**Characterization of the Hanford 618-10 Burial Ground Trenches,
Washington, USA - 11119**

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ABSTRACT

The 618-10 Burial Ground is a solid waste landfill located on the Department of Energy (DOE) Hanford Site. The trenches associated with this burial ground are scheduled for remediation beginning in Fiscal Year 2011. To support the remediation effort, a quantitative assessment of the inventory of radioactive material in the trenches was performed using waste shipment records and exposure rate to activity modeling.

INTRODUCTION

The 618-10 Burial Ground, located on the Department of Energy (DOE) Hanford Site and managed by Washington Closure Hanford (WCH), is a solid waste landfill that operated from March 1954 to September 1963. Wastes received at the 618-10 Burial Ground were generated primarily by research facilities located in the Hanford 300 Area. Approximately 90% of the waste was generated in Buildings 325, 327, 329, and 3706.

The missions of these four buildings were as follows:

- 325 Building – Radiochemistry, development and improvement of chemical separations processes including REDOX and PUREX processes.
- 327 Building – Radiometallurgy, metallurgical examination of irradiated fuel rods and reactor hardware
- 329 building – Biophysics, counting of radiological samples take of air, vegetation, soil, wildlife, and water
- 3706 Building – Laboratory, development of separation processes

Lesser quantities of waste were generated in a number of other 300 Area buildings. A limited number of offsite sources also contributed waste to the 618-10 site.

The high-activity wastes included irradiated metallic uranium reactor fuel elements, plutonium, and high-concentration fission product streams from the development of radioisotope extraction and chemical separation processes such as the Plutonium Uranium Extraction (PUREX) process. Extremely high activity wastes were directed to underground standpipes called Vertical Pipe Units (VPUs). Other wastes were directed to open trenches in the burial ground.

The trenches in the 618-10 Burial Ground are scheduled for remediation beginning in Fiscal Year 2011. An accurate source term is a necessary starting point for remediation planning and safety basis development. Existing estimates of the inventory in the 618-10 trenches were based on operational history and personnel interviews, and were semi-quantitative in nature. An effort was initiated to develop a quantitative inventory to supplement the existing estimates.

A number of characterization methodologies were considered. The most promising methodology was to review the available documentation to determine the characteristics of the waste items disposed in the burial ground, utilize exposure rate to activity modeling to calculate the inventory in each waste item, and then sum the waste item inventories to determine the inventory in the burial ground.

REFERENCE REVIEW

Over 3700 radiation survey records (RSRs), monthly reports, and other technical documents were recovered from the site archives. Each reference was then reviewed in detail to determine the number and types of waste items that were disposed in the 618-10 burial grounds.

The reference review process also sought to determine the characteristics of each waste item, including information necessary to support exposure rate to activity modeling, such as the preferred exposure rate measurement, the distance associated with the preferred exposure rate measurement, the physical configuration when the exposure rate was measured, the radioactive material stream responsible for the measured exposure rate, the age of the radioactive material when the exposure rate was measured, and the physical composition of the non-radioactive material within each waste item.

References from the period of 618-10 operations, primarily RSRs, did not, in general, contain all of the information necessary to determine the number and type of the waste items disposed in the 618-10 burial ground, or to determine all of the characteristics necessary to support exposure rate to activity modeling. Instead, the reference reviewers had to infer the required information from what was recorded.

The exteriors of the waste containers were surveyed before the containers were transported to the 618-10 site. The actual contents of the containers are uncertain, but radiological survey records indicate the number of waste shipments and the types of containers used. Trenches generally received low-activity waste in cardboard boxes. Materials with higher radioactivity were packaged in concrete and lead-shielded drums. Contaminated materials were often carried to the burial ground in "load luggers," which could hold approximately 5.6 m^3 (200 ft^3) of loose waste.

The radioactivity of the waste disposed from the 325 and 327 Laboratory hot cells increased during the lifetime of 618-10 due to a general increase in the radioactivity of reactor fuel elements and laboratory samples. The use of a shielded cask to transport waste containers from the 327 Building for disposal to the VPUs began in late 1955. This cask was eventually replaced with more heavily shielded casks known as the "milk pail" cask and the "gatling gun" cask in mid to late 1959. Containers were remotely loaded into lead-shielded casks for transport to the burial grounds. The waste was remotely released from the casks to the VPUs.

