describing the packaging of approximately 100 waste (berry) cans (December 1988 to February 1989), logbook pages of a study verifying the contents of the containers (July 1991), and radioassay results. Limited information relating to projects and dates.	1994. August.
P018	High Level Cell JN-1A. Decontamination and Decommissioning Operations. Louis B. Myers, Max B. Berchtold, Paul D. Faust, and Paul A. Tomlin.	This report describes the operations and configuration of the High Level Cell (HLC) in JN-1A. Operations described include cutting irradiated fuel sections and defueling fuel rod sections. Attachments include a 1996 update, lists of materials and equipment stored in the cell, and a letter discussing material lost during the Sabotage Project. Limited information relating to projects and dates.	1994. August.
P019	Charpy Room JN-1A. Decontamination and Decommissioning Operations. Louis B. Myers, Carl A. Redd, Sr., Max B. Berchtold.	This report describes operations and materials stored in the Charpy Room in JN-1A. Operations described include sheer testing of irradiated specimens and cleaning of samples. Attachments include a 1996 update, drawings and diagrams of the area, and a health physics survey report and data. Limited information relating to projects and dates.	1994. June.
P020	Curie Content Determination and Package Classification of Low-Level Waste at Battelle's Hot Cell. Failey, Michael P.	This report describes the procedures and methods used to determine the curie content and classification of low-level waste packages generated at Battelle's hot cell, including the identification of gamma-emitting radioisotopes.	1986. June 11.
P021	Draft Report on the Characterization of Remote Handled TRU Waste Packages by Gamma-Ray Spectroscopy. Failey, Michael P., Ph.D.	Gamma-ray spectroscopy was used to measure the amount of gamma-ray emitting radioisotopes present in waste generated in the hot cells and combined with ratios obtained from a computer code to calculate the amount of TRU radionuclides present in the waste packages. Also included are a general hot cell facility description, year operations ceased, how the wastes were packaged during cleanup activities, instrumentation used to measure the radioactive contamination, and system calibration, validation, and verification.	1992. October 5.
P022	Appendices for the Intensive Audit Report for the Hot Cell. Author Unknown.	Applications for radioisotopic procurement and/or use for items transferred from the Research Reactor (JN-3) to JN-1. The document describes movement of capsules and other reactor materials from JN-3 to JN-1 during the 1967-1969 time frame.	1969. August.
P023	Course 7: Metals for Nuclear Power. Lesson Ten: Structural Materials. Metals Engineering Institute.	One of a series of correspondence course pamphlets relating to metals used for the nuclear power industry. Useful as a reference for the structural metals used in nuclear reactors especially zirconium.	Copyright 1958.
P024	Procedures Manual for Battelle's Radioisotope, Gamma, and Hot-Cell Laboratories.	Document describing Battelle's radioisotopes, gamma, and hot-cell laboratory operations and West Jefferson and King Avenue. Provides maps and floor plans of the facilities and brief descriptions of operations in JN-1 in 1965.	1965. November 24.

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Ref No.	Title / Author	Summary	Date
P025	Miscellaneous Materials Safety Data Sheets (MSDSs). Authored by Manufacturers.	Miscellaneous MSDS sheets collected from numerous sources collected during AK research at the West Jefferson site.	
P026	The U.S. Government and Battelle, Partners in Nuclear Research, 1943 – Present. Battelle Columbus Laboratories.	Document prepared primarily from thousands of questionnaires sent to Battelle Memorial Institute employees in 1985. The document describes the involvement of BMI in atomic research and development at several facilities. Includes a history of Battelle's involvement and the scope of operations performed in JN-1 and the amount of work performed for the DOE and other customers. The report does not specifically identify work performed for defense- related government work and focuses on special nuclear material projects.	1985.
P027	Battelle Columbus Laboratory Decommissioning Project. West Jefferson Category 1 and 3 Low-Level Waste Summary. Battelle Columbus Laboratory.	Summary of decommissioning project at West Jefferson including an historical overview and physical and radiological characterization of compactible and noncompactible waste from JN-1, JN-2, and JN-3. The report focuses on low-level waste and does not distinguish between wastes generated from JN-1, JN-2, and JN-3.	1997. Juiy 25.
P028	Shipping/Receiving Records for U.S. Air Force Project (Project Number G7656-3). Battelle Columbus Laboratory.	Radioactive Material Receipt documents receipt of reactor material from Sundance Air Force Station to JN-1. Demonstrates that defense related materials from the Air Force was introduced into JN-1 in 1967. The document gives no details relating to the scope of the project.	1967. October.
P029	Shipping/Receiving Records for U.S. Army Project. Battelle Columbus Laboratory.	Radioactive Shipment and Receipt form and other documents that show the receipt of reactor material from the Army MH-1A reactor to JN-1. Demonstrates that defense related materials from the Army were introduced into JN-1 in 1970.	1970. November/ December.
P030	Hot-Laboratory Facility and Summary of Capabilities. Battelle Memorial Institute.	Brochure describing capabilities of the JN-1 Hot Cell Laboratory published before 1975, because HEC capabilities are not included. Includes description of the facility, projects and methods used in the hot cells.	Date Unknown.
P031	Battelle Hot Cell Laboratory Brochure. Battelle Columbus Laboratory.	Brochure describing capabilities of the JN-1 Hot Cell Laboratory published after 1975, because HEC capabilities are described. Includes description of the facility, projects and methods used in the hot cells. Does not identify where the different methods are performed.	Date Unknown.
P032	Procedures Manual for Battelle's Radioisotope, Gamma, and Hot-Cell Laboratories. Sunderson, Duane N., and John E. Gates.	Document describing Battelle's radioisotopes, gamma, and hot-cell laboratory operations and West Jefferson and King Avenue. Provides maps and floor plans of the facilities and brief descriptions of operations in JN-1 in 1962 before the construction of the Alpha/Gamma cells. The procedure also describes a Waste Disposal Area that was never constructed in the basement of JN-1.	1962. February 20.
P033	Decontamination of Battelle Columbus' Plutonium Facility. Final Report to U.S. DOE Chicago. Rudolph, Ann, George Kirsch, and Harley L. Toy.	This report summarizes the decontamination and decommissioning of the Plutonium Laboratory. The report includes a historical description of the facility and operations. In addition, the report documents that all waste generated by the program was shipped off site.	1984. November 12.

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# ACCEPTABLE KNOWLEDGE INVENTORY

Ref No.	Title / Author	Summary	Date
P034	Finding of No Significant Impact and Environmental Assessment. Battelle Columbus Laboratories Decommissioning Project.	This FONSI includes the Environmental Assessment and describes the proposed actions and DOE responsibility for the D&D of 15 Battelle Columbus Laboratories. Provides brief descriptions of West Jefferson North operations.	1990. June.
	U.S. DOE Chicago Operations Office.	This was the suides a list and brief descriptions of nuclear	1085
P035	Representative Battelle-Columbus Projects with the Department of Energy, 1980–Present. Battelle Columbus Laboratories.	materials/process, fossil fuel, biomass/solar, and environmental studies performed for the DOE by Battelle from 1980 to 1985. The report does not identify the specific facilities supporting the projects.	
P036	Battelle-Columbus 40 Years of Energy Research for the U.S. Government. Battelle Columbus Laboratories.	This report provides an overview of operations and the historical missions of Battelle-Columbus. Provides a brief description of JN-1 operations and describes defense related Army, Navy, and Air Force Programs. General overview that does not identify the facility where the research was conducted.	1985.
P037	Decontamination Work Plan for Building JN-1 (partial document). Battelle Columbus Laboratories.	This portion of the work plan describes the hydraulic oil leak in the Hydraulic Room below the Low Level and High Level cells and provides an elevation drawing of the cell doors in these cells.	1990. November.
P038	Battelle Columbus Laboratory Decommissioning Project Baseline. Battelle Columbus Laboratories.	This portion of this document provides a brief history of Battelle Columbus Laboratories and a history of special nuclear material flow for the 15 buildings to be decontaminated and decommissioned. Information is extremely brief.	1992. November.
P039	SCS-300 Operating Manual. Bartlett Services, Inc.	This operating manual describes the operations of the SCS- 300 Sonatol cleaning system equipment. The manual also provides a summary of the theory of the systems operations.	1998.
P040	Waste Characterization, Classification and Shipping Support Technical Basis Document for the Battelle Columbus Laboratories Decommissioning Project (BCLDP) West Jefferson North Facility. Battelle Columbus Laboratories.	This report describes the methods used to identify and quantify the isotopes present in waste generated in JN-1. The report provides a radioisotope distribution based on smear samples taken from surface contamination in the Controlled Access Area (CAA). Since the CAA historically supported operations in all the cells, samples of contamination should be very representative of the general distribution of isotopes in JN-1 (see C004).	1998. May.
P041	Interim Guidance on Ensuring that Waste Qualifies for Disposal at the Waste Isolation Pilot Plant. U.S. DOE Carlsbad Area Office.	This interim guidance was developed to assist the TRU waste sites in establishing and demonstrating that only TRU waste generated by atomic energy defense activities is certified for disposal at WIPP.	1997. February 13.
U001	Miscellaneous Maps of Battelle Columbus West Jefferson Facility. Battelle Columbus Laboratories	Maps provide a layout of the West Jefferson site and the JN-1, JN-2, and JN-3 facilities. The map of JN-1 includes the HEC, pool, and High Bay.	Date Unknown.
U002	Description of the Battelle Hot Cell Laboratory. Battelle Columbus Laboratories.	Summary of operations in the JN-1 hot cells including the HEC, LLC, HLC, MTC, and alpha gamma cells, including a three-dimensional diagram of the cells. The source of the document is unknown and provides no historical operational information.	Date Unknown.

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# ACCEPTABLE KNOWLEDGE INVENTORY Appendix A to Building JN-1 Hot Cell Laboratory Acceptable Knowledge Document

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Ref No.	Title / Author	Summary	Date
U003	Battelle-Columbus Hot Cell Laboratory - Capability Summary. Battelle Columbus Laboratories.	Summary of JN-1 capabilities and operational history. The source of this document is unknown and must have been written after 1972, but before 1975, because it does not include a description of the HEC capabilities.	Date Unknown.
U004	Buildings JN-1, JN-2, and JN-3 Summaries. Battelle Columbus Laboratories.	Historical dates and brief history bullets for JN-1, JN-2, and JN-3. Dates are useful for creating a chronology of the history of operations at the site. Limited information relating to specific operations.	Date Unknown.
U005	DOE Contract Log.	Log book documents transfers of materials from Battelle relating to various defense-related Army and Air Force projects. There is no way to determine if any of these materials originated from JN-1, however the enriched uranium project for Babcock and Wilcox Company, Naval Nuclear Fuel Division was likely conducted in JN-1 based on conversations with Harley Toy.	1975 1987.
U006	ENG-92 Contract Projects Database 1 Printout.	Printout of the database created for the reference The U.S. Government and Battelle, Partners in Nuclear Research (P026). The printout provides a list of DOE, other government, and industrial contracts supported by Battelle from 1943 through 1985. Includes project number, date, funding, principal investigator, and scope. This list does not directly identify where the research project was conducted.	1985. Query Date May 20.
U007	Volume Estimates for Potential TRU Contents at WJN. Battelle Columbus Laboratories.	This table estimates the volume and type of potentially TRU waste contained in JN-1 cells, and other areas in JN-1 and JN-3. The volumes are rough estimates based only on visual observations and does not identify all materials that are included.	Date Unknown.
U008	Nuclear Fuel Inventory at West Jefferson North. Battelle Columbus Laboratories.	Inventory of nuclear fuel material located in JN-1 and JN-3, including cemented slugs of burn-up solution cemented in Alpha/Gamma Box 7. Information verifies that the cement slugs are primarily power reactor material.	Date Unknown.
U009	Miscellaneous JN-1 Waste Inventory Data. Battelle Columbus Laboratories.	This source consists of a number of inventory lists and Waste Package Loading Records describing the materials in drums and berry cans, including pool resins (with MSDS) and filters. Includes radiochemistry results for Tri-Nuc filters used to filter JN-1 pool water and an internal memo describing the method used to analyze the filters. This source contains information that can be used to describe the materials that may be contained in the TRU waste streams.	1988 – 1997.
U010	Hot Cell Receipts and Shipments Log Book. Battelle Columbus Laboratories.	This JN-1 logbook documents receipt and shipment of irradiated materials. The logbook identifies numerous shipments of highly enriched uranium (navy reactor) fuel (see C004) into and out of the Hot Cell Laboratory from 1960 through 1969.	3/21/60 – 5/14/73.
U011	Battelle Memorial Institute Laboratory Record Book of BMI Reports, No. 13561. Battelle Columbus Laboratories.	This Battelle Memorial Institute Logbook records the identification numbers assigned to BMI research reports, in addition to the author(s), report date, and distribution (1957-1960). The log documents that a majority of the reports and drafts were destroyed and unavailable. The log lists ongoing naval reactor research, however it does not identify where the research was conducted	June 1957 – December 1960.

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Ref No.	Title / Author	Summary	Date
U012	Battelle Memorial Institute Laboratory Record Book of BMI Reports, No. 18423. Battelle Columbus Laboratories.	This Battelle Memorial Institute Logbook records the identification numbers assigned to BMI research reports, in addition to the author(s), report date, and distribution (1959-1968). The log documents that a majority of the reports and drafts were destroyed and unavailable. The log lists ongoing naval reactor research, however it does not identify where the research was conducted.	July 1959 – June 1968.
U013	Battelle Memorial Institute Laboratory Record, Book of BMI Reports, No. 13117. Battelle Columbus Laboratories.	This Battelle Memorial Institute Logbook records the identification numbers assigned to BMI research reports, in addition to the author(s), report date, and distribution (1960-1965). The log documents that a majority of the reports and drafts were destroyed and unavailable. The log lists ongoing naval reactor research, however it does not identify where the research was conducted.	December 1960 – August 1965.
U014	ENG-92 Contract Projects Database 2 Printout. Battelle Columbus Laboratories.	Printout of the database created for the reference The U.S. Government and Battelle, Partners in Nuclear Research (P026). The printout provides a list of DOE, other government, and industrial contracts supported by Battelle from 1943 through 1985. Includes project number, dates, scope, and nuclear material type. Also includes group notes prepared by Harley Toy describing assumptions made for this database. Unlike U006 this reference identifies the building where the research was conducted and specifically identifies naval reactor research being conducted in JN-1 (see C004).	1986. Query Date October 17.
U015	Miscellaneous Documentation describing Radioanalytical Laboratory Operations. Battelle Columbus Laboratories.	Miscellaneous sources of information describing the mission and operations in the Radioanalytical Laboratory (RAL) in JN-2, including a list of procedures, pages from the RAL Administrative Operating procedure, and a RAL quality assurance discussion from the 1996 BCLDP Site Environmental Report.	1997 — 1998.
U016	Nuclear Materials Questionnaires. Battelle Columbus Laboratories.	These Nuclear Materials Questionnaires were distributed during research conducted for The U.S. Government and Battelle, Partners in Nuclear Research (P026) and identify defense related programs conducted in JN-1. This information was compiled in the database queried for reference U014. These forms document ongoing naval reactor research; however they provide very limited project information.	1985.
U017	Summary Report of Battelle 741 Reports. Battelle Columbus Laboratories.	Available reports were compiled by Kathy Hall (CEMP) and Craig Jensen during review of 741 shipping reports stored in classified files in DOE Chicago. Information includes shipment numbers, shipper, receiver, material type/description, date, weight, project number, and other miscellaneous information. The information in the reports was reviewed for classification and was cleared for uncontrolled distribution. Based on a review performed by George Kirsch, none of the material could be linked to operations in JN-1.	1998. March/June
U018	Battelle Columbus Laboratories Decommissioning Project (BCLDP); DCLDP Hot Cell Facility Presentation. Battelle Columbus Laboratories.	1995 presentation describing the BCLDP program associated with the JN-1 Hot Cell Laboratory. Provides a summary of the program history, JN-1 history and map and brief summaries and drawings of the cells.	1995. July.

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# ACCEPTABLE KNOWLEDGE INVENTORY

Appendix A to Building JN-1 Hot Cell Laboratory Acceptable Knowledge Document

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Ref No.	Title / Author	Summary	Date
U019	Miscellaneous JN-1 Construction Detail Reports. Battelle Columbus Laboratories.	Construction detail reports for the High Energy Cell, Hydraulic Door Room, Low Level Subcell, Mechanical Test Cell, Low Level Cell, High Level Cell, and the JN-1 Pool and Sumps. Most of this information is duplicated in references P002, P003, P014, P015, P016, P017, and P018. No operations or waste information is contained in these reports.	1997.
U020	DOE/Battelle Cost Share – Scope July, 1997. Battelle Columbus Laboratories.	This document describes the history and responsibilities of Battelle, NRC, and DOE for the D&D of Battelle Laboratories.	1997.
U021	TCLP metals data for leaded glass. Rocky Flats Environmental Technology Site.	Analytical data for TCLP leachable metals in glass windows. Demonstrates leaded glass leaches at regulated levels for lead.	1998. July.

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Attachment 4 Revised Appendix 4.10.2.1 (for distribution)

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Appendix 4.10.2.1

Compliance Methodology for RH-TRU Waste Form Battelle Columbus Laboratories (BCL) West Jefferson, OH
## 1.0 SUMMARY

The purpose of this appendix is to identify acceptable methods of preparation and characterization to qualify remote-handled transuranic (RH-TRU) waste, as defined by the U.S. Department of Energy (DOE) (Ref. 12.1), as payload for transport in the CNS 10-160B cask and to demonstrate that the RH-TRU waste forms at Battelle Columbus Laboratories (BCL), described in this appendix, comply with the payload requirements.

The payload parameters that are controlled in order to ensure safe transport of the RH-TRU waste in the CNS 10-160B cask are as follows:

- Restrictions on the physical and chemical form of RH-TRU waste.
- Restrictions on payload materials to ensure chemical compatibility among all constituents in a particular CNS 10-160B cask (including the parts of the cask that might be affected by the payload).
- Restrictions on the maximum pressure in the CNS 10-160B cask during a 60-day transport period. (As a conservative analysis, the maximum pressure calculations are performed for a period of one year. Attachment D discusses the transport period.)
- Restrictions on the amount of potentially flammable gases that might be present or generated in the payload during a 60-day transport period.
- Restrictions on the layers of confinement for RH-TRU waste materials in the waste containers packaged in the cask.
- Restrictions on the fissile material content for the cask.
- Restrictions on the hydrogen generation rates or the decay heat for the waste containers packaged in the cask.
- Restrictions on the weight for the waste containers and the loaded cask.

This appendix provides allowable methods to prepare payloads to meet these restrictions. The methods for determining or measuring each restricted parameter, the factors influencing the parameter values, and the methods used by BCL for demonstrating compliance, are provided in the following sections.

