Final Status Survey Report for the Diamond Ordnance Radiation Facility (DORF)

By:

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Report No. 2008012/G-103343
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SIGNATURES

The undersigned certify that they have reviewed and provided comments on the enclosed report that was prepared for the investigation of the Diamond Ordnance Radiation Facility (DORF) at the Walter Reed Army Medical Center, Forest Glen Annex, in Silver Spring, Maryland.[1]

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Francis M. Fota, LTC, MS Radiation Safety Officer Walter Reed Army Medical Center	5 Jan 2011 Date
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[1] The comment resolution record is shown in Appendix 12.1.

ACRONYMS AND ABBREVIATIONS

ARC - Army Reactor Council

ARL - Army Research Laboratory

ARO - Army Reactor Office

bgs - below ground surface

BRAC - Base Realignment and Closure

CFR - Code of Federal Regulations

CHP - Certified Health Physicist (American Board of Health Physics)

cpm - count per minute

CSM - Conceptual Site Model

DandD Code - A computer code, developed by the USNRC, for use in the screening phase of decommissioning

DCGL - Derived Concentration Guideline Level

DORF - Diamond Ordnance Radiation Facility

DP - Decommissioning Plan

dpm - disintegration per minute

DQO - Data Quality Objective

EMC - Elevated Measurement Criteria

FSS - Final Status Survey

GPS - Global Positioning System

IEM - Integrated Environmental Management, Inc.

LASP - IEM's Land Area Survey Program

LGBR - Lower Bound of the Gray Region

MARSSIM - Multi-Agency Radiation Survey and Site Investigation Manual

MDA - Minimum Detectable Activity

MDC - Minimum Detectable Concentration

MDE - Maryland Department of the Environment

mrem - Millirem

NIST - National Institute of Standards and Technology

PCB - polychlorinated biphenyls

pCi - picocurie

QA - Quality Assurance

QC - Quality Control

Reg. Guide - U. S. Nuclear Regulatory Commission Regulatory Guide

ROC - Radionuclide of Concern

RRPT - Registered Radiation Protection Technologist (National Registry of Radiation Protection Technologists

SAP - Sampling and Analysis Plan

SSHO - Site Safety and Health Officer

SSP - Site-Wide Survey Plan for Expedited Release

SVOC - semi-volatile organic compounds

TEDE - Total Effective Dose Equivalent

Th - Thorium

U - Uranium

µg - microgram

USACE - U. S. Army Corps of Engineers

USNRC - U. S. Nuclear Regulatory Commission

UST - Underground Storage Tank

VOC - volatile organic compounds

VSP - Visual Sample Plan

WMP - Waste Management Plan

WRAMC - Walter Reed Army Medical Center

OVERVIEW

The Diamond Ordnance Radiation Facility (DORF) was operated by the Department of the Army's Harry Diamond Laboratories (HDL) at the Walter Reed Army Medical Center's (WRAMC's) Forest Glen Annex. The U. S. Army Corps of Engineers (USACE) was tasked by the Army to conduct decommissioning studies of the DORF for the purpose of assessing the status of the Army Reactor Permit (No. DORF-1-97). In addition, the studies were to be designed to ensure the collected data could be used to support the release of the DORF from U. S. Nuclear Regulatory Commission (USNRC) License No. 08-01738-02, issued to WRAMC.

Integrated Environmental Management, Inc. (IEM) was contracted by the USACE to provide radiological support under Contract No. W912DR-D-0022, "Multiple Award Radiological Services (Environmental Services & Environmental Remediation Services)". Delivery Order No. 0001 of that contract is to assist with the decommissioning of the DORF. The scope of work for the Delivery Order is the performance of an investigation at the DORF in order to secure the data and information needed to evaluate/select decommissioning options and prepare a site-wide decommissioning plan.[2]

The IEM Team prepared a Project Planning Package that was reviewed and approved for implementation by the USACE and applicable stakeholders.[3] The approach used for the effort was to acquire measurement data of sufficient quantity and quality to meet the requirements for final status survey. If the results of the investigation show the release criteria could be met, that survey area would then be eligible for expedited release without the need for additional data acquisition. The Sampling and Analysis Plan (SAP) set the following requirements for release of a survey unit: (1) an adequate number of measurements are made; (2) the mean of the sample distribution does not exceed the applicable release criteria; (3) no further statistical tests are necessary and (4) the level of survey coverage is sufficient to meet elevated measurement criteria requirements if applicable.[4]

The final status survey data acquired at the DORF demonstrate that there is no residual radioactivity (total or removable) in excess of the conservatively-derived USNRC default screening criteria within OUs 2 (Warm Room), 3 (Connector Room), 4 (main floor of Building 516), 5 (Building 513), 6 (former UST location), 7 (Outdoor Area inside fence), 8 (Outdoor Area outside fence), 10 (Groundwater), 11 (Ventilation and Mechanical Rooms in Building 516) and 12 (roof of Building 516).[5] Furthermore, because all gross measurement data (i.e., background included) were below the applicable action levels, no statistical analyses were required. All exceedances of investigation levels were fully evaluated. Therefore, OUs 2, 3, 4, 5, 6, 7, 8, 10, 11 and 12 are eligible for release for unrestricted use.

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[2] Integrated Environmental Management, Inc., Report No. Report No. 2008012/G-103341, "Investigation Report for the Diamond Ordnance Radiation Facility (DORF)", Section 2.8, 2010.

[3] Integrated Environmental Management, Inc., "Investigation of the Diamond Ordnance Radiation Facility (DORF)", Report No. 2008012/G-102381, July 30, 2009.

[4] Integrated Environmental Management, Inc., Report No. 2008012/G-102379, "Sampling and Analysis Plan for the Diamond Ordnance Radiation Facility (DORF)", July 17, 2009.

[5] The screening criteria were taken from Tables H.1 and H.2 of NUREG-1757 (Vol. 2) and Tables 5.19 and 6.91 of NUREG-5512 (Vol. 3).

The residual radioactivity in OU 1 (exposure room) from activation of concrete during former reactor operations, exceeds the release criteria. Therefore, OU 1 is not currently eligible for expedited release.

The former reactor pool area was designated a special survey area during the investigation and has since been designated as OU 9. Investigation-derived and historical data for this area indicate residual radioactivity is below the applicable Derived Concentration Guideline Levels (DCGLs) and clearly support its designation as a Class 3 survey unit. However, the quantity of available data, including the results of concrete core sample analyses during the investigation, are insufficient to support a MARSSIM-based release of the survey unit because it was thought that the acquisition of the additional concrete cores needed to secure its release as a Class 3 area would impact the structural integrity of Building 516. Therefore, new OU 9 is also subject to further action.

Soil cores to a depth of the foundation base were collected from the perimeter of Building 516 and analyzed for the radionuclides of concern. All results were a small fraction of the applicable DCGLs even without taking credit for natural radioactivity (i.e., background).

The lower floor of the DORF is comprised of OUs 1, 2 and 3. Although OUs 2 and 3 meet the criteria for release, any action taken to reduce the residual radioactivity in OU 1 has the potential to impact the radiological status of other OUs. Therefore, decommissioning planning will need to preserve the radiological status of Building 513 (OU 5), the upper floor of Building 516 (i.e., OUs 4, 11 and 12) and the outdoor areas (OUs 6, 7, 8 and 10) in order to minimize the need for additional data acquisition.

1 INTRODUCTION

1.1 Background

The Diamond Ordnance Radiation Facility (DORF or "facility") is located at what was at one time the Forest Glen Annex of the Walter Reed Army Medical Center (WRAMC). The facility consists of Building 513, Building 516 and the surrounding (fenced) property. In Building 516, a small research reactor was in use until the late 1970's. By 1980, the reactor was de-fueled and decommissioned (with the core and other radioactivity shipped off-site), the building released for unrestricted use, and the Army Reactor Permit terminated.

Later, however, the WRAMC began using portions of the building to stage/store medical and research radioactive waste pending disposal pursuant to U. S. Nuclear Regulatory Commission (USNRC) License No. 08-01738-02 and Department of the Army Radiation Authorization (ARA) No. 08-01-97, which was renewed on November 30, 2005 as ARA No. 08-01-15. In addition, the radiological status of the former "Exposure Room" in light of present-day (i.e., dose-based) release criteria was questioned. Therefore, the Army Reactor Permit (No. DORF-1-97) was reinstated and radiation-related activities at the site today are performed pursuant to that permit and to the USNRC medical/research license provisions.

ARA No. 08-01-15 covers possession and use of radioactivity not under the jurisdiction of the USNRC (i.e., accelerator-produced material). However, on November 30, 2007, the USNRC assumed jurisdiction for these materials at federal facilities and elsewhere, meaning radioactivity at the DORF that was subject to the provisions of the authorization were captured in License No. 08-01738-02.[6] Therefore, the requirements of ARA 08-01-15 are not relevant to the investigation and final status survey of the DORF.

Appendix 12.2 contains a copy of Permit No. DORF-1-97 and Amendment 79 of License No. 08-01738-02, which was current at the time of the investigation. As of the date of this report, License No. 08-1738-02 has been amended seven additional times, but none of the changes were relevant to the investigation and final status survey of the DORF.[7]

The Forest Glen Annex was annexed by Fort Detrick on October 1, 2008 as part of the base realignment process (BRAC). Fort Detrick, an Army base in Frederick, Maryland, assumed command and control of the facility on that date. However, regulatory responsibility for the DORF remains with the WRAMC until the facility has been released for unrestricted use (i.e., without regard for radiological constituents). Therefore, the Baltimore District of the U. S. Army Corps of Engineers (USACE) was asked by WRAMC to decommission the facility and to facilitate termination of all existing radioactive materials licenses and permits. Fort Detrick will assume full responsibility for the DORF after decommissioning is complete.

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[6] Department of the Army, Form 348, "Authority for Selected Radioactive Materials Not Controlled by the Nuclear Regulatory Commission", Authorization No. 08-01-15, Amendment 1, 2009.

[7] Burton, D., (MEDCOM, WRAMC), e-mail communication to C. D. Berger (Integrated Environmental Management, Inc.), December 7, 2010, 3:35 p.m.

1.2 Recent Activities

The USACE was tasked with the performance of decommissioning studies of the DORF for the purpose of assessing the status of Army Reactor Permit No. DORF-1-97. In addition, the USACE was asked to ensure the collected data could be used to support the release of the DORF from USNRC License No. 08-01738-02. Integrated Environmental Management, Inc. (IEM) was contracted by the USACE to provide radiological support under Contract No. W912DR-D-0022, "Multiple Award Radiological Services (Environmental Services & Environmental Remediation Services)". Delivery Order No. 0001 of that contract is to assist with the decommissioning of the DORF. The scope of work for the Delivery Order was the performance of an investigation at the DORF in order to secure the data and information needed for its release.[8]

The IEM Team prepared a Project Planning Package that was reviewed and approved for implementation by the USACE and applicable stakeholders. The approach used for the effort was to acquire measurement data of sufficient quantity and quality to meet the requirements for final status survey. If the results of the investigation show the release criteria could be met in any of the survey areas, that area would then be eligible for expedited release without the need for additional data acquisition.

The results of previous radiation surveys at the DORF were used to prepare a preliminary conceptual site model or "CSM", which appeared as Attachment 7.4 of the Project Planning Package. The CSM, included in its entirety in Appendix 12.3, divided the site into 11 Operable Units (OUs), one of which was designated "non-impacted" (OU 8) due to the presence of a buffer zone (i.e., OU 7 which is the property inside the fence). Note that OU 9 was unassigned. Note also that the CSM with respect to area classifications was found to be incorrect, however it was subsequently revised after the on-site portion of the investigation was complete (see Section 12.1 and Appendix 15.21).

The rest of the operable units, designated Class 3 or Class 1 for the investigation purposes, were expected to exhibit minimal residual radioactivity and thus eligible for expedited release for unrestricted use. The following is a listing of OUs at the DORF at the time of the initial investigation. Maps showing the location of each at the site are also contained in Appendix 12.3. Table 10.1 contains a summary of the area classifications and a listing and description of the OUs, along with their classifications.

- OU 1 - DORF Building 516, lower floor, including the Exposure Room. The former reactor pool area, adjacent to the Exposure Room, was designated a special survey area for the investigation and not as a stand-alone operable unit.
- OU 2 - DORF Building 516, lower floor, Warm Room.
- OU 3 - DORF Building 516, lower floor, Connector Room.
- OU 4 - DORF Building 516, Mezzanine Rooms 1, 1/2, 3 and 5, the main floor, Rooms 101, 104, 105 and 106, and the truck dock.
- OU 5 - DORF Building 513, storage building.

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[8] Integrated Environmental Management, Inc., Report No. Report No. 2008012/G-103341, "Investigation Report for the Diamond Ordnance Radiation Facility (DORF)", Section 2.8, 2010.

- OU 6 - DORF Building 516 Outdoor Area in the location of the former USTs.
- OU 7 - DORF Outdoor Area within the boundary fence (4.2 acres), including the truck ramp and the generator room (storage shed).
- OU 8 - DORF Outdoor Area outside of the boundary fence.
- OU 9 - Not assigned.
- OU 10 - Groundwater below the 4.2-acre DORF site.
- OU 11 - DORF Building 516 Mechanical Room and Ventilation Room (main floor).
- OU 12 - DORF Building 516 Roof.

Residual radioactivity in the OUs, with one exception, was thought to be comprised of primarily beta/gamma-emitters. The exception was in OU 4 where, shortly before the IEM Team mobilized to the site, a single drum with a "depleted uranium" label affixed was briefly stored. The presence of this drum raised the possibility of alpha emitters being present. Due to the differing survey techniques for assessing residual beta/gamma and alpha radiations, and in light of the low probability of actually encountering alpha emitters in OU 4 (the drum and its immediate surroundings did not exhibit removable alpha activity), OU 4 was also given a Class 3 designation for alpha-emitting isotopes.

The Class 1 status triggered more intensive surveys, while the Class 3 status required only limited surveys. If the limited survey data confirmed the presence of alpha activity above the action level, the OU would have been re-classified and the more aggressive Class 1 surveys for alpha radiation would have been performed. As shown in Chapter 7, this did not occur. Although not required in the work plans, alpha survey data are reported for other OUs due to simultaneous acquisition during the measurement campaigns.

With the exception of OU 1, where the concrete became activated from former reactor operations, any residual radioactivity present at the site was likely to be in the form of surface contamination. Concrete cores were collected from within OU 1 to confirm the identity, quantity and distribution of residual radioactivity within the structural material. To address the volumetric contamination potential in the reactor pool area adjacent to OU 1, horizontal concrete cores into the pool and, in one case, through the pool wall were collected.

The IEM Team mobilized to the site in August, 2009. The preponderance of the field work was completed by the end of September, 2009, with additional visits to the site taking place in October, November and December of 2009, and again in March and July of 2010. A variety of measurement types were performed in each of the OUs, and all results were reviewed in light of the Data Quality Objectives (DQOs) set for the project. Once the DQOs were determined to have been met, the measurement data were compared to the applicable DCGLs. This comparison revealed that OUs 2, 3, 4, 5, 6, 7, 8, 10, 11 and 12 exhibited residual radioactivity levels that were below the DCGLs and that no additional final status survey data are required. No residual alpha activity was found in OU 4.

The radiation surveys performed in OU 1, on the other hand, revealed the presence of detectable activation products (Eu-152 and Eu-154 from former reactor operations) only that exceeded certain of the Derived concentration Guideline Levels (DCGLs) in some locations. Therefore, OU 1 was deemed ineligible for expedited release. Additional work and/or additional final status survey data are required before the release status of this OU should be considered.

Historical and investigation-derived data from the former reactor pool area (next to OU 1) imply no residual radioactivity above the applicable DCGLs is present. However, these data are insufficient to support a statistically-based release conclusion. Therefore, the former reactor pool area was designated as (new) OU 9 and a Class 3 survey unit for future work (i.e., the collection of sufficient data to support a MARSSIM-based release).

Soil cores to below the foundation base were collected from the perimeter of Building 516 and analyzed for the radionuclides of concern. All results were a small fraction of the DCGLs even without credit for the presence of natural radioactivity (i.e., background).

2 HISTORICAL SITE ASSESSMENT

2.1 Facility History

Building 516 at the DORF housed a TRIGA Mark F Reactor as the principal research tool in the study of neutron and gamma radiation effects on electrical and electronic components. The reactor, designed and built by Gulf General Atomics of San Diego, California, was operational from September of 1961 through September of 1977, at which time reactor operations were terminated.

A decommissioning plan was prepared and implemented between 1979 through 1980 by Rockwell International. That plan called for the removal of all special nuclear material and all residual radioactivity such that the site could be released for unrestricted use.

The reactor fuel was removed in the spring of 1979 and shipped to various locations. There is historical documentation showing that on April 2, 1979, at 8:22 a.m., packaging began for 46 of the fuel elements, which were then shipped to the University of Utah. On April 24, 1979, at 8:55 a.m., 18 of the fuel elements were packaged and shipped to Penn State University. The remaining elements were shipped shortly thereafter to the Hanford facility for eventual chemical reprocessing. There is no evidence of fuel element leakage in the historical record, and the 1997 scoping surveys were negative for residual alpha activity within Building 516.

In addition to fuel removal, a concrete parapet inside Building 516 was removed, the rubble was placed into the emptied reactor cavity, and concrete was used to fill in the voids to form a continuous surface for the first floor of the building. Verification that the decommissioning plan was implemented in its entirety and that the stated release objectives were met was performed by the Army Reactor Systems Health and Safety Committee, who certified the project complete according to the regulations in existence at the time. (The Committee name was later changed to the Army Reactor Council or ARC.) The reactor permit was subsequently terminated.

Building 516 was equipped with three (3) 5,000-gallon underground storage tanks, the status of which was not addressed in the 1980 final decommissioning report. Radioactive waste that was discharged into the sanitary sewer system made its way into those tanks. In October of 1999, the tanks were characterized, deemed free of residual radioactivity and removed.

During a 1996 review of the site by the Army Reactor Office (ARO), the radiological status of the DORF was questioned in light of recently released decommissioning standards (10 CFR 20 Subpart E). The ARO thus requested a radiation survey be performed at the DORF to verify the exposure environment was consistent with the new standards. The results of the survey revealed low but detectable ambient exposure rates in the former Exposure Room. In response to that finding, the ARO issued Army Radiation Permit No. DORF-1-97 in 1998 in order to establish additional controls and monitoring procedures to prevent removal or disturbance of the activated concrete until follow-up actions could be implemented.

The WRAMC later used Building 516 for storage, "delay for decay", processing, and packaging of short-lived radioactive waste from research and hospital operations. These operations were authorized by USNRC License No. 08-01738-02 and Department of the Army Radiation Authorization No. ARA 08-01-97. All containerized waste was removed under a separate contract action in advance of the investigation of the DORF.

The Forest Glen Annex was annexed by Fort Detrick on October 1, 2008 as part of the base realignment process. Fort Detrick, an Army base in Frederick, Maryland, assumed command and control of the facility on that date. However, regulatory responsibility for the DORF remains with the WRAMC until the facility has been released for unrestricted use by all applicable regulatory agencies. Fort Detrick will assume full responsibility for the DORF after decommissioning is complete and the two licenses (i.e. USNRC and Army) terminated.

2.2 Facility Description

The DORF, located at the former WRAMC Forest Glen Annex (now controlled by Fort Detrick) in Silver Spring, Maryland, is eight miles due north of the center of Washington, D.C. The facility consists of a 4.2-acre site with a 65-by-50-by-25-foot high building with a basement that contained the reactor (Building 516). Also on the property is an approximately 25-by-25-by-10-foot high instrumentation building (Building 513). Both structures are encircled by an exclusion fence that has a radius of about 240 feet. Access to the site is controlled through a single gated entrance. The location of the DORF within the WRAMC Forest Glen Annex is depicted on Figure 11.1.

Building 516 is a brick building with approximately 6,896 square feet of floor space. It is a two-story building with a two-sided mezzanine level that is accessed from the main floor. The basement contains the former Exposure Room and work areas referred to as the "Warm Room", the "Connector Room" and the outer walls of the former reactor cavity. The main floor is primarily open space, with a Mechanical Room, mezzanines above the main floor, the former control room, two small offices and a restroom.

A sanitary sewer system discharges from Building 516. It was used for occasional discharge of radioactive waste pursuant to 10 CFR 20.2003 for reactor operations and regular discharge as part of the USNRC-licensed medical/research program.

Prior to the start of the investigation, Building 516 housed a variety of waste processing equipment that was known or thought to be contaminated with residual radioactivity. This included the following:

- Drum compactors (an active unit on the main floor and a retired unit on the basement level);
- Vial crusher (used to separate the scintillation fluids from the glass and plastic vials);
- Inactive fume hood (for radiological activities) on the main floor. (The exhaust from the hood was also vented through dual HEPA filters located on top of the hood assembly.);
- A large walk-in cooler (located in the former Exposure Room/basement and used to store medical/isotope waste); and
- Numerous shelving units used to store radioactive waste material.

Also prior to the investigation, Building 516 contained an assortment of hazardous materials including the following:

- Stacks of lead bricks;

- Several storage "pigs", either in the form of enclosed solid lead or containing lead shot;
- Lead lined drums; and
- Lead lined penetrations in the ceiling of the Exposure Room.

Building 513 is a single-floor brick building with floor space of approximately 600 square feet. Prior to the investigation it was being used for non-radiological materials storage. Although once designated a "radioactive instrument" area, its actual radiological history was not clear when the FSS work plans were prepared (see Section 2.4.5, below).

The outside areas of the DORF are mostly sand, grass and vegetation. Within the perimeter fence is a small paved parking lot and the primary access road. There is also a small temporary building positioned between Buildings 513 and 516 that was at one time used for hazardous material storage.

2.3 Contaminant Identification

Radiological activities that took place at the DORF included research reactor operations, then later the storage, staging and packaging of medical radioactive waste. A master list of radionuclides associated with reactor operations (i.e., neutron interactions with steel, concrete, plastics, oil, etc.; fission products; fuel handling) and with the medical and research waste storage operations was prepared.[9] From this master list of radionuclides, the following radionuclides of concern (ROCs) were determined through historical facility documents, previous measurement results and other considerations:

- Manganese-54 (Mn-54);
- Cobalt-57 and Cobalt-60 (Co-57/60);
- Iron-55 (Fe-55);
- Cesium-134 (Cs-134);
- Europium-152 and Europium-154 (Eu-152/154);
- Hydrogen-3 or "tritium" (H-3);
- Carbon-14 (C-14); and
- Natural and depleted uranium (U-nat and DU).

After the publication of the work plans, additional discussions with WRAMC personnel and a closer review of the licensed activity inventory revealed the presence of natural uranium at the DORF due to former medical/research license activities was deemed unlikely and it was thus removed from the

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[9] Integrated Environmental Management, Inc., Report No. 2008012/G-102379, "Sampling and Analysis Plan for the Diamond Ordnance Radiation Facility (DORF)", July 17, 2009.

ROC listing (see Section 5.2.2, below). In addition, the historical record shows Co-57 being present at the DORF as a result of reactor operations, although such an association is not typical. It was based on the fact that the 1980 decommissioning report (Rockwell) contained analytical results from two post-remediation concrete samples from the Exposure Room that reported the presence of Co-57 (i.e., approximately 15 pCi/g each). It is possible that other activation products known to be present in the concrete or naturally-occurring photon emitters were inadvertently identified as Co-57 in those samples, although the original gamma spectra were not available for confirmation. However, it is important to note that none of the pre-remediation sample results showed the presence of Co-57. Nonetheless, Co-57 was retained on the list of ROCs and was designated as being associated with reactor operations in the master list. Table 10.2 contains a more detailed listing of the ROCs that were the subject of the investigation at the DORF and that were addressed during the FSS.