A team of reviewers was needed given the volume of references. To help ensure more consistent results from the team, a set of desk instructions for reference review were developed and made available to all members. The instructions provided reference reviewers with guidance on how to infer waste item numbers and characteristics from the limited information typically available. The instructions were maintained in draft format during much of the reference review process, and served as a location to store the experience accumulated by the team for use by all interested parties.

DATABASE

A Microsoft Access® database application was designed to store and organize the substantial volume of information generated by the reference review process. One of the primary goals of the database application was to ensure traceability of the information collected, primarily to support third party review and independent verification. Some of the design criteria incorporated into the database are listed below.

- The database was designed to store both the references and the data for waste items associated with each reference.
- The database was designed to indicate the disposal location (VPUs or the trenches).
- All references were scanned in Adobe Acrobat® portable document format (PDF) and stored in a subfolder of the database folder to simplify retrieval.
- Links were entered into the database to allow the PDF file of a reference to be opened from any page that contained information extracted from the reference.
- The database was structured to allow each reference to be associated with more than one waste item, and each waste item to be associated with more than one reference.
- A flexible category system was incorporated to allow waste items to be grouped together in any meaningful way.
- The database was structured to allow each waste item to be associated with multiple categories.

Initially, the database allowed for each waste item to be associated with a single radioactive waste stream responsible for the measured exposure rate. However, during the reference review process, it became clear that waste items could be associated with more than one waste stream, and that some waste streams would not have a measurable exposure rate. Therefore, the database was modified to allow for a single waste stream associated with the measured exposure rate, and multiple other streams that do not contribute to the measured exposure rate.

For example, there were a few waste items with measurable exposure rates attributable to fission product wastes that also contained plutonium spill cleanup materials with negligible exposure rates. The multiple waste stream structure allowed the exposure rates to be used to quantify the fission products, and the plutonium to be quantified by direct mass entries. There were also hundreds of drums of depleted uranium disposed in 618-10 with no measurable exposure rate where the uranium was quantified by direct mass entries.

It also became clear during the reference review process that a means was needed to allow for both source splitting and overpacking operations. There were numerous instances where a large waste container was generated, but only the exposure rate from the worst case item placed in the waste container was recorded. There were also instances where an increment of radioactive waste was split into multiple shielded disposal containers after the exposure rate was measured.

To allow for these situations, a multiplier field was added to the database. The multiplier field was designed to multiply the quantity of radioactive material determined by exposure rate to activity modeling. The multiplier was set to 1 by default, but could be set to greater than 1 to account for overpacking or less than one to account for splitting.

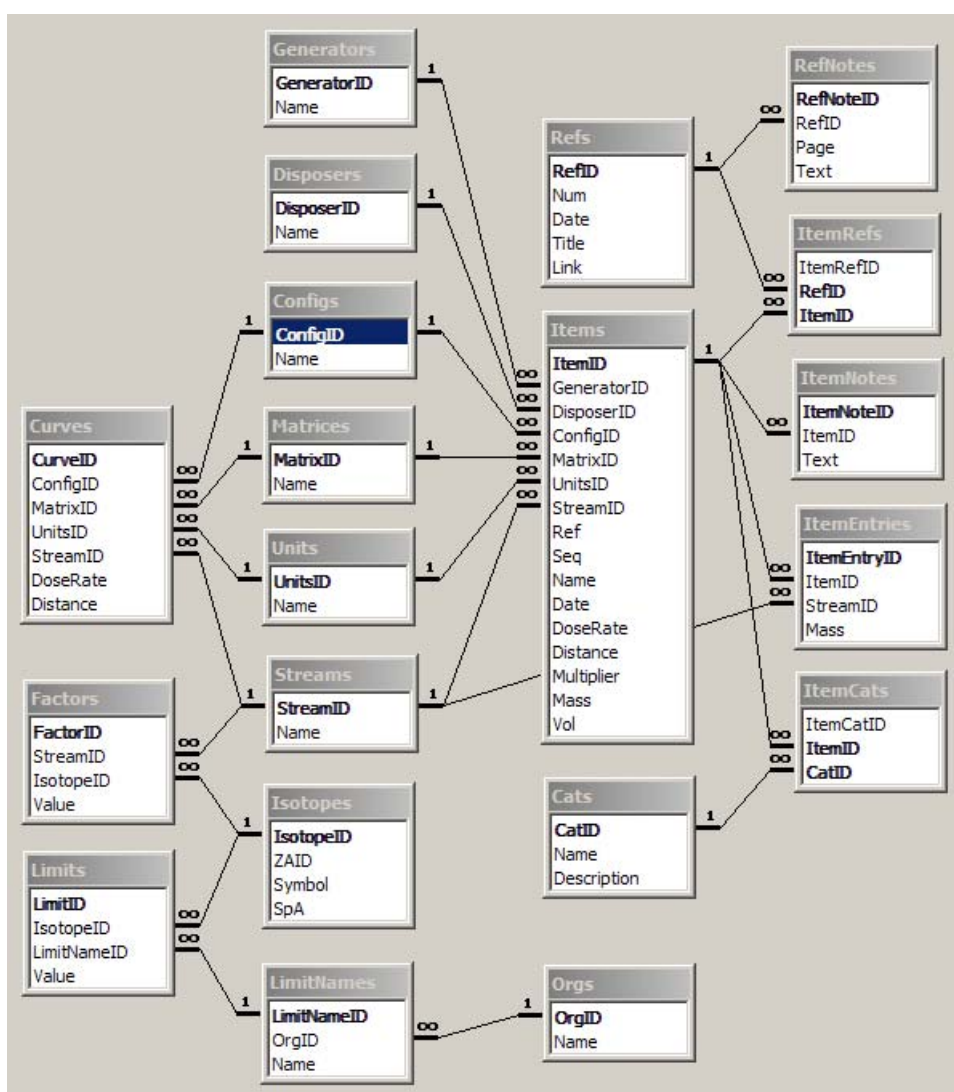The structure of the database is shown in Figure 1.