This appendix also includes the following as attachments:

- Content codes BC 312A, BC 314A, BC 321A, BC 321B, and BC 322A (Attachment A)
- Chemical Lists for the above mentioned content codes (Attachment A)
- Methods for Determining Gas Generation Rates and Decay Heat Values (Attachment B)
- Chemical compatibility analysis for the BCL content codes (Attachment C).
- Shipping Period for TRU Waste in the 10-160B Cask (Attachment D)

## 2.0 INTRODUCTION

## 2.1 Purpose

The purpose of this appendix is to describe the acceptable methods that shall be used to prepare and characterize the RH-TRU waste belonging to BCL prior to transport in the CNS 10-160B cask. It incorporates acceptable methods applicable to the content codes listed in Table 3-1 of this appendix. These methods will be expanded, as necessary, to incorporate additional waste content codes that may be identified in the future. Any additional content codes shall be submitted to the NRC for review and approval, with shipments under additional codes authorized only after NRC approval.

Section 2.2 lists the payload parameters that shall be determined for each payload. Section 3.0 describes the relationship between payload parameters and the classification of RH-TRU materials into CNS 10-160B cask payload content codes. Sections 4.0 through 11.0 discuss each payload parameter, the allowable method(s) for demonstrating compliance with the CNS 10-160B cask payload requirements, and the controls that are required for acceptable implementation of these method(s).

### 2.2 Payload Parameters

The payload parameters addressed in this document include:

- Physical form
- Chemical form and chemical properties
- Chemical compatibility
- Gas distribution and pressure buildup
- Payload container and contents configuration
- Isotopic characterization and fissile content
- Decay heat and hydrogen generation rates
- Weight.

## 3.0 TRU WASTE PAYLOAD FOR CNS 10-160B CASK

RH-TRU waste is classified into content codes, which give a description of the RH-TRU waste material in terms of processes generating the waste, the packaging methods used in the waste container(s), and the generating site. Content codes for the RH-TRU waste from BCL are provided in Attachment A and are listed in Table 3-1. For each content code, Attachment A provides a listing of all payload parameters, their corresponding limits and restrictions, and the methods used by BCL to meet these limits.

Table 3-1. BCL Content Codes			
Content Code	Waste Form Description		
BC 312A	Solidified Organic Waste (R&D operations)		
BC 314A	Cemented Inorganic Process Solids (R&D operations)		
BC 321A	Solid Organic Waste (D&D operations)		
BC 321B	Solid Organic Waste (Pool filters and resins)		
BC 322A	Solid Inorganic Waste (R&D operations)		

D&D = Decontamination and decommissioning.

R&D = Research and development.

The BCL has developed a formal TRU waste certification program that ensures the generation and packaging of waste under rigorous controls and documented procedures, in compliance with all governing regulations. In addition, complete documentation packages, along with quality assurance/quality control records, are generated for all payload containers. All TRU waste generated from the BCL will be packaged under a formal certification program. TRU waste generated from the BCL will comply with all transportation requirements using the following methods:

- Formally documented acceptable knowledge of the processes generating the waste
- Visual examination (VE), including audio/video surveillances of all packaging activities, conducted in accordance with approved procedures that ensure the absence of prohibited items and compliance with packaging requirements
- Data packages generated for all payload containers that document the contents and properties of the waste in the container
- Measurement of required parameters (weight, etc.) to ensure compliance with limits.

## 4.0 PHYSICAL FORM

### 4.1 Requirements

The physical form of waste comprising the CNS 10-160B cask payload is restricted to solid or solidified materials in secondary containers or activated reactor components. The total volume of residual liquid in a secondary container is restricted to less than 1% by volume. Secondary containers or components must be shored to prevent movement during accident conditions. Sharp or heavy objects in the waste shall be blocked, braced, or suitably packaged as necessary to provide puncture protection for the payload containers packaging these objects. Sealed containers greater than four liters in size are prohibited.

4.2 Methods of Compliance and Verification

All TRU waste from the BCLDP is newly packaged under procedures and plans that ensure compliance with transportation and other governing regulations. Pursuant to these procedures, compliance with the physical form requirements is ensured by documented acceptable knowledge (AK) and VE.

The BCL uses VE to verify the physical waste form descriptions documented as AK. As waste items are sorted, the BCL VE expert evaluates each waste item for consistency with the AK process description for the waste stream being packaged and determines and documents the physical form and description, and material type(s) and composition (percentage) of the item. The BCL AK expert independently reviews determinations made by the VE expert with respect to waste item assignments to waste streams as defined by the AK process descriptions during the AK confirmation process. AK discrepancy reports are generated with associated corrective actions, as necessary. The BCL also uses VE to ensure absence of prohibited items. As waste items are sorted, the BCL VE expert evaluates each waste item. As identified, any prohibited item is segregated for mitigation or other disposition and is not loaded into a waste container for shipment.

In addition to the generation of inventory loading records for each waste container, the VE documentation includes video/audio records. Video documentation of TRU waste packaging shall be performed at all times when TRU waste is being sorted and packaged under TC-OP-01.4 in the BCL

hot cells using two cameras and two videocassette recorders. A microphone feed is provided to verify and verbally note the identification of the waste stream and the container. This process duplicates the information recorded in hardcopy form on the waste container loading record. When TRU waste packages are in the cell, but packaging is not being performed, motion/light sensitive recording equipment shall be left running, with a videocassette in place, to document any movement in the packaging area. This will be used to verify that all packaging was recorded and that no packaging was performed without the proper VE documentation.

The BCLDP TRU WCP will use AK and VE to verify that the liquid content of the payload container complies with the requirements. Packaging personnel shall restrict the presence of free liquids to the extent that is reasonably achievable by pouring, pumping, or aspirating. Free liquids encountered during packaging shall be absorbed. Any liquid in nontransparent inner containers, including pumps or mechanical equipment that may contain an oil reservoir that is not solidified, will be handled by assuming that the container is filled with liquid and the volume will be added to the total liquid documented for the payload container in evaluating compliance with the 1% (volume) limit on free liquids.

BCLDP TRU WCP personnel shall ensure compliance with the requirement associated with sharp or heavy objects through visual examination at the time of packaging as described in TC-OP-01.4, Segregation and Packaging of TRU Waste. BCLDP packaging operations include the practice of size reduction and the use of a 0.015-inch thick steel liner in the 55-gallon drums. Following size reduction, items with the potential to puncture the liner and drum are blocked, braced, or suitably packaged to ensure container integrity.

BCLDP TRU WCP personnel shall ensure compliance with the requirement associated with sealed containers through visual examination at the time of packaging as described in TC-OP-01.4, Segregation and Packaging of TRU Waste. Sealed containers greater than 4 liters identified during the sorting process will be segregated for disposition and shall not be packaged for shipment.

As described in TC-AP-01.1, TRU Waste Data Package Generation, compliance with each of the restrictions on physical form shall be recorded in the payload container data package.

## 5.0 CHEMICAL FORM AND CHEMICAL PROPERTIES

### 5.1 Requirements

The chemical properties of the waste are determined by the chemical constituents allowed in a given content code. Specific requirements regarding the chemical form of the waste are as follows:

- Explosives, nonradioactive pyrophorics, compressed gases, and corrosives are prohibited.
- Pyrophoric radionuclides may be present only in residual amounts less than 1 weight percent.
- The total amount of potentially flammable volatile organic compounds (VOCs) present in the headspace of a secondary container is restricted to 500 parts per million (ppm).

5.2 Methods of Compliance and Verification

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Compliance with chemical form and chemical property restrictions is demonstrated through process knowledge or sampling programs, if required.

#### 5.2.1 Pyrophoric Materials

Nonradioactive pyrophoric materials (e.g., organic peroxides, sodium metal, and chlorides) shall be segregated and not be packaged into payload containers. Radioactive pyrophoric material (e.g., metallic plutonium and americium), if present in the waste stream, shall be limited to less than 1 percent (weight) of the payload container. In accordance with TC-OP-01.4, Segregation and Packaging of TRU Waste, qualified BCLDP TRU WCP personnel shall use AK information in conjunction with VE, as described in Section 4.2, during waste generation and packaging to verify the absence of nonradioactive pyrophoric materials and compliance with the restriction on radioactive pyrophoric material (e.g., according to records of waste generation processes, nonradioactive pyrophoric materials have not been used). As described in TC-AP-01.1, TRU Waste Data Package Generation, the absence of nonradioactive pyrophorics and compliance with the restriction on radioactive pyrophoric material shall be recorded in the payload container data package. Any nonradioactive pyrophorics encountered during examination shall be segregated and shall not be shipped.

#### 5.2.2 Explosives, Corrosives, and Compressed Gases

In accordance with TC-OP-01.4, Segregation and Packaging of TRU Waste, qualified BCLDP TRU WCP personnel shall use AK information in conjunction with VE, as described in Section 4.2, during waste generation and packaging to verify the absence of explosives, corrosives, and compressed gases. Any unvented compressed gas canisters (including aerosol cans) identified during the packaging of wastes shall be segregated as described in TC-OP-01.4, Segregation and Packaging of TRU Waste. Acids and bases, if found, shall be neutralized. The absence of explosives, unvented compressed gas canisters, and corrosives shall be documented in the payload container data packages by BCLDP TRU WCP personnel as described in TC-AP-01.1, TRU Waste Data Package Generation.

#### 5.2.3 Flammable VOCs

All TRU waste from the BCL is from research and development or decontamination and decommissioning related activities and will be packaged with the generation of complete data packages. The BCL wastes are not expected to have flammable VOCs based on the content codes, the waste packaging process (sorted and repackaged into drums as individual items, which minimizes the introduction of potentially flammable VOCs into the drums), and the lack of a source for potentially flammable VOCs.

#### 6.0 CHEMICAL COMPATIBILITY

### 6.1 Requirements

Each content code has an associated chemical list (Attachment A) based on AK information. Chemical constituents in a payload container assigned to a given content code shall conform to these approved chemical lists. Chemicals/materials that are not listed are allowed in trace amounts (quantities less than 1 percent [weight]) in a payload container provided that the total quantity of trace chemicals/materials is restricted to less than 5 percent (weight).

Chemical compatibility of a waste with its packaging ensures that chemical reactions will not occur that might pose a threat to the safe transport of a payload in the CNS 10-160B cask.

## 6.2 Methods of Compliance and Verification

Chemical compatibility analyses for all authorized payloads were performed according to a U.S. Environmental Protection Agency method (Ref. 12.2). Chemical compatibility for the content code chemical lists is ensured by these analyses. The results of these chemical compatibility analyses concluded that the chemicals/materials in each content code and between content codes are chemically compatible. The detailed evaluation of each content code is described in Attachment C. Qualified BCLDP TRU WCP personnel shall document the presence of any chemicals identified during the waste characterization process in the payload container data packages. TC-OP-01.4, Segregation and Packaging of TRU Waste, includes instructions for comparing chemicals noted in the payload container data packages against the chemicals listed in the appropriate content code to ensure the contents of payload containers are compatible.

## 7.0 GAS DISTRIBUTION AND PRESSURE BUILDUP

## 7.1 Requirements

Gas distribution and pressure buildup during transport of TRU waste in the CNS 10-160B cask payload are restricted to the following limits:

- The gases generated in the payload must be controlled to prevent the occurrence of potentially flammable concentrations of gases within the payload confinement layers and the void volume of the inner vessel (IV) cavity. Specifically, hydrogen concentrations within the payload confinement layers are limited to 5 percent by volume during a maximum 60-day shipping period (twice the expected shipping time of 30 days, see Attachment D).
- The gases generated in the payload and released into the IV cavity must be controlled to maintain the pressure within the IV cavity below the acceptable packaging design limit of 31.2 pounds per square inch gauge (psig).

### 7.2 Methods of Compliance and Verification

The primary mechanism for gas generation during TRU waste transportation in the CNS 10-160B cask is by radiolysis of the waste materials. Gas generation from other mechanisms such as chemical, thermal, or biological activity is expected to be insignificant for the TRU waste payload. As discussed in Section 6.0, the chemicals and materials in the TRU waste are compatible and inert, and the restrictions of the materials that can be present in each content code precludes the occurrence of chemical reactions that can produce excessive gas. Gas generation from biological activity is expected to be insignificant given the transportation time, the nature of the waste (solid or solidified), and the environment of the payload (lack of nutrients, lack of water content, etc.). The temperatures of the payload, given the decay heat limits applicable, are expected to be below the normal usage range for the payload materials, resulting in very little potential for gas generation due to thermal decomposition.

Compliance with the CNS 10-160B cask design pressure limit for each content code is analyzed by assuming that all gases generated are released into the IV cavity and by including the contributions from thermal expansion of gases and vapor pressure of atmospheric water.

Table 7-1. Maximum Pressure Increase Over 365-Day Shipping Period						
Content Code	G _{eff} (RT) ^a	Void Volume (Liters)	Activation Energy (kcal/mole)	Decay Heat Limit (Watts)	G _{eff} ^b	P _{max} ^c (psig)
BC 312Ad						
BC 314A	0.72	1938	0	7.38	0.72	12.09
BC 321A	8.4	1938	2.1	1.54	14.71	24.43
BC 321B	8.4/2.1e	1938	2.1	2.69	4.7	17.32
BC 322A	0.024	1938	0	221	0.024	12.08

Table 7-1 shows that the pressure increase during a period of 365 days is below the design pressure limit of 31.2 psig for all the BCL content codes.

^aG value for net gas (molecules per 100 eV) at room temperature (70°F).

^bEffective G value (molecules per 100 eV) at maximum operating temperature of 168°F calculated using the Arrhenius equation for which activation energy is an input.

^cMaximum pressure.

^dThis code consists of solidified organics; compliance with pressure limits will be shown by testing. ^eBC 321B reports 12% cellulosics and 80% resins (remainder being inorganic material) and is reflected in the calculation of the temperature-corrected G_{eff}.

Compliance with the restrictions on flammable gas concentration is discussed in Section 10.0.

## 8.0 PAYLOAD CONTAINER AND CONTENTS CONFIGURATION

### 8.1 Requirements

Fifty-five-gallon drums are authorized payload containers in a CNS 10-160B cask. Up to ten 55-gallon drums of RH-TRU waste may be packaged in the cask. Each 55-gallon drum to be packaged in the CNS cask must have a minimum of one filter vent. The minimum filter vent specifications for the 55-gallon drums and drum liners used to package waste inside the drums are provided in Table 8-1.

The test methods used to determine the compliance of filter vents with the performance-based requirements of flow rate, efficiency, and hydrogen diffusivity shall be directed by procedures under a quality assurance program.

Filter vents shall be legibly marked to ensure both (1) identification of the supplier and (2) date of manufacture, lot number, or unique serial number.

	Table 8-1.	Minimum Filter Ven	t Specifications	
Container Type	Number of Vents Required per Container	Flow Rate (ml/min of air, STP, at 1 inch of water) ^a	Efficiency (percent)	Hydrogen Diffusivity (mol/s/mol fraction at 25°C)
Drum	1	35	99.5	3.70E-6
Drum Liner Filter	1	35	NAb	3.70E-6

^a Filters tested at a different pressure gradient shall have a proportional flow rate (e.g., 35 ml/min at 1 inch of water = 1 L/min at 1 psi).

b Filters installed in containers that are overpacked are exempt from the efficiency requirement as the drum must exhibit a ≥99.5 percent efficiency.

NA = Not applicable

### 8.2 Methods of Compliance and Verification

Procured filter vents at BCL shall be inspected as directed by QD-AP-04.1, Documentation and Control of Purchased Items and Services, to verify compliance with the applicable filter vent specifications specified in the purchase requisition (i.e., visual inspection of certificate of conformance serial numbers to actual filter vents and inspection of filters for physical damage). Under WA-OP-006, Procurement and Inspection of Packagings for Hazardous Materials Shipments, payload containers and liners, if present, shall be visually inspected to ensure that they have been fitted with the required number of filter vents as specified above. Nonconforming filter vents shall be segregated in accordance with QD-AP-15.1, Nonconformance Reporting for Activities, Items and Materials. As described in TC-OP-01.4, Segregation and Packaging of TRU Waste, qualified BCLDP TRU WCP personnel also shall visually inspect payload containers during packaging to ensure that each has been fitted with the correct type and number of filter vents.

Prior to transport, payload container filter vents shall be visually inspected by the Transportation Certification Official (TCO) for damage or defect. If a defect is identified, a nonconformance report shall be issued in accordance with QD-AP-15.1, Nonconformance Reporting for Activities, Items and Materials, and the payload container shall be returned for repackaging or overpacking prior to certification.

## 9.0 ISOTOPIC CHARACTERIZATION AND FISSILE CONTENT

### 9.1 Requirements

The CNS 10-160B cask payload allows fissile materials, provided the mass limits of Title 10, Code of Federal Regulations, Section 71.53 are not exceeded. Plutonium content cannot exceed 0.74 Bq (20 curies) per cask.

### 9.2 Methods of Compliance and Verification

BCLDP TRU WCP personnel will calculate the fissile or fissionable radionuclide content of the payload container as Pu-239 (plutonium-239) fissile gram equivalents (FGE) and as plutonium curies as described in DD-98-04, Waste Characterization, Classification and Shipping Support Technical Basis Document, and TC-AP-01.2, Calculations Using Radioassay Data. These calculations are based on the waste generation source and configuration, which establishes the initial radionuclide compositions based on location and initial use. As described in DD-98-04, Waste Characterization, Classification and Shipping Support Technical Basis Document, assay of samples and dose rate measurements, along with the appropriate isotopic composition, are used to determine the isotopic inventory. The TCO shall evaluate the compliance of the total FGE value and the plutonium curies of payload containers with the maximum limits.

It should be noted that BCLDP accountability records indicate no more than approximately 50 grams of fissile material is dispersed throughout the BCL West Jefferson North facility in low isotopic enrichments (Reference 12.3). Therefore, the drum loading of fissile material will be much lower.

### 10.0 DECAY HEAT AND HYDROGEN GAS GENERATION RATES

This section describes the logic and methodology used in evaluating payload characteristics that meet the hydrogen gas concentration requirement for each of the RH-TRU content codes for the BCL RH-TRU wastes described in this section.

### 10.1 Requirements

The hydrogen gas concentration shall not exceed 5% by volume in all void volumes within the CNS 10-160B cask payload during transport up to twice the expecting shipping time of 30 days. A CNS 10-160B Cask payload must be assembled of payload containers belonging to the same or equivalent content code. Payload containers of different content codes with different bounding G values and resistances may be assembled together as a payload, provided the decay heat limit and hydrogen gas generation rate limit for all payload containers within the payload is conservatively assumed to be the same as that of the payload container with the lowest decay heat limit and hydrogen gas generation rate limit.

