Radiological Characterization Issues and Success Stories for the Melton Valley Scrap Yard and Homogeneous Reactor Experiment Evaporator Response Actions

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ABSTRACT

Proper characterization is essential for assuring that wastes meet any treatment or disposal site waste acceptance criteria (WAC). In addition characterization is vital for proper preparation of shipping papers for transporting the waste for treatment or disposal. Process knowledge was inadequate for characterization of legacy waste items from two sites involved in the Melton Valley Decontamination and Decommissioning (MVD&D) project at the Department of Energy's (DOE) Oak Ridge National Laboratory (ORNL): a scrap yard containing miscellaneous contaminated items, and a liquid waste evaporator used by an experimental nuclear reactor. Waste items at both sites were contaminated with Cs-137/Ba-137m. Through the use of scaling factors, the gamma radiation associated with this radionuclide pair was used to characterize these items for proper disposal. Application of scaling factors permitted successful remediation of these sites on an accelerated schedule and avoided radiation exposure concerns associated with traditional sampling and analysis.

INTRODUCTION

This paper describes the characterization approach for waste produced from the remediation of two response action sites in the Melton Valley area of the ORNL. One site contained equipment and metal items which were expected to be reused, scrapped, or salvaged. Because of many factors, not the least of which was lack of documentation, it was decided to dispose of the items present at this site as radioactive waste [1]. The other site contained a liquid waste evaporator that had last been used in the early 1960s. There was no documentation of likely contents of the evaporator and its ancillary piping, and high dose rates precluded customary sampling and analysis. Both the 7841 Scrap Yard waste and Homogeneous Reactor Experiment (HRE) Evaporator (see Figure 1) were required to be remediated in accordance with the Melton Valley Record of Decision [2] and resulting waste disposed by the end of Fiscal Year 2006 in accordance with the Bechtel Jacobs Company LLC completion contract with DOE. Characterization activities commenced in earnest in December 2005 and both projects were completed by September 2006.

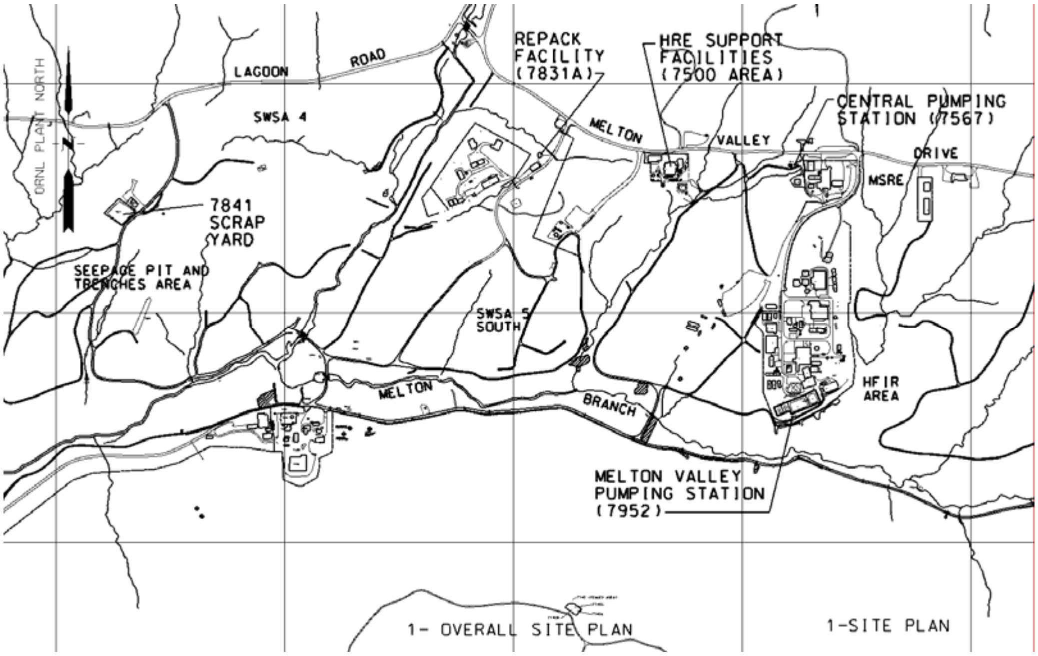

Fig. 1. Location of the HRE and 7841 Scrap Yard Facilities at ORNL.