2.4 Results of Previous Surveys

Historical survey information about the individual OUs established for the investigation is drawn from the (1980) Close-out Survey, (2009) Conceptual Site Model, (1996) Exposure Room Survey, (1999) ORISE Soil Sampling of UST Area and (1997) Building 513 Close-out Documentation. The following subsections are summaries as they pertain to the individual operable units. Note that OU 9 is not presented in the OU summaries below because it did not exist during the investigation. It was later added as a result of the investigation findings.

2.4.1 Operable Unit 1

Operable Unit 1 (OU 1) is comprised of the Exposure Room in the basement of Building 516. The former reactor pool area adjacent to the Exposure Room is not included in OU. Instead, it was subject to special investigation (see Chapter 8, below).

Early data on radioactivity levels in the Exposure Room were provided by Rockwell as part of the decommissioning effort that took place in the 1970's. Prior to remediation, the Rockwell Report documented exposure rates of up to 400 μR/hr on the walls, floor and ceiling of the Exposure Room due to activation of concrete. The primary activation products identified were the europium isotopes, although Co-60 was also reported. Ambient exposure rates in the area reportedly ranged from 20 μR/hr to 100 μR/hr. Core samples of the structural materials showed the presence of Co-60 in concentrations up to 400 pCi/g, Co-57 up to 15 pCi/g, Eu-152 up to 281 pCi/g, and Eu-154 up to 19 pCi/g, with the highest concentrations embedded in the shielding. Removable beta/gamma activity was reported to be less than 10 dpm/100 cm^2 in the Exposure Room.

The inside of the former reactor pool area prior to remediation exhibited gross beta concentrations ranging from 20 to 50 pCi/g, with the highest value noted at Rockwell Core Location No. 24 (see Figure 11.2). Only two isotopic analyses were performed on the concrete samples (at Core Locations 3 and 34), the results of which confirmed the presence of Co-60, Eu-152 and Eu-154. In general, the activity concentrations in the concrete samples were higher on the surfaces closest to the operating reactor, and they decreased with increasing depth away from the reactor. Total contamination on the pool walls was reported to be below the applicable release criterion (i.e., 5,000 dpm/100 cm^2 beta) and surface count rates at the core locations ranged from "background" to 400 counts per minute (at Core Location No. 24).

The area was remediated by removing six to 12 inches of concrete from the east wall of the Exposure Room (see Figure 11.3), followed by the collection of additional concrete samples. The Rockwell Report gave radionuclide concentrations and contact exposure rates that were reduced significantly as a result of the removal effort (see pages 47 through 50 of the Rockwell report), which again supported the assumption that activation products were non-uniformly distributed throughout the thickness of the concrete shielding, with higher concentrations located on surfaces nearest the reactor. Post-remediation contact dose rates in the Exposure Room reportedly ranged from 0.05 to 0.23 millirad per hour (north wall).

After remediation of the inside surface of the reactor pool area, additional concrete samples were collected from the eastern-most wall. Analytical results revealed significantly reduced Co-60, Eu-152 and Eu-154 concentrations from the pre-remediation results. Because all historical evidence points to activation product activities being higher on the surfaces closer to the reactor, it was reasonable for the authors of the Rockwell Report to assume that the low contact exposure rate data on the north and south walls of the pool were indicative of no radioactivity of significance in those locations (see Figure 11.4). Therefore, no samples were collected on the north and south walls of the pool area.

After remediation was complete, the former reactor pool was braced, backfilled with concrete from the demolished parapet located in the main room of Building 516, and then filled with additional concrete to grade. Contamination data on all surfaces within the main room were reported to be less than 5,000 dpm/100 cm^2 (beta), thus it is reasonable to assume that no residual radioactivity of significance was placed into the pool cavity via and demolished parapet. However, data specific to the parapet's radiological condition were not reported.

The Exposure Room was surveyed again in 1996 by the Army Research Laboratory (ARL). These findings were generally consistent with the conditions documented by the Rockwell authors after taking into account 16 years of radioactive decay. The ambient exposure rates inside the Exposure Room were reported to be about 37 μR/hr above background at that time. A follow-up survey of the Exposure Room (by the Army) took place in 2008, which again demonstrated ambient exposure rates above background (i.e., a few tens of microR per hour).

2.4.2 Operable Unit 2

Operable Unit 2 (OU 2) is also in the basement of Building 516, and is referred to as the "Warm Room". Historical surveys in this area show that ambient exposure rates were not distinguishable from background. Removable activity was reported to be less than 7 dpm/100 cm^2.

2.4.3 Operable Unit 3

Operable Unit 3 (OU 3), also in the basement of Building 516, is referred to as the "Connector Room". Historical surveys in this area showed that ambient exposure rates were not distinguishable from background. Removable activity was reported to be less than 10 dpm/100 cm^2.

2.4.4 Operable Unit 4

Operable Unit 4 (OU 4) is the largest operable unit in the investigation, and is comprised of the main floor, truck dock and mezzanine level of Building 516. Included in this OU are the largest of the rooms (Room 101), three smaller rooms (Room 104, 105 and 106), the mezzanine (M-1, M-3 and M-5), as well as the level located between the basement and ground floor of the building. Historical surveys in OU 4 showed that exposure rates were not distinguishable from background. Removable activity was reportedly less than 15 dpm/100 cm^2 on the first floor and less than four (4) dpm/100 cm^2 on the mezzanine level.

2.4.5 Operable Unit 5

Operable Unit 5 (OU 5) is the small storage facility (Building 513) located within the perimeter fence of the DORF. A 1997 radiation survey of the building included direct surveys for alpha, beta, and gamma radiation and removable activity surveys over a pre-gridded floor.[10] The results indicated no elevated alpha, beta or gamma-bearing contamination in the form of either fixed or removable activity.

During the planning phase of the investigation, the references to "radioactive instrument storage building" noted in the historical documentation resulted in the designation of OU 5 as an impacted area. Later, however, documentation was found stating that close-out surveys were performed in Building 513 and that it "could be classified as non-impacted, with no further surveys required".[11,12] It was further confirmed that Building 513 was never used by WRAMC for waste storage or processing.[13]

2.4.6 Operable Unit 6

Operable Unit 6 (OU 6) is an outdoor area in the location of the former Underground Storage Tanks (USTs). During the planning phase of the investigation, there was insufficient information available to rule out potential residual radioactivity in this location. Later it was discovered that in September 1999, ORISE collected soil samples in contact with and adjacent to the three tanks removed from this site. The subsurface soil samples were collected at varying depths up to 11 feet bgs and were evaluated for gamma-emitting activity, with the terminal depth representing the base or "bottom" elevation of the tanks during their operational lives. The analytical results revealed the presence of background-equivalent radionuclides only (i.e., uranium, thorium and potassium), which supported the conclusion that the soils surrounding the former USTs contained no residual radioactivity (see Appendix 12.4).

2.4.7 Operable Unit 7

Operable Unit 7 (OU 7) is the outdoor area within the 4.2-acre fence that includes the truck ramp, hazardous materials storage trailer and the generator room. In 1980, the Army collected and analyzed soil samples from this area to a depth of up to seven inches.[14] Analytical results showed concentrations averaging about 1.3 pCi/g for Pb-212, 17 pCi/g for K-40 and 0.2 pCi/g for Cs-137, all of which were typical of natural background radioactivity in soil.

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[10] Cabrera Services, "Historical Site Assessment and Addendum to Environmental Condition of Property, Walter Reed Army Medical Center, Washington, DC", Contract No. W912-DR-05-D-0024, Delivery Order 0002, 2006.

[11] Morton, Arthur R. WRAMC, Memorandum for Record, "Decommissioning Survey of Building 513 Forest Glen Annex,", July 7, 1997.

[12] Shanbaky, Mohamed M., U.S. Nuclear Regulatory Commission, "Correspondence Regarding No Further Radiological Use for Buildings 149A, 188, 500, 506, 508, 511, 512, and 513", August 18, 2000.

[13] Burton, D., (MEDCOM, WRAMC), e-mail communication to C. D. Berger (Integrated Environmental Management, Inc.), December 3, 2010, 8:12 a.m.

[14] Department of the Army, Environmental Hygiene Agency, "Radiation Protection Special Study Number 28-43-0982-80, Close-out Survey of the Diamond Ordnance Radiation Facility (DORF), 25-28 February, 1980", report to the Commander, U. S. Army Materiel Development and Readiness Command, September 2, 1980.

2.4.8 Operable Unit 8
Operable Unit 8 (OU 8) is comprised of the property that surrounds the DORF perimeter fence. All available historical information supports the conclusion that this area was not impacted by former DORF operations.

2.4.9 Operable Unit 10
Operable Unit 10 (OU 10) is comprised of the groundwater beneath the 4.2-acre DORF property. There is no known historical information on the radiological status of this OU. However, during the planning phase of the investigation, there was insufficient information available to rule out potential residual radioactivity in the vicinity of the former underground storage tanks, thus the groundwater was designated as impacted. Later, however, historical documentation showing the soil around the former tanks did not contain residual radioactivity of significance was discovered, thus OU 10 was re-designated as non-impacted (see Appendix 12.4).

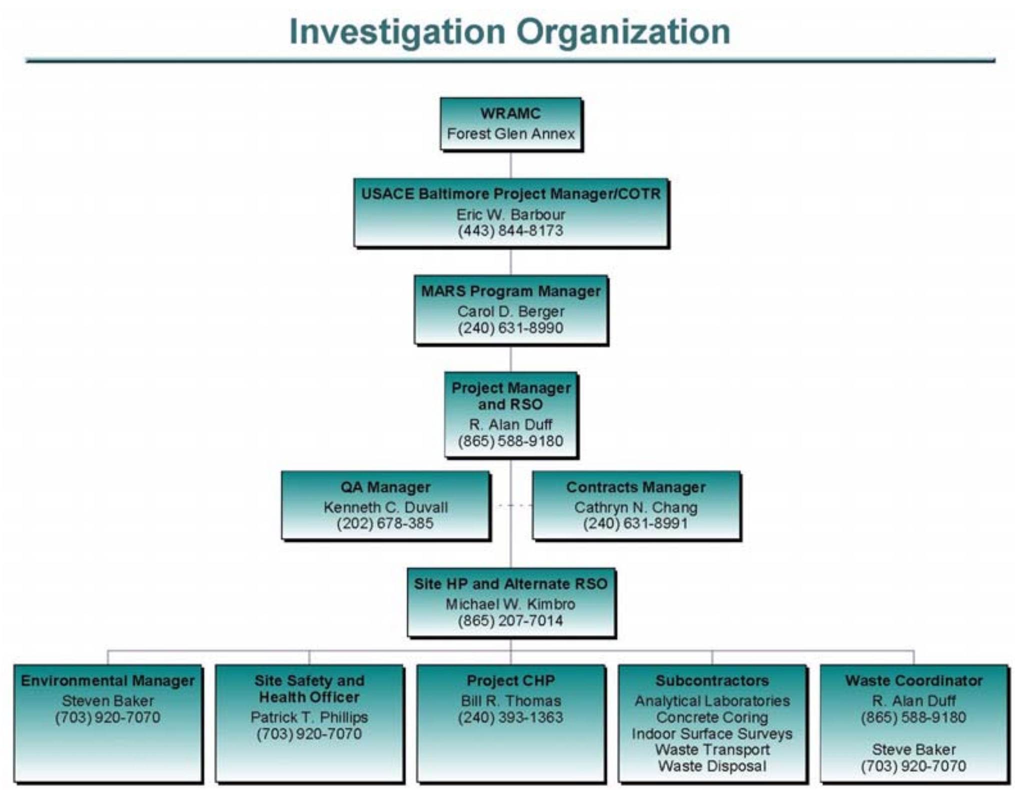
2.4.10 Operable Unit 11
Operable Unit 11 (OU 11), labeled as Room 102 or the Mechanical Room, is located on the main floor of Building 516. Above the Mechanical Room and also a part of OU 11 is the Ventilation Room. Historical surveys in these areas revealed exposure rates that were indistinguishable from background and removable activities of less than 15 dpm/100 cm^2.

2.4.11 Operable Unit 12
Operable Unit 12 (OU 12) is comprised of the roof of Building 516. There is no known historical information on the radiological status of this OU.

3 PROJECT OVERVIEW

3.1 Project Organization

The investigation was performed under the direction of the DORF Project Manager/COTR (USACE), Mr. Eric W. Barbour. Project Management was assigned to Mr. R. Alan Duff of IEM. Assisting Mr. Duff was Mr. Bill R. Thomas, who served as the Project CHP, Mr. Steven Baker the Environmental Manager for the project, Mr. Kenneth C. Duvall the Quality Assurance Manager, and Mr. Patrick Phillips, the Site Safety and Health Officer (SSHO). Appendix 12.5 contains the qualifications of key members of the IEM Team. The following figure shows the investigation organization:



Other participants in the project included representatives of the WRAMC, Fort Detrick, the BRAC and the ARO. Mr. David Burton (WRAMC/DORF Property Manager), Ms. Anne Delp (BRAC Environmental Coordinator for the WRAMC), Mr. Michael Jewett (Fort Detrick/Forest Glen Executive Officer), Mr. David Hudlow (Fort Detrick Safety Officer), Mr. Michael Borisky (ARL Health Physicist), COL Mark Melanson (WRAMC Radiation Safety Officer at that time), MAJ Andrew Scott (interim replacement for COL Melanson) and LTC Francis Fota (current WRAMC RSO), all provided input to the process.

Although the DORF is located in the State of Maryland, Federal facilities are typically regulated by the USNRC regardless of location (see 10 CFR 150.10). Therefore, work at the site was performed pursuant to the terms and conditions of IEM's radioactive materials license (MDE License No. MD-

31-281-01), as invoked by interstate reciprocity. In addition, the applicable provisions of Army Reactor Permit No. DORF-1-97, the WRAMC radioactive materials license (License No. 08-01738-02) and the Radiation Source Permit, issued to IEM by Fort Detrick as required by Army Pamphlet 385-24 (Section 204), were also enforced.

3.2 Approach

The objective of the investigation was to collect characterization data in order to determine the radiological conditions of the DORF. Areas with little detectable residual radioactivity would be subject to an expedited release process, thus the approach was to collect sufficient data of appropriate quality in order to meet final status survey requirements. In this case, characterization data could be used to support closure and release of eligible areas.

Figure 11.5 is a flow chart of the expedited release process. If excessive residual levels in areas were identified, the findings of the characterization would then be used to determine strategies/remedies and to outline the tasks necessary for site-wide decommissioning.

Table 10.1 contains a listing of impacted areas. The nature and extent of contamination at the DORF were thought to be limited to these areas. Most of the areas were expected to be suitable for release for unrestricted use without further remedial action. These expectations were based upon the findings of radiological assessments performed prior to the investigation. Therefore, the design of the investigation was to ensure the data acquired were of sufficient quality and quantity to permit "expedited release" of as much of the DORF as possible without the need to perform a subsequent Final Status Survey (FSS). (It is important to note that in order to decommission the DORF, the entire site must meet the criteria for release as defined in NUREG-1757, Vol. 1, Sect. 15.5.2.) Table 10.3 summarizes how the Data Quality Objectives (DQOs) were captured in the survey design.

The IEM Team went into the investigation with the understanding that if the actual radiological conditions anywhere at the site were not as anticipated, the expedited release process would be discontinued for the affected area, which would then subject to the more general approach whereby characterization data would be used as input to decommissioning planning and subsequent performance of the FSS. Additional investigation and biased measurements/sampling as necessary to identify the lateral and spatial extent of residual contamination would also be performed to ensure the data needed for developing site-wide decommissioning plans were available and of sufficient quality. Any further actions beyond those necessary for the acquisition of characterization data would be deferred to the decommissioning phase of the project.

4 RELEASE CRITERIA

4.1 Applicable Regulations

The ability to secure expedited release of the DORF was based on an assumption that remediation of an OU would not be necessary. It also assumed that the levels of residual radioactivity in an OU were generally less than appropriate release criteria.

The WRAMC is authorized to process and store waste pending disposal at the DORF by USNRC License No. 08-01738. The ARL is authorized to possess radioactivity associated with former reactor operations by Permit No. DORF-1-97 issued by the ARO. As such, possession and storage of the radioactivity allowed by the license or by the permit are authorized, and the approach used to release the DORF from both license and permit requirements must meet the license termination guidance of both the USNRC and the ARO.

Permit No. DORF-1-97 states, in part, that "All activities involving the residual reactor radioactivity at the DORF must be in compliance with applicable sections of Titles 10 and 40 of the Code of Federal Regulations, AR 50-7 and AR 385-11." Therefore, the USNRC requirements for decommissioning that appear in Title 10, Part 20 would apply for both license and permit termination.

The USNRC has established criteria for ensuring that facilities and property used for licensed operations present a tolerable radiological risk to people and the environment once licensed operations cease. The radiation dose that the USNRC believes presents a tolerable risk, as published in Title 10, Code of Federal Regulations, Part 20.1402, reads as follows:

> *"Decommissioning with license termination shall be limited to sites considered acceptable for unrestricted release where the residual radioactivity that is distinguishable from background radiation results in a total effective dose equivalent to an average member of the critical group that does not exceed twenty-five millirem per year (25 mrem/yr), including that from groundwater sources of drinking water, and the residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA)..."*

The level of residual radioactivity permissible at the DORF that would ensure compliance with USNRC's radiation dose objective may be determined by means of an exposure assessment. If the level of residual radioactivity in any of the OUs is less than the derived criteria from the exposure assessment, then they may be released for unrestricted use. These criteria are called Derived Concentration Guideline Levels or DCGLs.[15]

4.2 Derived Concentration Guideline Levels

The USNRC has prepared tables of screening values for release of facilities (structures) and for the release of land areas (soil) via exposure assessments that rely on highly conservative assumptions and parameters. These can be found in Tables H.1 and H.2 of NUREG-1757, Volume 2 and Tables 5.19 and 6.91 of NUREG-5512, Volume 3, respectively. If used in the place of site-specific DCGLs,

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[15] U.S. Nuclear Regulatory Commission, Multi-agency Radiation Survey and Site Investigation Manual, NUREG-1575, Revision 1, August, 2000.

and if conditions in the survey area are similar in nature to those assumed by the USNRC in developing the Table 5.19 and 6.91 values, the USNRC maintains the end result will be amply protective of human health and safety, thus regulatory approval for use of screening values is not required.[16,17]

The USNRC warns that users of screening values should recognize the appropriateness of embedded assumptions, parameters and scenarios. In line with that warning is the recognition that residual radioactivity resulting from reactor operations at the DORF consists of not only surficial, non-volumetric radioactivity on building surfaces, but also volumetric radioactivity from neutron-activated materials, particularly within the concrete walls of the Exposure Room. Therefore, the embedded aspects of the screening levels must be re-evaluated for volumetric activity using dose assessment methodologies and input parameters that are specific to the site conditions and reasonably likely exposure scenarios.

For surface soil in outdoor survey areas at the DORF, the USNRC screening values were deemed sufficiently-conservative for use as the DCGLs. For building surfaces subject to the medical waste source term, the USNRC screening values were also deemed sufficiently conservative for demonstrating their release. Residual radioactivity in excess of the screening values would not necessarily require remediation. Instead, a site-specific evaluation of the dose/risk would be performed. However, any operable units that do not meet the screening values would be eliminated from the expedited release option.

A listing of the source term applicable to the DORF is presented in Table 10.2, along with the applicable DCGLs. These take into account the radiological significance of both the demolition and the building re-use scenarios for building surfaces, soil and volumetrically-contaminated materials, as applicable, with the most conservative (i.e., the lowest) value from any scenario selected for survey planning.

The limiting screening values for surface scans and stationary measurements, taking into account detection efficiency and the most restrictive DCGL, was associated with the isotope Co-60 with its DCGL of 7,100 dpm (beta)/100 cm^2. The action level for use during data acquisition was set at 50% of the limiting DCGL (i.e., 3,500 dpm (beta)/100 cm^2). For practical reasons, a single scan data point may have exceeded the action level, but action was not initiated unless contiguous points were also in excess of the action level.

In order to ensure the direct exposure pathway for continuous presence (industrial use scenario) did not result in a dose potential in excess of the 25 millirem Total Effective Dose Equivalent (TEDE) in one year dose criterion, the ambient exposure rate within each survey area must be less than 25 millirem per year × 1000 microR/millirem ÷ 2000 work hours per year = 12.5 microR per hour. To ensure an element of conservatism, a value of 10 microR per hour was selected as the action level for ambient exposure rates.

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[16] U.S. Nuclear Regulatory Commission, *Consolidated Decommissioning Guidance Characterization, Survey, and Determination of Radiological Criteria,* Appendix H, NUREG 1757, Volume 2, Rev 1, September, 2006.

[17] U.S. Nuclear Regulatory Commission, *Residual Radioactive Contamination From Decommissioning - Parameter Analysis*, NUREG 5512, Volume 3, Draft, October, 1999.

If an action level was exceeded in a survey unit, that unit would be deemed ineligible for expedited release unless resolved by test decontamination. More on this topic appears in Section 8.4, below.

5 SURVEY OBJECTIVE

5.1 Data Quality Objectives

The Data Quality Objectives (DQO) process as it pertains to site investigations involves the evaluation of seven elements that address the fundamental decisions to be made and their inputs. These must be resolved before the overall project objectives can be met. The following are the seven DQO steps:

- State the problem;
- Identify the decision;
- Identify inputs to the decision;
- Define the study boundaries;
- Develop the decision rule;
- Specify tolerable limits on decision error, and
- Optimize the design.

The following subsections provide an overview of the DQO process as it was applied to the investigation of the DORF.

5.1.1 State the Problem

The global problem to be resolved at the DORF, and the end point of the decommissioning effort, is the release of the DORF for unrestricted use. The primary decision-makers in regard to the investigation were representatives of USACE, WRAMC, the ARL, Fort Detrick, the ARO and the BRAC. Other stakeholders are the USNRC and the State of Maryland. Resources available to address the problem were provided by the USACE, its contractor (IEM) and WRAMC.

5.1.2 Identify the Decision

The principal study question for the investigation is as follows: Can the DORF be released for unrestricted use by demonstrating compliance with applicable release criteria and using the MARSSIM-recommended methodologies? If the survey data acquired during the investigation demonstrate that release criteria can be met, then unrestricted release of the site can be accomplished. However, if the survey data do not support release, the investigation data will be used as input to the evaluation of remedial alternatives, the preparation of a decommissioning plan, and to the performance of a site-wide final status survey. Therefore, the primary decision statement for the investigation is: "Determine whether or not all of the survey units at the DORF satisfy applicable release criteria and are eligible for expedited release".

There are a number of interim decisions that needed to be resolved before the decision in regard to site release can be made. These include the following:

- Disposition of materials and equipment

- Areas eligible for expedited release
- Areas subject to test-decon activities
- Areas that may benefit from site-specific dose assessment
- Areas that require remediation efforts
- Areas that will be subject to decommissioning planning and the separate FSS
- Area classifications, including confirmation of non-impacted areas
- Areas that require additional alpha measurements
- Radionuclide measurements that are only subject to contaminant verification requirements
- Selection of judgmental and bias measurements
- Instrument selection
- Small areas of elevated activity (design of measurements and response to detection)
- Results of statistical analyses
- Verification of assumptions
- MDC and scan MDC requirements
- Decision error, LBGR, Δ/σ selection
- Design of sampling plan
- Special measurements and sampling of the sewer, groundwater, subsurface, concrete cores, ventilation systems, etc.
- Removable or fixed contamination levels
- Identifying operable units and survey units

5.1.3 Identify Inputs to the Decision

Inputs to decision-making for the DORF came from measurement data that provided information on residual radioactivity levels, small areas of elevated activity, radionuclides present, the amount of fixed and removable surficial contamination, the amount volumetric activity, and the spacial distribution of contamination. The data acquired reflected the nature and extent of contamination in land areas, on building surfaces and on the surfaces of materials and equipment subject to removal actions. Special measurements provided information on the extent of contamination in groundwater, subsurface soil, bulk media such as concrete, sewage systems and ventilation units.