### 10.2 Methodology of Ensuring Compliance with Flammable Gas Concentration Limits

As stated in Section 7.2, chemical, biological, and thermal gas generation mechanisms are insignificant in the CNS 10-160B cask. In addition, as shown in Section 5.1, potentially flammable VOCs are restricted to 500 ppm in the headspace of the CNS 10-160B cask secondary containers. Therefore, the only flammable gas of concern for transportation purposes is hydrogen. The concentration of hydrogen within any void volume in a layer of confinement of the payload or in the cask IV has been evaluated during a 60-day shipping period (i.e., twice the expected shipping duration, see Attachment D).

Attachment A provides the RH-TRU waste content codes for the BCL RH-TRU wastes that are included in the authorized payload for the CNS 10-160B cask. Each content code has a unique and completely defined packaging configuration. Modeling the movement of hydrogen from the waste material to the payload voids, using the release rates of hydrogen through the various confinement layers, defines the relationship between generation rate and void concentration. This modeling allows determination of the maximum allowable hydrogen generation rate for a given content code to meet the 5% concentration

limit, as detailed in Section 10.3. Based on hydrogen gas generation potential, quantified by hydrogen gas generation G values, the gas concentration limit can be converted to a decay heat limit, as detailed in Section 10.4. The maximum allowable hydrogen generation rates and decay heat limits for the RH-TRU content codes for BCL wastes are listed in Table 10-1.

Table 10-1. Maximum Allowable Hydrogen Gas Generation Rates, Decay Heat Limits, and Total Activity Limits					
Content Code	Maximum Allowable Hydrogen Gas Generation Rate, mole/second/drum	Maximum Allowable Hydrogen Gas Generation Rate, moles/second/cask	Maximum Allowable Decay Heat Limit, Watts/Drum	Maximum Allowable Decay Heat Limit, Watts/Cask	Maximum Total Activity Limit Curies/Cask ^a
BC 312A	3.5082E-8	3.5082E-7	_b	_b	_b 
BC 314A	3.5082E-8	3.5082E-7	0.738	7.38	1.45E+3
BC 321A	4.093E-8	4.093E-7	0.154	1.54	3.01E+2
BC 321B	4.093E-8	4.093E-7	0.269	2.69	5.76E+2
BC 322A	3.5082E-8	3.5082E-7	22.1	221	4.34E+4

^aOther limits applicable to the cask (not related to gas generation) shall also be met. ^bNo decay heat limit or activity limit due to unknown G value.

Parameters that govern the maximum allowable hydrogen generation rates and maximum allowable decay heat limits are listed below:

- Waste packaging configuration (i.e., the number and type of confinement layers).
- Release rates of hydrogen from each of these confinement layers.
- Void volume in the IV available for gas accumulation.
- Operating temperature and pressure for the payload in the CNS cask IV during the maximum shipping period.
- Duration of the shipping period (see Attachment D).
- Fraction of the gamma energy absorbed by waste materials that could potentially generate hydrogen.
- Hydrogen generation rates quantified by the G value of a waste material (the number of molecules of hydrogen produced per 100 eV of energy absorbed).
  - 10.3 Determination of Maximum Allowable Hydrogen Generation Rates for Content Codes

The modeling for determination of the maximum allowable generation rates is described below.

10.3.1 Input Parameters

The model parameters that must be quantified include the following:

**Waste Packaging Configuration and Release Rates**: Each content code has a unique packaging configuration that is completely defined. The waste described by content codes BC 312A, BC 314A, BC 321A, and BC 322A will be placed directly into a 55-gallon drum lined with a steel liner. The waste described by content code BC 321B will also be placed directly into a 55-gallon drum that may be lined with a steel liner or a polyethylene liner. Ten drums will then be placed into the CNS 10-160B cask. Release rates of hydrogen through the drum filters and drum liner filters have been quantified, and are summarized in Table 10-2. (Note that, if used, the polyethylene liner is punctured with a 1-inch diameter hole. The release rate associated with the puncture hole is conservatively assumed to be the same as that of the steel liner filter.) These are based on release rates obtained for filters (Ref. 12.4) at room temperatures. The release rates used in the calculations are the minimum measured values in each case. The release rates in Table 10-2 are shown for two different temperatures. The temperature dependence of these release rates is discussed later in this section.

**Void Volumes in Confinement Layers**: The void volumes in the confinement layers are content code specific. The void volumes for the layers in the different content codes are based on waste generation processes and the specific contents within the content codes. This section summarizes the void volumes within the confinement layers for the various content codes. In all cases, a conservative (i.e., minimal) void volume has been used.

Table 10-2. Release Rates of Hydrogen						
	Release Rate					
Content Code	Filter Type	(mol/sec/mol fraction)				
		T = 233K	T = 348.6K			
BC 312A	Drum Liner Filter	2.46 x 10-6	4.98 x 10-6			
	Drum Filter	2.46 x 10-6	4.98 x 10-6			
BC 314A	Drum Liner Filter	2.46 x 10-6	4.98 x 10-6			
	Drum Filter	2.46 x 10-6	4.98 x 10 ⁻⁶			
BC 321A	Drum Liner Filter	2.46 x 10-6	4.98 x 10-6			
	Drum Filter	2.46 x 10-6	4.98 x 10-6			
BC 321B	Drum Liner Filter	2.46 x 10-6	4.98 x 10-6			
	Drum Filter	2.46 x 10 ⁻⁶	4.98 x 10-6			
BC 322A	Drum Liner Filter	2.46 x 10-6	4.98 x 10-6			
	Drum Filter	2.46 x 10-6	4.98 x 10-6			

Cask Void Volume: The cask will have a payload of 10 drums and a drum carriage. The interior volume of the cask,  $V_{cask}$ , is 4438 liters. The volume occupied by the drum carriage,  $V_{carriage}$ , is 143.2 liters. The external volume of a single drum,  $V_{drum}$ , is 235.7 liters. The void volume within the cask is calculated as:

 $V_{V,cask} = V_{cask} - V_{carriage} - 10 V_{drum}$ 

*V*_{V.cask} = 4438 liters - 143.2 liters - 10 (235.7 liters)

 $V_{V,cask} = 1938$  liters

Drum Headspace Void Volume: The internal height of the cylindrical 55-gallon drum is 33.25 inches (in.) and the inside diameter is 22.5 in. The internal volume of the drum is thus calculated as:

$$V_{drum,Internal} = \pi r^{2}_{drum,Internal} h_{drum,Internal}$$

$$V_{drum,Internal} = \pi \left(\frac{22.5 in}{2} \times 2.54 cm / in\right)^{2} (33.25 in \times 2.54 cm / in)$$

$$V_{drum,Internal} = 216,644 cm^{3} = 216.6 \ liters$$

The external height of the cylindrical drum liner is 32.250 in., and the outside diameter is 19.500 in. The external volume of the drum liner is calculated as:

 $V_{liner,external} = \pi r^2_{liner,external} h_{liner,external}$ 

$$V_{liner,external} = \pi \left(\frac{19.5 \text{ in}}{2} \times 2.54 \text{ cm/in}\right)^2 (32.25 \text{ in } \times 2.54 \text{ cm/in})$$

 $V_{liner,external} = 157,830 \text{ cm}^3 = 157.8 \text{ liters}$ 

The void volume within the drum headspace,  $V_{V,hs}$ , is calculated as:

$$V_{V,hs} = V_{drum,Internal} - V_{liner,External}$$

 $V_{V hs} = 216.6 \, liters - 157.8 \, liters$ 

 $V_{V,hs} = 58.8$  liters

Drum Liner Void Volume:

#### Content Codes BC 321A and BC 321B

These content codes are comprised of combustible, debris type of waste materials. The drum liner thickness is 0.105 in. The external height of the cylindrical drum liner is 32.250 in., and the outside diameter is 19.500 in. The internal dimensions of the rigid liner are thus a height of 32 in. and a diameter of 19.3 in. The internal volume of the drum liner is calculated as:

$$V_{liner,Internal} = \pi r^2_{liner,Internal} h_{liner,Internal}$$

$$V_{liner,lnternal} = \pi \left(\frac{19.3 \, in}{2} \, x \, 2.54 \, cm/\ln\right)^2 (32 \, in \, x \, 2.54 \, cm/\ln)$$

 $V_{liner, Internal} = 153,400 \text{ cm}^3 = 153.4 \text{ liters}$ 

For content code BC 321A, BCL waste drum data (waste masses and volumes) for the waste result in a range of bulk densities of the waste material of 0.30 g/cm3 to 0.55 g/cm³. The bulk density of the waste material,  $\rho_{\text{bulk}}$ , is conservatively set to the maximum bulk density of this range, 0.55 g/cm³. The lowest solid material density,  $\rho_{\text{solid}}$ , for the dominant material in BC 321A, cellulosics, is 1.20 g/cm³ from Reference 12.5. Thus the minimum void ratio,  $\varepsilon$ , is calculated as:

$$\varepsilon = 1 - \frac{\rho_{bulk}}{\rho_{solid}}$$
$$\varepsilon = 1 - \frac{0.55}{1.20}$$

 $\epsilon = 0.542 = 54.2\%$ 

Based on the range of densities of 0.09 g/cm³ to 0.36 g/cm³ for residential, commercial, and combustible mixed construction debris from Reference 12.5, the conservatively assumed maximum bulk density of the waste material,  $\rho_{\text{bulk}}$ , in content code BC 321B, is 0.36 g/cm³. The solid material density,  $\rho_{\text{solid}}$ , for the dominant material in BC 321B, ion exchange resins, is 0.70 g/cm³ from Reference 12.5. As above, the minimum void ratio for BC 321B is calculated as:

$$\varepsilon = 1 - \frac{0.36}{0.70}$$

 $\epsilon = 0.486 = 48.6\%$ 

To verify that the assumed bulk density is conservative, bulk densities of the waste material were calculated from BCL representative waste drum data (waste masses and volumes) and compared to the assumed bulk density value. The calculated bulk density for the BCL drums of content code BC 321B ranged from 0.19 g/cm³ to 0.31 g/cm³. The calculated void ratio using this maximum bulk density and the solid material density of 0.70 g/cm³ is:

$$\varepsilon = 1 - \frac{0.31}{0.70}$$

 $\epsilon = 0.557 = 55.7\%$ 

To ensure that the total flammable gas concentration remains below 5% of the free gas volume in any confined region of the packaging during the shipping period, the smallest void ratio calculated above (48.6%) is used for both BC 321A and BC 321B content codes to conservatively calculate the minimum void volume within the rigid liner.

Thus, the minimum void volume within the rigid liner, VV.liner, is calculated as:

 $V_{V,liner} = \gamma V_{liner, Internal}$ 

 $V_{V \ liner} = 0.486 \ (153.4 \ liters)$ 

 $V_{V liner} = 73.6 liters$ 

## Content Codes BC 312, BC 314, and BC 322

A conservative void volume of 1 liter within the drum liner is used in the calculations of maximum allowable hydrogen gas generation rates for these content codes.

**<u>Pressure</u>**: The pressure is assumed to be isobaric and equal to one atmosphere. The mole fraction of hydrogen in each void volume would be smaller if pressurization is considered and would result in a greater maximum allowable hydrogen gas generation rate. Furthermore, the amount of hydrogen gas generated during a sixty day shipping period would be negligible compared to the quantity of air initially present at the time of sealing the CNS 10-160B cask.

<u>Temperature</u>: The input parameter affected by temperature is the release rate through the different confinement layers in the payload containers and the G values for hydrogen. For the RH-TRU waste content codes, these are the filters in the inner containers. These release rates increase with increasing temperature (Ref. 12.6). Therefore, the minimum release rates would be those at the lowest operating temperature. These are the release rates indicated in Table 10-2 for 233K. The minimum decay heat limits are determined by the ratio of the release rates and the G values. In other words, the higher the release rates, the higher the decay heat limit; the higher the G value, the lower the decay heat limit. The dependence of G values on temperature is documented in Section 10.4. For determining the decay heat limit, the temperature that yielded the minimum decay heat limit for each content code was used as the input parameter.

In summary, the temperature dependence of the input parameters was accounted for in the calculation so that, in each case, the minimum possible limit (hydrogen generation rate or decay heat limit) was obtained. This provides an additional margin of safety in the analysis for each content code.

These are the important input parameters for determining the maximum allowable hydrogen generation limits. Other assumptions used in the mathematical analysis are included in Section 10.3.2.

10.3.2 Mathematical Analysis For Determining the Maximum Allowable Hydrogen Gas Generation Rates

The maximum allowable gas generation rate for each content code was calculated using numerical solutions to differential equations, which describe the unsteady-state (i.e., transient) mass balances on

hydrogen within each confinement volume of the CNS 10-160B cask. The hydrogen generation rate which will yield 5 volume percent within the innermost layer of confinement is not known a priori and is calculated using an iterative scheme which is described below.

The assumptions that have been made in deriving the governing equations are as follows:

- Hydrogen is an ideal gas and the ideal gas law applies.
- The hydrogen is assumed to be nonreactive with any materials in the payload container.
- Hydrogen gas generation rates are not reduced by depletion of the waste matrix.
- The concentration of hydrogen within each of the layers of confinement prior to transport in the CNS 10-160B cask is assumed to be at steady-state (generation rate equals release rate). This assumption provides an additional margin of safety since the concentrations of hydrogen are maximum at steady-state conditions.

The following list defines the variables which are used in the description of the mathematical framework.

<b>X</b> ₁	-	Mole fraction hydrogen within the innermost confinement volume.
x _i	=	Mole fraction hydrogen in the confinement volume "i".
R _{df}	=	The effective release rate of hydrogen across the filter on a drum (L/day).
R _{lh}	=	The effective release rate of hydrogen across the drum liner filter (L/day).
vi	=	The void volume within confinement layer "i" (L).
t	=	Time (days).
R	=	The gas law constant (0.08206 atm L mol ⁻¹ K ⁻¹ ).
Т	=	Absolute temperature (K).
Р	=	Absolute pressure (atm).
CG	=	The hydrogen gas generation rate per innermost confinement layer in one drum (mol/sec).

For brevity in subsequent discussions, a parameter C₁ will be defined as:

 $C_1 = CG \times R \times T / (P \times V_1)$ 

The evaluation of maximum allowable hydrogen gas generation rates was performed through iterative calculations until the appropriate hydrogen gas generation rate per drum yielded a concentration of 0.05 mole fraction within the innermost layer of confinement (i.e., within the drum liner void) at the end of the 60-day shipping period.

The generation of hydrogen within the innermost layer, release across confinement layers and accumulation within the confinement volumes during transport were simulated by numerically solving the system of hydrogen mass balance differential equations for each void volume. The applicable systems of differential equations are listed below. The derivation of these equations follows the system of equations.

#### SYSTEM OF DIFFERENTIAL EQUATIONS

#### EQUATION

#### VOID VOLUME

[drum headspace]

 $dX_1/dt = C_1 - R_{lh} (X_1 - X_2) / V_1$ 

 $dX_2/dt = R_{lh} (X_1 - X_2)/V_2 - R_{df} (X_2 - X_3) / V_2$ 

 $dX_3/dt = 10R_{df}(X_2 - X_3) / V_3$ 

[CNS 10-160B cask IV]

[drum liner]

Derivation of the System of Differential Equations

Mass Balance for the Drum Liner Void Volume

$$\frac{dn_1}{dt} = CG - k_{lh}(P_1 - P_2)$$

where,

 $n_1$  = Moles of hydrogen inside drum liner void volume (mol)

 $k_{lh}$  = Effective release rate of hydrogen across the drum liner filter (mol/day⁻¹ atm H₂⁻¹)

 $P_1$  = Partial pressure hydrogen inside drum liner void volume (atm H₂)

 $P_2$  = Partial pressure hydrogen inside drum headspace void volume (atm H₂).

Applying the ideal gas law and assuming isobaric conditions such that P is constant total system pressure, yields:

$$\frac{PV_1}{RT}\frac{dX_1}{dt} = CG - k_{lh}P(X_1 - X_2).$$

Rearranging terms and defining R_{lh} as k_{lh}RT yields the first equation:

$$\frac{dX_1}{dt} = C_1 - \frac{R_{lh}(X_1 - X_2)}{V_1}.$$

#### Mass Balance for the Drum Headspace Void Volume

$$\frac{dn_2}{dt} = k_{lh} (P_1 - P_2) - k_{df} (P_2 - P_3)$$

where,

n2	=	Moles of hydrogen inside canister void volume (mol)
kdf	=	Effective release rate of hydrogen across the filter on the drum (mol/day ⁻¹ atm $H_2^{-1}$ )
P3	=	Partial pressure hydrogen inside CNS 10-160B cask IV void volume (atm H ₂ )

and the other variables are as defined earlier.

Applying the ideal gas law and assuming isobaric conditions such that P is constant total system pressure, yields:

$$\frac{PV_2}{RT}\frac{dX_2}{dt} = k_{lh} P(X_1 - X_2) - k_{df} P(X_2 - X_3).$$

Rearranging terms and defining Rdf as kdfRT yields the second equation:

$$\frac{dX_2}{dt} = \frac{R_{lh}(X_1 - X_2)}{V_2} - \frac{R_{df}(X_2 - X_3)}{V_2}.$$

Mass Balance for the CNS 10-160B Cask Void Volume With Ten Drums

$$\frac{dn_3}{dt} = 10 k_{df} \left( P_2 - P_3 \right)$$

where,

 $n_3 =$  Moles of hydrogen inside CNS 10-160B cask IV void volume (mol).

Applying the ideal gas law and assuming isobaric conditions

$$\frac{PV_{3}}{RT}\frac{dX_{3}}{dt} = 10 \ k_{df} \ P(X_{2} - X_{3}).$$

Rearranging terms yields the third equation:

$$\frac{dX_3}{dt} = \frac{10 R_{df} (X_2 - X_3)}{V_3}$$

### **Derivation of the Steady-State Concentrations in Confinement Layers**

Prior to transport in the CNS 10-160B cask the concentration of hydrogen within each of the layers of confinement is assumed to be at steady-state. At steady-state the release rates of hydrogen across each layer are equal to the hydrogen gas generation rate. The steady-state concentrations of hydrogen within each volume of confinement were evaluated from the relations below.