Figure 1. Database Structure

N REACTOR FUEL DATA

Much of the high-activity waste disposed in the 618-10 burial grounds contained irradiated reactor fuel from the Hanford single-pass reactors, or was derived from single-pass reactor fuel. Unfortunately, complete composition information was not available for single-pass reactor fuel. Instead, composition information for N Reactor fuel was substituted for single-pass reactor fuel. The most authoritative source of composition information for N Reactor fuel was a series of Oak Ridge Isotope Generation (ORIGEN) calculations documented in PNNL-11555 [1].

To support waste stream development, N Reactor fuel data was needed in electronic format suitable for constructing ORIGEN input files. Unfortunately, the specially-compiled burnup-dependent irradiation cross-sections used to generate PNNL-11555 were no longer available. Consequently, it was no longer possible to model the irradiation of relatively simple initial fuel compositions to re-create the post-irradiation data in PNNL-11555. Instead, the necessary post-irradiation data was extracted from scanned hardcopies and checked by hand.

The ORIGEN input files in PNNL-11555 utilized the CUT card to limit the size of the output by ignoring low mass or low activity isotopes. As a result, neither the mass or activity outputs in PNNL-11555 contained a complete composition. The mass and activity outputs were extracted, and then merged into a single vector that gave the most complete possible fuel composition.

As plutonium recycle and breeder programs were introduced, interest in high-exposure fuel continued to build at the Hanford site, until by 1960 a significant portion the single-pass reactor capacity was committed to producing high-exposure fuel. Therefore, both weapons grade (6% by weight Pu-240) and high-exposure (12% by weight Pu-240) compositions were extracted from PNL-11555.

Once extracted, N reactor fuel data was entered into ORIGEN input files to support waste stream development.

WASTE STREAMS

For the purposes of this effort, waste streams were defined as sources of radioactive material with established isotopic ratios affected only by radioactive decay. A separate ORIGEN input file was generated for each waste stream.

For some streams, the ORIGEN input file used a known initial composition, such as the fuel compositions from PNNL-11555 or natural uranium and thorium. Other waste streams were formed by starting with the reactor fuel compositions and then modeling different reprocessing steps using the capability built into ORIGEN. Each reprocessing model was supplied with user-defined elemental efficiencies to produce the required end product.

The initial composition was decayed to the age when the dose rate was measured, as well as to September 30, 2010 to reflect the current composition. The waste streams are shown in Table I.