The measurement systems selected to provide these data included direct measurements and scans using beta, gamma and alpha sensitive instrumentation. Gamma spectroscopy data were also acquired and samples for laboratory analysis were collected. Swipes were obtained to assess removable activity levels on surfaces and concrete core samples were obtained to assess volumetric contamination (level and distribution). All measurement results were compared to USNRC screening criteria, which are highly-conservative, dose-based values that do not require additional regulatory defense.

5.1.4 Define the Study Boundaries

The investigation applied specifically to the DORF site as it is situated within its boundary fence. Included in the study boundary are Building 516, Building 513 and the land areas within the fence. The land areas outside of the fence line are considered to be non-impacted and are excluded from study other than as mentioned herein. However, if the investigation reveals the radiological status of the land area within the fence is not as anticipated, the extent of the investigation may extend beyond the boundary fence, as necessary and as determined from the investigation findings.

Each area was classified according to MARSSIM guidance as either Class 1, 3 or non-impacted (see Table 10.1). The site was subdivided into operating units consistent with area classifications, and then further subdivided into survey units based on MARSSIM-based spatial limitations. The CSM, as described in Section 1.2, above, describes the extent of contamination in areas and baseline risks to be mitigated, as well as the levels of effort and resources needed for characterization and measurement. It also identifies end-points in meeting closure requirements for specific areas.

The investigation addressed concerns regarding risks from residual radioactivity. However, in regard to non-radiological constituents, the investigation scope included only identification of chemical hazards at the site. A radionuclide master list was developed based on the types of nuclear activities carried out at the site. A realistic list of potential radionuclides of concern (i.e., source term) was extracted from the master list by eliminating radionuclides that are not expected to be present based upon half-life, contributions expected to the overall dose potential, Historical Site Assessment findings, operational records and discussions with former employees. Release criteria for each radionuclide in the source term were based upon USNRC screening values as shown in NUREG-1757 or using the *DandD* computer code with default input parameters. Army and USNRC guidance will be referred to when implementing decommissioning activities, however, the most stringent criteria will be employed where the guidance overlaps.

The investigation included a planning phase, field mobilization, and an assessment phase. During the planning phase, plans were developed for data acquisition/analysis, waste management, accident prevention, and site surveys. Mobilization to the field involved equipment/material staging and removal activities, radiation measurements, coring, sample collection and inspections. Evaluation of the characterization data was conducted and those areas that were candidates for the expedited release process were identified. All other areas will be addressed during site-wide decommissioning.

5.1.5 Develop the Decision Rule

The investigation of the DORF was intended to acquire data and information for use in assessing radiological conditions. The decisions to be made during this activity was associated with the type and extent of characterization measurements, with the data used later for decisions on expedited release, remediation options, the FSS, and decommissioning strategies. Decisions on the type and extent of measurement were incorporated in the survey design, which met FSS requirements in order to ensure the viability of the expedited release option. In addition, some operable units were subject to surveys to verify assumptions and complete the knowledge base. Decisions on survey design were set during the planning phase of the investigation and incorporated into the Project Planning Package.

The decision rules used during the investigation were based on individual action levels and their associate actions. Some of the activities for which action levels were defined were to identify small areas of elevated activity, implement test-decon activities, and confirm the radionuclides of concern. Stationary measurement action levels for identifying small areas of elevated activity or the need for test-decon activities were set at 50% of the most limiting DCGL.

5.1.6 Specify Tolerable Limits on Decision Error

The decision error accumulated during the investigation was associated with decisions on survey design and decisions relating to the response to action levels during characterization measurements. Decisions on the survey design that met the requirements for expedited release addressed the Statistical Analysis and Elevated Measurement Criteria decisions in MARSSIM.

Tolerable limits for Type I and Type II decision error were set at 5 %. MDC and Scan MDC values were calculated to ensure that measurement sensitivity is sufficient to demonstrate that the $DCCL_w$ and $DCGL_{EMC}$ values are not exceeded. Detection methods were confirmed to be adequate to meet these MDC and Scan MDC requirements.

The decision error related to action level response had minimal impact on decisions during the investigation except for the response to the radionuclide of concern assessment and investigation. In addition to total beta measurement, gamma spectroscopy and laboratory sampling and analysis, alpha measurements were employed to ensure a comprehensive investigation. A full evaluation of the decision error associated with radionuclides of concern is expected to result from an independent assessment to be conducted as part of closure activities.

5.1.7 Optimize the Design

Optimization to minimize decision error during the investigation required revisiting the decision-making process. Re-evaluating parameters such area classifications, operable unit and survey unit designations, the use of screening or site-specific release criteria, the selection of scanning, sampling and direct measurements for characterization, action levels, detection limits, grid specifications, Type I and Type II error rates, etc. for optimal design took place if additional information became available. Approval of additional costs and resources were also utilized to optimize the design of characterization measurements.

5.2 Survey Design

The investigation of the DORF was designed to ensure sufficient characterization data were acquired to demonstrate compliance with release criteria and to meet appropriate data quality and quantity requirements. While the operable units at the DORF were expected to exhibit residual radioactivity that is below the DCGLs, the probability of this actually being the case was determined during the investigation.

This section describes the survey design for the investigation such that resulting data may be used to release eligible operable units. The design approach generally follows the recommendations of MARSSIM. Each operable unit was divided into "survey units" to meet MARSSIM spacial limits on areas and to facilitate a more manageable assessment of the various areas. The type and minimum number of direct surveys and samples was determined using a combination of MARSSIM tables and Visual Sample Plan.[18] The scanning specifications for each survey unit were designed pursuant to MARSSIM guidance.

There are a number of special entities at the DORF, including sewer systems, drainage systems, storm water areas, ventilation systems, and the groundwater, that were characterized and evaluated as part of the investigation. These evaluations required alternate methods to those utilized for Operable Units 1 through 7 and 11. Section 2.6 of MARSSIM offers provisions for alternate methods and recognizes the need for flexibility in addressing special entities as part of the release process. Therefore, pursuant to that guidance, decisions on release of the special entities at the DORF were based upon professional judgment in regard to measurement locations, frequencies, and comparisons of data to release criteria.

NUREG-1757, Volume 2, Appendix G does state that, with respect to sewers, floor drains, and ducts, the office worker scenario is not appropriate for exposure assessments, thus the USNRC screening values are not directly applicable to the data acquired from these areas. All measurement data acquired from penetrations, sewers, floor drains and ducts in all OUs other than OU 1 exhibited only background levels.[19]

5.2.1 Survey Unit Identification and Reference Coordinates

Each operable unit was separated into survey units and classified as described in Table 10.1. A reference coordinate system was established in each room or land area and the survey map reflected the grid boundaries. For Class 3 building surface areas, there was no limit on survey unit dimensions, but a manageable survey unit size was established. For Class 1 building surface areas, a maximum area of 100 square meters limited the size of the survey unit. Class 1 survey units for land areas were limited to 1,000 square meters. There were no Class 2 building surfaces or land areas at the DORF.

5.2.2 Survey Unit Classification

Sites scheduled for final status survey were divided into discrete survey units of a specific size and shape for which separate decisions relative to the DCGL were made. Impacted areas are those areas with a potential of being contaminated. Non-impacted areas are those that do not have a potential for being contaminated and were thus not surveyed as part of the final survey.

Each survey unit was classified as Class l or 3. In general, a Class 1 survey unit was an impacted area where there was expected to be locations with concentrations of residual radioactivity that exceed the DCGL. A Class 3 survey unit was an impacted area where there was no expectation of residual radioactivity greater than a fraction of the DCGL, or a buffer zone around Class 1 areas with a low contamination potential. Previous remediation precluded an area from being designated a Class 2 or Class 3 area.

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[18] Visual Sample Plan (Version 5.3.1), written and distributed by Pacific Northwest National Laboratory, is a software tool for the development of sampling plans that are based on statistical sampling theory and the statistical analysis of sample results to support decision-making (i.e., release for unrestricted use). The software couples site, building, and sample location visualization capabilities with optimized sampling design and statistical analysis strategies.

[19] Integrated Environmental Management, Inc., "Investigation of the Diamond Ordnance Radiation Facility (DORF)", Report No. 2008012/G-102381, Chapter 8.

It is important to note that residual radioactivity in the OUs, with one exception, was thought to be comprised of primarily beta/gamma emitters. The exception was in OU 4, which was used to temporarily store a single drum with a "depleted uranium" label affixed just before the investigation began. This action raised the possibility of alpha emitters being present in that OU only. Due to the differing survey techniques for assessing residual beta/gamma and alpha radiations, in light of the low probability of actually encountering alpha emitters in this OU (the drum and its immediate surroundings did not exhibit detectable removable alpha activity), and in addition to its Class 1 status for the presence of other medical/research waste, OU 4 was given a Class 3 designation for uranium isotopes in addition to its Class 1 status for beta/gamma emitters. The Class 1 status in OU 4 called for the performance of more intensive surveys, while the Class 3 status required only limited surveys. If the limited survey data confirmed the presence of alpha activity above the action level, the OU would have been re-classified and the more aggressive Class 1 surveys for alpha radiation would have been performed. As shown in Chapter 8, below, this did not occur. (Although not required in the work plans, alpha survey data were also collected and reported for other OUs due to their simultaneous acquisition during the measurement campaigns.)

5.2.3 Statistical Tests

Compliance with the DCGL for building surfaces was demonstrated by collecting direct measurements of the residual radioactivity present. For the DORF, these measurements were radionuclide-specific and statistical testing of results for comparison to the DCGLs was not necessary.

For Class 3 areas, only a residual level that was a small fraction of the DCGL was expected. Therefore, visual inspection of the survey data, which indicated that they were at least three standard deviations below the DCGL, was sufficient to demonstrate that the release criteria were met. For Class 3 areas, there was an opportunity to forego the use of complicated and rigorous statistical tests and to facilitate the process of release early in the decommissioning effort. However, the surveys were planned to meet the FSS design so that the data could be utilized for expedited release decisions.

There were provisions for Class 1 areas to be subject to more extensive analysis, based on the Sign Test, for areas where the contaminants of concern are not present in the natural background. The Elevated Measurement Criteria (EMC) evaluation was more significant in this case because small areas of elevated activity could have been present. As long as remediation of the area was not required and the residual levels were below the $DCGL_w$ (i.e., non-parametric statistical test), expedited release of the area based on characterization data in lieu of a separate final status survey was deemed acceptable.

For Class 1 areas, direct measurements of a specific number were performed within each survey unit, with provisions in the SAP for evaluating results using the Sign Test ($S+$). The Sign Test is designed to assess uniform residual levels throughout a survey unit and draws direct comparisons between the survey unit data and the chosen release criteria, i.e. $DCGL_w$. The null hypothesis was assumed to be true unless the statistical test indicates that it should be rejected in favor of the alternative.

The null hypothesis takes the assumption that the mean of the sample distribution exceeds the DCGL (H0) is rejected. Because the sample data represent a distribution, some of the data points may be greater than the DCGL or the median of the survey data are greater than the $DCGL_w$. The result of the hypothesis test determines whether or not the sample distribution as a whole meets the release criteria.

If all of the data in the distribution were less than the $DCGL_w$ then visual inspection of the data was deemed sufficient to determine compliance and no statistical evaluation (i.e., Sign Test) would be required. If required, however, the Sign Test would be applied as follows:[41]

- List the survey unit measurements, X_i , I = 1, 2, 3..., N.
- Subtract each measurement, X_i, from the $DCGL_w$ to obtain the differences:

$$D_i = DCGL_W - X_i \quad (where\ i = 1,\ 2,\ 3\ \ldots\ N)$$

- Discard each difference that is exactly zero and reduce the sample size, N, by the number of such zero measurements.
- Count the number of positive differences. The result is the test statistic *S+*. A positive difference corresponds to a measurement below the $DCGL_w$ and contributes evidence that the survey unit meets the release criterion.
- The value of *S+* is compared to the critical values in NUREG-1575, Table I.3.[42] If *S+* is greater than the critical value, k, in that table, the null hypothesis is rejected and the survey unit is eligible for release. Large values of *S+* indicate that the null hypothesis is false and the survey unit exceeds the release criterion.

5.2.4 Number of Measurements

The number of stationary measurements made within each survey unit depended on the non-parametric statistics used to test the null hypothesis, acceptable decision errors, and the relative shift. A minimum number of measurement locations were required in each survey unit to obtain sufficient statistical confidence that the conclusions drawn from the measurements represent the entire survey unit.[43] The minium number of measurements made in each of the survey units is shown in Table 10.4. The following subsections describe the measurement types.

5.2.4.1 Direct Alpha Measurements

In those operable units where uranium was one of the radionuclides of concern, stationary (fixed) alpha measurement were made on the structural surfaces of each survey unit. Measurements were

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[41] U.S. Nuclear Regulatory Commission, *Multi-agency Radiation Survey and Site Investigation Manual*, NUREG-1575, Revision 1, Section 8.3, August, 2000.

[42] U.S. Nuclear Regulatory Commission, *Multi-agency Radiation Survey and Site Investigation Manual*, NUREG-1575, Revision 1, Appendix I.3, August, 2000.

[43] U.S. Nuclear Regulatory Commission, *Multi-agency Radiation Survey and Site Investigation Manual*, NUREG-1575, Revision 1, Section 5.5.2.3, August, 2000.

conducted by integrating the total counts over one (1) minute count times.[44] They were made at the nodes of the grids, using a square grid pattern.

5.2.4.2 Direct Beta Measurements

Stationary beta measurements and stationary low-energy beta measurements were made on the structural surfaces of each survey unit. Measurements were conducted by integrating the total counts over a one (1) minute count time. Measurements were made at the nodes of the grids, using a square grid pattern. Although not specified in the SAP, gross gamma count rate and exposure rate measurements were also performed in a variety of the areas.

No beta measurements were made for soil areas. Instead direct measurement of gross gamma levels were performed.

5.2.4.3 Removable Activity Measurements

Smears for removable radioactivity were taken at each direct measurement location and analyzed for beta and alpha radiation by direct counting. Although alpha measurements were not always specified, alpha data were provided as function of the counting instrument and were therefore recorded. In addition, smears were also analyzed for the presence of H-3 and C-14. These data were reported in units of dpm/100cm^2. The USNRC screening values for surface contamination assume a 10% removable fraction. Thus the use of these screening values as DCGLs requires a demonstration that the removable fractions meet this criterion.

5.2.4.4 Probability of Exceeding the DCGL

The probability that a random stationary measurement from a survey unit would exceed the $DCGL_w$ was defined as Sign P. The variable, P, was used to determine the number of measurements to be performed during the survey and was selected from Table 5.4 of MARSSIM. If the value of the relative shift was not provided, the next lower value was selected.

5.2.4.5 Decision Error Percentiles

Before selecting the number of data points, the percentiles, $Z_{1-\alpha}$ and $Z_{1-\beta}$ were determined. These were represented by the selected decision errors, α and β, respectively, both of which were set at 0.05.

5.2.4.6 Number of Data Points

The number of data points to be obtained from each survey unit for the Sign Test was determined using the Visual Sample Plan, then confirmed using the information in Table 5.5 of MARSSIM. Based on this table, the relative shift, "Δ/σ", and the decision errors were used to determine the number of samples required to evaluate a given survey unit as shown in the SSP. Table 10.4 contains a summary listing.

5.2.5 Location of Measurements and Grid Spacing

Once the number of stationary measurements necessary for demonstrating compliance with the release criteria was determined, it was important to determine the locations of each measurement point based on the survey grid developed for the spacing L. It was also important to determine whether the minimum detectable activity for scanning (MDA for beta radiation), which is calculated

[44] A count time of up to two minutes may be conducted as necessary to attain appropriate detection levels. The survey record will be modified, as necessary.

in accordance with Chapter 5, is below the $DCGL_{EMC}$. If this condition was not met, the number of measurements collected in each survey unit needed to be increased to account for the lack of scanning sensitivity. Therefore, if the scanning MDA exceeded the $DCGL_{EMC}$, an Area Factor, AF, was to be calculated as follows:

$$AF = \frac{MDA_{scanning}}{DCGL}$$

The size of the area of elevated activity that corresponded to the Area Factor, AF, is consistent with the values in Table 5.7 of MARSSIM. The number of measurements required to account for the lack of scanning sensitivity, N_s, was calculated as follows:

$$N_S = \frac{A}{A_S}$$

where A = Survey Unit area; and A_s = the size of the area of elevated activity that corresponds to the Area Factor. If N_s exceeded the number of measurements calculated as shown in Table 10.4, then N_s represented the minimum number of measurements to be collected from each survey unit.

Grids were established for the purpose of referencing locations of measurements and sampling, relative to structure and/or site features. The grid spacing for the measurements and samples were determined assuming a square grid pattern as follows:

$$L = \sqrt{\frac{A}{N}}$$

where L = grid spacing, A = survey unit area (square meters), and N = the number of measurements. The starting point for the survey was established for each survey unit by selecting a reference point within the unit (i.e., a corner of the room). A random number generator was used to provide a number between 0 and 1 for an initial offset from the reference point in both the x and y coordinates. The random number pair was multiplied by the calculated grid spacing providing the offset from the reference point for the measurement location in that grid square.

Upon establishing the first grid location, the aforementioned grid spacing was used to establish a grid system throughout the survey unit. If a survey unit included floor or walls, the grid was extended to all surfaces from the initial point. Once gridded, the surveyors verified that the number of grid locations satisfied the calculated number of measurements. If not, a smaller grid spacing was used to ensure the necessary number of measurements and samples were obtained.

5.2.5.1 Relative Shift

The relative shift is defined as "Δ/σ" where "Δ" is the DCGL minus the Lower Bound of the Gray Region (LBGR) and the"σ" is the standard deviation of the contaminant distribution. In order to calculate the relative shift, the DCGL must be determined and assumptions made to estimate the LBGR and the standard deviation of the measurement distribution.

MARSSIM suggests that the LBGR be approximately 50% of the DCGL but that value can be adjusted to provide relative shift between the range of 1 to 3. For the Class 1 areas, the LBGR was set to 0 and the relative shift was thus 1.5.

The standard deviation may be estimated from preliminary survey data, prior surveys of similar areas and materials, or the standard deviation of a reference background area. However, it is important to note that "σ" represents the standard deviation in data acquired prior to release and when all area decontamination is thought to be complete. If no reference data are available to make a reasonable estimate, MARSSIM suggests using 30% of the mean survey unit background value.

In this characterization plan, the LBGR was conservatively assumed to be zero. The value $\Delta/\sigma \geq 3$ was assumed to be applicable for the Class 3 statistical analysis and $\Delta/\sigma = 1.5$ for Class 1 analyses. After the investigation was complete, the data spread in all of the OUs was evaluated, with re-calculated Δ/σ values all exceeding the assumed values, thus the number of measurements performed in each survey unit was adequate.[45]

5.2.5.2 Decision Error

There are two types of decision errors applicable to the survey and analytical results. These are Type I (α) and Type II (β) errors. A Type I error, or false positive, refers to the probability that a survey result/measurement is above the release criteria when in fact it is not. A Type II error, or false negative, is the probability of determining that a result/measurement is below the release criteria when it is not.

The probability of making decision errors can be controlled by adopting an approach called hypothesis testing. In this case, the null hypothesis (Ho) will be treated like a baseline condition. As specified in MARSSIM, Ho is that residual radioactivity in the survey unit which exceeds the applicable release criterion.[46] This means that the site or survey area will be assumed to be contaminated until proven otherwise. For testing the survey data from Operable Units 4 and 5, both Type I (a) and Type II (β) were set at 0.05 or 5 percent, meaning 95% confidence in the final conclusions.

5.2.6 Elevated Measurement Criteria

Measurements made in a Class 1 survey unit required that the area from which an elevated result was obtained be evaluated using the EMC. The EMC provides assurance that small areas of elevated activity receive proper attention and that any area having the potential for significant dose contribution is identified. It is intended to identify potential failures in the remediation process. If measurement results exceeded the elevated measurement criteria, the survey unit in question was rejected for meeting the criteria for release for unrestricted use, meaning it would require additional remediation.

Scanning measurements were intended to flag measurement data that exceed an action level. If a data point was flagged, then an investigation of that data point took place, which could indicate potential failures in the area classification process. If the scan MDC was not low enough to detect the $DCGL_{EMC}$, then an additional number of measurements must be added to the sampling data set to compensate for the lack of sensitivity in the scans.

It was recognized that the residual radioactivity resulting from reactor operations at the DORF facilities might consist of not only surficial, non-volumetric radioactivity on building surfaces, but

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[45] Duvall, K. C., "Verification and Quality Review of DORF Investigation Data", March 16, 2010.

[46] U.S. Nuclear Regulatory Commission, Multi-agency Radiation Survey and Site Investigation Manual, NUREG-1575, Revision 1, August, 2000.

also volumetric radioactivity from neutron-activated materials, particularly in the concrete walls. Normally, volumetric activity from neutron activation exhibits uniform spatial distribution when measured on the face of the concrete walls. However, it was known that the DORF research reactor operated with thermal neutron columns to test instruments and materials. Therefore, the IEM Team anticipated that collimated beams of neutrons may have produced localized areas of activation in the concrete, and created non-uniformities in the spatial distribution of volumetric radioactivity in the walls. While there were provisions in the SAP for EMC testing of small areas of elevated activity, such comparisons were not necessary after data acquisition because gross (i.e., background included) measurement results, with the exception of OU 1, were below the applicable DCGLs.

5.2.7 Surface Scanning

For Class 1 survey units, beta scans (depending on surface roughness) were performed over 100% of the accessible building surfaces (interior) using a proportional detector while listening to the audible output of the instrument. The scan was designed to detect small areas of elevated residual radioactivity that may not be detected by the static measurements, using a systematic pattern. The detector was maintained within one (1) centimeter of the surface or less, if surface conditions permitted. The scanning speed was determined based on detector sensitivity (see Appendix 12.6), and locations of elevated direct radiation levels were identified for further investigation.

Class 3 survey units required judgmental scanning coverage. Beta scans were performed for less than 30 percent (30%) of the surface area. Those areas with the highest potential for elevated residual radioactivity, based on professional judgement, were selected for scanning. The scan MDC was less than 50% of the $DCGL_w$, with the action level for flagging data points for further investigation based on the scan MDC. Surface scans were performed at 30% of the floor and/or lower wall locations by creating a temporary grid square and extending from the location where the smear and direct reading were taken. The location of the scan was documented on a scale drawing of the survey unit to ensure reproducibility. If obstacles interfered with the selected pattern, the shape of the area was changed or an alternate location selected, at the discretion of the technician. Any modifications were documented in the survey package for the affected survey unit(s), along with the location and results of any additional measurements performed at locations of concern by the health physics technician or the Project Manager (e.g., high-traffic areas, or areas where discoloration or other indicators were present).

Class 3 survey units where uranium was one of the ROCs were also subject to total gross alpha measurements. A total of 18 measurements were made in locations that were selected based on professional judgement, with all locations documented on a drawing of the survey unit to facilitate reproducibility.

5.2.8 Investigation Levels

The SAP had provisions for investigation to be performed or a survey unit reclassified, which would trigger a re-survey of the area. The following is a brief summary of the investigation actions applicable to the investigation:

- If survey measurements in a Class 3 survey unit exceeded a fraction of the limiting DCGL, that survey unit was to be re-evaluated and, if necessary, reclassified and subject to further survey accordingly.

- If the investigation demonstrated that any OU contained residual radioactivity above the applicable DCGL, that OU would no longer be eligible for expedited release under the investigation. Instead the area was to be addressed further (i.e., remediated, additional study) and subject to additional final status surveys as part of the DORF-wide decommissioning.

During the performance of the surveys at the DORF, none of these conditions were encountered except in OU 1 and OU 4, both of which were designated Class 1 areas for beta/gamma contamination. In OU 4, a test decon process removed the radioactivity of interest, but the area classification for beta/gamma contamination remained the same.