At steady-state, the concentration of hydrogen outside the drum is zero (i.e.,  $X_3 = 0$ ) and there is no accumulation of hydrogen inside the drum, thus the steady-state hydrogen mass balance for the drum liner void volume becomes:

$$\frac{dX_1}{dt} = 0 = C_1 - \frac{R_{lh}(X_1 - X_2)}{V_1}.$$

For the drum headspace, the steady-state hydrogen mass balance becomes:

$$\frac{dX_2}{dt} = 0 = \frac{R_{lh}(X_1 - X_2)}{V_2} - \frac{R_{df}X_2}{V_2}.$$

Rearranging the terms in the mass balance equation for hydrogen at steady-state inside the drum liner,

$$C_1 V_1 = R_{lh} (X_1 - X_2).$$

Substituting this relation in the steady-state equation for the hydrogen mass balance inside the drum headspace and rearranging terms

$$\frac{R_{lh}(X_1 - X_2)}{V_2} = \frac{C_1 V_1}{V_2} = \frac{R_{df} X_2}{V_2}.$$

Simplifying the previous equation and solving for the steady-state concentration in the drum headspace, X₂, yields

$$X_2 = \frac{C_1 V_1}{R_{df}}.$$

Rearranging the relation for the steady-state hydrogen mass balance inside the drum liner yields:

$$C_1 V_1 + R_{lh} X_2 = R_{lh} X_{l}$$

Rearranging terms in the previous equation and solving for the steady-state hydrogen concentration inside the drum liner,  $X_1$ , yields

$$X_1 = \frac{C_1 V_1}{R_{lh}} + X_2.$$

The steady-state concentrations were then used to define the initial state of the system (i.e., hydrogen mole fractions within each confinement volume) at the time the CNS 10-160B cask is sealed for transport with a payload of 10 drums.

These systems of differential equations were solved numerically using the Runge-Kutta Fourth Order numerical integration method. Numerical solution implies obtaining the mole fractions of hydrogen in each void volume as a function of time.

For the initial assumed gas generation rate, if the concentration of hydrogen in the innermost void volume is below 0.05 mole fraction, then the hydrogen gas generation rate for the next iteration is increased and the above set of calculations is repeated. If the concentration is greater than 0.05 mole fraction, the gas generation rate for the next iteration is decreased. The evaluation of steady-state hydrogen mole fractions for a given gas generation rate, and simulation of hydrogen generation during transport by numerical integration, continues until two values of the H₂ gas generation are evaluated such that for one rate the hydrogen mole fraction in the innermost volume is less than 0.05 and the other results in a mole fraction greater than 0.05. The maximum allowable hydrogen gas generation is estimated to an accuracy of  $10^{-13}$  mole/sec by refining the gas generation rate interval using the technique of interval halving (Ref. 12.7).

10.4 Determination of Maximum Allowable Decay Limits for Content Codes

The maximum allowable decay heat limit for a content code is determined using RadCalc Software (Reference 12.8). RadCalc for Windows 2.01 is a windows-compatible software program with applications in the packaging and transportation of radioactive materials. Its primary function is to calculate the generation of hydrogen gas by radiolytic production in the waste matrix of radioactive wastes. It contains a robust algorithm that determines the daughter products of selected radionuclides. The various functions in RadCalc for Windows can be used separately or together. The procedure is outlined below.

The first step in the evaluation of decay heat limits involves determining the activities of the radionuclides and daughters and the associated hydrogen gas generation rate at the time of sealing based on an initial isotopic ratio for the waste. The generation of hydrogen gas by radiolysis is a function of the energy absorbed by the waste. The second step in the evaluation of decay heat limits involves iterating on the total activity (decay heat limit) given the activity fractions from step one until the allowable hydrogen gas generation rate is obtained.

- 10.4.1 Databases and Input Parameters Used For Calculation of Maximum Allowable Decay Heat Limits
- 10.4.1.1 Radionuclide Databases

RadCalc uses radionuclide information, calculated gamma absorption fractions for selected container types, and G values to determine decay heat values. Radionuclide information is taken from FENDL/D-1.0 database (Ref. 12.9). The following are a list of radionuclide parameters taken from FENDL/D-1.0 and the values they are used to calculate:

- Radionuclide half-lives are used in calculating specific activity
- Average heavy particle, beta-type radiation, and gamma radiation energies per disintegration are used in decay heat and hydrogen gas generation calculations
- Discrete gamma energies and abundances are used in hydrogen gas generation calculations.

RadCalc uses the ORIGEN2 (Ref. 12.10) database for decay calculations. The decay algorithms calculate the activity of the user specified source and daughter products over a specified period of time and the total number of disintegrations accumulated over this same time interval for each radionuclide. Parameters relevant to these calculations include atomic mass, atomic number, and state. These parameters are used for radionuclide identification and conversions. The decay constant and the branching ratios for decay modes are also used in the decay algorithms.

### 10.4.1.2 Gamma Absorption Fraction Input Parameters

RadCalc uses the total energy emitted by heavy particle and beta-type decay in calculating the volume of hydrogen produced. However, only a percent of gamma energy will be absorbed in the package and the waste. The absorbed gamma energy is a function of energy, waste density, material type, and geometry. The gamma energy absorbed by the waste is a function of the gamma emission strength, the quantity of gamma ray energy which is absorbed by collision with a waste particle, and the number of particles which interact with the gamma ray. Therefore, gamma energy absorption increases with increasing waste density. For a given waste density, a larger container will contain more particles, and therefore a higher percentage of the gamma ray energy would be absorbed than in a smaller container. The total cumulative absorbed dose for all nuclides and decay modes at time, t is evaluated as:

$$D_{\text{total}}(t) = \prod_{i=1}^{NR} A_i (0.82E_i^{\forall} + E_i^{\exists} + E_i^{\flat} + E_i^{x})[1 - \exp(-\&_i t)]$$

where,

 $D_{total}(t) = Total cumulative absorbed dose at time, t (rad)$ 

A = A proportionality constant equal to  $1.84 \times 10^{10}$  rad gram MeV⁻¹ yr⁻¹ Ci⁻¹

C_i = The specific activity of the "i"th nuclide in Curies/gram of waste

$$\&_i$$
 = The decay constant of the "i"th radionuclide (yr-1)

NR = Number of radionuclides

$$E_i \forall = \forall$$
 energy in MeV of the "i"th radionuclide extracted from Flaherty et al. (Ref. 12.11)

 $E_i^{\exists}$  = Average beta energy in MeV of the "i"th nuclide. The average beta energy is approximately one-third of the sum of the possible beta emissions multiplied by the relative abundance of each emission and were obtained from Flaherty et al. (Ref. 12.11).

 $E_i^X$  = The absorbed secondary energy in MeV of the "i"th radionuclide. The secondary radiations result from the transition of a radionuclide from an excited state to the ground state and were obtained from Flaherty et al. (Ref. 12.11).

 $E_i^{\mathbf{j}}$  = The absorbed gamma ray energy in MeV of the "i"th nuclide. The fraction of gamma energy that is absorbed by the waste is a function of the waste density and waste container geometry, and is evaluated for each radionuclide "i" as:

$$E_i^{\mathfrak{d}} = \Gamma_j n_{ij} f_{ij} E_{ij}^{\mathfrak{d}}$$

where,

Гj	=	the summation of the fractions of the gamma ray energies absorbed for all gamma emissions of the "i"th nuclide.
n _{ij}	=	the abundance of the "j"th gamma ray per decay of the "i"th nuclide
f _{ij}	=	the fraction of energy, of the "j"th gamma ray of the "i"th nuclide that is absorbed in the waste.
E _{ii} ∍	=	the energy in MeV, of the "j"th gamma ray of the "i"th nuclide.

RadCalc uses curve fits obtained from Flaherty et al. (Ref. 12.11) and recalculated using the Monte Carlo N-Particle (MCNP) transport code (Ref. 12.12) for ten containers, for obtaining the absorbed gamma dose.

The CNS cask is not currently recognized by the RadCalc. Therefore, another container with dimensions directly proportional to the cask was used for undertaking the calculations.

## 10.4.1.3 G Value Data

G values for RH-TRU waste are content specific. G values are determined based on the bounding materials present in the payload. The following G values were used for each of the content codes based on the presence of the bounding materials. The G values at 70°F are adjusted to the maximum operating temperature of the CNS cask (168°F) using the Arrhenius equation.

## <u>BC 312A</u>

This content code represents solidified organics and does not have a defined G value.

## <u>BC 314A</u>

This content code represents cemented inorganic process solids consisting of solidified cement slugs. It is assumed that water is the dominant hydrogen gas generating material in the waste form and will therefore be the bounding material. The G value for hydrogen from water is 1.6 molecules/100eV. It is also assumed that the moisture content of the waste is 30% and therefore the G value is 30% of the G value for water or 0.48 molecules/100eV (at 70°F).

## <u>BC 321A</u>

This content code represents solid organic debris consisting of various combustible and non-combustible items. The dominant material present in this waste is cellulosics (95%) and is therefore considered as the bounding material. The G value for hydrogen associated with cellulosics is 3.2 molecules/100eV (at 70°F).

## BC 321B

This content code represents organic pool filter and resin waste consisting of ion exchange resins. The dominant material present in this waste is organic resins (80%). The waste also consists of cellulosics (12%). The effective G value for hydrogen for this content code is the sum of 80% of the G value for

organic resins (1.7 molecules/100eV at 70°F) and 12% of the G value for cellulosics (3.2 molecules/100eV at 70°F), which is 1.74.

### <u>BC 322A</u>

This content code represents waste consisting of glass, metal, and solidified and other inorganic materials. It is conservatively assumed that residual water is the dominant hydrogen gas generating material in this waste form and will therefore be the bounding material. The G value for hydrogen from water is 1.6 molecules/100eV. It is also assumed that the moisture content of the waste is 1% and therefore the G value is 1% of the G value for water, or 0.016 molecules/100eV (at 70°F).

### 10.4.2 Input Parameters

The input parameters can be placed in three groups: (1) container data, (2) waste data, and (3) source data.

### 10.4.2.1 Container Data

RadCalc requires as input the following parameters associated with the container for which the maximum allowable decay heat limit is being calculated:

Container Type - The payload container for the waste material Container Dates - Date of generation, date of sealing, and shipping period Package Void Volume - void volume of the payload container.

A 6- by 6-foot liner with a volume equal to the CNS cask is used to represent the payload container in the RadCalc input file as the RadCalc database does not include the CNS cask. The package void volume for a CNS 10-160B cask is 1938 liters as shown earlier.

## 10.4.2.2 Waste Data

RadCalc requires as input the following parameters associated with the waste for which the maximum allowable decay heat limit is being calculated:

- Physical Form liquid, solid, or gas
- Waste Volume volume of the waste, cm³
- Waste Mass mass of the waste, g
- G Value G value of the waste, molecules per 100 eV

Liquids and gas wastes are prohibited in the CNS 10-160B cask. The volume of the waste is assumed to be 217 liters per drum (the external volume of the waste drum) and 2170 liters for 10 drums in the cask. The waste volume is used by RadCalc, along with the waste mass, to determine the volume of hydrogen generated in the cask. The mass of the waste is calculated based on the assumed bulk density of the waste. The volume of hydrogen generated is directly proportional to the mass of the waste, as discussed in Reference 12.11. The most conservative estimate of the volume of hydrogen (greatest volume) would occur at the highest possible bulk density of the waste. As presented in Section 10.3.1 above, the waste bulk densities for content codes BC 321A and BC 321B are conservatively assumed to be 0.55 g/cm³ and 0.36 g/cm³, respectively. A conservative bounding waste bulk density of 1.5 g/cm³, obtained from Reference 12.5, is used for content codes BC 314A and BC 322A, consisting of cement and metal scrap as bounding materials, respectively. Representative waste drum data for these content

codes provide waste bulk densities well below the  $1.5 \text{ g/cm}^3$  bounding bulk density used to calculate the decay heat limits. It may also be relevant that an additional level of conservatism is observed in the calculation of the mass of the waste. This mass is calculated based on the total volume of the 10 waste drums (2170 liters) rather than the available internal waste volume of 153.4 liters per drum shown in Section 10.3.1.

Section 10.4.1.3 provides a discussion of the G values for each of the content codes.

10.4.2.3 Source Data

RadCalc requires as input the following parameters associated with the source for which the maximum allowable decay heat limit is being calculated:

Isotopic Composition - List of radionuclides present in the waste Activity - Reported activities of the listed radionuclides in curies or Becquerel.

10.4.3 Procedure For Determining Maximum Allowable Decay Heat Limits

The necessary inputs are provided to the code prior to initiating a run. A time period of 60 days is conservatively assumed between date of beginning of decay and date of analysis to reflect the shipping period (see Attachment D). The model is run with the initial isotopic composition and activity and the corresponding hydrogen gas generation rate is obtained. It is compared with the maximum allowable hydrogen gas generation rate as obtained from Section 10.3, and the scaling factor is obtained by dividing the maximum allowable hydrogen gas generation rate by the RadCalc obtained rate. The isotopic composition is scaled by this differential factor. This is done on the basis of the assumption that the maximum decay heat occurs at the time of maximum activity which will result in the maximum hydrogen gas generation rate. The model is now run so that the hydrogen gas generation rate will now be equal to the maximum allowable hydrogen gas generation rate. The associated decay heat value will be the maximum decay heat limit as the decay heat limit shares a direct relationship with the hydrogen gas generation rate, independent of time.

The results of the decay heat limit modeling are shown in Table 10-1. Methods for demonstrating compliance of the BCL TRU waste with the decay heat and hydrogen gas generation rate limits are shown in Attachment B.

10.5 Methodology for Compliance with Payload Assembly Requirements

The TCO shall ensure that the CNS 10-160B Cask payload consists of payload containers belonging to the same or equivalent content code. In the event that payload containers of different content codes with different bounding G values and resistances are assembled together in the CNS 10-160B Cask, the TCO shall ensure that the decay heat and hydrogen gas generation rate for all payload containers within the payload are less than or equal to the limits associated with the payload container with the lowest decay heat limit and hydrogen gas generation rate limit.

### 11.0 WEIGHT

### 11.1 Requirements

The weight limit for a 55-gallon drum is 1,000 pounds. The weight limit for the loaded cask is 14,500 pounds.

11.2 Methods of Compliance and Verification

In accordance with TC-OP-01.4, Segregation and Packaging of TRU Waste, BCLDP shall weigh each payload container and contents on a calibrated scale to determine the total weight of the payload container. Based on the total measured weight of the individual payload containers, BCLDP shall calculate total assembly weight and evaluate compliance with the maximum loaded cask weight limit.

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## ATTACHMENT A

## REMOTE-HANDLED TRANSURANIC CONTENT CODES AND CHEMICAL LISTS FOR BATTELLE COLUMBUS LABORATORIES

4.10.2.1-A-1

## CONTENT CODE: BC 312A

CONTENT DESCRIPTION: Solidified Organic Waste

WASTE DESCRIPTION: This waste consists of solidified organic and inorganic liquid wastes.

**GENERATING SOURCES**: This waste is generated during research and development activities conducted in Building JN-1.

**WASTE FORM**: The waste consists primarily of inorganic and organic liquids that have been solidified using Floor Dry. The inorganic liquids included acids and acid solutions, and elemental mercury. The organic liquids included hydraulic oil, waste water, sludge of sand and mixed fission products (dust, small fragments); small items such as tools may also be present; nonhalogenated organic liquids such as glycols, oils, and alcohols.

**WASTE PACKAGING**: The waste will be placed directly into a 55-gallon drum with no layers of confinement. The drum is lined with a steel liner. Ten drums will then be placed into the CNS 10-160B cask.

**METHODS FOR DETERMINATION OF ISOTOPIC CHARACTERIZATION**: The isotopic information required to demonstrate compliance with the limits on fissile content, decay heat, and curie content will be determined based on the waste generation source and configuration, which establishes the initial radionuclide compositions based on location and initial use. A combination of assay of samples and modeling the isotopic generation processes results in the establishment of a mixture that characterizes the waste in the content code and the majority of waste at the BCLDP. Using shipping package modeling, dose rate and weight measurements based on the mixture then allows the BCLDP to determine the isotopic inventory. As required, additional radioassay (e.g., confirming gamma spectrometry) will be performed.

**FREE LIQUIDS**: Liquid waste is prohibited in the drums except for residual amounts in well-drained containers. The total volume of residual liquid in a payload container shall be less than 1 volume percent of the payload container. Waste packaging procedures ensure that free liquids are less than 1 volume percent of the payload container. Absorbents such as Radsorb or diatomaceous earth (e.g., Floor Dry) will be added to any waste matrix that has the potential to dewater after packaging.

**EXPLOSIVE/COMPRESSED GASES**: Explosives and compressed gases in the payload containers are prohibited by waste packaging procedures. If present, pressurized cans shall be punctured and emptied prior to packaging.

**PYROPHORICS**: Nonradioactive pyrophorics in the payload containers are prohibited by waste packaging procedures. Waste packaging procedures shall ensure that all pyrophoric radioactive materials are present only in small residual amounts (less than 1 weight percent) in payload containers.

**CORROSIVES**: Corrosives are prohibited in the payload container. Acids and bases that are potentially corrosive shall be neutralized and rendered noncorrosive prior to being a part of the waste. The physical form of the waste and the waste generating procedures ensure that the waste is in a nonreactive form.

**CHEMICAL COMPATIBILITY**: A chemical compatibility study has been performed on this content code, and all waste is chemically compatible for materials in greater than trace (>1% by weight) quantities.

**ADDITIONAL CRITERIA**: Each drum is fitted with a minimum of one filter vent. The steel liner is fitted with a filter with a hydrogen diffusivity of 3.7E-06 mole/second/mole fraction.

**MAXIMUM ALLOWABLE HYDROGEN GENERATION RATES - OPTION 1**: The maximum allowable hydrogen generation rate limit is 3.5082E-08 moles per second per drum and 3.5082E-07 moles per second per CNS 10-160B cask.

**MAXIMUM ALLOWABLE DECAY HEAT LIMIT - OPTION 2**: There is no decay heat limit for this content code as no G values have been established. Waste cannot be transported under Option 2.

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# BATTELLE COLUMBUS LABORATORIES CONTENT CODE BC 312A SOLIDIFIED ORGANIC WASTE

MATERIALS AND CHEMICALS >1%

DIATOMACEOUS EARTH (FLOOR DRY) ACIDS AND ACID SOLUTIONS MERCURY HYDRAULIC OIL, GLYCOLS, OILS, AND ALCOHOLS SAND VERMICULITE RADSORB AQUA-SET/PETRO-SET

## MATERIALS AND CHEMICALS <1%

METALS (including stainless steel, aluminum, iron, copper, lead, beryllium, and zirconium)

## CONTENT CODE: BC 314A

## CONTENT DESCRIPTION: Cemented Inorganic Process Solids

**WASTE DESCRIPTION**: This waste consists of slugs produced from dissolving fuel specimens in an acid solution that was then diluted several times and mixed with cement and water and allowed to solidify in foam cups.

**GENERATING SOURCES**: This waste is generated during repackaging of the waste materials generated from research and development activities conducted in Building JN-1.

**WASTE FORM**: The waste consists of slugs produced from dissolving fuel specimens in an acid solution which was then diluted several times and mixed with cement and waster and allowed to solidify in foam cups. The slugs will contain limited amounts of radionuclides from fuel because of this dilution. The waste matrix will also include Floor Dry added during repackaging to absorb any water from condensation or dewatering.