Table I. Waste Streams

Name	Description	Ref
Activated Graphite	Irradiated graphite from the single-pass reactor blocks	[2]
Activated Metal	Cladding and supports from reactor fuel irradiated to 6% Pu-240	[1]
Fuel, Weapons Grade	Uranium metal reactor fuel irradiated to 6% Pu-240	[1]
Fuel, High Exposure	Uranium metal reactor fuel irradiated to 12% Pu-240	[1]
Pu, Weapons Grade	Plutonium extracted from reactor fuel irradiated to 6% Pu-240	[1]
Pu, High Exposure	Plutonium extracted from reactor fuel irradiated to 12% Pu-240	[1]
PUREX 1WW	Fission and activation products associated with irradiated fuel with most of the actinides removed by reprocessing	[1]
Strontium	Strontium extracted from reactor fuel irradiated to 6% Pu-240	[1]
Thorium	Thorium metal refined from natural ore	[3]
U, Depleted	Uranium with reduced U-235, a by-product of the enrichment process	[4]
U, Natural	Uranium that has not been enriched	[4]

WASTE AGE

The photon spectrum generated by a radioactive material will change in both total photon rate and energy distribution as the material decays. Consequently, the age of a radioactive material must first be determined before the exposure rate can be accurately modeled.

For some materials, the exposure rate may increase significantly during the initial phase of radioactive decay due to the in-growth of certain daughter isotopes. Eventually, the exposure rate for all materials will decrease with age. When used in exposure rate to activity modeling, a longer waste age is generally conservative.

A single waste age was suitable for most waste streams. Typically, the waste was assumed 0.50 years old at the time of exposure rate measurement based on general practices at the 300 Area laboratories. However, for the thorium stream a waste age of 5.0 years was used based on a description of the generating process [3].

A single waste age was found to be overly conservative for irradiated fuel research conducted at the Hanford 325 and 327 buildings. Fuel that had been discharged only a few days was regularly examined at the 327 building as part of efforts to reduce the rate of fuel failures. Some of the green fuel was also dissolved and sent to the 325 building for burn-up and plutonium yield analyses. Therefore, waste ages of 0.25 and 0.50 years were used for the reactor fuel waste streams to provide a wider range of waste ages.

PHOTON SPECTRA

The initial composition of each waste stream was decayed to the age when the exposure rate was measured. At the end of this decay step, the photon spectrum for the material was calculated by ORIGEN. The photon spectra were used in Monte Carlo N-Particle (MCNP) models to calculate the exposure rate as a function of distance from each waste item.

Programs such as Microshield are able to calculate photon spectra, but only from radioactive decay processes. ORIGEN, on the other hand, also calculates photons generated by internal and external bremsstrahlung, spontaneous fission, and alpha-neutron reactions. In recently irradiated reactor fuel, all of these processes are significant. Therefore, ORIGEN was used to calculate all photon spectra.

The photon spectrum generated by bremsstrahlung is dependent on the elemental composition of the radioactive material. ORIGEN has separate libraries for water and uranium oxide. The water library produces exposure rates approximately 15% below the uranium oxide library, which is conservative for exposure rate to activity calculations. The ORIGEN photon library for water was used for all waste streams.

MCNP MODELS

MCNP models were used to calculate the exposure rate as a function of distance from each waste item. A separate MCNP model was generated for each unique combination of configuration, waste stream, waste age, and waste matrix. A total of 134 separate models were required.

The configuration captures the size, shape, and materials of construction of the container. The waste stream and waste age determine the photon rate and energy spectrum generated by the contents of the container. The waste matrix determines the self-shielding associated with the waste item. Together, these three parameters uniquely determine the exposure rate case.

Each MCNP input file was normalized to1 gram of the waste stream uniformly distributed over the payload volume of the container. Therefore, dividing the measured exposure rate by the results from the MCNP model yields the mass of waste in grams directly.

Each MCNP input file was used to calculate the exposure rate along the axis of the payload at intervals from 5.08 cm (2 in) to 304.80 cm (10 ft) from the surface of the package. Exposure rates were calculated using point detectors positioned at the centroid of the detector. All exposure rate measurements were assumed to have been measured by a Hanford CP (Cutie Pie) instrument, which had a cylindrical detector 7.62 cm (3 in) in diameter and 15.24 cm (6 in) long. The Hanford CP was the most commonly used exposure rate instrument during the period of 618-10 operations.

In the waste shipment records, the distance associated with an exposure rate measurement is the straight-line distance between the front edge of the instrument detector and the surface of the

item. Therefore the point detectors were located at the exposure rate measurement distance plus one half of the detector length.