For surface soil in outdoor operable units, the USNRC screening values with the unity rule applied as follows were deemed sufficiently-conservative for use as DCGLs:

$$\frac{C_1}{DCGL_1} + \frac{C_2}{DCGL_2} + \frac{C_3}{DCGL_3} + \ldots \frac{C_n}{DCGL_n} < 1$$

where C = the concentration of ROC_n. For building surfaces subject to the medical waste source term, the USNRC screening values were also deemed sufficiently conservative for ensuring their release.

A listing of the source term applicable to the DORF is presented in Table 10.2, along with the applicable DCGLs. These take into account the radiological significance of both the demolition and the building re-use scenarios for building surfaces, soil and volumetrically-contaminated materials, as applicable, with the most conservative (i.e., the lowest) value from any scenario selected for survey planning.

The limiting screening value for surface scans and stationary measurements, taking into account detection efficiency and the most restrictive DCGL, was associated with the isotope Co-60 with its DCGL of 7,100 dpm (beta)/100 cm^2. The action level for use during data acquisition was set at 50% of the limiting DCGL (i.e., 3,500 dpm (beta)/100 cm^2).

In order to ensure the direct exposure pathway for continuous presence (industrial use scenario) did not result in a dose potential in excess of the 25 millirem per year dose criterion, the ambient exposure rate within the area must be less than 25 millirem × 1000 microR/millirem ÷ 2000 work hours = 12.5 microR per hour. To ensure an element of conservatism, a value of 10 microR per hour, above the instrument background, was selected as the action level.

For the beta-sensitive instruments, the detection levels from the daily checks (see Chapter 6, below) were examined in light of the specified action level of 3,500 dpm/100 cm^2, which was based on half of the most limiting DCGL (7,100 dpm/100 cm^2 for Co-60). The following are the parameters used as input to the assessment:

- The average beta energies for ROCs Co-60, Eu-152 and Eu-154 are 96, 300 and 225 keV, respectively. Only the beta energy/yield for Co-60 via beta emission (half-life of 5.3 years) was included in this evaluation. The 1,550 keV (max) beta emission with the 0.23% yield for Co-60 via the isomeric transition decay path (half-life of 10.5 minutes) was not considered.

- The corresponding emission abundances (yields) for Co-60, Eu-152 and Eu-154 are 100%, 28% and 100%, respectively.

- The average beta energy and yield for the radionuclide used for daily efficiency checks (Tc-99) are 85 keV and 100%, respectively.

The relationship between instrument detection level (MDA), efficiency (ε) and yield (Y) is generally defined as follows:

$$MDA \propto \frac{C}{\varepsilon \times Y}$$

where C = instrument minimum detectable activity (dpm) as determined during daily checks. For beta counting, the efficiency of the instrument generally increases with increasing beta energy, resulting in a decrease in detection level. For example, nominal efficiencies for 85, 563 and 1,410 keV betas are 16, 36 and 55 percent.[47] On the other hand, the smaller the yield, the lower the detection level. For C = 240 dpm for a 100 square centimeter probe, the MDA based upon the use of the Tc-99 check source with its 85 keV average beta energy would be generally proportional to the following:

$$MDA_{Tc-99} \propto \frac{240}{0.16 \times 1.00} \propto 1{,}500\ dpm/100\ cm^2$$

As shown in Appendix 15.17, the minimum detectable count rate for the stationary beta instruments during the investigation ranged from 231 to 238 counts per minute, thus an assumption of 240 is conservative.

For the ROC with the lowest yield (i.e., Eu-152), with its average beta energy of 300 keV, the MDA would be conservatively proportional to the following:

$$MDA_{Eu-152} \propto \frac{240}{0.36 \times 0.28} \propto 2{,}381\ dpm/100\ cm^2$$

Therefore, the MDA for the instruments as determined during the daily checks was actually 2,381 ÷ 1500 = 1.6 times higher for Eu-152, which became the most limiting isotope. This value of 240 × 1.6 = 384 dpm/100 cm^2 is well-below the action level of 3,500 dpm dpm/100 cm^2, thus the limiting isotope assumption remained valid throughout the investigation.

[47] U. S. Nuclear Regulatory Commission, NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions", Table 4.2, June, 1998.

6 INSTRUMENTATION

6.1 Selection Criteria

In general, the radiation detection instrumentation used for the investigation was selected and operated according to the type of analysis being performed, and to ensure sensitivities are sufficient to detect the identified radionuclides at the minimum detection requirements. Table 10.5 provides a list of the instrument types that were used for the DORF final status survey, along with the types of radiations they detect, and the necessary calibration sources. The following subsections provide additional information.

It is important to note that the measurement of gross beta radiation using a plastic scintillation detector (Ludlum 43-93) or a gas flow proportional detector (Ludlum 43-68), as intended for this effort, ensured reasonable detection of all isotopes with beta energies ranging from an average of approximately 65 keV to a maximum of 1,450 keV. Detector efficiencies were assessed using a known quantity of Tc-99, with its average beta energy of 84 keV and its maximum of 294 keV. Because detector efficiency increases with increasing beta energy, measurement response for those with elevated energies was higher, thus the survey design was conservatively set. For example, for Eu-152, the calculated detector efficiency was approximately three times higher than the efficiency for Co-60.[48,49] However, if one assumes it is equal to that of Tc-99, and because the limiting DCGL for the ROCs is associated with Co-60, Eu-152 detectability was ensured. The following is a listing of the radiation energies associated with the ROCs at the DORF:

Radionuclide	Beta Energy (keV)	
	Average	**Maximum**
H-3	6	19
C-14	49	156
Mn-54	No beta emission Gamma 834 keV	NA
Fe-55	No beta emission EC decay	NA
Co-57	No beta emission Gamma 122 keV	NA
Co-60	96	318
Cs-134	210	658
Eu-152	704	1865

48 U.S. Nuclear Regulatory Commission, Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions, NUREG 1507, June, 1998.

49 The detector efficiency for Tc-99 and Co-60 is calculated to be 0.16 cpm/dpm for the Ludlum Model 43-37 probe (Ludlum Floor Monitor 239-1F). The same detector is reported to have a detector efficiency of 0.36 for a beta maximum energy of 1,415 keV, similar to Eu-152.

Radionuclide	**Beta Energy (keV)**	
	Average	**Maximum**
Eu-154	176	569
U (depleted)	4.1 MeV (alpha)	NA

Surveys for gross alpha activity were performed using either a gas-flow proportional counter or a zinc sulfide detector, the selection of which depended on accessibility issues. Detection efficiency for the alpha energies from the uranium decay series was excellent in light of surface roughness.

6.2 Instrument Calibration

All instrumentation used during the investigation were calibrated, checked and used in a controlled manner, with performance documented. All portable instruments were calibrated by a licensed commercial calibration service using National Institute of Standards and Technology (NIST) traceable sources and calibration equipment. Instrument calibration included:

- High voltage calibration;
- Discriminator threshold calibration;
- Window calibration;
- Alarm operation verification;
- Scaler calibration verification.

The calibration of the detectors included:

- Operating voltage determination;
- Calibration constant determination; and
- Dead time correction determination.

Labels showing the instrument identification number, calibration date and calibration due date were attached to all portable instruments. A copy of all relevant calibration records are shown in Appendix 12.6.

6.3 Calibration Sources

All sources used for on-site instrument calibration, daily checks or efficiency determinations were representative of the instrument's response to the identified radionuclides and traceable to NIST. These sources included ^{99}Tc and ^{137}Cs sealed sources for beta and gamma radiation detection, respectively, and ^{230}Th sources for alpha radiation.

The Project Manager and the Site HP controlled the use and storage of the radiation sources used for instrument response checks and efficiency determination. They were stored securely and signed out when needed. Possession of the sources at the DORF was authorized by Fort Detrick in an

Army Radiation Permit. At the end of the on-site effort, all sealed sources were accounted for and removed from the site.

6.4 Response Checks

Periodic instrument response checks were conducted to assure constancy in instrument response, to verify the detector is operating properly, and to demonstrate that the measurement results were not the result of detector contamination or failure. Instrument response was checked each day before the instrument was used. The check sources were used to duplicate the same type of radiation that was being measured with the particular instrument using a specified source-detector alignment that could be easily repeated. If the instrument failed its response check, it was not be used until the problem was resolved.

6.5 Minimum Detectable Activity

Minimum Detectable Activity (MDA) is defined as the smallest amount or concentration of radioactive material that will yield a net positive count with a 5% probability of falsely interpreting background responses as true activity. The MDA is dependent upon count times, geometry, sample size, detector efficiency, background, and for scanning the scanning rate and the efficiency of the surveyor.[50] Nominal detection sensitivities were calculated using the guidance in NUREG-1507. From there, instruments were selected to achieve detection sensitivities of less than the $DCGL_w$ for direct, static measurements and less than the $DCGL_{EMC}$ for scan surveys.

The required MDAs for direct measurements, surface scanning and removable activity measurements were set based on detection sensitivity. Appendix 15.17 shows the calculated values. Since the MDAs for scanning were equal to or less than the applicable DCGL, the scanning MDAs did not effect the number of measurements or samples required to evaluate a specific survey unit for compliance with release criteria. In all survey units, the MDA requirements were met.

6.5.1 Direct Alpha and Beta Measurements

The equation that was used for calculating the MDA for direct measurements of alpha or beta activity is:

$$MDA = \frac{\frac{2.71}{t_s} + 3.29\sqrt{\frac{R_b}{t_s} + \frac{R_b}{t_b}}}{E \times \frac{A}{100}}$$

where MDA = Minimum detectable activity (dpm/100 cm^2), R_b = Background count rate (cpm), t_b = Background count time (minutes), t_s = Sample count time (minutes), A = Detector area (cm^2), and E = Detector efficiency (counts/disintegration).

[50] U.S. Nuclear Regulatory Commission, *Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions*, NUREG/CR-1507, December, 1997.

6.5.2 Beta Scans

The equation that was used for calculating the MDA for beta scans (MDA_{SCAN}) is:

$$MDA_{SCAN} = \frac{d' \times \sqrt{b_i} \times \frac{60}{I}}{E_I \times E_S \times \sqrt{\rho} \times \frac{A}{100}}$$

where MDA = Minimum detectable activity (dpm/100 cm^2), d' = Decision error assumed to be 3.28 for α =0.05 and β=0.95, I = Observation counting interval (scan speed divided by detector width), b_i = Background count per observation interval, E_I = Detector efficiency, E_s = Surface efficiency, ρ = Surveyor efficiency (Assumed to be 0.5), and A = Detector area (cm^2). ISO-7503 recommends using a surface efficiency based on the type of radiation and radiation energy in the absence of experimentally derived values.[51] A surface efficiency of 0.25 is recommended for beta radiation with a maximum beta energy between 150 keV and 400 keV.

6.6 Measurement Positioning

6.6.1 Outdoor Areas

IEM's Land Area Survey Program (LASP) was used to perform the walk-over scans and stationary measurements in outdoor areas. The LASP uses a global positioning system for precise data acquisition, coupled to a Ludlum Model 44-10 detector and Ludlum Model 2241 rate meter. Position, time and instrument read-outs are acquired once per second and stored away for later processing.[52]

Survey data were acquired by walking over the footprint of the areas to be surveyed with the sensitive area of the survey instrument held approximately one foot above the soil surface. This detector height was selected as one that would maximize detection capability yet minimize geometry (i.e., source-to-detector) effects. As the surveyor progressed, marks were made on the ground using a long-handled paint wand to facilitate tracking of paths.

When the on-site work was complete, the data were transferred directly into a computer-based Geographical Information System (GIS) that displayed them in tabular form for statistical analysis and in map form that was laid over a photographic image of the property. (Individual data points can be located and re-located to as close as six inches in most cases after post-processing is complete using the GPS coordinates given in the final report.) The measurement results from the survey area, which included the contribution from background, were compared to the applicable DCGL in order to demonstrate the radiological status of the outdoor survey area.

6.6.2 Indoor Areas

A spatial locating system was used to locate survey/sample points and also to perform scan surveys inside of the buildings. The system consisted of an auto-tracking total station with a laser-range finder capable of automatically tracking radiation detectors as measurements are being acquired. Position and radiation data are recorded every second. The system was coupled with conventional

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[51] International Organization for Standardization (ISO), *Evaluation of Surface Contamination*, ISO 7503, 1988.

[52] For more information on the LASP, see http://www.iem-inc.com/iemland.html.

radiation detectors (i.e., gas-flow proportional counters) for beta scans of floors and walls. Hand-held instruments were used for over-head areas, with results recorded separately.

7 PROCEDURES

7.1 Radiological Measurement Methods

7.1.1 Total Radioactivity Measurements

Handheld survey instrument backgrounds were acquired daily while instruments were in use. The measurement location was the field office trailer positioned on the east side of Building 516. Reference background measurements for surfaces were collected inside of Building 513. Although there was no evidence radioactivity ever having been used in Building 513, it was designated an OU and a Class 3 survey unit only because an old facility map referred to it as a "Radioactive Instrument Area". However, it was assumed to have the least potential for contamination as a result of DORF operations (see Section 2.4.5, above) because it exhibited no contamination above background during the 1997 scoping surveys, and it contained surfaces similar to those found within Building 516, thus the reason for its selection as a reference background area.

Background data were also collected over cinder block and concrete surfaces by acquiring 10 one-minute measurements for gross beta, gross alpha, and gross low-energy beta activity. All measurements were made with the detector in close proximity to the measurement surface. Appendix 15.7 contains the background survey information.

All other material types (i.e. metal, vinyl, and wood) were considered by the project team to contain no natural radioactivity of significance. Therefore, the daily ambient background measurements for each detector type that were performed as part of the daily QC checks were considered to be applicable background values for these materials and surfaces.

It is important to note that no credit for background was taken during the interpretation of characterization findings. All measurement results are conservative in that they include the contribution of background radionuclides and thus over-estimate the actual ROC activity.

Direct measurements for residual radioactivity were acquired using the instrumentation described above and with the following implementation considerations:

- Background was determined with a detection sensitivity and accuracy that will support MDA levels low enough to meet project goals, and by using the same instruments and techniques as used for assessing final site conditions.

- Because building design and construction had a marked influence on the survey efforts, the time required to conduct a survey of a building interior surface was directly proportional to the total surface area.

- Consideration for the possibility that contamination seeped into porous surfaces or has been sealed in place was given.

- As necessary, furnishings or other equipment were removed in order to achieve access to survey locations.

- The possibility that process and/or structural modifications were made was investigated to ensure that all possible sources of contamination and radionuclides, including those covered up by construction, were identified.

- Where voids exist in walls (e.g., drywall on studs, cinder block, or tile), the potential for residual contamination to have accumulated in these voids was considered.

If the fixed or direct measurement results were less than the $DCGL_w$, no further action was required. If greater than the $DCGL_w$ but less than the $DCGL_{EMC}$, the measurement set for that survey unit was to be increased by 10% in order to improve the statistical base for decision-making. If greater than the $DCGL_{EMC}$, professional judgment was to be used to collect whatever additional information necessary to prepare a remedial action plan. During the surveys, however, no measurement results exceeded the $DCGL_w$ except for OU 1 and one location within OU 4, where a test decon served to reduce the level of residual radioactivity to well-below the action level (see Section 8.4, below).

7.1.2 Removable Radioactivity Measurements

Background for removable activity measurements was determined by counting blank (i.e., unused) smears by each of the measurement methods outlined below. At least five (5) percent, but no less than one, of each counting batch was comprised of blanks.

Smears for determination of beta activity were counted (analyzed) using a plastic scintillator equivalent to a Ludlum Model 2929 smear counter. The analytical method was as described in one of IEM's procedures.[53] Smears for determination of low-energy beta activity were counted by the method of liquid scintillation at a nearby but off-site laboratory.

If the removable activity results were less than the $DCGL_w$, no further action was required. If greater than the $DCGL_w$ but less than the $DCGL_{EMC}$, the sample set for that survey unit was increased by 10% in order to improve the statistical base for decision-making. If greater than the $DCGL_{EMC}$, professional judgment was used to collect whatever additional information was necessary to prepare a remedial action plan. No removable activity measurement results exceeded the applicable DCGLs.

7.1.3 General Area Dose Rate Measurements

Background for assessing general area dose rates or photon exposure rates was assessed at locations within the DORF that were unlikely to have been impacted by previous licensed operations. Measurements were made at both indoor and outdoor locations, with the instruments held at the applicable measurement height above the ground surface and at least one (1) meter from any wall or fixture. General area dose rates were acquired using the instrumentation described in Chapter 6, above, and pursuant to the provisions of IEM's procedures.[54]

If the general area dose rate results were equivalent to the ambient gamma background, no further action was required. If greater than background but less than 20 microrem per hour above background, the sample set for that survey unit was to be increased by 10% in order to improve the statistical base for decision-making. If greater than 20 microrem per hour above background, professional judgment was to be used to collect whatever additional information might be necessary to prepare a remedial action plan.

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[53] Integrated Environmental Management, Inc., Radiation Safety Procedure No. RSP-001, "Radiation Protection Program Plan", Section 5.1.

[54] Integrated Environmental Management, Inc., Radiation Safety Procedure No. RSP-001, "Radiation Protection Program Plan", Section 5.8.

7.1.4 Outside Area (Walkover) Surveys
Walkover survey data (i.e., count rates) over land areas were acquired using the instrumentation described in Chapter 6, above. The measurement method was consistent with the requirements of IEM's procedures, as modified for the investigation.[55] Coverage of a minimum of 40% of the land area was specified in the SAP.

To the greatest extent practical, walk-over surveys were performed over land areas only, even though multiple surfaces were present within the perimeter fence and outside of the buildings. Survey unit data planning and assessment assumed a common DCGL (i.e., counts per minute and, in the case of soil samples, picocuries per gram).

7.1.5 In-Situ Gamma Spectroscopy
A portable gamma spectrometer, coupled to a 2-in. by 2-in. sodium iodide detector, was utilized in the Exposure Room for qualitative assessment of the type and general magnitude of radionuclides contributing to the ambient exposure environment. The assessment method was consistent with the requirements of IEM's procedures, as modified for the investigation.[56]

7.1.6 Sample Collection and Analysis
Concrete cores were collected from OU 1 and analyzed at an off-site analytical laboratory for the applicable ROCs. In addition, surface and subsurface soil samples collected from OU 7 were also sent off-site for analysis. Materials from selected locations (e.g., conduit and sump debris, paint chips, etc.) were also sampled and analyzed. Finally, a number of samples were collected for chemical analysis.

A commercial lab with a Quality Assurance (QA) plan, National Environmental Laboratory Accreditation Conference (NELAC) accreditation and compliance with the Department of Defense Quality Systems Manual for Environmental Laboratories (v. 3), was selected to analyze the radiological and non-radiological samples. Prior to submitting the samples, a letter of specification for the necessary measurement results and analytical methods was prepared and included, by reference, in all purchase orders. The detection limits specified for the analyses were sufficiently low to ensure the sum of fractions criterion for release could be met (i.e., less than 10% of the applicable DCGL).

Samples sent for hazardous (or non-radiological) analysis were screened on-site to verify that they did not contain residual radioactivity. All such samples were appropriately packaged, preserved, secured (e.g., Chain-of -Custody documentation and custody seals) and sent to a NELAC-accredited lab for evaluation.

7.2 Data Assessment
For each survey unit, the surveyor developed a package or portfolio by performing a walk-down and preparing a worksheet/tracking sheet that outlined the general survey instructions, location codes, and any specific survey instructions for any abnormal conditions within the survey area. Each room was cleared of all loose equipment and materials to the maximum extent possible. Completion and

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[55] Integrated Environmental Management, Inc., Radiation Safety Procedure No. RSP-001, "Radiation Protection Program Plan", Section 5.7 and 5.8.

[56] Integrated Environmental Management, Inc., Radiation Safety Procedure No. RSP-001, "Radiation Protection Program Plan", Section 5.7 and 5.8.

review signature blocks were used to track the progress of the surveys. During the survey, the surveyors updated the survey package(s) with the survey data and results of any special surveys or sample analyses performed.

Once all surveys were complete, the data were reviewed and evaluated in accordance with the "Data Interpretation and Assessments" and "Decision-Making" sections of the SAP to demonstrate that the residual radioactivity on building surfaces is less than the applicable $DCGL_w$. All areas exceeding the $DCGL_w$ were reassessed in terms of area classification and removed from the process of expedited release during the investigation, as applicable.

Data sets were evaluated to ensure the proper number of measurements were made. If insufficient, the grid size was reduced and the area re-surveyed. Individual volumetric results that were below the applicable DCGLs were then subject to the following sum of fractions test to adjust the DCGLs for the potential presence of other ROCs to ensure the total dose potential remains below 25 millirem per year if multiple radionuclides were present:

$$\frac{C_1}{DCGL_1} + \frac{C_2}{DCGL_2} + \ldots \frac{C_n}{DCGL_n} < 1$$

where C = the measured concentration of the ROC and DCGL = the DCGL for that ROC.

7.3 Data Validation

The survey data were reviewed by the Project Manager, the Project CHP and the Quality Assurance Manager to verify that they were authentic, appropriately documented and technically defensible. (Data that were input to spreadsheets were also reviewed by IEM's Quality Assurance Officer for transcription errors.) The review criteria for data acceptability included the following items:

- The instruments used to collect the data were capable of detecting the radiation of interest at or below the DCGL and less than 0.5 DCGL for Class 3 areas.
- The calibration of the instruments used to collect the data was less than twelve (12) months old;
- Instrument response was checked with satisfactory results before the instrument was used;
- The MDAs and assumptions used to develop them were appropriate for the instruments and the survey methods used to collect the data;
- The final survey data set consisted of qualified measurements that were representative of the current facility status and collected as prescribed by the survey design; and
- The data were properly recorded.

A discrepancy existed if one or more of these criteria were not met. In that case, the discrepancy was to be reviewed by the Project Manager and the reasons for the acceptability of the data or the corrective actions taken to restore data acceptability were to be documented. In fact, all of the aforementioned criteria were met.

7.4 Requirements for Release

A survey unit would meet the requirements for release for unrestricted use provided (1) an adequate number of measurements were taken; (2) the statistical tests were passed and (3) the EMC evaluation, as applicable, was passed.

7.5 Statistical Evaluation

The results of the statistical testing allowed one of two conclusions to be drawn:

Conclusion 1: The survey unit meets the dose-based release criteria.

Conclusion 2: The survey unit fails to meet the dose-based release criteria.

The first means the data do indeed provide statistically significant evidence that the mean level of residual radioactivity in the survey unit is less than the applicable criteria, $DCGL_w$, that small areas of elevated activity do not exceed the $DCGL_{EMC}$, and that the decision to release this area can be made with confidence and without further analysis.

In the case of the second conclusion, the data do not provide sufficient statistically significant evidence that the mean level of residual radioactivity is acceptable, or that pockets of small areas of elevated activity may exist, thus further analysis to determine why will be required. The survey unit might then require re-survey and/or collection of another set of discrete measurements for the statistical analysis. Alternatively, the survey unit may have been reclassified and removed from the expedited release process during the investigation. The following shows the conclusion that is associated with each result type from comparison with the DCGL:

Survey Result	Conclusion
Largest survey unit measurement is less than release criteria	Survey unit automatically meets release criterion
A fraction of the survey unit measurements are greater than release criteria	Conduct Sign Test and elevated measurement criteria comparison
Results of the Sign Test exceed the critical value established for the survey unit	Survey unit does not meet release criterion

7.6 ALARA Considerations

The USNRC's tables of screening criteria for reuse of facilities (structures) and for the release of land areas (soil) were used as the DCGLs for the final status survey of the DORF. In light of the inherent conservatism built into these criteria during their development, an analysis to demonstrate that they present a dose potential that is as low as reasonably achievable (ALARA) is not required.[57]

7.7 Survey Records

The DORF Project Manager maintained records of surveys in the survey packages for each area. Each survey package included the following records depending upon the survey design and protocols:

- Survey package worksheet giving the package identification, survey location information, general survey instructions and any specific survey instructions;
- Survey Unit Diagram of the area to be surveyed as available;
- Photographs of the survey area to show special or unique conditions; and
- Data sheets to record the results of the surveys and analyses.