**WASTE PACKAGING**: The waste will be placed directly into a 55-gallon drum with no layers of confinement. The drum is lined with a steel liner. Ten drums will then be placed into the CNS 10-160B cask.

**METHODS FOR DETERMINATION OF ISOTOPIC CHARACTERIZATION**: The isotopic information required to demonstrate compliance with the limits on fissile content, decay heat, and curie content will be determined based on the waste generation source and configuration, which establishes the initial radionuclide compositions based on location and initial use. A combination of assay of samples and modeling of the isotopic generation process used in the establishment of a mixture that characterizes the waste in the content code and the majority of waste at the BCLDP. Using shipping package modeling, dose rate and weight measurements based on the mixture then allows the BCLDP to determine the isotopic inventory. As required, additional radioassay (e.g., confirming gamma spectroscopy) will be performed.

**FREE LIQUIDS**: Liquid waste is prohibited in the drums except for residual amounts in well-drained containers. The total volume of residual liquid in a payload container shall be less than 1volume percent of the payload container. Waste packaging procedures ensure that free liquids are less than 1 volume percent of the payload container. Absorbents such as diatomaceous earth will be added to any waste matrix that has the potential to dewater after packaging.

**EXPLOSIVE/COMPRESSED GASES**: Explosives and compressed gases in the payload containers are prohibited by waste packaging procedures. If present, pressurized cans shall be punctured and emptied prior to packaging.

**PYROPHORICS**: Nonradioactive pyrophorics in the payload containers are prohibited by waste packaging procedures. Waste packaging procedures shall ensure that all pyrophoric radioactive materials are present only in small residual amounts (less than 1 weight percent) in payload containers.

**CORROSIVES**: Corrosives are prohibited in the payload container. Acids and bases that are potentially corrosive shall be neutralized and rendered noncorrosive prior to being a part of the waste. The physical form of the waste and the waste generating procedures ensure that the waste is in a nonreactive form.

**CHEMICAL COMPATIBILITY**: A chemical compatibility study has been performed on this content code, and all waste is chemically compatible for materials in greater than trace (>1% by weight) quantities.

**ADDITIONAL CRITERIA**: Each drum is fitted with a minimum of one filter vent. The steel liner is fitted with a filter with a hydrogen diffusivity of 3.7E-06 mole/second/mole fraction.

**MAXIMUM ALLOWABLE HYDROGEN GENERATION RATES - OPTION 1**: The maximum allowable hydrogen generation rate limit is 3.5082E-08 moles per second per drum and 3.5082E-07 moles per second per CNS 10-160B cask.

**MAXIMUM ALLOWABLE DECAY HEAT LIMIT - OPTION 2**: The maximum allowable decay heat limit is 0.738 watts per drum and 7.38 watts per CNS 10-160B cask.

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# BATTELLE COLUMBUS LABORATORIES CONTENT CODE BC 314A CEMENTED INORGANIC PROCESS SOLIDS

MATERIALS AND CHEMICALS >1%

DIATOMACEOUS EARTH (FLOOR DRY) CEMENT SLUGS

MATERIALS AND CHEMICALS <1%

NITRIC ACID WATER

## CONTENT CODE: BC 321A

CONTENT DESCRIPTION: Solid Organic Waste

WASTE DESCRIPTION: This waste consists of a variety of combustible and noncombustible items.

**GENERATING SOURCES**: This waste is generated from activities supporting the decontamination and decommissioning of Building JN-1 under the Battelle Columbus Laboratories Decommissioning Project (BCLDP).

**WASTE FORM**: The waste may include combustible items such as cloth and paper products (e.g., from the cleanup of spills), rags, coveralls and booties, plastic, cardboard, rubber, wood, surgeons gloves, and Kimwipes. The waste may also include filter waste (e.g., dry box filters, HEPA filters, and filter cartridges); noncombustible Benelex and Plexiglas neutron shielding, blacktop, concrete, dirt, and sand; leaded gloves and aprons comprised of Hypalon rubber and lead oxide impregnated neoprene; and small amounts of metal waste. The waste may also include particulate and sludge-type organic process solids immobilized/solidified with Portland cement, vermiculite, Aqua-Set, or Petro-Set.

**WASTE PACKAGING**: The waste will be placed directly into a 55-gallon drum with no layers of confinement. The drum is lined with a steel liner. Ten drums will then be placed into the CNS 10-160B cask.

**METHODS FOR DETERMINATION OF ISOTOPIC CHARACTERIZATION:** The isotopic information required to demonstrate compliance with the limits on fissile content, decay heat, and curie content will be determined based on the waste generation source and configuration, which establishes the initial radionuclide compositions based on location and initial use. A combination of assay of samples and modeling of the isotopic generation process, results in the establishment of a mixture that characterizes the waste in the content code and the majority of waste at the BCLDP. Using shipping package modeling, dose rate and weight measurement based on the mixture then allows the BCLDP to determine the isotopic inventory. As required, additional radioassay (e.g., confirming gamma spectroscopy) will be performed.

**FREE LIQUIDS**: Liquid waste is prohibited in the drums except for residual amounts in well-drained containers. The total volume of residual liquid in a payload container shall be less than 1 volume percent of the payload container. Waste packaging procedures ensure that free liquids are less than 1 volume percent of the payload container. Absorbents such as Radsorb or diatomaceous earth will be added to any waste matrix that has the potential to dewater after packaging.

**EXPLOSIVE/COMPRESSED GASES**: Explosives and compressed gases in the payload containers are prohibited by waste packaging procedures. If present, pressurized cans shall be punctured and emptied prior to packaging.

**PYROPHORICS**: Nonradioactive pyrophorics in the payload containers are prohibited by waste packaging procedures. Waste packaging procedures shall ensure that all pyrophoric radioactive materials are present only in small residual amounts (less than 1 weight percent) in payload containers.

**CORROSIVES**: Corrosives are prohibited in the payload container. Acids and bases that are potentially corrosive shall be neutralized and rendered noncorrosive prior to being a part of the waste. The physical form of the waste and the waste generating procedures ensure that the waste is in a nonreactive form.

**CHEMICAL COMPATIBILITY:** A chemical compatibility study has been performed on this content code, and all waste is chemically compatible for materials in greater than trace (>1% by weight) quantities.

**ADDITIONAL CRITERIA**: Each drum is fitted with a minimum of one filter vent. The steel liner is fitted with a filter with a hydrogen diffusivity of 3.7E-06 mole/second/mole fraction.

**MAXIMUM ALLOWABLE HYDROGEN GENERATION RATES - OPTION 1**: The maximum allowable hydrogen generation rate limit is 4.093E-08 moles per second per drum and 4.093E-07 moles per second per CNS 10-160B cask.

**MAXIMUM ALLOWABLE DECAY HEAT LIMIT - OPTION 2**: The maximum allowable decay heat limit is 0.154 watts per drum and 1.54 watts per CNS 10-160B cask.

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# BATTELLE COLUMBUS LABORATORIES CONTENT CODE BC 321A SOLID ORGANIC WASTE

MATERIALS AND CHEMICALS >1%

**BLACKTOP (ASPHALT) CELLULOSICS RUBBER** DIATOMACEOUS EARTH (FLOOR DRY) GLASS **IRON-BASED METAL/ALLOYS** PAPER PLASTIC RADSORB CLOTH CARDBOARD WOOD **KIMWIPES FILTERS** BENELEX PLEXIGLAS NEOPRENE PORTLAND CEMENT VERMICULITE AQUA-SET/PETRO-SET OTHER INORGANICS

## MATERIALS AND CHEMICALS <1%

METALS (including aluminum, lead, zirconium, stainless steel, and carbon steel) CONCRETE SOIL
#### CONTENT CODE: BC 321B

#### CONTENT DESCRIPTION: Solid Organic Waste

WASTE DESCRIPTION: This waste consists of a variety of combustible and noncombustible items.

**GENERATING SOURCES**: This waste is generated during the change-out of resins in the Transfer/Storage Pool filtering system in Building JN-1 (Hot Cell Laboratory).

**WASTE FORM**: The waste may include filter waste (e.g., pool filters); nuclear grade resin, resin bags, paper, rubber gloves, Floor Dry bags, seals, hoses, valves, and clamps.

**WASTE PACKAGING**: The waste will be placed directly into a 55-gallon drum with no layers of confinement. The drum may be lined with a steel or polyethylene liner. Ten drums will then be placed into the CNS 10-160B cask.

**METHODS FOR DETERMINATION OF ISOTOPIC CHARACTERIZATION**: The isotopic information required to demonstrate compliance with the limits on fissile content, decay heat, and curie content will be determined based on the waste generation source and configuration, which establishes the initial radionuclide compositions based on location and initial use. A combination of assay of samples and modeling the isotopic generation process results in the establishment of a mixture that characterizes the waste in the content code. Using shipping package modeling, dose rate and weight measurements based on the mixture then allows the BCLDP to determine the isotopic inventory. As required, additional radioassay (e.g., confirming gamma spectrometry) will be performed.

**FREE LIQUIDS**: Liquid waste is prohibited in the drums except for residual amounts in well-drained containers. The total volume of residual liquid in a payload container shall be less than 1 volume percent of the payload container. Waste packaging procedures ensure that free liquids are less than 1 volume percent of the payload container. Absorbents such as Radsorb or diatomaceous earth will be added to any waste matrix that has the potential to dewater after packaging.

**EXPLOSIVE/COMPRESSED GASES**: Explosives and compressed gases in the payload containers are prohibited by waste packaging procedures. If present, pressurized cans shall be punctured and emptied prior to packaging.

**PYROPHORICS**: Nonradioactive pyrophorics in the payload containers are prohibited by waste packaging procedures. Waste packaging procedures shall ensure that all pyrophoric radioactive materials are present only in small residual amounts (less than 1 weight percent) in payload containers.

#### 4.10.2.1-A-11

**CORROSIVES**: Corrosives are prohibited in the payload container. Acids and bases that are potentially corrosive shall be neutralized and rendered noncorrosive prior to being a part of the waste. The physical form of the waste and the waste generating procedures ensure that the waste is in a nonreactive form.

**CHEMICAL COMPATIBILITY**: A chemical compatibility study has been performed on this content code, and all waste is chemically compatible for materials in greater than trace (>1% by weight) quantities.

**ADDITIONAL CRITERIA**: Each drum is fitted with a minimum of one filter vent, and the steel or polyethylene liner, if present, is either punctured or fitted with a filter with a hydrogen diffusivity of 3.7E-06 mole/second/mole fraction.

**MAXIMUM ALLOWABLE HYDROGEN GENERATION RATES - OPTION 1**: The maximum allowable hydrogen generation rate limit is 4.093E-8 moles per second per drum and 4.093E-7 moles per second per CNS 10-160B cask.

**MAXIMUM ALLOWABLE DECAY HEAT LIMIT - OPTION 2**: The maximum allowable decay heat limit is 0.269 watts per drum and 2.69 watts per CNS 10-160B cask.

## BATTELLE COLUMBUS LABORATORIES CONTENT CODE BC 321B SOLID ORGANIC WASTE

MATERIALS AND CHEMICALS >1%

CELLULOSICS (≤12 weight %) RUBBER DIATOMACEOUS EARTH (FLOOR DRY) ION EXCHANGE RESIN (≤80 weight %) IRON-BASED METAL/ALLOYS RADSORB RESIN BAGS FILTERS OTHER INORGANICS

#### MATERIALS AND CHEMICALS <1%

METALS (including aluminum, lead, zirconium, stainless steel, and carbon steel)

#### CONTENT CODE: BC 322A

CONTENT DESCRIPTION: Solid Inorganic Waste

WASTE DESCRIPTION: This waste consists of a variety of glass and metal materials.

**GENERATING SOURCES**: This waste is generated during repackaging of the waste materials generated from research and development activities conducted in Building JN-1.

**WASTE FORM**: The waste consists primarily of glass and metal debris. Glass debris includes laboratory glassware, windows, leaded glass windows, and various glass apparatus. Metal items may include deteriorated berry cans, cable, wire, planchets, signs, valves, piping, strapping, tools, foils, sheeting, fixtures, equipment (e.g., pumps or motors that have had all oil or any other free liquids removed up to an allowance of 1%), hardware (e.g., nuts, bolts, brackets), specimen vials, fuel rod cladding, metallurgical mounts, and lead lined tubing. Metals of construction include stainless steel, aluminum, iron, copper, lead, beryllium, and zirconium.

**WASTE PACKAGING**: The waste will be placed directly into a 55-gallon drum with no layers of confinement. The drum is lined with a steel liner. Ten drums will then be placed into the CNS 10-160B cask.

**METHODS FOR DETERMINATION OF ISOTOPIC CHARACTERIZATION:** The isotopic information required to demonstrate compliance with the limits on fissile content, decay heat, and curie content will be determined based on the waste generation source and configuration, which establishes the initial radionuclide compositions based on location and initial use. A combination of assay of samples and modeling of the isotopic generation process used in the establishment of a mixture that characterizes the waste in the content code and the majority of waste at the BCLDP. Using shipping package modeling, dose rate and weight measurements based on the mixture then allows the BCLDP to determine the isotopic inventory. As required, additional radioassay (e.g., confirming gamma spectroscopy) will be performed.

**FREE LIQUIDS**: Liquid waste is prohibited in the drums except for residual amounts in well-drained containers. The total volume of residual liquid in a payload container shall be less than 1 volume percent of the payload container. Waste packaging procedures ensure that free liquids are less than 1 volume percent of the payload container. Absorbents such as diatomaceous earth (e.g., Floor Dry) will be added to any waste matrix that has the potential to dewater after packaging.

## 4.10.2.1-A-14

**EXPLOSIVE/COMPRESSED GASES**: Explosives and compressed gases in the payload containers are prohibited by waste packaging procedures. If present, pressurized cans shall be punctured and emptied prior to packaging.

**PYROPHORICS**: Nonradioactive pyrophorics in the payload containers are prohibited by waste packaging procedures. Waste packaging procedures shall ensure that all pyrophoric radioactive materials are present only in small residual amounts (less than 1 weight percent) in payload containers.

**CORROSIVES**: Corrosives are prohibited in the payload container. Acids and bases that are potentially corrosive shall be neutralized and rendered noncorrosive prior to being a part of the waste. The physical form of the waste and the waste generating procedures ensure that the waste is in a nonreactive form.

**CHEMICAL COMPATIBILITY**: A chemical compatibility study has been performed on this content code, and all waste is chemically compatible for materials in greater than trace (>1% by weight) quantities.

**ADDITIONAL CRITERIA**: Each drum is fitted with a minimum of one filter vent. The steel liner is fitted with a filter with a hydrogen diffusivity of 3.7E-06 mole/second/mole fraction.

**MAXIMUM ALLOWABLE HYDROGEN GENERATION RATES - OPTION 1**: The maximum allowable hydrogen generation rate limit is 3.5082E-8 moles per second per drum and 3.5082E-7 moles per second per CNS 10-160B cask.

**MAXIMUM ALLOWABLE DECAY HEAT LIMIT - OPTION 2**: The maximum allowable decay heat limit is 22.1 watts per drum and 221 watts per CNS 10-160B cask.

## BATTELLE COLUMBUS LABORATORIES CONTENT CODE BC 322A SOLID INORGANIC WASTE

MATERIALS AND CHEMICALS >1%

CEMENT DIATOMACEOUS EARTH (FLOOR DRY) GLASS METALS (including stainless steel, aluminum, iron, copper, lead, beryllium, and zirconium) IRON-BASED METAL/ALLOYS OTHER INORGANICS

## MATERIALS AND CHEMICALS <1%

CARBON TETRACHLORIDE 1,1,1-TRICHLOROETHANE TRICHLOROETHYLENE

#### ATTACHMENT B

#### METHODOLOGY FOR DETERMINATION OF DECAY HEATS AND HYDROGEN GAS GENERATION RATES FOR REMOTE-HANDLED TRANSURANIC (RH-TRU) CONTENT CODES FOR BATTELLE COLUMBUS LABORATORIES

#### 1.0 INTRODUCTION

All Battelle Columbus Laboratories Decommissioning Project (BCLDP) remote-handled transuranic (RH-TRU) waste to be transported in the CNS 10-160B cask shall comply with the 5% (by volume) limit on hydrogen concentration during transport. If a bounding G value and decay heat limit have been established for the approved content code, compliance with the decay heat limit shall be evaluated pursuant to this attachment for the individual containers under the content code. If compliance with the decay heat limit cannot be demonstrated, the hydrogen generation rate of the container shall be determined as outlined in this attachment and compared to the hydrogen gas generation rate limit specified for that approved content code. If the container meets the limit, it is eligible for shipment if all other transportation requirements are met. If the container does not meet the limit, it cannot be shipped and shall be segregated for repackaging or other mitigation measures.

#### 2.0 DECAY HEAT METHODOLOGY

This section describes the general features of nondestructive assay methods used in conjunction with acceptable knowledge by the BCLDP.

The overall methodology for the determination of the radioassay properties is described in DD-98-04, Waste Characterization, Classification, and Shipping Support Technical Basis Document and is summarized in Figure 1. Under the methodology, the isotopic content for an identified TRU waste stream is determined by a combination of (1) representative waste stream sample analyses, (2) conservative application of the Oak Ridge Isotope Generation and Depletion (ORIGEN2) code values for isotopes expected to be present, but not represented by the sample analyses, and (3) assessment of cesium (Cs)-137 content of a payload container based on external radiation field measurements and calculation of TRU isotopic content using a ratio of radionuclides based on known Cs-137 content. The determinations are verified on an approved, periodic basis by sample submission to the BCLDP Radioanalytical Laboratory for gamma and/or alpha spectroscopy. The results of the implementation of the DD-98-04 methodology provide the data inputs to the computer program (spreadsheet) used by the TRU Waste Transportation Certification Official to determine the parameters of interest for each payload container (including fissile grams equivalent and decay heat).

Since the gamma rays emitted by radionuclides can be readily detected and quantified by common measurement techniques, i.e., as a dose rate, emitted gamma are used to model the quantity of isotopes present in a standard waste stream. Verifying samples are analyzed for both gamma and alpha emitters. Because isotopes other than gamma emitters are known to be present, laboratory measurements of the isotopic distribution are combined with a computer-generated distribution of account for required isotopes, e.g., per U.S. Department of Transportation requirements. The measured isotopic distribution is based on laboratory analysis (alpha and gamma spectroscopy) of air, smear and material samples taken from throughout the accessible work areas of Building JN-1. Using the measured distribution as a base, the remaining isotopes are scaled according to the distribution generated by the ORIGEN2 computer code, which models the production and decay of fission and activation products of commercial nuclear power plant fuel. Commercial fuel best characterizes the overwhelming majority of the isotopes present, by isotope and relative ratio.