TALLY CONVERSION FACTOR

MCNP tallies are reported in units of particle fluence, while exposure rates are recorded in units of Roentgen per hour. Often MCNP calculations utilize fluence-to-dose factors to convert tally results to dose, and then assume that 0.01 Sv (1 rem) of dose is equivalent to one Roentgen of exposure. Use of this assumption introduces an error that may be significant depending on the source of the conversion factors used and the photon spectrum of the material in question. For photons from Cs-137, the errors are shown in Table I.

Table I. Fluence to Dose Errors

Fluence-Dose Factors	Error
ANSI/ANS-6.1.1-1977 [5]	15.6%
ICRP 21 [6]	-1.6%
ANSI/ANS-6.1.1-1991 [7]	-7.0%
ICRP 51 [8]	-7.0%

The tally multiplier (FM) card provided in MCNP may be used to convert fluence tallies directly to Roentgen, thereby avoiding the errors introduced by using fluence-to-dose conversion factors. The FM card approach was used for 618-10 characterization to improve accuracy. The factor required by the FM card is derived as follows.

MCNP solves radiation transport problems by generating simulated particles with randomly-selected properties, and then propagating the particles through the physical model. The event history experienced by each particle is recorded by MCNP and used to calculate the contribution of the particle to the total fluence. The total fluence is then divided by the number of histories to calculate the particle fluence per history, which is reported as the tally (T) value. A tally is essentially the fluence generated by an "average" particle. By simulating many particles, the tally may be determined with increasing accuracy.

A tally may be multiplied by the number of histories per unit time to convert from a per history basis to a per unit time basis. This may be done using the FM card constant, or by starting each particle with a weight (WGT) value equal to the particle rate per unit time. It is convenient to use the WGT parameter for this conversion because it is problem dependent, while most other conversion factors are problem independent.

$$\text{T} \cdot \text{WGT} = \left(\frac{\text{particles}}{\text{cm}^2 \cdot \text{history}} \right) \cdot \left(\frac{\text{histories}}{\text{sec}} \right) = \frac{\text{particles}}{\text{cm}^2 \cdot \text{sec}} \qquad \text{(Eq. 1)}$$

A tally may be multiplied by the energy-dependent total reaction cross-section (σ_t) to convert to total reaction rate. Cross-sections in MCNP are microscopic quantities, and are recorded in units of barns. To obtain the macroscopic reaction rate in a material, the tally must also be multiplied by the atom density of the material (ρ_a). In MCNP, atom densities are expressed in units of barn-centimeters to simplify the multiplication.

$$T \cdot WGT \cdot \sigma_t \cdot \rho_a = \left(\frac{particles}{cm^2 \cdot history} \right) \cdot \left(\frac{histories}{sec} \right) \cdot (b) \cdot \left(\frac{atoms}{b \cdot cm} \right) = \frac{reactions}{cm^3 \cdot sec} \qquad \text{(Eq. 2)}$$

A reaction rate may be multiplied by the energy-dependent heating cross-section (σ_h) to convert to energy absorbed. Heating cross-sections are stored in units of MeV per collision. Since dosimetric quantities are stated in terms of energy absorbed per gram of material, the reaction rate may also be divided by the mass density (ρ_m) of the material in grams per cubic centimeter.

$$T \cdot WGT \cdot \sigma_t \cdot \sigma_h \cdot \left(\frac{\rho_a}{\rho_m} \right) = \left(\frac{particles}{cm^2 \cdot history} \right) \cdot \left(\frac{histories}{sec} \right) \cdot (b) \cdot \left(\frac{MeV}{collision} \right) \cdot \left(\frac{atoms}{b \cdot cm} \right) \cdot \left(\frac{cm^3}{g} \right) = \left(\frac{MeV}{g \cdot sec} \right) \qquad \text{(Eq. 3)}$$

The energy absorbed may be converted to Roentgen (R) if air is the absorbing material. The energy absorbed may be divided by the energy required to produce an ion pair in air (W) to convert to charge deposited. The charge deposited may be divided by the definition of a Roentgen to convert to exposure in units of Roentgen. The time unit may also be converted to the more conventional hours. The value for the charge of an electron (e) was taken from CODATA 2006 [9], the value of W was taken from ICRU Report 31 [10], and the value of R was taken from NIST Special Publication 811 [11].