The Site HP generated a report that presented all raw data, converted data, and information by survey location. The site HP and the DORF Project Manager reviewed them for completeness, accuracy, suspect entries, and to compare results to the applicable DCGLs. Any changes to the data tables, such as detector efficiency and background, that could affect survey results required the DORF Project Manager approval. In addition, changes to data in primary tables required a written explanation, to be attached to the survey report and maintained as a permanent project record.

Data and document control included the maintenance of the raw data files, translated data files (spreadsheets), and documentation of all corrections made to the data. All electronic databases were backed up daily.

[57] U. S. Nuclear Regulatory Commission, “Consolidated Decommissioning Guidance”, NUREG-1757, Vol. 2, Rev. 1, Section 6.2, September, 2006.

8 SURVEY RESULTS

8.1 OU 1 (Building 516, Exposure Room)

The Exposure Room on the lower floor of Building 516, was subject to surface beta scans, measurements of ambient gamma radiation at a height of three feet from the floor and from all vertical surface measurement locations, fixed contamination surveys (beta and tritium) and removable contamination surveys (beta and tritium). Although not required, the removable contamination smears were analyzed for both alpha and beta activity.

Table 10.5 is a listing of the instrumentation used for these measurements. Appendix 12.7 contains a graphical representation of the surface beta scans and fixed beta measurements for the four walls and the floor. The individual data points are contained in the spreadsheets in Appendix 12.8.

It is important to note that the fixed beta measurements and the beta scans are reported in gross activity units, meaning background was not subtracted. Because two different instruments were used for these measurements, with the scanning instrument being the larger of the two, the results do not compare directly. They do, however, show that both instruments were measuring generally similar results at the same measurement point, with the fixed results being somewhat lower than the scan results.

Table 10.6 contains a summary of the results along with the applicable DCGLs. In general, detectable residual radioactivity exists on the concrete surfaces, and elevated concentrations of Eu-152 and Eu-154 only were noted in the concrete cores. Low levels of natural radioactivity (i.e., uranium/thorium decay series and K-40) typical of concrete and cement were also noted.

A series of concrete cores were collected from this OU at the locations shown in Figure 11.6. The coring work was performed by A. E. Brice, a local (Baltimore, Maryland) concrete cutting/coring firm. Work commenced on August 5, 2009 and was completed on August 7, 2009. Each of the 20 cores drilled into the ceiling, walls, and floor of the Exposure Room and the four deep cores drilled into the former reactor pool area were advanced using three-inch bits (2.5-inch bits for the pool cores) to cut and recover the concrete. The analytical results are listed in Appendix 12.8, and the radiological Certificates of Analysis in Appendix 12.9.

The four deep cores were collected from the former reactor pool area in order to determine if the materials used to backfill the pool (i.e., the parapet) were significant. One of the cores (identified as Pool 3) was advanced through the eastern wall of the cavity in the approximate location of Rockwell Core Location No. 24 (see Figures 14.6 and 14.7). All of the cores were scanned using hand-held instrumentation with unremarkable results (see Appendix 12.10).

The dose assessment parameter used to ensure compliance with the 25 millirem per year radiation dose objective for release of OU 1 for unrestricted use is the DCGL. For this investigation, the DCGLs were set to be equal to the USNRC's screening value for each radionuclide potentially-present in this OU (see Table 10.2). Measurements of ambient gamma dose rates were also utilized to estimate the external dose potential for an industrial worker scenario.

The most limiting DCGL for surface scans and stationary measurements is that associated with the isotope Co-60. That value is 7,100 dpm/100 cm^2 (beta/gamma). In order to ensure the direct exposure pathway for continuous presence (industrial use scenario) did not result in a dose potential in excess of the 25 millirem per year dose criterion, the ambient exposure rate within the area must be less than 25 millirem × 1000 microR/millirem ÷ 2000 work hours = 12.5 microR per hour. To ensure an element of conservatism, a value of 10 microR per hour, above action levels, was selected as the action level. The applicable DCGLs for samples collected from this OU are as shown in Table 10.2.

Surface scans and stationary measurement results in OU 1 were occasionally greater than the DCGL for Co-60, in particular on the south and west walls of the Exposure Room, although average values for the OU were less than the DCGL. The average results were observed to be approximately 3,700 dpm/100 cm^2 gross beta. The maximum value indicated on the west wall was 8,427 dpm/100 cm^2. Removable activity results were all less than the DCGLs for gross beta, H-3 and C-14.

One of the five ceiling cores, two of five floor cores and 12 of 15 wall cores from within the Exposure Room had analytical results that exceeded the volumetric screening value in Table 10.2 for Eu-152 or Eu-154. In general, the highest concentrations were located within in the top one-inch of concrete from wall and ceiling cores. Scans of the cores revealed count rates that dropped from approximately 9,000 counts per minute to approximately 8,000 counts per minute (ambient background) as the detector moved two and five inches from the inner surface (i.e., nearest to the wall/ceiling/floor surface). Beyond five inches, all results were at background. No statistically-significant Co-60 was found in any of the analytical results.

The most dramatic evidence of activity at depth beyond one inch was observed on the Exposure Room wall that directly intercepted the neutron beam from the former reactor core. On this wall, the average concentration of Eu-152 over a the first five inches of depth was 21 pCi/g. The remaining walls exhibited lower average concentrations (i.e., seven, 10, nine and six pCi/g for the east wall, west wall, floor and ceiling), the preponderance of which was confined to the first two or three inches of depth.

The ambient gamma exposure rate inside of the Exposure Room averaged approximately 21 microR per hour (excluding instrument background). When applied to a 2,000-hour working year, this results in an external dose to a hypothetical industrial worker within the Exposure Room of 42 millirem, which exceeds the 25 millirem per year dose limit for release for unrestricted use.

In order to release OU 1 for unrestricted use, the dose potential from all applicable pathways must be less than 25 millirem per year and all measurement data must be below the applicable DCGLs, or will pass when subject to MARSSIM statistical tests and EMC comparisons. Because the investigation results show that assumptions made during the planning stage about the expected low residual levels of radioactivity in OU 1 were not correct, OU 1 was deemed ineligible for expedited release. Additional action with regard to remediation strategies and/or the development site-specific release criteria are required, as are additional final status survey data, before release can be further considered.[58]

[58] The walls, ceiling and floor of the Exposure Room were modeled using the computer code MicroShield (v. 7.0, Grove Software) as six separate rectangular volumes containing Eu-152 in the concentrations measured in the concrete cores. The calculated dose rate at a location in the approximate center of the room, at a height of approximately one (1) meter above the floor and eight (8) meters below the ceiling, with a room average Eu-152 concentration of 8.77 pCi/g was 20.46 microR per hour, which is in good agreement with the average measured exposure rate of 21 microR per hour. In order to reduce the ambient exposure rate such that the 25 millirem per year dose limit would be met, one approach (continued...)

Historical survey data from the walls that surrounded the former reactor pool area, acquired before the pool was backfilled, revealed no radioactivity of significance. The analytical results from the horizontal concrete cores collected from the pool walls during the investigation are consistent with those findings. However, additional cores to the north and south of the pool walls were not acquired due to structural concerns. As a result, data sufficient for subsurface statistical testing in this area are not yet available. An engineering evaluation performed after the on-site portion of the investigation confirmed that additional cores can in fact be collected without concern for negative structural impacts.

In light of the fact that additional measurement data are required in order to assess the release status of the former reactor pool area, it has been designated OU 9 for future study. (For the FSS, OU 9 was not assigned.) The inside surface of the east wall of the Exposure Room is a part of OU 1, while the east wall itself, plus the former reactor pool area, are the new OU 9.

8.2 OU 2 (Building 516, Warm Room)

The Warm Room was subject to surface beta scans, measurements of ambient gamma radiation at a height of three feet from the floor and from all vertical surface measurement locations, total (fixed plus removable) contamination surveys (beta and low-energy beta) and removable contamination surveys (gross beta, C-14 and H-3). Although not required, the removable contamination smears were counted for both alpha and beta activity. The upper walls and the ceiling of this OU was classified as non-impacted.

Appendix 12.7 contains a graphical representation of the surface beta scans and fixed beta measurements for the four walls and the floor. The individual data points are shown in the spreadsheets in Appendix 12.8. Table 10.7 contains a summary of the results along with the applicable DCGLs. Any analytical results from this OU are shown in Appendix 12.8, with the radiological Certificates of Analysis in Appendix 12.9. All results are unremarkable.

The sump in OU 2 was investigated. Water pumped from the sump contained low but detectable H-3 concentrations which was later determined to be residual contamination in a pipe that connected the Warm Room sump to a small (shallow) sump that was co-located with the lead shield hoist in the adjacent Exposure Room (OU 1). The shallow sump was filled with concrete during the 1980 decommissioning of the DORF. Removable activity in accessible areas of the pipe revealed H-3 activity that was well-below the action level. Removable and total contamination surveys were also performed in the sump, the results of which were well-below applicable DCGLs.

Surface scan and stationary measurement results in OU 2, which ranged from 441 to 722 dpm/100 cm^2, were below the action levels. Thus there is no reason to believe the dose contribution from all of the ROCs approaches the applicable dose limit. Removable activity results were likewise unremarkable.

In accordance with MARSSIM, no statistical evaluation is necessary when all survey values in the survey unit are less than the DCGL. Furthermore, the analytical results for all samples collected exhibited trivial concentrations of residual radioactivity.

[58] (...continued)
is to remediate the area such that the effective Eu-152 concentration is reduced to less than 5.3 pCi/g, thus reducing the ambient dose rate. By way of example only, that level could be reached by removing at least five (5) inches of concrete from the south wall only.

8.3 OU 3 (Building 516, Connector Room)

The Connector Room was subject to surface beta scans, measurements of ambient gamma radiation at a height of three feet from the floor and from all vertical surface measurement locations, total (fixed plus removable) contamination surveys (beta and low-energy beta) and removable contamination surveys (gross beta, C-14 and H-3). Although not required, the removable contamination smears were counted for both alpha and beta activity. The upper walls and the ceiling of this OU were classified as non-impacted.

Appendix 12.7 contains a graphical representation of the surface beta scans for the four walls and the floor and the individual data points are shown in the spreadsheets in Appendix 12.8. Table 10.7 contains a summary of the results along with the applicable DCGLs. Any analytical results from this OU are shown in Appendix 12.8, with the radiological Certificates of Analysis in Appendix 12.9. None of the results exceeded the applicable DCGL.

Surface scan and stationary measurement results in OU 3 were less than the limiting DCGL and removable activity results were likewise unremarkable. In accordance with MARSSIM, no statistical evaluation is necessary when all survey values in the survey unit are less than the DCGL. Furthermore, the analytical results for all samples collected exhibited trivial concentrations of residual radioactivity.

8.4 OU 4 (Building 516, Main Floor)

The Main Floor of the DORF was subject to surface beta scans, measurements of ambient gamma radiation at a height of three feet from the floor and from all vertical surface measurement locations, total (fixed plus removable) contamination surveys (alpha, beta and low-energy beta) and removable contamination surveys (gross alpha, gross beta, C-14 and H-3). The upper walls and the ceiling of this OU were classified as non-impacted.

Appendix 12.7 contains a graphical representation of the surface beta scans for the four walls and the floor and the individual data points are shown in the spreadsheets in Appendix 12.7. Table 10.7 contains a summary of the results along with the applicable DCGLs. Any analytical results from this OU are shown in Appendix 12.7, with the radiological Certificates of Analysis in Appendix 12.9.

In accordance with MARSSIM, no statistical evaluation is necessary when all survey values in the survey unit are less than the DCGL. Furthermore, the analytical results for all samples collected exhibited trivial concentrations of residual radioactivity with respect to the DCGLs. The following are additional findings:

- Surface beta scan results elsewhere within OU 4, and the stationary beta and low-energy beta measurement results, were less than the limiting DCGLs. All removable activity results were likewise unremarkable.[59]

[59] The beta measurements in OU 4 appeared to be elevated when compared to similar locations in other OUs. This, however, is an artifact of the instrumentation used to make the measurements. The radiation measurements on the floors, lower walls and upper walls were performed using three (3) different detector types, each of which had a different sensitive area and thus a different ambient background (i.e., the larger the area the higher the background count rate). Because the values shown in Table 10.7 and Appendix 12.8 reflect gross results only, the appearance of elevated activity in some locations is actually due to the elevated background count rate associated with the larger detectors.

- Total and removable alpha activities in OU 4 were below their applicable DCGLs. This finding demonstrates that the temporary presence of the drum with the "depleted uranium" label in this OU did not impact its radiological status. In addition, the maximum measured surface alpha and beta activity, when compared to their applicable DCGLs as follows, is less than unity, meaning the 25 millirem TEDE in a year dose objective is met:

$$\frac{A_\alpha}{DCGL_\alpha} + \frac{A_\beta}{DCGL_\beta} < 1$$

where A = the measured alpha or beta activity. The following maximum measured activities were used as input to the assessment:

Activity Type	A (dpm/100 cm^2)	Measurement Location No.
Total Alpha	20	Survey Unit 4.6, Location 11
Total Beta	941	Survey Unit 4.6, Location 3

The following is the result of the calculation, demonstrating that none of the analytical results from OU 4 exceed a sum of ratios of one (1):

$$\frac{20}{273} + \frac{941}{7100} = 0.21$$

- Elevated photon count rates were noted in three six-inch-diameter U-shaped conduit runs which, at one time, may have routed cables from the top of the reactor to cable runs under metal floor plates. An investigation into the source of the count rates revealed the presence of soil at the lowest point of the conduit. The natural radioactivity in the soil, which was confirmed by sampling, was thus determined to be the source. None of the conduit itself, the concrete in which it was embedded, or the gravel that surrounded it exhibited residual radioactivity that even approached the action levels.

8.5 OU 5 (Building 513)

Building 513 was subject to surface beta scans, measurements of ambient gamma radiation at a height of three feet from the floor and from all vertical surface measurement locations, total (fixed plus removable) contamination surveys (beta and low-energy beta) and removable contamination surveys (gross beta, C-14 and H-3). Although not required, the removable contamination smears were counted for both alpha and beta activity. The upper walls and the ceiling of this OU were classified as non-impacted.

Appendix 12.7 contains a graphical representation of the surface beta scans for the four walls and the floor and the individual data points are shown in the spreadsheets in Appendix 12.8. Table 10.7 contains a summary of the results along with the applicable DCGLs. Any analytical results from this OU are shown in Appendix 12.8, with the radiological Certificates of Analysis in Appendix 12.9. None of the results exceeded the applicable DCGL.

Surface scans, stationary measurement results and removable activity results in OU 5 were unremarkable, thus there is little potential for any combination of ROCs to exceed the applicable limit. In accordance with MARSSIM, no statistical evaluation is necessary when all survey values in the survey unit are less than the action level.

8.6 OU 6 (Former UST Area)

Subsurface soil samples from the former Tank Farm footprint were collected using Geoprobe® sample collection techniques. The sampling crew collected and logged continuous soil cores from the five pre-determined sampling points in this locale. The purpose of this action was to confirm that neither radiological nor chemical contaminants of significance were present in native (or backfill) soils remaining in the tank farm footprint. The length of the recovered cores were field screened using a Bicron Microrem meter to determine gamma dose rates, a pancake GM detector to determination of the general magnitude and uniformity of beta/gamma emissions and a photoionization detector (PID) for evidence of possible volatile chemical compounds. No indication of contamination was observed using these field screening methods.

In addition to the field screening methods referenced above, soil samples were also collected from the terminal depth of each boring and sent for fixed laboratory analysis. These samples were evaluated for both radiological isotopes as well as a broad range of chemical contaminants. The chemical analytes included volatile organic compounds (VOCs), semi-volatile organic compounds (SVOCs), target analyte list (TAL) metals, and polychlorinated biphenyls (PCBs). Only a small number of inorganics (metals) were detected above comparative EPA screening concentration limits; however, these elevated metals levels were not considered to be outside the expected range of native metals variability in soils. Table 10.8 contains a summary of the results along with the applicable DCGLs.

The level of residual radioactivity permissible in OU 6 that would ensure compliance with the 25 millirem per year radiation dose objective for release for unrestricted use is equal to the DCGLs shown in Table 10.2. All of the analytical results from the Geoprobe® samples collected from this area were below the applicable action levels. In accordance with MARSSIM, no statistical evaluation is necessary when all survey values in the survey unit are less than the applicable DCGLs.

8.7 OU 7

8.7.1 Generator Room

The Generator Room was subject to surface beta scans, measurements of ambient gamma radiation at a height of three feet from the floor and from all vertical surface measurement locations, total (fixed plus removable) contamination surveys (beta and low-energy beta) and removable contamination surveys (gross beta, C-14 and H-3). Although not required, the removable contamination smears were counted for both alpha and beta activity. The upper walls and the ceiling of this OU were classified as non-impacted.

Appendix 12.7 contains a graphical representation of the surface beta scans for the four walls and the floor and the individual data points are shown in the spreadsheets in Appendix 12.8. Table 10.7 contains a summary of the results along with the applicable DCGLs.

8.7.2 Outdoor Area Sampling

Surface soil samples were collected at a variety of locations within OU 7. Figure 11.7 shows the collection locations. Table 10.7 contains a summary of results.

As a special study, the IEM Team also advanced 10 individual borings around all sides of Building 516, using Geoprobe® direct push collection techniques, to depths equivalent to, or slightly below, the basement floor level of the building. The purpose of this effort was to determine whether any identifiable contaminants were migrating from the interior of the building. Subsurface (greater than 1-foot bgs) sampling within other areas of this OU was not included in this effort given the nature of historical site operations. Figure 11.8 shows the boring locations.

Due to the varied topographic conditions immediately surrounding the building as well as the structure's physical design, boring depths varied from sample point to sample point. Depths ranging from 20 to 24 feet bgs were reached in each location. Soil samples, like those recovered from OU 6 (Former Tank Farm Site), were collected for analysis of radiological isotopes as well as chemical contaminants. The chemical analyses included a broad "suite" of potential contaminants associated with former site operation that include VOCs, SVOCs, TAL metals, and PCBs. The length of each soil core recovered from the subsurface was field screened using a Bicron Microrem meter to determine gamma dose rates, a pancake GM detector for determination of the general magnitude and uniformity of beta/gamma emissions and a photoionization detector (PID) for evidence of possible volatile chemical compounds. No indication of any contamination was observed using these field screening methods.

In addition to the field screening methods referenced above, soil samples were also collected from the terminal depth of each boring and sent for fixed laboratory analysis. These samples were evaluated for both radiological isotopes as well as a broad range of chemical contaminants. As discussed in Section 8.6, only a small number of inorganics (metals) were detected above comparative EPA screening concentration limits. However, these elevated metals levels were not considered to be outside the expected range of native metals variability in soils.

Table 10.7 is a summary of the radioanalytical results along with the applicable DCGLs. All statistically-positive individual measurement results were less than the DCGLs. To adjust the individual DCGLs for the potential presence of the others, the following unity rule was applied to the five ROCs applicable to OU 7:

$$\frac{C_{H-3}}{DCGL_{H-3}} + \frac{C_{C-14}}{DCGL_{C-14}} + \frac{C_{Cs-137}}{DCGL_{Cs-137}} + \frac{C_{Eu-152}}{DCGL_{Eu-152}} + \frac{C_{Eu-154}}{DCGL_{Eu-154}} < 1$$

where C = the concentration of the ROC. The following maximum measured concentrations of each ROC from Table 10.7 were used as input to the assessment:

ROC	C (pCi/g)	Measurement Location No.
H-3	4.50	8
C-14	0.21	11
Cs-137	0.55	1
Eu-152	0.15	11
Eu-154	0.12	12

The following is the result of the calculation, demonstrating that none of the analytical results from OU 7 exceed a sum of ratios of one (1):

$$\frac{4.50}{110} + \frac{0.21}{12} + \frac{0.55}{5.7} + \frac{0.15}{8.7} + \frac{0.09}{8.6} = 0.18$$

8.7.3 Outdoor Area Surveys

A walkover survey of OU 7 was also performed as part of the investigation. For this survey, a Ludlum Model 2241 with a Ludlum Model 44-10 2" x 2" sodium iodide detector were used. The equipment was response-checked prior to use with a known radiation standard. The Model 2241 was attached via an RS-232 port to a backpack GPS survey system that automatically records the surveyor's position and the corresponding survey meter reading at that position every second as the surveyor walked over the measurement area. The GPS backpack unit used was a Trimble Model GeoXH with a Zephyr antenna, which is accurate to within six (6) inches after post-processing of collected data.

Walkover survey count rates in the location where a background soil sample was collected (outside of the perimeter fence) averaged 11,030 $\pm$ 430 counts per minute, with a lower confidence level (95%) of 10,714 counts per minute. Appendix 12.8 contains the listing of the data set, which is comprised of 6,294 validated measurement points that range from 4,523 to 15,130 counts per minute, with a mean of 9,536 $\pm$ 1,770 counts per minute.

The limiting DCGL for scans and stationary measurements is associated with the isotope Co-60, which is 7,100 dpm/100 cm^2 (beta/gamma). The limiting screening level for walkover surveys is approximately 3,100 counts per minute above background, which is nominally equivalent to the limiting screening concentration of 3.8 picocuries of Co-60 per gram. Figure 11.9 shows a color-coded survey map of the area, and Figure 11.10 shows the Z-scored values for the measurement points.[60]

8.7.4 OU 7 Findings

The following are the key findings of the OU 7 investigation:

- Surface scan and stationary measurement results in the Generator Room of OU 7 were less than the applicable DCGL. Removable activity results in this area were likewise unremarkable.

- The analytical results from the surface soil and Geoprobe® samples collected from OU 7 exhibit radionuclides typical of the natural background and in concentrations that are unremarkable. None of the ROCs were identified in concentrations that even approached the applicable DCGLs.

- The walkover survey results are generally uniform over the entirety of OU 7, with Z-scores that are predominantly less than three. The measured count rates at the majority of the locations with Z-scores greater than three are significantly lower than the population mean (i.e., the survey points were over the top of asphalt or other

[60] The method used for variability testing is the "z-score", which expresses the divergence of a measurement result from the most probable value as the number of standard deviations.

coverings that shielded the detector from the natural radioactivity in soil). Exceptions are the survey points on the south and east perimeter Building 513, where several concrete blocks with elevated (above background) gamma levels were buried near the surface. These blocks were similar in appearance to a collection of blocks that were found inside of the Connector Room of Building 516 (OU 3), which were sampled and found to contain K-40 and isotopes of the natural decay series in concentrations of just a few picocuries per gram. All of the blocks, including those at building 513, were later collected, packaged and disposed of as radioactive waste. The locations with the Z-scores greater than three (3) were then re-surveyed, with results that were consistent with the data set mean (i.e., Z-scores of 1 and 2). Figure 14.13 shows the pre- and post-removal survey results at these measurement locations. The only other exception is a measurement point at the south east corner of the DORF property where the sloped grade in this location exposed the detector to a larger surface soil area and thus elevated the count rate. The background soil sample was collected close to this measurement point and was shown to contain only isotopes in the natural decay series in concentrations of just a few picocuries per gram.++

- The historical aerial photography review could only generally confirm that the asphalt-covered surfaces, which comprise approximately 15.4% of OU 7, were present prior to the start of research reactor operations.[61,62] The fact that the walkover survey results, soil sampling (from cores collected through the asphalt) and surveys of the concrete areas at the exit points from the building (i.e., truck dock), as well as the historical data, were negative for the ROCs, is considered to be a reasonable basis for concluding the soil below the asphalt surfaces is likewise non-impacted.