7841 SCRAP YARD

The ORNL 7841 Scrap Yard consisted of a gravel pad (52 m x 55 m [170 ft × 180 ft]) with a 2.4-m-high (8-ft) security fence surrounding the area, which is approximately 2830 square meters (0.7 acres). It was established when DOE and its contractors began to seek recycle/reuse opportunities for assets such as steel boxes, carriers, and other scrap metal of real or material value. The exact start of the 7841 Scrap Yard is somewhat uncertain, however, it is known to be around 1990. Facility use had been mostly restricted to container and equipment storage. A photograph of the facility is included as Figure 2.

Fig. 2. Aerial view of the 7841 Scrap Yard Facility.

While the initial intent of the facility was to stage scrap metal material for resale and/or reuse, at some point the facility required management and oversight for its radiological content, so the material became defined as low level waste (LLW). The wastes generated by remediation of the Scrap Yard consisted of hundreds of waste items, many of which were disposed off-site at the Energy Solutions disposal facility in Clive, Utah. A subpopulation of 220 items was disposed at the Environmental Management Waste Management Facility (EMWMF) located in Bear Creek Valley on the Oak Ridge Reservation. Waste items included the following:

- 81 items of general equipment (e.g., tables, cabinets, boxes, trailers, cylinder overpacks, duct sections, dumpsters, glove boxes, fume hoods, pipes);
- 64 tanks;
- 57 carriers (e.g., boxes, cylinders, casks);
- 7 drums;
- 5 high integrity containers (HICs);
- 4 shields (e.g., boxes, blocks); and
- 2 miscellaneous (bag house exhaust unit and metal ring).

In addition this waste lot contained minor volumes of the following waste forms necessary to complete the response action:

- miscellaneous small volumes of wood;

- secondary waste such as personal protective equipment (PPE) generated during the execution of this response action;
- very minor amounts of concrete and/or soils generated as part of the Scrap Yard cleanup; and
- miscellaneous debris generated during the response action.

The waste forms varied considerably in size and dimension. As the waste items were inspected and assessed against the EMWMF physical WAC, a determination was made to either crush or shred the waste item with readily available field equipment used for that purpose. Alternately, many waste items were filled with grout or other flowable non-compressible material to mitigate void space.

Little knowledge of radionuclide content of waste lot items existed prior to the year 2000. The initial characterization of these waste items used data obtained in the latter part of that year from non-destructive assay (NDA) (using High-Purity Germanium (HPGe) detectors) for gamma-ray-emitting radionuclides. The activity of radionuclides associated with each item measured by NDA was estimated by correcting the detected gamma emission rate for geometrical loss, matrix attenuation, and radionuclide branching ratios. Thus, NDA measurement results were normalized across varying waste item geometries (e.g., drums, carriers, boxes, etc.). Also, as an independent check to test for the presence of alpha-emitting TRU isotopes, confirmatory passive neutron measurements were conducted on 28 items, and no indications of the presence of alpha-emitting TRU isotopes were observed [3, 4, 5].

The bullets below summarize the data relationship between the waste lot of 220 items and the Phase II field NDA investigation [3, 4, 5].

- 135 items (61%) – Phase II field screening determined items had no significant gamma-ray (NSG) activity (i.e., activity less than three times background);
- 54 items (25%) – Phase II field screening determined items had significant activity (i.e., activity greater than three times background), all items subsequently measured by NDA; and
- 31 items (14%) – No measurement data.

An actual minimum detectable activity was not calculated for each of the 135 items that were determined to have no significant activity [3, 4, 5] because it would have been inefficient to perform this calculation for each geometry of the items surveyed. For characterization purposes, the project conservatively assumed that the median Cs-137 activity for the 54 items with NDA results represented the activity on each of the 135 NSG items. Activities of other radionuclides were scaled from this Cs-137 value.

The 2000 scanning of the exterior of items could have detected any major concentrations of gamma-ray emitters (e.g., sources remaining in carriers). However, low levels of residual contamination inside the items could have been undetected by the screening. Specifically, it was recognized that some radionuclides which are not gamma-ray emitters (e.g., Sr-90) or which emit only a weak gamma or X ray (e.g., Pu-239) could be present and escape detection by HPGe detectors. Hence, selected items from this group were smear sampled to confirm that this characterization was adequate and scaling factors were needed to relate the non-gamma-ray-emitters to those measured by NDA.