#### Isotopic Composition

- ORIGEN2 output
- Smear samples alpha and gamma
- spectroscopic analysis
- Determines Isotopic Mix

Dose Estimates for 1 mCi of Mix

- Waste configuration (trash bags, B-25 boxes,

- 55-gallon drums, D-boxes, misc. metal objects)
- QAD run to determine dose for 1 mCi - Isotopes with activities >1.0E-5 mCi
- Matrix used cellulose (bags and D-boxes with plastics, wood, clothes, etc.); iron (B-25 boxes with dense materials including concrete)

Isotopic Quantity Estimates for Items

- Derive equations and graphs for dose rate as a function of weight
- Sort criteria in graphs determine split between TRU and LLW
- Net weight and dose rate entered in a spreadsheet determine isotopic quantity

Determination of Cs-137

- External radiation field measurement
- Cs-137 content indicates TRU content based on radionuclide ratio

Determination of Transportation Parameters

- Isotopic quantity and composition yield FGE, decay heat, TRU alpha activity and PE-curies

## Figure 1 BCLDP Methodology for Determination of Radioassay Properties for RH-TRU Waste

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The "JN standard isotopic mixture" used in the model is representative of the composition of the majority of radioactive waste generate in all areas of the BCL facility, except the pool. Waste from the pool is separately characterized based on sample results. In addition to the pool waste, other waste streams may be encountered that do not match the JN standard isotopic mix. In such cases, the newly characterized waste will be characterized based on specific analytical results (alpha and gamma spectroscopy).

A given quantity of the JN standard isotopic mixture is used as the radioactive material source with the QAD computer shielding code to generate external gamma ray interaction rates for various package and form weights. These interaction rates are used to generate interaction rates-to-weight conversion equations for each package and waste form. The equations are incorporated into spreadsheets so that activity content, in millicuries, for individual packages and waste forms can be calculated. Spreadsheets are also used to calculate TRU interaction rate levels, and plots of these values as a function of net container weight are provided for each container type to simplify field sorting and packaging.

Required QAD inputs include source and package dimensions including any shielding materials, and quantities of individual isotopes that make up the source (i.e., JN standard isotopic mixture). QAD calculations are performed for a range of representative weights for each package. Specific package models include field sort waste bag, metal case, IP-2 147-cubic-feet box, standard D-box, 55-gallon drum, and standard B-25 box models. The final packaging configuration for BCL TRU waste is the 55-gallon drum.

Waste matrices are modeled as either cellulose or iron. The cellulose matrix represents the varied composition of the bag and D-box models, which are composed of plastics, wood, cloth, etc., and are similar to cellulose in their electronic configuration. The iron matrix is used for the B-25 box and 55-gallon drum models, which include a range of more dense materials, including concrete. The choice of waste matrices is conservative as the physical properties of cellulose and iron relative to radiological parameters are well characterized. It is important to note that the representative weights for 55-gallon drum, for example, correspond to a density much less than the density of iron (7.86 g/cm³), on the close order of less than  $1.0 \text{ g/cm}^3$ .

As detailed in DD-98-04, estimated uncertainties associated with the container weight, Cs-137 activity based on measurement of decay gammas emanating from the container, and estimation of inventories of other radionuclides and total transuranics based on measured or predicted ratios to Cs-137 activity have been factored into the determinations of an upper bounding uncertainty for the methodology.

The application of the ORIGEN2 code in the proposed BCLDP TRU Waste Certification Program methodology for determining radioassay properties for RH-TRU wastes is conservative. In addition, associated measurement errors and assumptions have been conservatively estimated to determine a total error that is bounding for the methodology. The following subsection provides details on the application of the ORIGEN2 code for determining the radioassay properties.

#### **Application of ORIGEN2 Code**

The ORIGEN2 code (RSIC Code Package CCC-371) is used in the DD-98-04 methodology. Characterization of the JN standard isotopic distribution depended upon whether available data existed to permit estimation of the normalized activity ratio (to Cs-137 activity) for the isotopes of interest. Where

sufficient data were available, a lognormal fit was used. Where insufficient data was available, the results of a series of ORIGEN2 software analyses were employed.

For Am-241, Cm-244, Co-60, Cs-134, Eu-154, Np-237, Pu-238, Pu-239/-240, Sb-125, Sr-90, U-234, and U-238, as many as 69 samples from the anticipated waste stream were available. A two-parameter ( $\mu$ ,  $\sigma$ ) lognormal distribution was fitted to these data. The mean parameter ( $\mu$ ) estimated for each studied isotope represented its assumed normalized activity ratio (to Cs-137) in the standard isotopic distribution. The estimated spread parameter ( $\sigma$ ) was used in considering the total uncertainty associated with the waste characterization.

For the isotopes of interest, the computer code ORIGEN2 was used to estimate their normalized activity ratio (to Cs-137). Specifically, values were assumed for enrichments, burn-up, and decay consistent with the processed used to generate the waste stream being classified. These values were then applied as parameters within the ORIGEN2 software, producing estimates of the activities of the various isotopes of interest.

Professional judgement consensus center, low, and high values were identified for each of these three parameters: enrichment, burn up, and decay. Taken together, these three values were meant to represent the central tendency and distribution (i.e., practical range) of the enrichment, burn up, and decay of potential waste streams. A deliberate choice was made to underestimate the decay time so as to make the resultant values conservative.

Twenty-seven iterations of ORIGEN2 software code would be required to consider each combination of these three values for each of three parameters (i.e.,  $3^3 = 27$ ). Additional code runs would be necessary, moreover, to provide some measure of the uncertainty associated with the application of ORIGEN2 in estimating the normalized activity ratios of the remaining isotopes of interest. Latin Hypercube sampling is an alternative approach, allowing for effective integration of computer code but with fewer runs. In order to apply these values in the context of a Latin Hypercube design, an assumed distribution is required for each parameter considered in the design. Because the low and high values for each parameter were not symmetric in relation to the center value, a skewed distribution was selected. The lognormal represents a skewed distribution that can be readily applied without additional mathematical complication. The log-transformed center value was assumed to represent the distribution's log mean, and its log standard deviation was derived by averaging the deviations of the log-transformed low and high values from the log mean. Specifically, the average deviation was assumed to represent 1.645 (i.e., the 0.95 quantile of a standard normal distribution) times the log-standard deviation. Doing so is equivalent to assuming the low and high values represent, on average, a range from the 5th to the 95th percentile of the distribution.

In using the ORIGEN2 software code to characterize the normalized activity ratios for the isotopes without available data, the Latin Hypercube employed in the DD-98-04 methodology assumed that values for enrichment, burn up, and decay and software together represent a 'black-box' estimation of normalized activity ratios. Using this approach, a series of replicate designs is applied. The mean result from each replicate design is considered when estimating the mean and variance in normalized activity.

Four replicates of a five-sample Latin Hypercube design were developed thereby providing 20 analysis runs. The distribution of each parameter was divided into five partitions of equal probability. Latin Hypercube sampling, then, insures that a random value of each partition is included in each of the five replicated designs, while minimizing the total number of required analysis runs.

The mean result across the 20 ORIGEN2 runs (or equivalently, the mean of the mean results determined for the four replicated designs) estimated for each studied isotope represents its assumed normalized activity ratio (to Cs-137) in the standard isotopic distribution. Though ORIGEN2 reports activities for all the isotopes of interest, only the results for those isotopes without sufficient available sample data are retained. The ORIGEN2 results and those based on available data are comparable. The estimated variance in mean result across the four replicate designs—a measure of the uncertainty associated with using the ORIGEN2 software to characterize isotope activity—is used in considering the total uncertainty associated with the waste characterization.

#### 3.0 OBJECTIVES OF THE GAS GENERATION TESTING

The maximum allowable hydrogen gas generation rates for the RH-TRU content codes for the BCLDP are provided in Table 10-1 of this appendix. Compliance with the hydrogen gas generation rate shall be demonstrated by testing. Compliance with the requirements of this test plan should be documented in site-specific procedures under a documented quality assurance program.

#### 4.0 GAS GENERATION TEST METHODOLOGY

The following sections describe how compliance with the limit on the hydrogen gas generation rate will be implemented for each authorized content code for BCLDP.

#### Demonstration of Compliance With Hydrogen Gas Generation Limit

During the course of the testing, the headspace gas of the selected waste containers will be sampled and analyzed to determine the concentrations of hydrogen and other gases that are produced by radiolysis or present when the waste was packaged. Sampling lines that communicate with the headspace of the waste containers will be installed. Samples of the headspace gas will be withdrawn periodically and analyzed using a gas chromatograph and/or a mass spectrometer. The analytical results will be used to calculate the hydrogen gas generation rate. The measured hydrogen gas generation rate will be compared to the appropriate hydrogen gas generation limits for each content code to evaluate compliance with transportation requirements.

Because all layers of confinement in all the containers have been vented since the time of generation and the containers have been in a vented condition for a period of time, steady-state hydrogen concentrations exist within all void volumes inside a container. At steady-state conditions, the rate of gas generation by radiolysis equals the release rate of gas across each layer of confinement. The measured hydrogen gas concentration in the headspace gas will be used to calculate the hydrogen gas generation rate.

The hydrogen gas generation rate of the waste container is calculated from the measured hydrogen gas concentration using the following relationship:

$$C_g = X_H x L_{CF}$$

where,

 $C_g =$  the hydrogen gas generation rate (mole/sec)  $X_H =$  the measured concentration of hydrogen gas in the waste container headspace (mole fraction)  $L_{CF} =$  diffusion characteristic of the waste container filter.

The rate shall be compared to the appropriate limit for the content code. The container shall be qualified for shipment only if the limit is met.

Another method may also be used when the final waste form is a solid monolith of evaporated/solidified inorganic wastes (BC 312A or 314A) that will be directly placed into drums. Process controls will be used to ensure homogeneity of the sludge. A small sample of the waste will be analyzed for its gas generation properties. The hydrogen gas generation rate for the drum can then be determined based on the mass of waste in the drum. For example, a sludge sample can be placed in a sealed test chamber of known volume. The concentration of hydrogen will be measured in the chamber after an elapsed period of time, and the following relationship will be used to calculate the hydrogen gas generation rate from the sample:

$$C_{g, sample} = \frac{X P V_{chamber}}{R T \Delta t}$$

where,

C _{g.sample}	=	hydrogen gas generation rate from sample (mol/sec)
X		mole fraction hydrogen in the test chamber
Р	=	absolute ambient pressure (atm)
V _{chamber}	=	volume of the test chamber (L)
R	=	gas law constant (0.08206 atm L mol ⁻¹ K ⁻¹ )
Т	=	absolute ambient temperature (K)
)t	=	elapsed time (sec).

The hydrogen gas generation rate will be calculated on a drum basis using the following relationship:

$$C_{g, drum} = C_{g, sample} \frac{m_{drum}}{m_{sample}}$$

where,

Cg.drum	=	hydrogen gas generation rate in drum (mol/sec)
mdrum	=	mass of waste form in drum (g)
m _{sample}	=	mass of sample (g).

The actual drum hydrogen gas generation rate will be compared to the maximum allowable hydrogen generation rate limit in Table 10-1 of this appendix.

## ATTACHMENT C

#### CHEMICAL COMPATIBILITY ANALYSIS OF BATTELLE COLUMBUS LABORATORIES CONTENT CODES

4.10.2.1-C-1

#### **1.0 INTRODUCTION**

This attachment describes the method used for demonstrating chemical compatibility within a given content code, and among all content codes at Battelle Columbus Laboratories (BCL) for the CNS 10-160B Cask payload. The chemical compatibility analyses cover normal conditions of transport as well as hypothetical accident conditions, as described in this attachment.

#### 2.0 METHODOLOGY FOR CHEMICAL COMPATIBILITY ANALYSES

All information for the chemical compatibility study is maintained in databases. The chemicals reported for each content code by BCL are classified into reaction groups as defined by the U.S. Environmental Protection Agency (EPA) document "A Method for Determining the Compatibility of Hazardous Wastes" (EPA 600/2-80-076, 1980, Ref. 3.1).

A database program was developed to evaluate the chemical compatibility of the waste. With this program, potential incompatibilities can be assessed by content code or by other parameters. Potential incompatibilities are those defined in the EPA document, which identifies combinations of chemical groups that may react to create adverse conditions, listed in Section 2.1. All such combinations have been entered into a reference database to be used in assessing the chemical compatibility of a given list of chemicals. The logic of the program used in evaluating the chemical compatibility by content code is described in detail below.

As an initial step, the program indexes the databases according to content code. The program will then locate the first reaction group in the content code and pick the highest concentration of any chemical in that group. The group is then paired with every other group in the database to check for incompatibility. If a potential incompatibility is found, it is printed out along with the corresponding content codes. After finding all potential incompatibilities for that content code, the program moves on to another content code until all content codes have been processed.

To ensure accuracy, the databases were printed and checked against the original submittal forms from BCL. The list of potential chemical incompatibilities reported by the program was hand checked using the EPA document as a reference to assure proper functioning of the program. All potential chemical incompatibilities were then evaluated on a case-by-case basis to identify which, if any, of the reactions could occur, given the nature of the waste and the chemical constituents.

#### 2.1 Chemical Compatibility Evaluation of RH-TRU Waste Content Codes

Content codes are classified as chemically "incompatible" if the <u>potential</u> exists for any of the following reactions:

- Corrosion
- Explosion
- Heat generation
- Gas generation (flammable gases) by chemical interactions
- Pressure build up (nonflammable gases) by chemical interactions
- Toxic by-product generation.

As stated earlier, a comprehensive list of all possible chemicals present in the waste for each content code has been produced. These chemical components are determined by examining the process technology, and by comprehensive analyses of the process knowledge. Under this system, all chemical inputs into the system are accounted for, even though all of these components may not be a part of the waste. For example, a chemical list might include both acids and bases, even though the two groups have been neutralized prior to placement in a payload container.

The chemical concentration levels are reported as either Trace (T) (< 1% by weight), Minor (M) (1-10%), or Dominant (D) (> 10%). Each chemical list is divided into groups based on chemical properties and structure (e.g., acids, caustics, metals, etc.). Table 1 lists all the groups and their number designations. As noted in the table, the groups and examples listed are only for illustrative purposes, and do not necessarily represent components of waste materials in a CNS 10-160B Cask payload. If incompatible groups are combined, the possibility exists for the reactions listed above. For example, a reaction between Group 1 (Acids, Mineral, Non-oxidizing) and Group 10 (Caustics) could result in heat generation. Incompatibilities have been defined within each content code, and where mixing of content codes could occur under hypothetical accident conditions.

Interactions between compounds present in trace quantities (<1 percent by weight) and compounds present in concentrations  $\geq$  1 percent by weight (i.e., D x T, M x T, or T x T) do not pose an incompatibility problem for the following reasons:

- Most trace chemicals reported by the sites are in concentrations well below the trace limit of 1 weight percent.
- The trace chemicals are usually dispersed in the waste, which further dilutes concentrations of these materials.
- Total trace chemicals within a payload container are limited to less than 5 weight percent.
- Trace chemicals that might be incompatible with minor and dominant materials/chemicals would have reacted during the waste treatment process prior to placement in payload containers.

GROUP NUMBER	GROUP NAME	EXAMPLE
1	Acids, Mineral, Non-Oxidizing	Hydrochloric Acid
2	Acids, Mineral, Oxidizing	Nitric Acid (>1%)
3	Acids, Organic	Acetic Acid
4	Alcohols and Glycols	Methanol
5	Aldehydes	Formaldehyde
6	Amides	Acetamide
7	Amines, Aliphatic and Aromatic	Aniline
8	Azo Compounds, Diazo Compounds and Hydrazines	Hydrazine
9	Carbamates	Carbaryl
10	Caustics	Sodium Hydroxide
11	Cyanides	Potassium Cyanide
12	Dithiocarbamates	Maneb
13	Esters	Vinyl Acetate
14	Ethers	Tetrahydrofuran
15	Fluorides, Inorganic	Potassium Fluoride
16	Hydrocarbons, Aromatic	Toluene
17	Halogenated Organics	Carbon Tetrachloride
18	Isocyanates	Methyl Isocyanate
19	Ketones	Acetone
20	Mercaptans and other Organic Sulfides	Carbon Disulfide
21	Metals, Alkali and Alkaline Earth, Elemental	Metallic Sodium

 TABLE 1

 EPA LIST OF CHEMICAL GROUPS AND MATERIALS*

*Modified from "A Method for Determining the Compatibility of Hazardous Wastes," Reference 3.1.

<u>NOTE</u>: The chemical groups and materials listed in this table are a comprehensive listing of chemical compounds that may be incompatible. This is not meant to infer that all the listed chemical compounds and materials are present in TRU waste.

TABLE 1
EPA LIST OF CHEMICAL GROUPS AND MATERIALS*
(CONTINUED)

GROUP NUMBER	GROUP NAME	EXAMPLE
22	Metals, other Elemental and Alloys in the form of Powders, Vapors or Sponges	Titanium
23	Metals, other Elemental and Alloys as Sheets, Rods, Moldings, Drops, etc.	Aluminum
24	Metals and Metal Compounds, Toxic	Beryllium
25	Nitrides	Sodium Nitride
26	Nitriles	Acetonitrile
27	Nitro Compounds	Dinitrobenzene
28	Hydrocarbons, Aliphatic, Unsaturated	Butadiene
29	Hydrocarbons, Aliphatic, Saturated	Cyclohexane
30	Peroxides and Hydroperoxides Organic	Acetyl Peroxide
31	Phenols, Cresols	Phenol
32	Organophosphates, Phosphothioates, and Phosphodithioates	Malathion
33	Sulfides, Inorganic	Zinc Sulfide
34	Epoxides	Epoxybutane

^{*}Modified from "A Method for Determining the Compatibility of Hazardous Wastes," Reference 3.1.

<u>NOTE</u>: The chemical groups and materials listed in this table are a comprehensive listing of chemical compounds that may be incompatible. This is not meant to infer that all the listed chemical compounds and materials are present in TRU waste.

# TABLE 1 EPA LIST OF CHEMICAL GROUPS AND MATERIALS*

#### (CONTINUED)

GROUP NUMBER	GROUP NAME	EXAMPLE
101	Combustible and Flammable Materials, Miscellaneous	Cellulose
102	Explosives	Ammonium Nitrate
103	Polymerizable Compounds	Acrylonitrile
104	Oxidizing Agents, Strong	Hydrogen Peroxide
105	Reducing Agents, Strong	Metallic Sodium
106	Water and Mixtures Containing Water	Water
107	Water Reactive Substances	Sulfuric Acid (>70%)

*Modified from "A Method for Determining the Compatibility of Hazardous Wastes," Reference 3.1.