- Figures 11.9 and 11.10 (lower left corner) show the photon count rate over a typical asphalt surface in OU 7 to be significantly lower than the count rate over the soil portion of OU 7 (i.e., Z-scores are greater than three over the asphalt). As a result, the MDA for the ROCs over the asphalt-covered surfaces is lower than the MDAs over the soil because the asphalt shields the detector from the natural radioactivity in the oil. Even though there are two surface media in OU 7, none of the measurement results, including soil samples collected from below the asphalt (soil boring No. OU7-SB04 and OU7-SB05) are indicative of the presence of the ROCs at even a fraction of the applicable DCGLs.

In light of these findings, and in accordance with the guidance in MARSSIM, no statistical evaluation of the data is necessary. All measurement results in OU 7 are less than the applicable DCGLs.

8.8 OU 8 (Outdoor Area Outside Perimeter Fence)

As shown in the project planning documents, surveys and sampling of OU 8 would only be required if radioactivity of significance was present within OU 6 or OU 7. Because the results of the

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[61] Phillips, P., BMT Entech, e-mail communication to C. D. Berger, with attachment, August 20, 2010, 11:28 a.m..

[62] Integrated Environmental Management, Inc., Report No. Report No. 2008012/G-103341, "Investigation Report for the Diamond Ordnance Radiation Facility (DORF)", Section 2.8, 2010.

investigation of OU 6 and OU 7 show there is no source of DORF-related radiological contaminants in or on the soil outside of Buildings 513 and 516, the potential for impact of the property outside of the perimeter fence is unlikely. In addition, historical documentation did not reveal evidence of atmospheric release of radionuclides from DORF operations. Furthermore, radiation surveys of filter banks that were in the DORF at the time of the investigation showed detectable activity on the "inside" filter surface, with only instrument background levels on the "outside" surface.

8.9 OU 10 (Groundwater)

Subsurface soil sampling using Geoprobe® direct push sample collection methods yielded no evidence of groundwater (either perched or aquifer waters) during investigative activities conducted within OUs 6 and 7. Although borings were limited to the first 30 feet below grade, groundwater resources are generally expected to be considerably deeper given the variable (incised) topography located north and west of the DORF. Additionally, as discussed in Sections 7.6 and 7.7, residual radioactivity in surface and subsurface soils within the 4.3-acre footprint of the DORF property is below action levels. Given these findings, the potential for groundwater impact from former operations at the DORF is unlikely.

8.10 OU 11 (Building 516, Mechanical and Ventilation Rooms)

The Mechanical Room was subject to surface beta scans, measurements of ambient gamma radiation at a height of three feet from the floor and from all vertical surface measurement locations, total (fixed plus removable) contamination surveys (beta and low-energy beta) and removable contamination surveys (gross beta, C-14 and H-3). The upper walls and the ceiling of this OU were classified as non-impacted.

The Ventilation Room was also subject to the same measurements with the exception of the surface beta scans. The boiler and ventilation equipment in this room rendered it too congested to perform the scans, although stationary measurements were performed. Appendix 12.7 contains a graphical representation of the surface beta scans for the four walls and the floor and the individual data points are shown in the spreadsheets in Appendix 12.8. Table 10.7 contains a summary of the results along with the applicable DCGLs. Any analytical results from this OU are shown in Appendix 12.8, with the radiological Certificates of Analysis in Appendix 12.9.

Surface scan and stationary measurement results in OU 11 were less than the DCGL and removable activity results were likewise unremarkable. In accordance with MARSSIM, no statistical evaluation is necessary when all survey values in the survey unit are less than the DCGL. Furthermore, the analytical results for all samples collected were exhibited trivial concentrations.

A coolant sump in OU 11 was subject to special investigation, including the collection of samples. Removable and total contamination surveys were also performed in the sump, the results of which were below action levels.

8.11 OU 12 (Building 516, Roof)

During the investigation, it was noted that the roof of Building 516 is a tar-type roof with a gravel covering on top. This construction did not permit the performance of the various survey types specified in the Project Planning Package. Therefore, the performance of beta scans and low-energy beta measurements were dropped from the survey plan. Instead, Class 3-type ambient gamma measurements and both beta and H-3/C-14 smears were performed at locations selected by professional judgement but with a bias towards emission points. Analytical results from this OU are shown in Appendix 12.8, with the radiological Certificates of Analysis in Appendix 12.9. No residual radioactivity above the applicable DCGLs was found, thus OU 12 is eligible for release.

9 FINDINGS AND CONCLUSIONS

A final status survey of the DORF was performed as part of the WRAMC effort to terminate Army Reactor Permit No. DORF-97-1. In addition, sufficient data were collected to support the release of the majority of the DORF from USNRC License No. 08-01738-02.

The USNRC has established criteria for ensuring that facilities and property that were used for licensed operations present negligible radiological risk to people and the environment once licensed operations cease. The radiation dose that the USNRC believes presents a negligible risk, as published in Title 10, Code of Federal Regulations, Part 20.1402, reads as follows:

> *"Decommissioning with license termination shall be limited to sites considered acceptable for unrestricted release where the residual radioactivity that is distinguishable from background radiation results in a total effective dose equivalent to an average member of the critical group that does not exceed twenty-five millirem per year (25 mrem/yr), including that from groundwater sources of drinking water, and the residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA)..."*

In other words, if the level of residual radioactivity in any of the operable units at the DORF is less than DCGLs that are equivalent to the USNRC's dose basis of 25 millirem per year, those operable units are eligible for release for unrestricted use. (Radionuclides known to be present at the site but eliminated from consideration after characterization must be accounted for in the dose basis.)

A characterization survey of the DORF was performed in 2009. The design of the survey was not just to understand the radiological constituents therein, but included assuring the data collected would be of sufficient quality and quantity to meet MARSSIM-based final status survey requirements. For each operable unit, the requirements for release were met if (1) an adequate number of measurements were made; (2) the mean of the sample distribution is less than the applicable DCGL; and (3) no further statistical tests were required.

The final status survey information reported herein demonstrate that there is no residual radioactivity in excess of the USNRC's screening criteria within OUs 2, 3, 4, 5, 6, 7, 8, 10, 11 and 12. In addition, because all gross activity results for these OUs, both individually and combined, were below the DCGLs, there is no need for statistical analysis. Although background data were acquired, corrections for background contribution to the measurement results were not applied when comparing the results to the DCGLs, thus adding an additional level of conservatism to the process. All exceedances of action levels were fully evaluated and no area classification changes were necessary.

Measurement results for removable activity within the OUs were all below the USNRC screening criteria for each surface measured. Therefore, none of the aforementioned OUs exceeded the removable activity DCGLs.

Because the DCGLs for the DORF were set to be equivalent to the USNRC's conservatively-derived screening values in NUREG-1757, an analysis to demonstrate the resulting dose potential is ALARA is not required. Therefore, OUs 2, 3, 4, 5, 6, 7, 8, 10, 11 and 12 are eligible for release for unrestricted use.

The residual radiological constituents in OU 1, due to activation of concrete from former reactor operations, exceeded certain of the USNRC's screening criteria, and thus likely contribute to the elevated ambient exposure rate in this OU. As a result, OU 1 is not eligible for expedited release and additional work and final status survey data are required before it may be released for unrestricted use. The preponderance of the residual radioactivity in this OU is located within the top five inches of concrete.

The former reactor pool area was designated a special survey area for the investigation and has since been named OU 9. Investigation and historical data for this area indicate residual radioactivity is below the applicable DCGLs, but a Class 1 designation is applicable because it was previously remediated. However, the quantity of data acquired to date, including the results of concrete coring performed during the investigation, are insufficient data to support a MARSSIM-based release of the survey unit. Therefore, OU 9 also requires further action before it may be released for unrestricted use.

The CSM was modified based upon the findings of the investigation. Included in the modification was the designation of the former reactor pool as new OU 9. It also shows the ROCs for OUs 1 and 9 to be Eu-152, Eu-154 and H-3 only from former reactor operations because none of the other radionuclides on the aforementioned listing were identified in statistically-significant quantities during the investigation. However, future activities that require the use of DCGLs may need to take into account the potential presence of the de-selected radionuclides by reducing the dose limit and corresponding DCGLs for Eu-152, Eu-154 and H-3 accordingly. NUREG-1757 (Vol. 2, Appendix O) permits de-selection of radionuclides from the listing of ROCs if their potential presence would contribute less than 10% of the total dose. In addition, de-selection is permitted for undetected radionuclides as long as the detection levels are sufficient to conclude that the dose contribution is less than 10 % of the dose criterion (i.e., with the assumption that the radionuclides are present at the MDC concentrations).

In summary, the data and information in this report serve as certification that OUs 2, 3, 4, 5, 6, 7, 8, 10, 11 and 12 meet the USNRC's conservatively-derived screening values for release and are eligible for release for unrestricted use. OUs 1 and 9 will require further action before they are eligible for release. Appendix 12.4 contains the revised CSM for the DORF.

10 TABLES

Table 10.1 - Operable Units and Area Classifications

Operable Unit No.	Description	ROCs	Class 1 Survey	Class 3 Survey	Class 3 Survey (alpha)	Focused Study
1	DORF Building 516 - Lower Floor (Exposure Room)	Mn-54, Co-57, Fe-55, Co-60, Cs-134, Eu-152, Eu-154, H-3, C-14	yes	no	no	no
--	DORF Building 516 - Lower Floor	Mn-54, Co-57, Fe-55, Co-60, Cs-134, Eu-152, Eu-154, H-3, C-14	no	no	no	yes (volumetric assessment of reactor pool area)
2	DORF Building 516 - Lower Floor (Warm Room)	Mn-54, Co-57, Fe-55, Co-60, Cs-134, Eu-152, Eu-154, H-3, C-14	yes	no	no	no
3	DORF Building 516 - Lower Floor (Connector Room)	Mn-54, Co-57, Fe-55, Co-60, Cs-134, Eu-152, Eu-154, H-3, C-14	yes	no	no	no
4	DORF Building 516 - Mezzanine Level (main floor, truck dock)	Mn-54, Co-57, Fe-55, Co-60, Cs-134, Eu-152, Eu-154, H-3, C-14, U	yes	no	yes	yes (alpha assessment for DU drum)
5	DORF Storage Building 513	Mn-54, Co-57, Fe-55, Co-60, Cs-134, Eu-152, Eu-154, H-3, C-14	no	yes	no	no
6	DORF Outdoor area (in location of former Underground Storage Tanks)	Mn-54, Co-57, Fe-55, Co-60, Cs-134, Eu-152, Eu-154, H-3, C-14	no	yes	no	no
7	DORF Outdoor area (within 4.2-acre fence line, including truck ramp, generator room and shield hoist penetration)	Mn-54, Co-57, Fe-55, Co-60, Cs-134, Eu-152, Eu-154, H-3, C-14	no	yes	no	no
8	DORF Outdoor area (outside fence line)	--	Non-impacted	no	no	no
10	Ground water	Mn-54, Co-57, Fe-55, Co-60, Cs-134, Eu-152, Eu-154, H-3, C-14	no	yes	no	no
11	DORF Building 516 - Main Floor - Source Room and ventilation system	Mn-54, Co-57, Fe-55, Co-60, Cs-134, Eu-152, Eu-154, H-3, C-14	yes	no	no	no
12	Roof	Mn-54, Co-57, Fe-55, Co-60, Cs-134, Eu-152, Eu-154, H-3, C-14	no	yes	no	no

Table 10.2 - Radionuclides of Concern (ROCs)

Radio-nuclide	Possible Contaminant Source	Radiation(s)	Expedited Release Criteria/DCGL				Limiting Activity	Note
			Surface, total (dpm/100 cm^2)	Surface, removable (dpm/100 cm^2)	Volumetric* (pCi/g)	Equipment (dpm/100 cm^2)***		
Mn-54	Reactor Operations	Photon	32,000	3200	15	5,000 ave; 15,000 max; 1,000 remov		
Co-57	Reactor Operations	Photon	211,800	21180	150	5,000 ave; 15,000 max; 1,000 remov		
Fe-55	Reactor Operations	Photon	4,500,000	450000	10,000	5,000 ave; 15,000 max; 1,000 remov		Surrogate
Co-60	Reactor Operations	Beta, Photon	7,100	710	3.8	5,000 ave; 15,000 max; 1,000 remov	Yes (Beta, Photon)	
Cs-134	Reactor Operations	Beta, Photon	12,700	1270	5.7	5,000 ave; 15,000 max; 1,000 remov		
Eu-152	Reactor Operations	Beta, Photon	12,600	1260	8.7	5,000 ave; 15,000 max; 1,000 remov		
Eu-154	Reactor Operations	Beta, Photon	11,400	1140	8.8	5,000 ave; 15,000 max; 1,000 remov		
H-3	Medical & Research License and Reactor Operations	Low-energy Beta	120,000,000	12000000	110	120,000		
C-14	Medical & Rese;h License	Low-energy Beta	3,700,000	370000	12	3,700	Yes (Low-energy Beta)	
U (depleted)	Medical & Research License	Alpha	273	27.3	0.5	5,000 ave; 15,000 max; 1,000 remov	Yes (Alpha)	
--		Ambient exposure rate	10 microR per hour**					
--		Walkover survey count rate	3,100 cpm ****					

*NUREG-1757, Vol. 1 states that the screening values are based on: (1) The contamination on building surfaces (e.g., walls, floors, ceiling) should be surficial and non-volumetric (e.g., < 10 mm (0.4 in)); and (2) The initial residual radioactivity (after decommissioning) is contained in the top layer of the surface soil (e.g., approximately 15 cm (6 in)), and that the use of the screening values in cases where volumetric contamination is thought to be present is not appropriate. However, the use of the DandD Code with input parameters for demolition, disposal and facility reuse scenarios produces higher DCGLs than for the surface soil scenario. Therefore, to ensure an element of conservatism in the assessment a surface soil exposure scenario is assumed.
**Assumes an industrial worker scenario and a dose limit of 25 millirem in one year.
***NUREG-1757-recommended criteria used to release materials and equipment from within the DORF.
****Assuming the limiting radionuclide is Co-60, the DCGL of 3.8 pCi/g is equivalent to a walkover survey count rate of approximately 3,100 counts per minute.

Table 10.3 - Summary of Data Requirements

OU	Area Classification	Fixed Beta	Fixed Low-energy Beta	Fixed Alpha	Fixed Gamma	Scan Beta	Removable Beta	Removable C-14 and H-3	Removable Alpha	Gamma Walkover	Soil	Subsurface Soil	Ground water	Cores
1	Class 1-Beta	N=36 AL=3500 dpm/100cm2, MDC<DCGL	N=36 AL=35000 dpm/100cm2		N=15 AL= 10µìR/hr	100% coverage, 5 cm/sec speed, Scan MDC<0.5 DCGL, AL=3500 dpm/100cm2	N=36 AL=71 dpm/100cm2	N=36 AL=36000 dpm/100cm2		--	--	--	--	N=12,walls N=3, floor N=3, ceiling
2, 3, 4, 11	Class 1- Beta	N=18 AL=3500 dpm/100cm2, MDC<DCGL	N=18 AL=35000 dpm/100cm2		N=18 AL= 10µìR/hr	100% coverage, 5 cm/sec speed, Scan MDC<0.5 DCGL, AL=3500 dpm/100cm2	N=18 AL=71 dpm/100cm2	N=18 AL= 1200000 dpm/100cm2 (OU-2 & 11), AL= 3700 dpm/100cm2 (OU-3 & 4)		--	--	--	--	--
	Class 3- Alpha (OU 4 only)			N=14 random AL=20 dpm/100cm2, MDC<DCGL					N=14 random AL=5 dpm/100cm2	--	--	--	--	--
5, 12	Class 3	N=14 random AL=3500 dpm/100cm2, MDC<DCGL	N=14 random AL=35000 dpm/100cm2		N=14 random AL= 10µìR/hr	30% coverage, 5 cm/sec speed, Scan MDC<0.5 DCGL, AL=3500 dpm/100cm2	N=14 AL=60 dpm/100cm2	N=14 AL= 1200000 dpm/100cm2 (OU-12), AL= 3700 dpm/100cm2 (OU-5)		--	--	--	--	--
6	Class 3	--	--	--	N=5 random AL= 10µìR/hr	--	--	--	--	--	N=14 random	Biased	--	--
7	Class 3	--	--	--	N=10% random AL= 10µìR/hr	--	--	--	--	40-100% coverage, 5 cm/sec speed AL= 20 pCi/g of gross gamma	N=14 random	Biased	--	--
10	--	--	--	--	--	--	--	--	--	--	--	--	Biased	--

Table 10.4 - Survey and Sample Summary
Expedited Release

Operable Unit No.	Size (m2)	Class	No. SUs	Scans, Walkover Surveys and Ambient Gamma				No. Fixed Contamination Surveys (dpm/100 cm2)			No. Removable Contamination Surveys (dpm/100 cm2)			Scan Speed Beta (cm/sec)	No. Samples	Analytical Methods (specify)
				Surface Beta or Scan Coverage	Action Level* (dpm/100 cm2)	No. of Ambient Gamma Measurements (3 ft from floor)	Action Level*	Beta	H-3/C-14	Action Level*	Beta	H-3/C-14	Action Level*			
2	52	1	1	100% floors and walls	3500	18	10 uR/hr	18	18	3500	18	18	71 (beta) 1,200,200 (LSC)	5	Biased samples***	Gamma Spectroscopy, Fe-55, Iso-U
3	90	1	1	100% floors and walls	3500	18	10 uR/hr	18	18	3500	18	18	71 (beta) 3,700 (LSC)	5	Biased samples***	Gamma Spectroscopy, Fe-55, Iso-U
4	680	1	9	100% floors and walls	3500	162	10 uR/hr	162	162	3500	162	162	71 (beta) 3,700 (LSC	5	Biased samples***	Gamma Spectroscopy, Fe-55, Iso-U
4	680	3	1	NA				18 random		20	18 (gross alpha)		5	n.a.	Biased samples***	Iso-U
5	143	3	1	100% floors and wall/grids	6300	14	10 uR/hr	14	14	6300	14	14	60 (beta) 3,700 (LSC)	5	Biased samples***	Gamma Spectroscopy, Fe-55, Iso-U
6	74	3	1	None	NA	5	10 uR/hr	0	0	NA	0	0	NA	5 (gamma)	1 surf. soil; 5 subsurf soil	Gamma spectroscopy, H-3, C-14
7	16,900**	3	2	40-100% ground surface (gamma)*****	3 pCi/g gross gamma	10%	10 uR/hr	14	0	NA	0	0	NA	5 (gamma)	Biased samples***	Gamma spectroscopy, H-3, C-14
8	--	Not Impacted	1	None	NA	0	NA	0	0	NA	0	0	NA	5 (gamma)	Biased samples***	Gamma spectroscopy, H-3, C-14
10	--	Not Impacted	1	None	NA	0	NA	0	0	NA	0	0	NA	NA	Biased water samples***	H-3, C-14
11	121	1	2	100% exposed floors and walls	3500	36	10 uR/hr	36	36	3500	36	36	71 (beta) 3,700 (LSC	5	Biased samples***	Gamma spectroscopy, Fe-55, Iso-U , H-3, C-14
12	280	3	1	100% roof surface	3500	14	10 uR/hr	14	14	3500	14	14	71 (beta) 1,200,000 (LSC)	5	Biased samples***	Gamma spectroscopy, H-3, C-14

* Above background. The survey unit becomes ineligible for expedited release if results exceed 100 times the Action Level. This "remediation level" was selected to ensure a survey unit isn't prematurely removed from the expedited release stream (i.e., it may be possible to release the area with respect to site-specific DCGLs that will not be determined/approved until after the on-site portion of the investigation is complete).
** Approximate dimensions.
***Samples will be collected if action level exceeded, with the number of samples based on surveyor judgement to ensure sufficient characterization information for the Operable Unit.
****Gross gamma activity based upon detection of the most limiting radionuclide (i.e., Co-60) with the detector at a height of 30 cm above the ground surface (see Table 7.9 of the SAP)
*****40 to 100% coverage of this area, with actual areas surveyed selected by surveyor judgement.

SURVEY AND SAMPLE SUMMARY
Characterization

Operable Unit No.	Fixed Contamination Surveys			Removable Contamination Surveys		No. Samples	Analytical Methods (specify)
	Coverage	Number	Action Level** (dpm/100 cm2)	Number	Action Level** (dpm/100 cm2)		
1*	Beta (100% of all grids, including ceiling)	36	3500	36	71	3 each; approx 15' long concrete cores into former pool area, 18 8" concrete cores into Exposure Room walls/floor/ceiling; biased samples	Gamma Spectroscopy, Fe-55, Iso-U, H-3, C-14
	100% beta scan (including ceiling)	5 cm/sec (scan speed)	3500	NA	NA		
	H-3/C-14	36	35,000	36	3700		H-3 and C-14
	Gamma (distribute locations evenly throughout the room volume)	15	10 uR/hr	NA	NA		
	Alpha*** (10% of floor grids only)	15	20	36	5		Gross alpha

* Operable Unit 1 is comprised of approximately 163 square meters and is designated a Class 1 MARSSIM area for beta-emitters and a Class 3 area for alpha emitters. A single survey unit may not exceed 100 square meters in area.
** Above background.
*** The number of fixed measurements is a minimum level of effort. Additional measurements may be made at the discretion of the survey technician in order to adequately characterize the Operable Unit. Samples will be collected from random locations in order to satisfy requirements for expedited release of a Class 3 area (alpha).

Operable Unit No.	Size (m2)	Surface Coverage*	No. Fixed Alpha Contamination Surveys**	Action Level** (dpm/100 cm2)	No. Removable Alpha Contamination Surveys*	Action Level** (dpm/100 cm2)	Analytical Methods*** (specify)
2****	52	10% floor grids	6	20	10	5	Gamma Spectroscopy, Fe-55, Iso-U
3****	90	10% floor grids	10	20	10	5	Gamma Spectroscopy, Fe-55, Iso-U
4	680	10% floor grids	45	20	20	5	Gamma Spectroscopy, Fe-55, Iso-U
11****	121	10% floor grids	20	20	10	5	Gamma Spectroscopy, Fe-55, Iso-U

* The number of fixed measurements is a minimum level of effort. Additional measurements may be made at the discretion of the survey technician in order to adequately characterize the Operable Unit. Samples will be collected from random locations in order to satisfy requirements for expedited release of a Class 3 area.
** Above background.
*** Samples will be collected only if action levels are exceeded, with the number of samples based on surveyor judgement to ensure adequate characterization of the Operable Unit.
****Alpha measurements not required but collected nonetheless.