None of the 31 items for which no NDA or screening data exists were sampled for isotopic analysis. These items were assumed to be consistent with the characterization that existed for the rest of the waste lot, which were confirmed or adjusted based on results of the sampling and analysis developed for the project. These 31 items, as all items in the waste lot, were subjected to the inspection requirements of the Waste Anomaly Detection Plan developed by the project to ensure no free liquid or other anomalous conditions existed that would have been undetected by NDA measurement.

Scaling factors had been developed for radionuclides contained in the supernatant and sludge associated with the liquid low-level waste (LLLW) system at ORNL [6]. The LLLW system serves, or served, all major radionuclide producers or generators at ORNL for decades. Generators had to meet a WAC to discharge into the system. Hence, any radionuclide of significance that was generated by, or passed through, ORNL would end up in the LLLW system and specifically sludge generated by the system. Use of these scaling factors for Scrap Yard items provided a reasonable and conservative approach to include all radionuclides that might possibly be present in or on those items. Scaling factors are appropriate only if they relate radionuclide activity to a dominant gamma-ray emitter [7]. Cs-137/Ba-137m was the dominant gamma-ray-emitting radionuclide measured during NDA of Scrap Yard items and was also the dominant gamma-ray emitter in the LLLW system.

The LLLW consisted of sludge and supernatant components. The maximum ratios for each radionuclide in sludge were pooled with the maximum ratios for supernatant into a single value that became the scaling factors used for Scrap Yard items. Table I provides the scaling ratios determined through this exercise. Scaling could then be done directly from the Cs-137 measured during NDA. This mitigated the need to scale from calculated dose rates for various item geometries; the geometry and dose rate issues had been incorporated into the NDA measurements.

Scaling performed in this manner provided a bounding condition (upper 95% confidence level) for radionuclide inventories in or on Scrap Yard items. Sampling and analysis of representative items using an approved Sampling and Analysis Plan (SAP) [8] was used to confirm the bounding nature of the characterization of the 7841 Scrap Yard items. Because of the nature of the items being disposed, it was anticipated that analytical results could show anomalous concentrations of some radionuclides. At least two of the 220 Scrap Yard items (~ 1%) were shown to contain radionuclide inventories (primarily Pu-239) outside the predicted bounds. These items were eventually disposed using the radionuclide inventories estimated by specifically sampling and analyzing smears of the items. Use of scaling factors to estimate radionuclide inventories saved a substantial amount of time in performing the Scrap Yard response action. Representative sampling and analysis insured that items sent for disposal were indeed within the WAC limits of the disposal facility [9].

Table I. Radionuclide ratios (maximums) used for scaling factors for WL 149.9

Radionuclide	Sludge	Supernatant	Pooled
Am-241	8.25E-02	1.58E-02	1.91E-02
Am-243		1.58E-05	1.58E-05
C-14		1.16E-04	1.16E-04
Cm-242		4.19E-06	4.19E-06
Cm-244	8.25E-01	2.11E-01	2.44E-01
Co-60	3.19E-01	8.89E-02	1.02E-01
Cs-134	3.76E-02	6.39E-01	6.40E-01
Cs-137	1.00E+00	1.00E+00	1.00E+00
Eu-152	5.81E+00	9.47E-01	1.18E+00
Eu-154	2.06E+00	8.11E-01	8.93E-01
Eu-155	5.63E-01	2.21E-01	2.44E-01
H-3		1.67E-04	1.67E-04
I-129	1.32E-07	2.64E-07	2.69E-07
Ni-63	8.13E-02		3.25E-03
Np-237	3.50E-04		1.40E-05
Pu-238	1.23E-01	1.05E-03	5.95E-03
Pu-239/Pu-240	7.00E-02	8.32E-04	3.63E-03
Pu-241	6.25E-01	6.21E-03	3.12E-02
Pu-242	2.07E-04	3.57E-06	1.18E-05
Pu-244		1.05E-06	1.05E-06
Ru-106/Rh-106	1.37E-01	1.61E+00	1.62E+00
Sr-90/Y-90	2.56E+01	9.16E-01	1.94E+00
Tc-99	3.75E-03	5.16E-03	5.31E-03
Th-232	3.00E-04		1.20E-05
U-232		4.25E-05	4.25E-05
U-233	5.31E-02	1.89E-02	2.11E-02
U-234	1.51E-03	2.84E-04	3.45E-04
U-235	7.52E-05	2.78E-05	3.08E-05
U-236		2.78E-05	2.78E-05
U-238	2.06E-03	5.26E-04	6.09E-04