$$T \cdot WGT \cdot \sigma_t \cdot \sigma_h \cdot \left(\frac{\rho_a}{\rho_m \cdot W} \right) = \frac{\left(\frac{MeV}{g \cdot sec} \right) \cdot \left(\frac{cm^2}{b} \right) \cdot \left(\frac{e}{33.8 \; eV} \right) \cdot \left(\frac{10^6 \; eV}{MeV} \right) \cdot \left(\frac{1.602176487 \cdot 10^{-19} \; C}{e} \right) \cdot}{\left(\frac{R}{2.58 \cdot 10^{-4} \; C/kg} \right) \cdot \left(\frac{1000 \; g}{kg} \right) \cdot \left(\frac{3600 \; sec}{hr} \right)} = \frac{R}{hr} \qquad \text{(Eq.4)}$$

In the above equation, T is calculated by MCNP, WGT is supplied in the SDEF card, and the cross-sections are applied using the -5 and -6 reactions on the FM card. The remaining quantities constitute the constant C needed for the FM card to complete the conversion to Roentgen per hour.

$$C = \frac{\rho_a}{\rho_m \cdot W} \qquad \text{(Eq. 5)}$$

The quantity (ρ_a/ρ_m) is simply the Avogadro constant (N_{av}) divided by the formula mass of the material in question and converted to the appropriate units. The formula mass of the material may be calculated from the composition and density of the material, and the associated elemental masses. The formula mass calculation for air is shown in Table II. The elemental masses were taken from NUBASE 2003 [12], and the composition and density of air were taken from PNNL-15870 [13].

Table II. Formula Mass of Air

Air Composition		Element Mass (g/mol)	Formula Mass (g/mol)
Element	Mass Fraction		
C	0.000124	12.010736	0.001489
N	0.755268	14.006743	10.578845
O	0.231781	15.999405	3.708358
Ar	0.012827	39.947677	0.512409
Total	1.000000		14.801101

Atom densities in MCNP stated are in units of atoms per barn centimeter, and mass densities in MCNP are stated in units of grams per cubic centimeter, so the quantity (ρ_a/ρ_m) must be stated in units of square centimeters per barn gram and not simply inverse grams. The conversion factor necessary to calculate exposure in units of Roentgen per hour from a fluence tally is given by:

$$\frac{\rho_a}{\rho_m} = \left(\frac{6.02214179 \cdot 10^{23} \text{ atoms}}{\text{mol}} \right) \cdot \left(\frac{\text{mol}}{14.801101 \text{ g}} \right) \cdot \left(\frac{10^{-24} \text{ cm}^2}{\text{b}} \right) = 4.069 \cdot 10^{-2} \frac{\text{cm}^2}{\text{b} \cdot \text{g}} \quad \text{(Eq. 6)}$$

$$C = \frac{\left(4.069 \cdot 10^{-2} \frac{\text{cm}^2}{\text{b} \cdot \text{g}} \right) \cdot \left(\frac{1000 \text{ g}}{\text{kg}} \right) \cdot \left(\frac{e}{33.8 \text{ eV}} \right) \cdot \left(\frac{10^6 \text{eV}}{\text{MeV}} \right)}{\left(\frac{1.602176487 \cdot 10^{-19} \text{ C}}{e} \right) \cdot \left(\frac{R}{2.58 \cdot 10^{-4} \text{ C/kg}} \right) \cdot \left(\frac{3600 \text{ sec}}{\text{hr}} \right)} = 2.691 \cdot 10^{-6} \left(\frac{\text{sec}}{\text{MeV}} \right) \cdot \left(\frac{\text{cm}^2}{\text{b}} \right) \cdot \left(\frac{R}{\text{hr}} \right) \quad \text{(Eq. 7)}$$

INTERPOLATION

The waste shipment records contain exposure rates measured at a wide range of distances. The only effective way of dealing with the range of measurement distances was to create a series of exposure rate versus distance curves calculated by MCNP, and then interpolating the exposure rate value for a particular measurement distance.

Because Microsoft Access does not include built-in interpolation routines, specific code methods were written to perform interpolation. A standard log-log interpolation routine was prepared for the database in Visual Basic for Applications using a bracket and bisect search algorithm. The methods were placed in a module built into the database application.

INVENTORY CALCULATIONS

Inventory calculations were performed in four separate steps. First, the mass of waste in each waste item was calculated from the measured exposure rate, measurement distance, and waste item characteristics. Next, the waste stream masses for each item were summed by waste mass for all items. The summed waste stream masses were then multiplied by the composition of each waste stream to calculate individual isotope inventories. Finally, the isotope inventories were summed to provide a single vector of isotope activities for the 618-10 burial ground.