Table 10.5 - Instrumentation Listing

Make	Rate Meter Model	Detector Model	Detector Type	Radiation Detected	Calibration Source	Use
Ludlum	2224	43-68	Gas flow Proportional	Alpha, 1-5 MeV Beta, 65-1,450 keV	230Th, 99Tc	Direct alpha and beta surveys; Alpha and beta scan on solid surfaces
Ludlum	3	44-9	Geiger Mueller	Beta, 65-1,450 keV	99Tc	Direct beta surveys; Beta scan on solid surfaces
Ludlum	2929	43-10	Dual scintillation (alpha/beta phoswich)	Alpha, 1-5 Mev Beta	230Th, 99Tc	Removable activity surveys
Ludlum	3	44-2	Gamma scintillation	Photon	137Cs	Gamma radiation surveys in drains and small penetrations
Ludlum	2221	43-94	Gas flow proportional	Alpha, Beta	99Tc	Direct beta surveys in drains and small penetrations
Ludlum	2241	44-10	Gamma Scintillation	Photon	137Cs	Gamma radiation walk-over surveys
Eberline/Thermo Scientific	E-600	380AB	Dual scintillation (alpha/beta)	Beta	99Tc	Beta scans on solid surfaces
Eberline/Thermo Scientific	E-600	BP17A	Beta Scintillation	Beta	99Tc	Large area beta scans on solid surfaces
Eberline/Ludlum	E-600	43-37	Gas flow proportional	Beta	99Tc	Large area beta scans on solid surfaces
Ludlum	12	44-110	Windowless gas flow proportional detector	Low-energy beta (i.e., H-3 and C-14)	Secondary calibration between H-3 and Tc-99	Stationary measurements and scans on solid surfaces.
Ludlum	19 MicroR meter		Sodium iodine	Photon	137Cs	General area dose rates
Bicron	Microrem		Plastic Scintillator	Photon	137Cs	General area dose rates
Packard	Tri-Carb		Liquid scintillation counter	5-1,500 keV beta	14C and 3H	Removable activity surveys

Table 10.6 - Summary of Measurement Results (OU 1)

Table 10.6 - SUMMARY OF MEASUREMENT RESULTS
Operable Unit 1 (Exposure Room)[63]

Survey Unit Identifier	Scans (dpm/100 cm2 β - Gross Activity)			Stationary (dpm/100 cm2 β - Gross Activity)			Stationary (dpm/100 cm2 α - Gross Activity)			Stationary (dpm/100 cm2 Low Energy β - Gross Activity)			Removable (dpm/100 cm2 β - Net Activity)			Removable (dpm/100 cm2 α - Net Activity)			Removable (dpm/100 cm2 H-3 - Net Activity)			Removable (dpm/100 cm2 C-14 - Net Activity)			Ambient (microR/hr γ - Gross Rate)			Stationary (cpm Shielded γ - Gross Rate)		
	DCGL	Average	Max	DCGL	Average	Max	DCGL	Average	Max	DCGL	Average	Max	DCGL	Average	Max	DCGL	Average	Max	DCGL	Average	Max	DCGL	Average	Max	DCGL	Average	Max	DCGL (nominal equivalent)	Average	Max
Ceiling	7100	3767	6836	7100	1308	1631	273	14	25	3700000	2070	2549	710	40	88	27	1	3	120,000,000	0	5	3700000	0	5	10	26	30	9000	13500	20000
Floor		1146	3338		1271	1715		7	15		1797	2353		66	152		2	6		3	6		1	6		25	30		--	--
East Wall		2794	4640		1408	1662		4	10		1922	2157		37	52		2	5		4	7		2	7		26	30		12800	16000
North Wall		3698	6794		1696	2092		8	20		2020	2549		57	108		2	3		1	4		0	7		30	30		11700	18000
South Wall		3616	8257		1357	1865		5	10		1980	2157		53	112		1	3		-1	4		0	2		22	28		11500	15000
West Wall		3300	8427		1485	1658		10	20		2108	2745		27	40		1	3		-5	-1		3	10		28	30		12300	16000

63 Data not censored for results that were less than the MDA for the measurement in question.

Table 10.7 - Summary of Measurement Results (OU 2, 3, 4, 5, 7, 8, 10, 11, 12)

Table 10.7 - SUMMARY OF MEASUREMENT RESULTS
Operable Units 2, 3, 4, 5, 8, 10, 11, and 12[64]

Operable Unit	Area (SU)	Stationary (dpm/100 cm2 β - Gross Activity)			Stationary (dpm/100 cm2 α - Gross Activity)			Stationary (dpm/100 cm2 Low Energy β - Gross Activity)			Removable (dpm/100 cm2 β - Net Activity)			Removable (dpm/100 cm2 α - Net Activity)			Removable (dpm/100 cm2 H-3 - Net Activity)			Removable (dpm/100 cm2 C-14 - Net Activity)			Ambient (microR/hr γ - Gross Rate)		
		DCGL	Average	Max	DCGL	Average	Max	DCGL	Average	Max	DCGL	Average	Max	DCGL	Average	Max	DCGL	Average	Max	DCGL	Average	Max	DCGL	Average	Max
2	--	7100	512	722	273	8	20	3700000	3350***	17647***	710	5.5	108	27	2	9	120000000	2	11	3700000	0	3	10	3	3
3	--		512	659		4	10		1531	1961		-9	31		-2	0		2	8		0	5		4	9
4	4.1.1		733	856		3	15		623	902		-2	62		1	3		0	9		-1	4		4	4
4	4.1.2		714	885		2	15		667	922		30	119		1	5		-2	8		0	6		4	4
4	4.1.3		731	859		1	5		908	2157		5	77		1	5		-1	12		-1	3		4	4
4	4.1.4		702	841		2	15		639	843		-5	58		0	5		-3	9		1	8		4	4
4	4.1.5		727	885		4	15		731	1863		-11	46		0	5		-1	8		1	6		4	4
4	4.2		747	1254		8	50		703	1176		49	92		1	3		2	10		1	9		3	4
4	4.4		581	823		3	15		663	882		-10	15		-2	0		2	27		0	6		2	2
4	4.5		620	823		4	15		777	980		51	128		1	6		2	14		3	10		3	3
4	4.6		785	941		7	20		668	824		-14	8		-2	0		1	9		1	8		2	2
4	Mezz. 1		392	548		4	15		721	843		-4	27		-2	0		3	12		1	6		2	2
4	101 Over-head**		1038	1131		5	19		596	804		0	108		1	5		--	--		--	--		--	--
5	--		782	963		16	25		--	--		47	92		2	6		4	16		0	8		6	6
7	Gen. Rm.		772	996		4	19		--	--		-21	12		1	8		1	9		1	7		4	4
11	Vent Rm.		702	793		9	20		--	--		21	140		-2	0		2	11		-2	7		2	2
11	Mech Rm.		628	681		6	15		656	824		52	88		1	3		3	13		1	9		4	4
12	--		546	648		8	15		--	--		26	88		-2	3		0	5		-3	5		6	6

**A different detector type, with a higher background, was used to measure the overhead areas in OU 4. Therefore, the gross beta activity results are somewhat higher than for the other surfaces.

***A possible transcription problem (i.e., decimal point in the wrong location) is alleged for two measurements in this OU. However, the original field notes are not sufficiently legible to support a change, thus the data as entered into the data base (i.e., 9000 and 7000 cpm, as opposed to 900 and 700 cpm) are included in this data summary.

64 Data not censored for results that were less than the MDA for the measurement in question.

Operable Unit	Area (SU)	Surface	Scans (dpm/100 cm2 β - Gross Activity)		
			DCGL	Average	Max
2	--	Floor	7100	467	676
		West		892	3047
		East		1315	4457
		North		1081	4127
		South		894	1681
3	--	Floor	7100	655	1683
		West		788	2330
		East		1780	4272
		North		808	1636
		South		852	1776
4	4.1.1, 4.1.2, 4.1.3, 4.1.4 and 4.1.5	Floor	7100	510	23507 (test decon area)
		West (2)		1277	2619
		East		163	2160
		North		331	1270
		South		1082	1973
		North (2)		1282	2226
		South (2)		1330	2098
		West		224	2410
4	Mezz. 1	Floor	7100	309	414
		West		--	--
		East		788	6056
		North		760	6238
		South		756	6227
4 (Mezzanine 3 and hall) - SU 4-4	4.4	Floor	7100	474	975
		West		1066	6023
		East		1222	3831
		North		979	5516
		South		1126	4304
4 (Mezzanine 5) - SU 4-6	4.6	Floor	7100	555	861
		West		1455	1758
		East		1672	3164
		North		1576	3071
		South		--	--

Operable Unit	Area (SU)	Surface	Scans (dpm/100 cm2 β - Gross Activity)		
			DCGL	Average	Max
4	4.5	Floor	7100	498	1142
		West		1122	1876
		East		1368	1918
		North		--	--
		South		1181	1762
4 (R106) - SU 4-2	4.2	Floor	7100	488	627
		West		1220	2136
		East		1322	2203
		North		1128	1871
		South		1291	2298
4 (R104) - SU 4-2	4.2	Floor	7100	441	932
		West		1334	2182
		East		1310	2353
		North		1206	2319
		South		1252	2183
4 (R105) - SU 4-2	4.2	Floor	7100	2036	2916
		West		1892	2894
		East		1874	2976
		North		1709	3185
		South		1876	3022
5	--	Floor	7100	635	1483
		West		1425	2331
		East		1543	2759
		North		1507	2971
		South		1571	2282
7	Gen. Rm.	Floor		437	905
		West		940	1987
		East		918	1633
		North		891	1759
		South		1172	2765
11 (R102, Mechanical Room)	Mech. Rm.	Floor	7100	488	1278
		West		1029	1939
		East		1083	2665
		North		1037	1766

Operable Unit	Area (SU)	Surface	Scans (dpm/100 cm2 β - Gross Activity)		
			DCGL	Average	Max
		South		974	1582
11	Vent. Rm.	Equipment in area did not permit the performance of scans			
12 (Roof)	--	Not scanned (see Section 8.9 of the report)			

Table 10.8 - Summary of Soil Sample Results (OU 6, 7)[65]

Operable Unit 6 - Former UST Area

Radionuclide	DCGL (pCi/g Gross Activity)	Average (pCi/g Gross Activity)	Max (pCi/g Gross Activity)
C-14	12	0.8	0.5
Eu-152	8.7	0.7	0.3
Eu-154	8.8	0.0	0.1
H-3	110	1.1	1.7

Operable Unit 7 - Outdoor Area[66]

Radionuclide	DCGL (pCi/g Gross Activity)	Average (pCi/g Gross Activity)	Max (pCi/g Gross Activity)
C-14	12	0.8	2.3
Cs-137	5.7	0.7	2.3
Eu-152	8.7	0.0	0.3
Eu-154	8.8	0	0.1
H-3	110	1.2	4.5

[65] Data not censored for results that were less than the MDA for the radionuclide in question. Only radionuclides positively identified by the analytical method are reported.

[66] The generator room in OU 7 was scanned and results presented in Table 10.7.

11 FIGURES

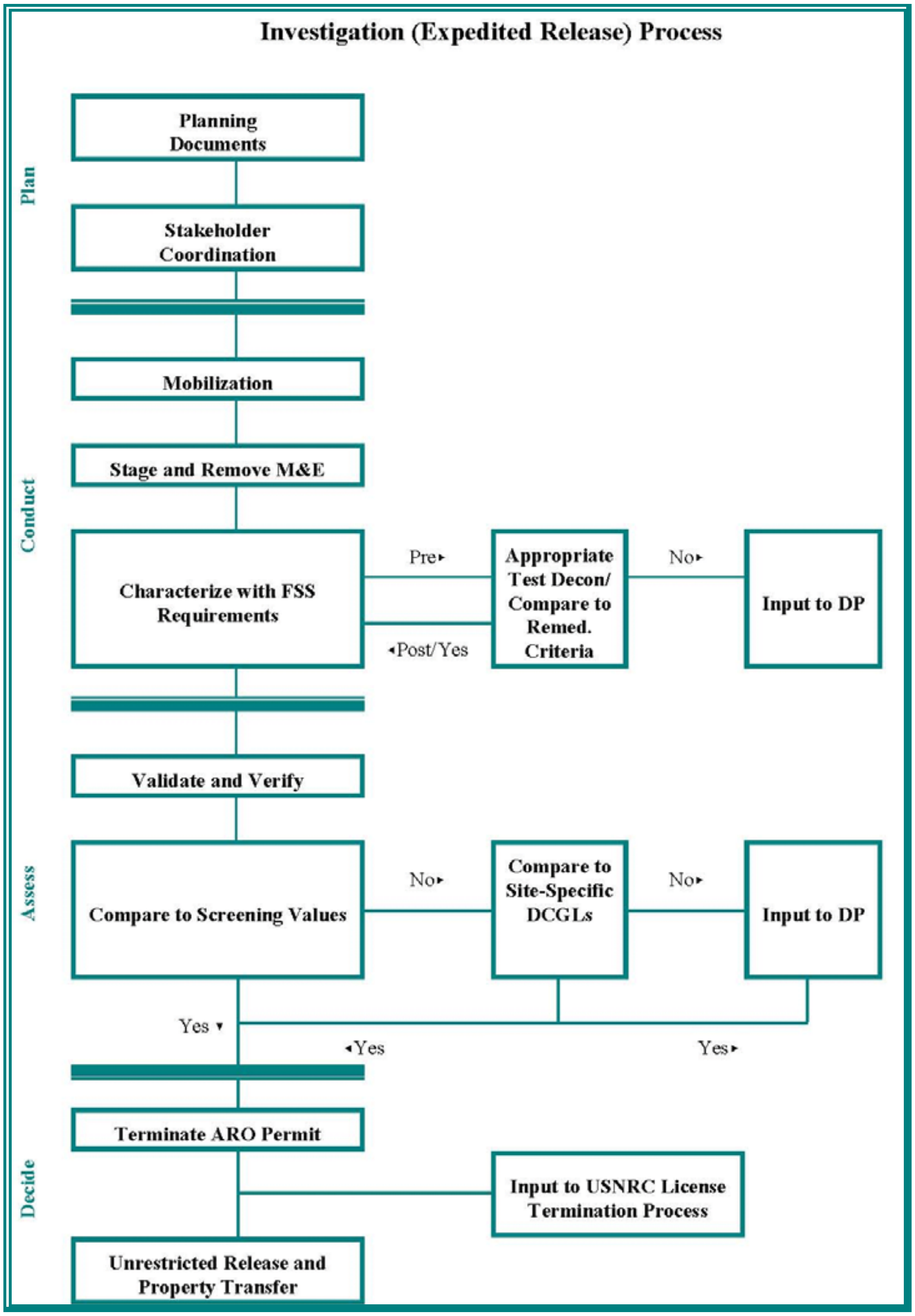
11.1 - Spatial Location of Buildings

11.2 - Rockwell Concrete Core Locations

11.3 - Rockwell Excavation Plan

11.4 - Rockwell Final Survey of East Wall and Pool Cavity

11.5 - Investigation Process



11.6 - OU 1 Concrete Core Locations

11.7 - OU 7 Soil Sample Collection Locations

11.8 - OU 7 Boring Locations

11.9 - OU 7 Walkover Survey Map

11.10 - OU 7 Z-Score Map

12 APPENDICES

12.1 - Comment Resolution Record

12.2 - DORF Licenses and Permits

12.3 - Conceptual Site Model and Operable Unit Maps

12.4 - Revised Conceptual Site Model

12.5 - Personnel Qualifications

R. Alan Duff - Project Manager and Radiation Safety Officer

Professional Qualifications

Mr. Duff has over 30 years of experience in nuclear and hazardous materials project management, design support, surveillance, operational health physics, training, and decommissioning activities. He has prepared numerous plans, procedures, and license documents for U. S. Department of Energy facilities, U. S. Department of Defense facilities, U. S. Nuclear Regulatory Commission licensees, and commercial client facilities that are regulated by agreement states. Mr. Duff is well versed in the area of civilian and government radioactive and mixed waste transport and disposal requirements. He is registered by the National Registry of Radiation Protection Technologists (NRRPT).

Education and Training

Advanced Radioactive Material Transportation and Disposal Classes, 1989, 1993, 2003, and 2007.

GPS/GIS Backpack Survey Training Classes, 2007.

IT Corporation Project Management Course (40 hours), 1992.

40-Hour OSHA HAZWOPER (29 CFR 1910.120) Training, 1987.

Eight-hour Supervisor Training, 1990

Eight-hour OSHA Annual Refresher (29 CFR 1910.120), 2008.

Operational Water Chemistry and Radiological Controls, U.S. Navy, 1982

Engineering Laboratory Technician School, U.S. Navy, 1980.

Nuclear Power Training Unit (prototype), U.S. Navy, 1980.

Naval Nuclear Power School, U.S. Navy, 1978.

Registrations/Certifications

Registered Radiation Protection Technologist (RRPT), National Registry of Radiation Protection Technologists

Radiation Safety Officer - MDE Radioactive Materials License No. MD-31-281-01.

Authorized User - MDE Radioactive Materials License No. MD-31-281-01.

Experience and Background

2002-Present *Vice President of Nuclear Services, Integrated Environmental Management, Inc., Knoxville, Tennessee* - As the director of IEM's Nuclear Services Division, which operates as a compliment to our consulting capability by providing support services and on-site project management for major client initiatives, Mr. Duff is responsible for turn-key decontamination and decommissioning of nuclear facilities - including the preparation of all planning documentation, characterization surveys and sampling, facility and equipment decontamination, final status survey performance, waste packaging/transport/disposal coordination, routine facility surveillance services, emergency response, leak testing of sealed sources, instrument rental, employee monitoring services for internal and/or external exposures, training, and a host of other applied health physics operations. Mr. Duff also serves as the Radiation Safety Officer (RSO) for IEM operations pursuant to Maryland Department of the Environment Radioactive Materials License No. MD-31-281-01.

1995-2002 *Program/Project Manager, Integrated Environmental Management, Inc., Knoxville, Tennessee* - Provided high-quality project management and remediation services to commercial and government clients. As a member of the client's response team, worked with clients to: Develop scopes-of-work and bid packages for specialty subcontractors handling highly focused assignments; identify those subcontractors who will provide the greatest value to the client; manage teams of specialty subcontractors to ensure that the client's goals and expectations (technical, regulatory, and financial) are met from the beginning until project completion; provide insights into future regulatory issues and their impact as input to the client's long-range business planning and cost forecasting process; provide site remediation/decommissioning services for radioactive and hazardous materials; advise and train clients on waste transportation and disposal issues; and develop project specific plans and procedures to conduct on site activities.

1994-1995 *Senior Environmental Specialist, AWK Consulting Engineers, Inc., Pittsburgh, Pennsylvania* While assigned to the Oak Ridge, Tennessee office, was responsible for performing technical and administrative duties required to satisfy customer needs on site characterization and pre-remedial design support projects and for all aspects of D&D projects. Responsible for preparing project plans, project work plans, task specific Health & Safety Plans, and budgets/schedules for these projects. Also responsible for identifying and implementing decommissioning and decontamination methods for these projects.

1987-1994 *Project Manager, Health Physics Supervisor, Nuclear/Mixed Waste Engineering Services, IT Corporation, Knoxville, Tennessee*. Provided project management and health physics support services for nuclear and mixed waste projects throughout the United States.

1978-1987 *Engineering Laboratory Technician (ELT), Leading Petty Officer, Radiological Controls Shift Supervisor*, United States Navy Supervised a division of 40 personnel, provided support for nuclear powered submarines, and performed over 250 error-free shipments of radioactive materials. Served as Leading ELT and Engine Room Supervisor on the USS Grayling, SSN 646.

Professional Society Memberships

Health Physics Society (Plenary Member)

American Nuclear Society

Conference of Radiation Control Program Directors (Advisor to the Radioactive Waste Management Committee E-5 and to the D&D Committee E-24)

Awards

Navy Achievement Medal for conducting the first Trident Class submarine ion exchange resin discharge and solidification.

IT Corporation Project Management Associate

Example Project Descriptions

Project Manager for health physics field activities during characterization, remediation and survey of several oil production sites with soil contaminated with Naturally-Occurring Radioactive Materials (NORM) for multiple clients in support of litigation defense.

Project Manager for the first contaminated soil remediation project conducted in Venezuela. Soil was contaminated with cesium-137 from a sealed source that was suspected to be improperly disposed of and had exposure rates up to 10 mR/hr on contact with the soil.

Project Manager for the radiological characterization (MARSSIM surveys) of a facility that manufactured thorium fluoride for use as an optical surfacing product. Conducted radiation and contamination surveys and obtained analytical samples of building materials. Returned to the facility to conduct surveys in support of property ownership transfer. Supervised radiological remediation of facility including floor and wall contamination, underground tank removal, drain line removal, roof decontamination, and equipment demolition including ventilation systems, fume hoods, and scrubber systems. Responsible for coordination for treatment and disposal of radioactive and mixed wastes generated during the project and conducted final status surveys at the facility upon completion of work.

Project Manger for Phase 1 Environmental Assessments conducted at six radioactive waste processing and disposal facilities and investigative characterization activities at two of those facilities including coring through concrete floors to obtain soil samples under buildings.

Radioactive waste broker and DOT shipper for multiple client sites for shipping and disposal of client's sealed sources and radioactive process wastes.

Project Manager and Health Physicist for the remediation and final status surveys/sampling of a former oilfield pipe scale facility. Supervised the demolition of the site building, excavation and disposal of twelve truckloads of NORM- contaminated soil, and excavation and release of over 175 truckloads of clean soil. Interfaced with the client and state regulators on the planning and final release of the facility. Work performed under the terms/conditions of License No. MD-31-281.01.

Project Manager and Health Physicist for the remediation and final status survey of a pharmaceutical company's radiological laboratories contaminated with Hydrogen-3 and Carbon-14. Supervised the on site demolition of the labs including fume hoods, lab furniture and ventilation systems. Supervised the disposal of radioactive and mixed wastes from the site and the performance of the final status survey of the facility.

Project Manager for the decommissioning of an oven contaminated with mercury and thorium (mixed waste). Arranged for subcontractors to conduct decontamination and disposal activities, prepared project plans, supervised all field activities, and conducted all radiological surveys during the decommissioning. Responsible for coordination for treatment and disposal of mixed and hazardous wastes generated during the project. Later conducted removal of a central vacuum system that was contaminated with mercury and thorium at the same facility.

Conducted audits of a client's radiation protection program including tour of the site, interviews with employees to verify radiological and respirator training, review of shipping, waste disposal, sealed source, training, and survey records. Also conducted leak tests of client's radioactive sealed sources.

Project Manager for escalated decommissioning a State-licensed site that manufactured, tested, and distributed gauging devices in anticipation of the sale of the company and the possibility of its moving its operations to another location. Responsible for preparation of work plans, negotiations with regulatory agencies, decontamination of indoor and outdoor areas, performance and documentation of a final status survey, shipment of waste, and project-specific health and safety.

Project Manager and health physicist for the remediation of a building foundation drainage system and the processing of over 100,000 gallons of water contaminated with cobalt-60 up to levels of one (1) microcurie per liter for a commercial client. Responsible for coordination of a water processing subcontractor, an excavation subcontractor, and off-site analytical laboratory activities. Also interfaced with on-site U. S. Nuclear Regulatory Commission, U. S. Environmental Protection Agency, and a variety of state and local agencies. Follow up work at the same facility included development of decommissioning funding plans and site decommissioning plans.

Technical writer for the development of a logic flow diagram for identifying radioactive and mixed wastes at the U. S. Department of Energy's Portsmouth (Ohio) Gaseous Diffusion Plant.

Technical writer for the Fernald Remedial Investigation/Feasibility Study (RI/FS). Provided technical guidance to engineering staff, generated reports on radioactive and mixed waste packaging, transport, and disposal.

Site Manager for the characterization survey of an EPA Superfund site three story warehouse that had been used in the past as a lantern mantle manufacturing facility and had been contaminated with thorium. Assisted in the development of project plans and final reports, supervised a crew of Health Physics technicians performing characterization surveys, interfaced with the facility owner and EPA personnel while on site.

Project Manager for the decommissioning and decontamination of three facilities at Sandia National Laboratory contaminated with radioactive and mixed waste. Responsible for the coordination of resources for the development of project plans, development of Project Work Plan, and maintaining project budget and schedule commitments.

Health Physics Supervisor for a transuranic (TRU) waste repackaging project. Supervised the characterization, repackaging and shipment of 130 containers of high-activity americium-241 and plutonium-238 hot cell waste. The waste was packaged to meet the WIPP waste acceptance criteria and was transported (highway route controlled quantity) to the Idaho National Engineering Laboratory (INEL) for storage.

Project Manager for the excavation and disposal of radium waste cells for the Corps of Engineers at Bergstrom Air Force Base in Austin, TX. Developed all project plans, supervised field efforts, and coordinated waste transport and disposal activities.

Project Manager for the decontamination and final release survey of a 70,000 ft^2 facility that manufactured cesium-137 level gauges. Decontamination efforts involved overhead areas, work area concrete floors, and removal of soil under the floor slab. Facility was released from their license following a verification survey by the state radiological licensing agency. Developed state approved decommissioning plan and final status survey report.