<u>NOTE</u>: The chemical groups and materials listed in this table are a comprehensive listing of chemical compounds that may be incompatible. This is not meant to infer that all the listed chemical compounds and materials are present in TRU waste.

• The waste is either solidified and immobilized (solidified materials) or present in bulk form as a solid (solid materials) with very little residual liquids. Any interactions in the waste would be limited to solid or solidified materials, which are in a stable form. In almost all cases, any possible reactions take place before the waste is generated in its final form.

Potential incompatibilities between minor and dominant compounds have been analyzed on a case-by-case basis. As mentioned earlier, some chemicals listed as being present in the waste would have reacted prior to becoming part of the waste. For example, a site listing a caustic (Group 10) and an acid (Group 1) in their waste has only the neutralized product present in an immobilized form. Further reactions of this type do not occur once the waste is generated in its final form.

Waste content codes that are allowed for transportation in the CNS 10-160B Cask are described in the Remote-Handled Transuranic Content Codes and Chemical Lists for BCL (Attachment A). Chemical incompatibilities do not exist in content codes listed in Attachment A. This has been ensured by a knowledge of the processes generating the wastes and the chemical compatibility analysis. The following section details the chemical compatibility analysis for the different content codes in Attachment A. This analysis includes the chemical compatibility of each content code (i.e., within individual waste containers) and the compatibility of the CNS 10-160B Cask payload during hypothetical accident condition (i.e., between individual waste canisters with different content codes).

#### 2.2 Chemical Compatibility of Each Content Code

Chemical compatibility has been analyzed for the different content codes based on the chemical constituents within each content code. Tables 2 through 6 list the chemicals present in each of the content codes in Attachment A.

Because the TRU waste from the sites is very well characterized, chemical reactions and incompatibilities are not anticipated in the final waste form. Waste is typically segregated prior to or upon placement in payload containers, and the segregated wastes are then assigned to different appropriate content codes. The similarity of wastes placed in a single container for a given content code reduces the potential for chemical reactions and incompatibilities. In addition, transportation requirements on the waste form prevent many potential incompatibilities. The restriction on the presence of free liquid requires liquid waste constituents to be solidified and/or present only in trace quantities. Prohibition of caustic substances requires all acids and bases to be neutralized prior to or in conjunction with solidification processes, further precluding potential reactivity of waste constituents. Furthermore, prior to transportation, the wastes have typically been emplaced in the payload containers for a sufficient period of time so that any potential reactions would have already taken place.

Despite the fact that chemical reactions and incompatibilities are not expected, the use of the EPA methodology allows for a complete analysis of all potential incompatibilities. For each content code, potential incompatible combinations were identified, and each of these was evaluated on a case-by-case basis. It was determined that the required processing, prior to placing the waste in the waste containers, and the final form of the waste, obviated the potential incompatibility in each case. Therefore, it was concluded that the chemicals/materials in each content code are chemically compatible. The detailed evaluation of each content code and resolution of the potential incompatibilities is described below.

Attachment A lists five content codes from BCL: BC 312A, BC 314A, BC 321A, BC 321B, and BC 322A. The chemicals present in these content codes are listed in Tables 2 through 6. A detailed chemical compatibility evaluation for each content code is presented below.

#### CONTENT CODE: BC 312A - SOLIDIFIED ORGANIC WASTE

<u>Brief Description</u> - This waste content code describes solidified organic and inorganic wastes, including acids, acid solutions, elemental mercury, hydraulic oil, waste water, sludge (sand and mixed fission products), glycols, oils, and alcohols solidified with Floor Dry. Small items such as tools may be included in the waste.

<u>Chemical Compatibility Analyses</u> - Eighteen potential chemical incompatibilities, listed in Table 7, were found in this content code. The first sixteen potential chemical incompatibilities involve the presence of either inorganic or organic acids with other waste constituents. Because all corrosives are prohibited in the waste, all acids will be neutralized prior to being placed in the waste. The restriction on free liquids also precludes any potential chemical incompatibilities involving acids, as all acids or acid solutions are solidified with Floor Dry. Furthermore, five of these first sixteen potential incompatibilities involve trace constituents, which are not a chemical compatibility concern, as previously stated. The remaining two potential incompatibilities involve water and mixtures containing water (Group 106), and the restriction on free liquids precludes these incompatibilities from being an issue in this waste form. Therefore, there are no incompatibilities for the transportation of this waste.

#### **CONTENT CODE: BC 314A - CEMENTED INORGANIC PROCESS SOLIDS**

<u>Brief Description</u> - The waste described by this content code includes slugs produced from dissolving fuel specimens in an acid solution that was then diluted several times and mixed with cement and allowed to solidify in foam cups. The slugs contain limited amounts of radionuclides from fuel because of the dilution. The waste matrix also includes Floor Dry added during packaging to absorb any water from condensation or dewatering.

<u>Chemical Compatibility Analyses</u> - Only one potential incompatibility, listed in Table 7, was identified for this content code. The combination of oxidizing mineral acids (Group 2) and water or mixtures containing water (Group 106) has the potential for generating heat. The restriction placed on free liquids, however, would preclude such a reaction from occurring. The acids would also be neutralized prior to placement in the payload container. Furthermore, both constituents are present only in trace concentrations, and therefore are not a significant concern. There are no incompatibilities for the transportation of this waste.

#### CONTENT CODE: BC 321A - SOLID ORGANIC WASTE

<u>Brief Description</u> - The waste in this content code may include combustible items such as cloth and paper products (e.g., from the cleanup of spills), rags, coveralls and booties, plastic, cardboard, rubber, wood, surgeon's gloves, and Kimwipes. The waste may also include filter waste (e.g., dry box filters, HEPA filters, and filter cartridges); noncombustible Benelex and Plexiglas neutron shielding, blacktop, concrete, dirt, and sand; leaded gloves and aprons comprised of Hypalon rubber and lead oxide impregnated neoprene; small amounts of metal waste; particulate and sludge-type organic process solids immobilized/solidified with Portland cement, vermiculite, Aquaset, or Petroset.

<u>Chemical Compatibility Analyses</u> - No potential chemical incompatibilities were identified for this content code.

#### CONTENT CODE: BC 321B - SOLID ORGANIC WASTE

<u>Brief Description</u> - The waste in this content code may include filter waste (e.g., pool filters), nuclear grade resin, resin bags, paper, Floor Dry bags, seals, hoses, valves, and clamps.

<u>Chemical Compatibility Analyses</u> - No potential chemical incompatibilities were identified for this content code.

#### CONTENT CODE: BC 322A - SOLID INORGANIC WASTE

<u>Brief Description</u> - The waste described by this content code consists primarily of glass and metal debris. Glass debris includes laboratory glassware, windows, leaded glass windows, and various glass apparatus. Metal items may include deteriorated berry cans, cable, wire, planchets, signs, valves, piping, strapping, tools, foils, sheeting, fixtures, equipment (e.g., pumps or motors that have had all oil or other free liquids removed to <1%), hardware (e.g., nuts, bolts, brackets), specimen vials, fuel rod cladding, metallurgical mounts, and lead lined tubing. Metals of construction include stainless steel, aluminum, iron, copper, lead, beryllium, and zirconium.

<u>Chemical Compatibility Analyses</u> - Two potential chemical incompatibilities, listed on Table 7, were identified for this code. Both potential incompatibilities involve the reaction of halogenated organic liquids (Group 17) with metals in various forms (Groups 22 and 23). In both cases, the potential reaction will not take place because the halogenated organics will be solidified to meet the restriction on free liquids in the payload container. Furthermore, the halogenated organics are found only in trace concentrations, and are therefore, not a significant chemical reaction concern. Therefore, there are no incompatibilities for the transportation of this waste.

#### 2.3 Chemical Compatibility During Hypothetical Accident Conditions

The hypothetical accident condition for chemical compatibility is defined as a situation where all the individual waste containers within the CNS 10-160B Cask are breached during an accident. The waste from individual waste containers is assumed to intimately mix, but there is no breaching of the CNS 10-160B Cask inner vessel containment boundary. The following analysis of chemical compatibility is extended across all BCL content codes.

As with the chemical compatibility analysis within each content code, chemical reactions and incompatibilities are not anticipated between content codes or waste forms. Restrictions on the presence of free liquid requires liquid waste constituents to be solidified and/or present only in trace quantities. Prohibition of caustic substances requires all acids and bases to be neutralized prior to or in conjunction with solidification processes, further precluding potential reactivity of waste constituents.

The database management program was used to evaluate the potential chemical incompatibility conditions that may occur during an accident. The analysis assumed that mixing of any one content code could occur with any other content code during the accident.

Twenty-five potential incompatibilities, listed on Table 8, were identified for BCL under the hypothetical accident conditions of mixing all site content codes. These exclude the potential incompatibilities that can be present within a content code, since these were covered in Section 2.2. Each of the potential incompatibilities was evaluated on a case-by-case basis. It was determined that the required processing, prior to placing the waste in the waste containers, and the final form of the waste, obviated the potential incompatibility in each case. Therefore, it was concluded that the chemicals/materials across the BCL content codes are chemically compatible for the transportation of waste from BCL. The detailed evaluation and resolution of the potential incompatibilities is described in subsequent text.

The first 21 potential incompatibilities involve either inorganic acids (Groups 1 and 2) or organic acids (Group 3). As all corrosives have been neutralized prior to placement in the payload container and all liquids are either solidified or meet the restriction on free liquids, these reactions cannot occur. Potential incompatibilities 22 and 23 involve the presence of halogenated organics (Group 17) which must be solidified or meet the restriction on free liquids. Furthermore, the halogenated organics are present only in trace quantities, and therefore would not be a reactivity concern even if present due to the low concentration. Potential incompatibilities 24 and 25, similarly, involve the presence of water or mixtures containing water, which would be solidified to meet transportation requirements, and would therefore not be available in a chemically reactive form. Therefore, none of the 25 potential incompatibilities is a concern for the transportation of BCL waste under hypothetical accident conditions.

This is the comprehensive chemical compatibility analysis for the waste content codes listed in Attachment A. This analysis demonstrates that only compatible content codes will be transported in the CNS 10-160B Cask.

#### **3.0 REFERENCES**

3.1 Hatayama, H. K., Chen, J.J., de Vera, E.R., Stephens, R.D., Storm, D.L., "A Method for Determining the Compatibility of Hazardous Wastes," EPA-600/2-80-076, EPA, Cincinnati, Ohio, 1980.

#### TABLE 2

#### BATTELLE COLUMBUS LABORATORIES LIST OF CHEMICALS AND MATERIALS IN RH-TRU WASTE CONTENT CODES

#### CONTENT CODE BC 312A

#### SOLIDIFIED ORGANIC WASTE

GROUP 1:	ACIDS, MINERAL, NON-OXIDIZING (Constituents reacted prior to loading in payload containers.)	D
GROUP 2:	ACIDS, MINERAL, OXIDIZING (Constituents reacted prior to loading in payload containers.)	D
GROUP 3:	ACIDS, ORGANIC (Constituents reacted prior to loading in payload containers.)	D
GROUP 4:	ALCOHOLS AND GLYCOLS	D
GROUP 22:	METALS, OTHER ELEMENTAL AND ALLOYS, IN THE FORM OR SPONGES	OF POWDERS, VAPORS,
	ALUMINUM ZIRCONIUM	T T
GROUP 23:	METALS, OTHER ELEMENTAL AND ALLOYS, AS SHEETS, ROETC.	DDS, MOLDINGS, DROPS
	ALUMINUM COPPER IRON LEAD STAINLESS STEEL ZIRCONIUM	T T T T T
GROUP 24:	METALS AND METAL COMPOUNDS, TOXIC	
	BERYLLIUM COPPER LEAD MERCURY ZIRCONIUM	T T D T

D = Dominant Component (>10% by wt.)

- M = Minor Component (1 10% by wt.)
- T = Trace Component (<1% by wt.)
- T1 = Trace Component (<0.1% by wt.)
- T2 = Trace Component (low ppm range)
- T3 = Trace Component (<1 ppm range)

## TABLE 2

## (CONTINUED)

#### BATTELLE COLUMBUS LABORATORIES LIST OF CHEMICALS AND MATERIALS IN RH-TRU WASTE CONTENT CODES

#### CONTENT CODE BC 312A

#### SOLIDIFIED ORGANIC WASTE (CONTINUED)

GROUP 101:	COMBUSTIBLE AND FLAMMABLE MATERIALS, MISCELLA		
	HYDRAULIC OIL OIL	D D	
GROUP 106:	WATER AND MIXTURES CONTAINING WATER		
	ACID SOLUTIONS	D	
OTHER INORGANIC			
	SAND	D	
OTHER SOLID	IFICATION MATERIAL/ABSORBENTS		
	DIATOMACEOUS EARTH (Floor Dry)	D	

- T1 = Trace Component (<0.1% by wt.)
- T2 = Trace Component (low ppm range)
- T3 = Trace Component (<1 ppm range)

D = Dominant Component (>10% by wt.)

M = Minor Component (1 - 10% by wt.)

T = Trace Component (<1% by wt.)

#### TABLE 3

#### BATTELLE COLUMBUS LABORATORIES LIST OF CHEMICALS AND MATERIALS IN RH-TRU WASTE CONTENT CODES

#### CONTENT CODE BC 314A

#### CEMENTED INORGANIC PROCESS SOLIDS

GROUP 2:	ACIDS, MINERAL, OXIDIZING (Constituents reacted prior to loading in payload containers.)	
	NITRIC ACID	T
GROUP 106:	WATER AND MIXTURES CONTAINING WATER	
	WATER	Т
OTHER INORG	ANIC	
	CEMENT SLUGS	D
OTHER SOLIDI	FICATION MATERIAL/ABSORBENTS	
	DIATOMACEOUS EARTH (Floor Dry)	D

D = Dominant Component (>10% by wt.)

M = Minor Component (1 - 10% by wt.)

T = Trace Component (<1% by wt.)

T1 = Trace Component (< 0.1% by wt.)

T2 = Trace Component (low ppm range)

T3 = Trace Component (<1 ppm range)

#### TABLE 4

#### BATTELLE COLUMBUS LABORATORIES LIST OF CHEMICALS AND MATERIALS IN RH-TRU WASTE CONTENT CODES

#### CONTENT CODE BC 321A

#### SOLID ORGANIC WASTE

GROUP 22: METALS, OTHER ELEMENTAL AND ALLOYS IN THE FORM OF POWDERS, VAPORS, OR SPONGES

ALUMINUM	Т
ZIRCONIUM	Т

GROUP 23: METALS, OTHER ELEMENTAL AND ALLOYS AS SHEETS, RODS, MOLDINGS, DROPS, ETC.

Т
Т
D
Т
Т
Т

GROUP 24: METALS AND METAL COMPOUNDS, TOXIC

LEAD	Т
ZIRCONIUM	Т

D

D

D

D

D

D

D

D

D

D

D

GROUP 101: COMBUSTIBLE AND FLAMMABLE MATERIALS, MISCELLANEOUS

BENELEX BLACKTOP (ASPHALT) CARDBOARD CELLULOSICS CLOTH FILTERS KIMWIPES NEOPRENE PAPER PLASTIC PLEXIGLAS

- T = Trace Component (<1% by wt.)
- T1 = Trace Component (<0.1% by wt.)
- T2 = Trace Component (low ppm range)
- T3 = Trace Component (<1 ppm range)

D = Dominant Component (>10% by wt.)

M = Minor Component (1 - 10% by wt.)

## TABLE 4

#### (CONTINUED)

#### BATTELLE COLUMBUS LABORATORIES LIST OF CHEMICALS AND MATERIALS IN RH-TRU WASTE CONTENT CODES

#### CONTENT CODE BC 321A

#### SOLID ORGANIC WASTE (CONTINUED)

## GROUP 101: COMBUSTIBLE AND FLAMMABLE MATERIALS, MISCELLANEOUS (CONTINUED)

	RUBBER	D
	WOOD	D
OTHER	NORGANIC	
	CONCRETE	Т
	FILTERS	D
	GLASS	D
	SOIL	Т
OTHER	ORGANIC	
	FILTERS	D
	NEOPRENE	D
	SOIL	Т
OTHER	SOLIDIFICATION MATERIAL/ABSORBENTS	
	AQUASET	D
	DIATOMACEOUS EARTH (Floor Dry)	D
	PETROSET	D
	PORTLAND CEMENT	D
	RADSORB	D

VERMICULITE

D

D = Dominant Component (>10% by wt.)

M = Minor Component (1 - 10% by wt.)

T = Trace Component (<1% by wt.)

T1 = Trace Component (<0.1% by wt.)

T2 = Trace Component (low ppm range)

T3 = Trace Component (<1 ppm range)

#### TABLE 5

#### BATTELLE COLUMBUS LABORATORIES LIST OF CHEMICALS AND MATERIALS IN RH-TRU WASTE CONTENT CODES

#### CONTENT CODE BC 321B

#### SOLID ORGANIC WASTE

GROUP 22: METALS, OTHER ELEMENTAL AND ALLOYS IN THE FORM OF POWDERS, VAPORS, OR SPONGES

ALUMINUM	Т
ZIRCONIUM	Т

GROUP 23: METALS, OTHER ELEMENTAL AND ALLOYS AS SHEETS, RODS, MOLDINGS, DROPS, ETC.

ALUMINUM	Т
CARBON STEEL	Т
IRON-BASED METAL/ALLOYS	D
LEAD	Т
STAINLESS STEEL	Т
ZIRCONIUM	Т

GROUP 24: METALS AND METAL COMPOUNDS, TOXIC

LEAD	Т
ZIRCONIUM	Т

GROUP 101: COMBUSTIBLE AND FLAMMABLE MATERIALS, MISCELLANEOUS

CELLULOSICS	D
FILTERS	D
ION EXCHANGE RESIN	D
PAPER	D
PLASTIC	D
RESIN BAGS	Μ
RUBBER	D

T1 = Trace Component (<0.1% by wt.)

T3 = Trace Component (<1 ppm range)

D = Dominant Component (>10% by wt.)

M = Minor Component (1 - 10% by wt.)

T = Trace Component (<1% by wt.)

T2 = Trace Component (low ppm range)

D

D

D D

## TABLE 5

#### (CONTINUED)

#### BATTELLE COLUMBUS LABORATORIES LIST OF CHEMICALS AND MATERIALS IN RH-TRU WASTE CONTENT CODES

#### CONTENT CODE BC 321B

#### SOLID ORGANIC WASTE (CONTINUED)

OTHER INORGA	ANIC
	FILTERS
OTHER ORGAN	IC
	FILTERS
OTHER SOLIDIFICATION MATERIAL/ABSORBENTS	
	DIATOMACEOUS EARTH (Floor Dry) RADSORB

D = Dominant Component (>10% by wt.)

M = Minor Component (1 - 10% by wt.)