A Structured Query Language (SQL) query was constructed to calculate the mass of the waste stream in each waste item. The appropriate exposure rate versus distance curve was obtained based on the configuration, waste stream, waste age, and waste matrix. The curve was provided to the interpolation routine along with the measured exposure rate and distance. The exposure rate for 1 gram of waste at the actual measurement distance was calculated by interpolation from the exposure rate versus distance curve. The measured exposure rate was then divided by the exposure rate for 1 gram to determine the mass of waste. Since this calculation was quite time consuming, the calculated mass was stored in the database, and the database was designed to update the value whenever one of the waste item characteristics was changed.

A second SQL query was constructed to sum the waste stream masses. A union query was used to combine waste stream masses calculated from exposure rate measurements with those entered by the user for waste streams without a measurable exposure rate. A third SQL query was constructed to multiply the summed waste stream masses by the waste stream compositions and sum the results by isotope. Both of these queries were designed to run as needed and produce electronic or hard copy reports for use by interested parties.

EXAMPLE

An example reference is shown in Figure 2. The number of waste items and characteristics were determined as follows:

RSR Information	Interpretation
Waste bottles were placed into multiple drums	Two waste items added to the database
Waste was generated in room 4-7A of the 325 building	Waste items were assumed to be placed in 208 L (55 gal) drums shielded with concrete, and were assumed disposed in the 618-10 trenches
Waste originated in the 325 building	Radioactive waste stream was assumed to be irradiated reactor fuel
Reference was dated prior to 1960	Reactor fuel was assumed irradiated to a weapons grade burnup as opposed to a high exposure burnup
Exposure rate on the waste bottles was relatively low	Age of the waste was assumed to be 0.50 years
Exposure rate on the waste bottles was relatively low	Assumed that 5 waste bottles were placed in each shielded drum. Therefore each waste item was assigned a multiplier of 5
Size of the waste bottles was not specified	Volume or each bottle assumed to be 1 liter
Waste originated in the 325 building laboratories	Non-radioactive material in the waste bottles was assumed to be water

Once the determination of waste item number and characteristics was complete, the measured dose rate was converted to waste mass as follows:

- The exposure rate versus distance curve was obtained from the MCNP results for a radioactive waste stream of 0.50 year old fuel irradiated to a weapons grade exposure, dissolved in a water matrix, and placed in a 1 liter metal can. The exposure rate curve is shown in Table III.

Table III. Exposure Rate vs Distance

Exposure Rate (mR/hr/g waste)	Distance (in)
1.779E+03	2
7.246E+02	6
3.039E+02	12
1.045E+02	24
3.123E+01	48
1.478E+01	72
8.563E+00	96
5.585E+00	120

- Since the measured dose rate of 30 mR per hour was obtained at a distance of 5.08 cm (2 in), the interpolated conversion factor was 1.779E+03 mR per hour per gram waste. Since 5 waste bottles were assumed loaded into each shielded drum, the total mass of waste in each shielded drum is given by:

$$5 \cdot \left(30 \ \frac{mR}{hr} \right) \div \left(1.779 \cdot 10^3 \ \frac{mR/hr}{g \ waste} \right) = 8.430 \cdot 10^{-2} \ g \ waste \qquad \text{(Eq. 9)}$$

The exposure rate to waste mass calculation was time consuming, primarily due to the effort required to lookup the appropriate exposure rate versus distance curve and to perform the distance interpolation step. Therefore, the resulting waste mass was saved to the database, and the database was designed to recalculate the waste mass if any one of the key waste item characteristics were changed.

Once the waste mass has been calculated, a number of useful manipulations may be readily performed. For example, the quantity of waste could be multiplied by isotopic composition factors for weapons grade fuel taken from the ORIGEN calculations to determine the isotopic activities on September 30, 2010. Specific activities from ORIGEN may also be used to determine the isotopic masses on September 30, 2010. The results for individual waste items may also be readily summed to generate the inventory in the 618-10 trenches.

The isotopic activities and masses, for individual items or in the trenches as a whole, may then be compared to facility safety limits, transportation limits, and waste acceptance criteria as appropriate to assist in remediation planning.

Figure 2. Example Reference

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