7502 LIQUID WASTE EVAPORATOR

Bldg. 7502, Waste Evaporator, consisted of a reinforced concrete evaporator cell built in 1951 and the surrounding enclosure that was constructed in the mid-1950s. The evaporator cell inside dimensions were approximately 3.4 m by 3.4 m by 2.4 m (11 ft by 11 ft by 8 ft) with 0.9-m-thick (3-ft) reinforced concrete walls. Baryte was added to some of the concrete to increase the shielding properties of the concrete. The top of the cell consisted of four 0.6-m-thick (2-ft) reinforced concrete removable shield plugs. One shield plug was removed to gain access to the Waste Evaporator and associated piping. The shield plugs were not in the scope of this waste lot. Surface dose rates associated with the Waste Evaporator ranged from 1.2 mSv/hr (120 mrem/hr) at the upper access panel to 10 mSv/hr (1000 mrem/hr) at the lowest bend in the evaporator legs.

The Waste Evaporator itself was a 0.6-m (2-ft) diameter by 3-m (10-ft) long tank equipped with three evaporator legs. Other items associated with the evaporator included the evaporator steam coil, evaporator feed tank, condenser catch tank, waste evaporator header, entrainment separator, two shell and tube condensers, shielded sampler, metering tank, piping and valves, utility supply systems, support structure, shielding, and shelter structure.

Most of the radiological contamination in the Waste Evaporator cell resulted from operating the HRE-2 reactor and subsequent reclamation of uranium materials in the evaporator unit during the late 1950s and early 1960s. The evaporator unit was flushed after each reactor run with resulting liquid waste discharged to the sample sink. The Waste Evaporator was removed from service in 1961, but the equipment remained in the cell. As a result of these operations these waste items were contaminated primarily with uranium isotopes and fission products which had decayed for over 40 years. The concrete shielding surrounding the evaporator unit and making up the walls of the 7502 facility are contaminated as well.

During planning for D&D activities, high dose rates were encountered within the waste evaporator cell indicating characterization of the waste evaporator facility by a typical physical sampling approach would result in increased radiological exposure to workers. Therefore, the decision was made to characterize the waste evaporator and associated piping and equipment using a modeling approach requiring minimal physical sampling and analysis coupled with the use of scaling factors. Microshield® calculations were used with the results of radiological surveys and process knowledge to determine inventories of gamma-ray emitters in the evaporator system. A smear was collected from the waste evaporator sample sink, which is located outside of the evaporator cell and assumed to be representative of the worst case contaminants associated with the waste evaporator and associated piping. The sample sink was chosen for ALARA purposes (i.e., it was accessible with minimal dose rate to the person taking the sample) and because it was the drain for solutions/waste running through the evaporator system. Thus, its distribution of radionuclides was expected to be representative of the distribution of isotopes in the evaporator process equipment and piping. Analytical results were used to determine radionuclide activities and relative ratio distributions, and to develop scaling factors.

The data required to perform the Microshield® calculations included both field measurements (e.g., dose rate) and estimated quantities. Dose rate measurements were obtained using a Teletector™ instrument at the locations shown in Figure 3. Because no engineering drawings of the evaporator facility could be found, piping type, lengths, and diameters were estimated from the photograph and field observations. Based on the fuels used in the reactor experiments, the calculations assumed the gamma-ray emitters in the radioactive source material to be 100% Cs-137 (in equilibrium with Ba-137). A small amount of Co-60 was found in the analysis of the smear taken from the evaporator sample sink (0.036% of Cs-137 inventory). This small amount of Co-60 contributed insignificantly to calculated dose rates. Using Microshield®, a determination of source activity was computed for the evaporator body and each of the piping elements for which an identifiable dose rate location could be determined from photographs.

Microshield® calculations assumed two geometries for each item: annular cylinder and shielded slab, with the purpose of using the more conservative in the final characterization computations. The Teletector™ instrument was used to measure dose rate at various locations around the evaporator legs, evaporator body, and ancillary piping. Microshield® calculations of dose rate

Fig. 3. 7502 Waste evaporator and associated piping with measured doses (Surface/30cm).

were normalized to those obtained from survey results. A summary of results is given in Table II. Scaling factors from the sample sink smear and process knowledge were then used in conjunction with the normalized activity to characterize the radionuclide inventory in the waste evaporator and associated piping. Assumptions for Microshield® calculations were as follows:

- Evaporator is 0.6-m (2-ft) diameter by 3-m (10-ft) long.
- Evaporator shell is Schedule 30 stainless steel. Ancillary pipes, including legs, are Schedule 40 stainless steel.
- Pipe diameters and lengths are those determined from photographs.
- The thickness of deposits inside the evaporator and associated piping are approximately 2mm. Specific gravity of the deposits is approximately 2.2, the same as that found in the deposits elsewhere at ORNL on the Melton Valley Storage Tanks Dip Legs.
- For shielding calculations, the electron density of stainless steel is approximately the same as that of iron. Density of stainless steel is assumed to be 8 g/cm3.
- Dose rates are calculated at distances of 30 cm from the exterior surface of the pipe. Dose measurement locations along the pipes are approximated to the nearest foot.
- Measured dose rates are those obtained in the radiological survey. An area background of 1.0 mSv/h (100 mR/h) (as determined by the field health physicist) was subtracted from the 30-cm measurements.
- For the slab geometry, thickness of the source was 2 mm. Width of the source was 2/3 of the diameter of the pipe being modeled.
- For all pipes other than the evaporator, legs, pipe K, and pipe L, the dimensions were assumed to be the same as those for pipe L. It was assumed that the dose rate at the mid point of the pipe was 0.5 mSv/h (50 mR/h).

Table II. Microshield® Summary Results.

	Dose Rate (mSv/hr) [mR/h]			Implied Cs-137 (Bq) [Ci]			Inventory Estimates (Bq) [Ci]	
		Microshield®						
Pipe	Measured	Cylinder	Slab	Cylinder	Slab	Component	Minimum	Maximum
F (30-cm from end)	3.5 [350]	3.8 [380]	3.8 [380]	2.56E+10 [6.91E-01]	2.93E+10 [7.92E-01]	Pipe F	1.10E+10 [2.96E-01]	8.81E+10 [2.38E+00]
F (center)	1.5 [150]	4.5 [450]	5.5 [550]	9.25E+09 [2.50E-01]	8.70E+09 [2.35E-01]			
F (30-cm from end)	0.5 [50]	3.8 [380]	3.8 [380]	3.65E+09 [9.87E-02]	4.18E+09 [1.13E-01]			
C (end)	1 [100]	2.8 [280]	2.8 [280]	1.98E+09 [5.36E-02]	2.64E+09 [7.14E-02]	Pipes C&D	1.19E+10 [3.21E -01]	1.80E+10 [4.86E-01]
D (15-cm from end)	1.5 [150]	3.2 [320]	3.7 [370]	2.60E+09 [7.03E-02]	3.00E+09 [8.11E-02]			
A (30-cm from end)	0.5 [50]	0.5 [50]	0.5 [50]	1.23E+10 [3.32E-01]	1.57E+10 [4.25E-01]			
A (center)	0.5 [50]	0.6 [60]	0.7 [70]	1.02E+10 [2.77E-01]	1.12E+10 [3.03E-01]	Pipe A	1.02E+10 [2.77E-01]	1.57E+10 [4.25E-01]
K	7 [700]	4.2 [420]	3.9 [390]	1.23E+10 [3.33E-01]	1.33+10 [3.59E-01]	Pipe K	1.23E+10 [3.33E-01]	1.33E+10 [3.59E-01]
L	4 [400]	3.9 [390]	3.7 [370]	1.14E+10 [3.08E-01]	1.20E+10 [3.24E-01]	Pipe L	1.14E+10 [3.08E-01]	1.20E+10 [3.24E-01]
Remaining[a]	0.5 [50]	3.9 [390]	3.7 [370]	1.42E+09 [3.85E-02]	1.50E+09 [4.05E-02]	Remainder	2.28E+10 [6.15E-01]	2.40E+10 [6.49E-01]
						Total	7.96E+10 [2.15E+00]	1.71E+11 [4.62E+00]

[a]Assumed 0.5 mSv/h (50 mR/h), length and diameter same as L

A waste profile was developed and approved enabling disposal of the waste evaporator in the EMWMF. After characterization and profile development, the project successfully addressed unique challenges in anomaly detection and transporting the evaporator to EMWMF in compliance with all State and Federal requirements.

CONCLUSION

Scaling factors provide a means to characterize a number of unique items and/or heterogeneous waste using appropriately applied measurements and or assumptions with the following primary benefits:

- Conservative and bounding characterization
- Fully consistent with ALARA
- Reducing the time and cost of traditional sampling and analysis techniques.

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