T = Trace Component (<1% by wt.)

T1 = Trace Component (<0.1% by wt.)

T2 = Trace Component (low ppm range)

T3 = Trace Component (<1 ppm range)

#### TABLE 6

#### BATTELLE COLUMBUS LABORATORIES LIST OF CHEMICALS AND MATERIALS IN RH-TRU WASTE CONTENT CODES

#### CONTENT CODE BC 322A

#### SOLID INORGANIC WASTE

GROUP 17: HALOGENATED ORGANICS

CARBON TETRACHLORIDE	Т
1,1,1-TRICHLOROETHANE	Т
TRICHLOROETHYLENE	Т

GROUP 22: METALS, OTHER ELEMENTAL AND ALLOYS, IN THE FORM OF POWDERS, VAPORS, OR SPONGES

ALUMINUM	D
ZIRCONIUM	D

GROUP 23: METALS, OTHER ELEMENTAL AND ALLOYS, AS SHEETS, RODS, MOLDINGS, DROPS, ETC.

ALUMINUM	D
COPPER	D
IRON	D
IRON-BASED METAL/ALLOYS	D
LEAD	D
STAINLESS STEEL	D
ZIRCONIUM	D

GROUP 24: METALS AND METAL COMPOUNDS, TOXIC

BERYLLIUM	D
COPPER	D
LEAD	D
ZIRCONIUM	D
COPPER LEAD ZIRCONIUM	D D D

D = Dominant Component (>10% by wt.)

M = Minor Component (1 - 10% by wt.)

T = Trace Component (<1% by wt.)

T1 = Trace Component (<0.1% by wt.)

T2 = Trace Component (low ppm range)

T3 = Trace Component (<1 ppm range)

## TABLE 6

## (CONTINUED)

#### **BATTELLE COLUMBUS LABORATORIES** LIST OF CHEMICALS AND MATERIALS IN RH-TRU WASTE CONTENT CODES

#### CONTENT CODE BC 322A

#### SOLID INORGANIC WASTE (CONTINUED)

#### OTHER INORGANICS

CEMENT GLASS	D D
OTHER SOLIDIFICATION MATERIAL/ABSORBENTS	
DIATOMACEOUS EARTH (Floor Dry)	D

DIATOMACEOUS EARTH (Floor Dry)

D = Dominant Component (>10% by wt.)

M = Minor Component (1 - 10% by wt.)

T = Trace Component (<1% by wt.)

T1 = Trace Component (<0.1% by wt.)

T2 = Trace Component (low ppm range)

T3 = Trace Component (<1 ppm range)

#### TABLE 7

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## SUMMARY OF POTENTIAL INCOMPATIBILITIES FOR BCL WASTE CONTENT CODES

Content Code	Potential Chemical Compatibility Reaction	Concentration of Reactants*	Reaction Code #
BC 312A	Acids, Mineral, Non-Oxidizing x Alcohols and Glycols (1 x 4)	D x D	Н
	Acids, Mineral, Non-Oxidizing x Metals, Other Elemental and Alloys as Powders, Vapors, or Sponges (1 x 22)	D x T	GF H F
	Acids, Mineral, Non-Oxidizing x Metals, Other Elemental and Alloys as Sheets, Rods, Drops, Moldings, etc. (1 x 23)	D x T	GF H F
	Acids, Mineral, Non-Oxidizing x Metals and Metal Compounds, Toxic (1 x 24)	D x D	S
	Acids, Mineral, Non-Oxidizing x Combustible and Flammable Materials, Miscellaneous (1 x 101)	D x D	H G
	Acids, Mineral, Non-Oxidizing x Water and Mixtures Containing Water (1 x 106)	D x D	Н
	Acids, Mineral, Oxidizing x Acids, Organic (2 x 3)	D x D	G H
	Acids, Mineral, Oxidizing x Alcohols and Glycols (2 x 4)	D x D	ΗF
	Acids, Mineral, Oxidizing x Metals, Other Elemental and Alloys as Powders, Vapors, or Sponges (2 x 22)	D x T	GF H F
	Acids, Mineral, Oxidizing x Metals, Other Elemental and Alloys as Sheets, Rods, Drops, Moldings, etc. (2 x 23)	D x T	GF H F
	Acids, Mineral, Oxidizing x Metals and Metal Compounds, Toxic (2 x 24)	D x D	S
	Acids, Mineral, Oxidizing x Combustible and Flammable Materials, Miscellaneous (2 x 101)	D x D	H F GT

## TABLE 7(CONTINUED)

## SUMMARY OF POTENTIAL INCOMPATIBILITIES FOR BCL WASTE CONTENT CODES

Content Code	Potential Chemical Compatibility Reaction	Concentration of Reactants*	Reaction Code #		
BC 312A (Continued)	Acids, Mineral, Oxidizing x Water and Mixtures Containing Water (2 x 106)	D x D	Н		
	Acids, Organic x Alcohols and Glycols (3 x 4)	D x D	НР		
	Acids, Organic x Metals, Other Elemental and Alloys as Powders, Vapors, or Sponges (3 x 22)	D x T	GF		
	Acids, Organic x Metals and Metal Compounds, Toxic (3 x 24)	D x D	S		
	Metals, Other Elemental and Alloys as Powders, Vapors, or Sponges x Water and Mixtures Containing Water (22 x 106)	ΤxD	GF H		
	Metal Compounds, Toxic x Water and Mixtures Containing Water (24 x 106)	D x D	S		
BC 314A	Acids, Mineral, Oxidizing x Water and Mixtures Containing Water (2 x 106)	ТхТ	Н		
BC 321A	No Potential Chemical Reactions Identified				
BC 321B	No Potential Chemical Reactions Identified				
BC 322A	Halogenated Organics x Metals, Other Elemental and Alloys as Powders, Vapors, or Sponges (17 x 22)	ΤxD	ΗE		
	Halogenated Organics x Metals, Other Elemental and Alloys as Sheets, Rods, Drops, Moldings, etc. (17 x 23)	T x D	H F		
* Concentration of Reactants: $T = Trace$ (<1% by wt.), $T1 = Trace$ (<0.1%), $T2 = Trace$ (low ppm range), $T3 = Trace$ (<1 ppm range), $M = Minor$ (1-10%), and $D = Dominant$ (>10%).					

[#] Reaction Code: H = heat generation, S = solubilization of toxic substances, F = fire, GF = flammable gas generation, G = non-flammable gas generation, P = violent polymerization, GT = toxic gas generation, E = explosion.

#### 4.10.2.1-C-22

#### TABLE 8

## SUMMARY OF POTENTIAL INCOMPATIBLE CONDITIONS FOR THE HYPOTHETICAL ACCIDENT FOR BCL WASTE CONTENT CODES

Case History No.	Content Codes	Potential Chemical Compatibility Reaction	Concentration of Reactants*	Reaction Code #
1	BC 312A	Acids, Mineral, Non-Oxidizing (Group 1)	D	
	BC 322A	Alcohols and Glycols (Group 4)	Т	Н
2	BC 312A	Acids, Mineral, Non-Oxidizing (Group 1)	D	
	BC 322A	Halogenated Organics (Group 17)	Т	H GT
3	BC 312A	Acids, Mineral, Non-Oxidizing (Group 1)	D	
	BC 322A	Ketones (Group 19)	Т	н
4	BC 312A	Acids, Mineral, Non-Oxidizing (Group 1)	D	
	BC 321A BC 321B BC 322A	Metals, Other Elemental and Alloys as Powders, Vapors, or Sponges (Group 22)	T T D	GF H F
5	BC 312A	Acids, Mineral, Non-Oxidizing (Group 1)	D	
	BC 321A BC 321B BC 322A	Metals, Other Elemental and Alloys as Sheets, Rods, Drops, Moldings, etc. (Group 23)	D T D	GF H F
6	BC 312A	Acids, Mineral, Non-Oxidizing (Group 1)	D	
	BC 321A BC 321B BC 322A	Metals and Metal Compounds, Toxic (Group 24)	T T D	S
7	BC 312A	Acids, Mineral, Non-Oxidizing (Group 1)	D	
	BC 321A BC 321B	Combustible and Flammable Materials, Miscellaneous (Group 101)	D D	H G
# TABLE 8

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# (CONTINUED)

# SUMMARY OF POTENTIAL INCOMPATIBLE CONDITIONS FOR THE HYPOTHETICAL ACCIDENT FOR BCL WASTE CONTENT CODES

Case History No.	Content Codes	Potential Chemical Compatibility Reaction	Concentration of Reactants*	Reaction Code #
8	BC 312A	Acids, Mineral, Non-Oxidizing (Group 1)	D	
	BC 314A	Water and Mixtures Containing Water (Group 106)	Т	Н
9	BC 314A	Acids, Mineral, Oxidizing (Group 2)	Т	
	BC 312A	Acids, Organic (Group 3)	D	GH
10	BC 312A BC 314A	Acids, Mineral, Oxidizing (Group 2)	D T	
	BC 312A BC 322A	Alcohols and Glycols (Group 4)	D T	H F
11	BC 312A BC 314A	Acids, Mineral, Oxidizing (Group 2)	D T	
	BC 322A	Hydrocarbons, Aromatic (Group 16)	Т	H F
12	BC 312A BC 314A	Acids, Mineral, Oxidizing (Group 2)	D T	
	BC 322A	Halogenated Organics (Group 17)	Т	H F GT
13	BC 312A BC 314A	Acids, Mineral, Oxidizing (Group 2)	D T	
	BC 322A	Ketones (Group 19)	Т	H F
14	BC 312A BC 314A	Acids, Mineral, Oxidizing (Group 2)	D T	
	BC 312A BC 321A BC 321B BC 322A	Metals, Other Elemental and Alloys as Powders, Vapors, or Sponges (Group 22)	T T D	GF H F

4.10.2.1-C-24

# TABLE 8

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#### (CONTINUED)

# SUMMARY OF POTENTIAL INCOMPATIBLE CONDITIONS FOR THE HYPOTHETICAL ACCIDENT FOR BCL WASTE CONTENT CODES

Case History No.	Content Codes	Potential Chemical Compatibility Reaction	Concentration of Reactants*	Reaction Code #
15	BC 312A	Acids, Mineral, Oxidizing (Group 2)	D	
	BC 314A		Т	
	BC 312A	Metals, Other Elemental and Alloys as	Т	GF H F
	BC 321A	Sheets, Rods, Drops, Moldings, etc.	D	
	BC 321B	(Group 23)	D	
	BC 322A		D	
16	BC 312A	Acids, Mineral, Oxidizing (Group 2)	D	
	BC 314A		Т	
	BC 312A	Metals and Metal Compounds, Toxic	D	S
	BC 321A	(Group 24)	Т	
	BC 321B		Т	
	BC 322A		D	
17	BC 312A	Acids, Mineral, Oxidizing (Group 2)	D	
	BC 314A		Т	
	BC 312A	Combustible and Flammable Materials,	D	H F GT
	BC 321A	Miscellaneous (Group 101)	D	
	BC 321B		D	
18	BC 312A	Acids, Mineral, Oxidizing (Group 2)	D	
	BC 314A		Т	
	BC 312A	Water and Mixtures Containing Water	D	Н
	BC 314A	(Group 106)	Т	
19	BC 312A	Acids, Organic (Group 3)	D	
	BC 322A	Alcohols and Glycols (Group 4)	Т	НР
20	BC 312A	Acids, Organic (Group 3)	D	
	BC 321A	Metals, Other Elemental and Alloys as	Т	GF
	BC 321B	Powders, Vapors, or Sponges (Group 22)	Т	
	BC 322A		D	

4.10.2.1-C-25

# TABLE 8

#### (CONTINUED)

SUMMARY OF POTENTIAL INCOMPATIBLE CONDITIONS FOR THE HYPOTHETICAL AC	CIDENT
FOR BCL WASTE CONTENT CODES	

Case History No.	Content Codes	Potential Chemical Compatibility Reaction	Concentration of Reactants*	Reaction Code #
21	BC 312A	Acids, Organic (Group 3)	D	
	BC 321A	Metals and Metal Compounds, Toxic	Т	S
	BC 321B	(Group 24)	Т	
	BC 322A		D	
22	BC 322A	Halogenated Organics (Group 17)	Т	
	BC 312A	Metals, Other Elemental and Alloys as	Т	ΗE
	BC 321A	Powders, Vapors, or Sponges (Group 22)	Т	
	BC 321B		Т	
23	BC 322A	Halogenated Organics (Group 17)	Т	
	BC 312A	Metals, Other Elemental and Alloys as	Т	H F
	BC 321A	Sheets, Rods, Drops, Moldings, etc.	D	
	BC 321B	(Group 23)	D	
24	BC 312A	Metals, Other Elemental and Alloys as	Т	
	BC 321A	Powders, Vapors, or Sponges (Group 22)	T	
	BC 321B		T	
	BC 322A		D	
	BC 312A	Water and Mixtures Containing Water	D	GF H
	BC 314A	(Group 106)	Т	
25	BC 312A	Metal Compounds, Toxic (Group 24)	D	
	BC 321A	L · · · · · · · · · · · · · · · · · · ·	Т	
	BC 321B		Т	
	BC 322A		D	
	BC 312A	Water and Mixtures Containing Water	D	S
	BC 314A	(Group 106)	Т	

* Concentration of Reactants: T = Trace (<1% by wt.), T1 = Trace (<0.1%), T2 = Trace (low ppm range), T3 = Trace (<1 ppm range), M = Minor (1-10%), and D = Dominant (>10%).

# Reaction Code: H = heat generation, S = solubilization of toxic substances, F = fire, GF = flammable gas generation, G = non-flammable gas generation, P = violent polymerization, E = explosion, GT = toxic gas generation.

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Attachment D

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Shipping Period for TRU Waste in the 10-160B Cask

#### **1.0 INTRODUCTION**

This Attachment presents the basis for the shipping period for Battelle RH-TRU wastes from the time of cask closure until cask opening. This shipping period is used in the analysis of the gas generation in the 10-160B cask.

Two different shipping plans are expected for the Battelle wastes. In Plan A, wastes will be shipped in the 10-160B cask from the Battelle West Jefferson, OH site to the DOE Hanford site for interim storage. When the WIPP site is ready to receive RH-TRU waste, the stored waste will be shipped from Hanford to WIPP. In Plan B, RH-TRU wastes will be shipped directly to the WIPP from Battelle with no interim storage. While the shipments are in transit, under either plan, a satellite tracking system will be operational to monitor progress and provide direct communication between the driver and the transport dispatcher.

#### 2.0 EXPECTED SHIPPING PERIOD

The expected shipping period is the amount of time from the sealing of the cask at the loading facility until the opening of the cask at the unloading facility. It consists of : the time from cask sealing to the release of the transport unit from the loading facility, the expected transit time, and the time from arrival at the unloading facility until the cask is opened. For assessing the expected shipping period, it will be assumed that there are no delays.

2.1 Loading

The loading process from cask sealing to unit release includes health physics surveys, installing the upper impact limiter, and vehicle inspections. The time from cask sealing until the unit is released for travel has been accomplished in less than four (4) hours. To be conservative, a one-day (24 hour) duration will be assumed.

2.2 Transit

The longest route for either Plan A or Plan B is from Battelle to Hanford, 2275 miles. All Battelle RH-TRU shipments will be made with two drivers. Using two drivers, on an appropriate rotational schedule, the truck can travel for twenty-four (24) hours per day for up to seven days. Assuming an average speed of 45 mph, which includes time for vehicle inspections, fueling, meals, and driver relief, the duration of the trip is expected to be 51 hours. Again, to be conservative, the transit duration will be assumed to be three days (72 hours).

2.3 Unloading

The unloading process includes receipt survey and security checks, positioning of the trailer in the RH unloading area, removal of the cask from the trailer to a transfer cart, positioning of the cask in the cask unloading room, and removal of the lid. This process has been accomplished in less than eight (8) hours. Again, to be conservative, the unloading duration will be assumed to be one day (24 hours).

2.4 Total

The total expected shipping period, with no delays, is less than 65 hours. For the purpose of this analysis, a conservative period of 5 days (120 hours) will be assumed.

#### 3.0 SHIPPING DELAYS

The maximum shipping time will be assumed to be the sum of the expected shipping time and the time for delays which could extend the shipping time. These delays are: loading delays; transit delays due to weather or road closures, shipping vehicle accidents, mechanical delays, or driver illness; and unloading delays. Each of these delays are assessed below.

3.1 Loading Delays

There are a number of situations that could extend the time between cask sealing and truck release. These include: loading preceding a holiday weekend, problems with a leak test, and handling equipment failure. Both the leak test problem and the handling equipment failure should be resolvable by replacing or obtaining temporary equipment. Each of these situations is unlikely to cause more than a two day delay. The holiday weekend could cause a delay of three days, i.e., from Friday afternoon until Tuesday. It is very unlikely that more than two of the three loading delays could occur on the same shipment, so a total of five days seems a reasonably conservative assessment for a loading delay.

#### 3.2 Transit Delays

Transit delays due to weather, e.g., a road closed due to snow, are unlikely to cause a delay of more than five days. A road closure due to a vehicle accident or a roadway or bridge failure would result in re-routing which could add up to two days to the transit time. A transit time delay due to weather or road closure will be assumed to be five days.

Transit delays due to an accident with the truck could cause a lengthy delay. Response time for notification and to take immediate corrective action is assumed to be one day. (The use of the on-board satellite communication system will facilitate an early response.) Accident mitigation may require transferring the cask to a different trailer using cranes and other heavy equipment. Mitigation is assumed to take five days for a total accident delay of six days.

Mechanical problems with the truck or trailer could also cause multi-day delays. Significant failures may require a replacement tractor or trailer. An appropriate response to a mechanical failure is assumed to take four days.

Driver illness could also cause transit delays. If a driver it too ill to continue, a replacement driver will be brought in. A two day delay is assessed for bringing in a replacement driver.

3.3 Unloading Delay

An unloading delay will occur if the truck arrives just before a holiday weekend. This could result in a four day delay. Additionally, a delay due to unloading equipment failure could occur. Repair of such equipment should not require more than four days. The unloading delay will be conservatively assumed to be five days. If an unanticipated situation occurs that would result in a much longer delay, the cask can be vented.

3.4 Total Delay

The total delay, i.e., the sum of the delay times for each of the delay types, is 27 days. This assumes that each type of delay occurs on the same shipment.

#### 4.10.2.1-D-3

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### MAXIMUM SHIPPING PERIOD

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The maximum shipping period, as the sum of the expected shipping period and the total delay, is 32 days. This period assumes that each of the possible shipping delays occurs on the same shipment, a very unlikely occurrence. Further, for additional conservatism, the assumed maximum will be nearly doubled to 60 days. Thus, a 60 day shipping period will be used in all analysis of gas generation in the sealed cask.