Project Manager for the packaging and disposal of 55,000 Curies of Cobalt-60 teletherapy sources. Sources were loaded into cask liners in the facility hot cell and loaded into Type B casks for shipment for disposal. Also supported the packaging and disposal of several low level waste drums and HEPA filters that required the use of shielded Type A and B shipping containers.

Project Manager for the decommissioning and decontamination of IT Corporation's Oak Ridge Mixed Waste Analytical Laboratory. Developed the decommissioning and decontamination plan that was approved by the State of Tennessee. Also supervised the field crew during final surveys of facility.

Project Manager for the decommissioning and decontamination of a magnesium-thorium waterfall grinding booth at Tinker Air Force Base in Oklahoma. Responsible for the development of project plans, schedule and budget management, and disposal of radioactive and mixed wastes.

Project Manager for the decommissioning of a commercial facility which had previously processed ores containing uranium and thorium. Generated the decommissioning plan submitted to and approved by the U. S. Nuclear Regulatory Commission, and was responsible for schedule, budget, and on site activities.

Project Manager for the removal of a 22 MeV particle accelerator from a major university medical center. Developed State-approved decommissioning and decontamination plans, arranged for waste disposal and transfer of the accelerator to a university in Beijing, China, and was responsible for budget, schedule and all on site activities.

Project Manager for the decommissioning and decontamination of two radioactive source manufacturing laboratories at Chevron Research and Technology. The laboratories housed a neutron generator and were contaminated with tritium, carbon-14, cesium-134, and cobalt-60. Negotiated plan approvals with the State agency, and was responsible for budget, schedule, and all on site activities including waste transport and disposal.

Project Manager for the routine quarterly surveillance and special radiological projects at a metallurgical facility licensed by the USNRC. Conducted radiation, contamination, and airborne radioactivity surveys as well as personnel bioassay and dosimetry program and environmental monitoring program each quarter. Provided health physics coverage for non-routine activities such as baghouse and stack testing, heats of specialty materials, final release surveys of an excavated road area, storage yard, and a warehouse formerly used for storage of radioactive materials, and recovery of radioactively contaminated equipment improperly released from site. Responsible for the generation of quarterly surveillance reports.

Project Manager for the development of a conceptual decommissioning plan for a maintenance facility located in South Carolina. The plan was generated to provide support for the facility’s decommissioning funding plan.

Health and Safety Manager/Project Manager at the U. S. Department of Energy’s Fernald site thorium silo and bins decommissioning and decontamination project. Developed the project-specific health and safety plan, and interfaced with the client on health physics and health/safety issues. This project received safety and quality awards from the client.

Health Physics Supervisor responsible for the sampling of underground storage tanks with radioactive and mixed wastes at Brookhaven National Laboratory.

Health and Safety Manager for the U. S. Department of Energy’s Fernald Plant K-65 Silo sampling project. Developed the health/safety and sampling plans. The silos contained up to 0.5 microcurie of Radium-226 per gram and were the largest single source of radon gas in the U.S.

D&D Technical Manager for the decommissioning of the U. S. Department of Energy’s LEHR facility at the University of California at Davis. Developed project decommissioning and decontamination plans and field procedures.

Health Physics Supervisor for the excavation of waste materials which included mixtures of uranium and explosives.

Project Manager for the MARSSIM type final status survey of a potentially contaminated 10 acre property on Staten Island, New York. Developed site characterization/survey plans, supervised the on site characterization survey and soil sampling at the site, and developed the project report for submittal to regulators.

Developed numerous business proposals for nuclear decommissioning and decontamination projects including job walk downs, cost estimation, scheduling, and technical content of proposals.

While in the US Navy, acted as radioactive materials shipper for the Trident Submarine Refit Facility. Performed over 250 error-free shipments of radioactive materials including Type B quantity radiography source shipments and radioactive waste shipments to the naval shipyard.

Billy R. Thomas - Project CHP

Professional Qualifications

Mr. Thomas has over 29 years of senior-level experience in radiological and industrial hygiene activities with emphasis on systems to minimize personnel exposures to radioactive and hazardous materials, compliance with federal and state regulations, site and facility audits. Mr. Thomas has developed and implemented comprehensive programs for radiation and chemical protection programs. Mr. Thomas is actively involved in all aspects of health and safety including regulatory compliance, site decommissioning, program evaluation, applied health physics, occupational safety, training and project management.

Education

M.S., Environmental Health, University of Oklahoma, 1981

B.S., Health Physics, Oklahoma State University, 1976

Certifications

Certified Health Physicist (Comprehensive Practice), American Board of Health Physics, 1988. Recertified: 2004.

Certified Industrial Hygienist (Comprehensive Practice), American Board of Industrial Hygiene, 1984. Recertified : 2007.

OSHA Hazardous Waste Operations and Emergency Response (HAZWOPER) Training. Initial training 1987 and updated each year.

Lead Abatement Training for Supervisors, University of Cincinnati. 1996.

Asbestos Abatement Supervisor Course, Asbestos Consulting and Training Systems, 1997.

Authorized User - Maryland Department of the Environment Radioactive Materials License No. MD-31-281-01.

Experience and Background

2002-Present *Vice President, Consulting Division, Integrated Environmental Management, Inc. Findlay, Ohio.* As the director of the company's consulting division, Mr. Thomas is responsible for selecting and coordinating the services of senior-level consultants in the areas of radiation safety and industrial hygiene. In addition, he maintains and ensures all members of the division maintain a track record of technical excellence, cost and schedule control, and innovation in solving environmental and health/safety problems for both government and commercial clients.

2008-Present *Adjunct Instructor, College of Science, University of Findlay, Findlay, Ohio.* Serves as instructor for Environmental, Safety and Occupational Health Management program in the College of Science. Presents classes for both the graduate and undergraduate in topics related to safety management and industrial hygiene.

1999-2002 *Senior Health Physicist, Integrated Environmental Management, Inc. Findlay, Ohio.* Provides high-quality radiation protection services to commercial and government clients. As a member of the client's response team, works with clients to promote an understanding of what is required to achieve and/or maintain compliance in the eyes of all pertinent regulatory agencies, individually or jointly; develop and overall strategy for achieving compliance and reduce liabilities in a technically-sound, legally defensible, and fiscally-conservative business manner; recommend specific solutions that are compatible with the client's operating philosophy; and provide insights into future regulatory issues and their impact as input to the client's long-range business planning and cost forecasting process.

Mr. Thomas served as the task manager to develop a baseline human heath risk assessment for a confidential client who previously processed enriched uranium and manufactured fuel pellets. The risk assessment was developed for potential exposures both hazardous chemicals and radioactive materials found in soil and groundwater. The assessment incorporated the requirements of the USEPA Risk Assessment Guidance for Superfund (RAGS) as well as requirements established by the State authorities.

Mr. Thomas developed a Emergency Response and Preparedness Manual for a Canadian client who manufactured uranium pellets for nuclear power reactors. The manual was prepared in accordance with the guidance provided by the Canadian Nuclear Safety Commission (CNSC) and the U.S. Nuclear Regulatory Commission (USNRC). The manual addressed the resources to mobilize to an emergency, involving both hazardous chemicals and radioactive uranium in several different chemical forms. The manual was implemented by the client and approved by the CNSC.

A commercial client, licensed by the Nuclear Regulatory Commission, required an evaluation of their internal dosimetry program. Mr. Thomas prepared a procedure to measure both internal and external exposure. The procedure satisfied the recommendations established by the NCRP and ANSI as well as requirements established by the USNRC.

Mr. Thomas worked as part of a project team to develop decommissioning plans for six (6) different facilities licensed to process radioactive materials. The decommissioning plans established the derived concentration guidelines levels for a variety of radioactive isotopes, including enriched uranium, thorium and byproduct radioactive materials. The potential exposures to future residents were limited to less than twenty-five millirem per year and evaluated over a period of 1,000 years. The plans were compliant with the requirements established by the USNRC and NUREG 1757. Each plan was approved by the USNRC and implemented by the client in order to decommission the facility and terminate the license.

A commercial client required a plan to survey, remediate and ultimately release the building surfaces for unrestricted use. Mr. Thomas established the release criteria using and developed a procedure to complete the radiation survey. The procedure was consistent with the requirements established by the USNRC and NUREG 1575, MARSSIM.

Mr. Thomas completed radiation surveys to evaluate potential exposures to electromagnetic frequency (EMF) radiation in commercial manufacturing facilities. The evaluation of personal exposures were compared to recommendations published by the ACGIH and OSHA. Recommendations were provided to the clients to limit personnel radiation exposures and verify that exposures were acceptable.

1993-1999 *Director of Health and Safety, The IT Group, Findlay, Ohio.* Originally joined OHM Remediation Services in 1993. The IT Group purchased OHM in 1998. Duties including conducting site and facility health and safety audits, determination of personal protective equipment and respiratory protection equipment, supervising the development and implementation of site specific health and safety plans, and providing industrial hygiene training and services. He had direct accountability for health and safety compliance, including regulatory compliance with federal, state and local agencies. He implemented a comprehensive health and safety program for demolition and remediation activities by the Midwest region, which accumulated 2.3 million man-hours from March, 1994 to July, 1997 without a single 1ost time injury.

Safety and Health Manager, Kansas City PRAC II, Kansas City District. Duties on this HTRW contract included the development of safety and health plans as well as procedures to be implemented at each of the KC PRAC projects. Developed SSHP for specific KC PRAC projects including, Ottawa, Illinois, Galena, Kansas, Mead Nebraska, and Fort Riley, Kansas. Mr. Thomas provided specific support on the KC PRAC projects including:

Project CIH, Project CHP, Ottawa Radiation Sites, Ottawa, Illinois September 1994 – August 1997. Developed the site specific health and safety plan and radiation protection plan to excavate soil contained radioactive radium generated by a luminous processing company. This project involved the excavation of radioactive contamination from nearby residences and selected sites in the city. Worked with State of Illinois and the EPA to implement an effective contamination control program, including air sampling and personnel monitoring for radium. Provided radiation worker training for the work crew and directed the on-site health physics and industrial hygiene program for the initial phases of the project. Conducted site inspections and project audits on a periodic basis.

Safety and Health Manager, USACE, Omaha District Rapid Response II. Duties on this HTRW contract included the development of program procedures and policies to work on multiple USACE projects. Developed SSHP for specific Rapid projects, including work at Joliet, Illinois, Ames, Iowa and Des Moines, Iowa. Mr. Thomas conducted site inspections and provided technical support for the implementation of the site safety and health program for RR/IR task orders. Mr. Thomas provided support on each Rapid project, including:

Project CIH, Project CHP; Ames Laboratory Chemical Disposal Site, Ames, Iowa. July 1994 – November 1994. Developed the site specific health and safety plan for the excavation and disposal of approximately 1,000 cubic yards of radioactive uranium wastes and contaminated soils. Developed the radiation protection program to be implemented by project employees to reduce exposures to ionizing radiation to as low as reasonable achievable. Contaminated materials were packaged and shipped for disposal in Clive, Utah.

Safety and Health Manager, USACE, TERC Number 1. Duties on this contract included the development of SSHP for work at Ellsworth AFB in Rapid City SD and KI Sawyer AFB in Michigan. Mr. Thomas provided support for some of the TERC projects including:

Project CIH, Ellsworth AFB, OU2 and OU7, Rapid City South Dakota. November 1996 – September 1997. Developed the site specific health and safety plan to excavate radioactive materials from disposal trenches at OU2 and OU 7. Developed radiation protection plan as well as the release criteria to be implemented to document that the site was free of contamination. Worked with the USAF Radiation Safety Committee to establish protocols to identify plutonium in soil and verify that debris was handled correctly.

Project CIH, Tarracorp Industries, Granite City, Illinois April, 1993 – May, 1997. USACE Omaha PRAC II. Developed the site specific safety and health plan for this project to excavate and treat lead-contaminated soil from smelter emissions. Treatment was completed by stabilizing the soil using a pugmill. This process delists the soils to a "special waste" classification, resulting in key cost savings in disposal. To date, over 300 residential sites have been remediated, and over 100,000 tons of soil have been processed. Excavation, transportation, and disposal of wastes containing battery chips have also taken place. Developed the elements of the air monitoring program. The air monitoring program was sufficient to evaluate the personnel exposures to airborne lead dust, as well as the fugitive emission from the exclusion zone. Performed periodic site visits to review results of the air sampling program and confirm that exposures were acceptable.

Health and Safety Manager, Department of Energy, Weldon Spring Site Remedial Action Program (WSSRAP), April 1993 – July, 1995. OHM was contracted to excavate contaminated construction debris from the WSSRAP quarry. Materials in the quarry were accumulated from a munitions manufacturing facility at Weldon Spring, as well as the demolition of buildings from the Mallinckrodt site used during the Manhattan project. Personnel exposures to uranium and thorium were documented, as well as nitroaromatics and asbestos. Mr. Thomas completed site inspections to evaluate the effectiveness of the health and safety plan and review the results of employee exposure monitoring.

Health and Safety Manager during the demolition of selected manufacturing buildings at the WSSRAP. The demolition projects involved the controlled demolition of nine buildings. Employees encountered radioactive uranium as well as asbestos containing materials and cadmium based paints. Mr. Thomas evaluated the construction safety program as well as industrial hygiene program during the demolition tasks.

Health and Safety Manager during the remediation of facilities at the Piketon Gaseous Diffusion Plant in Portsmouth, Ohio. OHM was contracted to remediate a chromic acid tank, including the removal of the lead liner in Building X700. OHM also demolished the incinerator in Building X705A. Mr. Thomas prepared the health and safety plan to document the methods necessary to reduce employee exposure to hazardous materials, both chemical and radiation exposures. OHM employees encountered hot environments in Building X700 where chromic acid and uranium were present.

Health and Safety Manager during the remediation of mixed waste that was buried in several burial pits at the Ames Laboratory in Ames, Iowa. Mr. Thomas participated in the planning and execution of the project, including presentations at the public hearings that were provided by the DOE to the public. The waste in the burial pits contained a variety of hazardous materials, including radioactive uranium, thorium, and asbestos as well as volatile organics including methyl ethyl ketone and trichloroethylene. Mr. Thomas prepared the health and safety plan for the project which described the industrial hygiene practice, the construction safety requirements, and the elements of the health physics program. Mr. Thomas evaluated the controls that were implemented and verified that employee exposures were reduced to as low as reasonably achievable.

1990-1993 *Health and Safety Manager, IT Corporation, St. Louis, Missouri.* Provided direction day-to-day for laboratory operations in the areas of health physics, industrial hygiene, hazardous waste management, and laboratory safety. Served as the Radiation Safety Officer for the USNRC Broad Scope license for the use of by-product and source material at the laboratory .

Collateral assignment as Department Manager of a radiochemistry laboratory to analyze samples from a variety of commercial and government facilities, including facilities operated by the DOE. Services were provided to a variety of DOE facilities including Fernald, Idaho National Energy Laboratory, Lawrence Livermore National Laboratory, Nevada Test Site, Oak Ridge National Laboratory, Paducah Gaseous Diffusion Plant, Rocky Flats, WSSRAP, and the Y-12 Production Facility. Supervised the analysis of various environmental media to be analyzed for specific radioactive isotopes including uranium, plutonium, thorium, and radium. Other analyses were performed for fission products and gross methods including alpha and beta analysis. Served as the RSO for the broad-scope license issued to the laboratory by the NRC.

Performed waste management assessment for four different DOE facilities. Principal investigator for hazardous and mixed waste policies, procedures and practices. Recommended program changes and upgrades. Worked at the following facilities, including: Portsmouth Gaseous Diffusion Plant, Piketon, Ohio; K-25 Gaseous Diffusion Plant, Oak Ridge, Tennessee; Paducah Gaseous Diffusion Plant, Paducah, Kentucky; and Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Served as project manager for the Industrial Hygiene department at Los Alamos National Laboratory (HSE-5). Responsibilities included reviewing and making recommendations for several of the programs being implemented by HSE-5 for the National Laboratory. These programs included asbestos controls, carcinogen control, sampling strategies and hazardous waste site characterization. Mr. Thomas also developed a sampling strategy to evaluate personnel exposures to hazardous materials. Mr. Thomas evaluated the asbestos management program at Los Alamos Laboratory. He reviewed the work performed by the IH department, including project oversight and air monitoring. He inspected work sites established by contractors including Pan American Services to assess compliance with LANL procedures and OSHA regulations.

Served as project manager to prepare mixed waste and radiative waste management plans and programs for waste generated during the remedial investigation at the Nevada Test Site. The programs required coordination between the Remedial Investigation contractor, the DOE Operations Area office and the facility receiving the waste for disposal.

1988-1990 *Director of Corporate Health and Safety, Burlington Environmental, Columbia, Illinois.* Responsible for designing and implementing health and safety programs to limit exposures to hazardous chemicals and radioactive material during sampling and remediation activities. Developed procedures and conducted training classes for field service personnel to correctly use personal protective equipment and perform air monitoring to evaluate personnel exposures.

Mr. Thomas also served on several audit teams to review the health physics programs at DOE site, including Rocky Flats, Los Alamos and the Nevada Test Site. The criteria for the audits were based on the DOE Technical Safety Appraisal objectives. Mr. Thomas worked with the program personnel to correct deficiencies and measure the effectiveness of the programs.

Member of Technical Advisory Group for Martin-Marietta Energy Systems. The Advisory Group provided oversight of the Federal Facility Agreement regarding the operation of the Low Level Radioactive Waste Tank Systems implemented for Oak Ridge National Laboratory. Made recommendations to implement standard industry practices for the purposes of reducing personnel exposures to hazardous and radioactive materials. Reviewed the elements of the industrial hygiene relating to the engineering controls and administrative controls implemented to reduce exposures to hazardous materials. Evaluated the effectiveness of the health physics programs for the purposes of reducing personnel exposures to radiation to as low as reasonably achievable.

Mr. Thomas reviewed the industrial hygiene and health physics programs being implemented at each facility. Used the Technical Safety Appraisal guidelines developed by DOE to critique the effectiveness of the programs begin implemented. Worked with each respective program managers, responsible for the H&S program, to develop an action plan to upgrade the program and track the progress of the changes.

Member of the Management Advisory Team for Martin Marietta Energy Systems Gaseous Diffusion Plants. The Advisory team reviewed the effectiveness of the Health and safety programs being implemented including the health physics and industrial hygiene programs. The Advisory Group was responsible for reviewing each of the health and safety programs and making recommendations for areas of improvement.

1983-1988 *Senior Health Physicist, IT Corporation, Oak Ridge, Tennessee.* Provided health physics and industrial hygiene consulting to government and commercial clients. Served as the project manager for several remedial decontamination projects involving hazardous and radioactive materials. His experience included:

Project CIH, Fernald Feed Materials Production Center, US Department of Energy Cincinnati, Ohio. May, 1987 – June, 1988. Performed health-and-safety review of engineering improvements at DOE uranium metals production facility. Improvements included new ventilation systems, radioactive materials handling systems, and decontamination of the facility. Recommended health physics and industrial hygiene controls to minimize worker's exposure, and updated air monitoring programs for both workplace exposures and effluent sampling.

Task Manager, Fernald Feed Materials Production Center, US Department of Energy Cincinnati, Ohio. August, 1985 – June, 1986. Mr. Thomas developed and implemented the collection and analysis of radiation measurement to assess the concentration of uranium in the soil surrounding the manufacturing facility. This work was performed as part of the site wide Remedial Investigation/ Feasibility study.

Health Physics Supervisor, Joliet, Illinois, Commonwealth Edison, September, 1984 – December, 1985. Provided support for the chemical cleaning of the primary cooling system at Dresden Nuclear Power Station, Unit 1. Mr. Thomas was responsible for assessment of engineering controls to reduce personnel exposures to radiation. The techniques were successful to remove more than 750 curies of cobalt-60 and other activation corrosion products. Personnel exposures were less than 7 man-Rems for the total project.

Health Physics Supervisor, Confidential Client, August 1983 - July, 1984. Provided support to decommission a facility that manufactured neutron sources (Am-Be) for nuclear power plants and radiography applications. The hot cells and glove boxes were segmented and packages in Type B shipping containers; the TRU waste shipped to Idaho Falls for storage and ultimate disposal by the USDOE. Drums of remote handled TRU were repackaged and characterized in order to satisfy the waste acceptance criteria for the USDOE. All work was performed in containments designed to minimize the spread of radioactive contamination, both airborne and surface contamination. Exposures to remediation workers was maintained below 1,000 millirem per person for the 15 month project; external exposures to gamma and neutron radiation were minimized. Internal exposures to TRU, including plutonium and americium were evaluated and verified to satisfy the requirements of the USNRC.

1976-1983 *Senior Research Industrial Hygienist, Dow Chemical, Midland, Michigan and Tulsa, Oklahoma.* Provided health and safety support for employees in manufacturing facilities, including plastic and other intermediate chemical production. Assigned as lead health physicist for decontamination projects at several nuclear power plants. From 1977 to 1980, Mr. Thomas served as the radiation safety officer for a NRC broad scope license to authorize the use of mixed fission products and special nuclear material used in manufacturing and research applications at Dow Chemical. The program included a TRIGA reactor, two small accelerators, sealed radioactive sources and tracers for a variety of research programs. Mr. Thomas directed all elements of the health physics program including training, standard operating procedures, exposure assessment and documentation. Mr. Thomas later (1981 - 1983) served as the radiation safety officer for the field services division where sealed sources and mixed fission products were used in treatment systems. This assignment had responsibilities in 22 states for approximately 3,000 employees. Mr. Thomas directed the use of radioactive materials licenses in 16 different states and a NRC license for the use of these radioactive materials.

Professional Society Membership

Health Physics Society (Plenary member)
American Academy of Health Physics
American Industrial Hygiene Association
American Academy of Industrial Hygiene

Bibliography

Mr. Thomas has authored/coauthored many papers and technical reports. In addition, he has developed/presented training courses in the field of health physics, industrial hygiene and safety.

Other Appointments/Awards

Ohio Radiation Advisory Council. Appointed by Governor Taft in 2002. Elected Chair of the Council each year from 2004 through 2008. Appointment expires in 2010.

Ohio Utility Radiological Safety Board, Citizen's Advisory Council. Elected Chair in 2001 and 2002.

Member of the Working Group for the ANSI/HPS N43.8 Standard, *Classification of Industrial Ionizing Radiation Gauging Devices*, 2006-2008.

Director of the State of Ohio Low Level Radioactive Waste Facility Development Authority Board. Appointment by the Speaker of the Ohio State Legislature in 1997.

Chairman's Award for Safety Excellence, OHM Remediation Services, 1996, 1997

Senior Technical Associate, International Technology Corporation, 1991.

Member of the People to People Ambassador Delegation visiting the People's Republic of China, 1987. Invited speaker to review health physics practices.

Michael W. Kimbro - Site Health Physicist

Professional Qualifications

Mr. Kimbro has over 21 years of experience in the radiation protection field, with emphasis on decontamination, decommissioning, site surveillance and applied health physics. His extensive field and management experience, design capabilities, training expertise, interpersonal skills, and technical abilities in the decontamination, decommissioning, and radiation protection fields are accompanied by excellent qualifications in project coordination, regulatory compliance, site characterization and radiological oversight and verification for U. S. Department of Energy, U. S. Army Corps of Engineers and U. S. Nuclear Regulatory Commission (or Agreement State) licensee sites.

Education

Santa Fe Community College, Gainesville, FL 1983, 1985-86

St. John's River Community College, Palatka, FL 1984-85

Miami-Dade Community College, Homestead, FL 1991

Florida Community College at Jacksonville, FL, Jacksonville, FL 1993

Columbus State Community College, Columbus, OH 1994-1996, 1999, 2001

Multiagency Radiation Survey and Site Investigation Manual (MARSSIM) Implementation and Design Course (40 hours), 2003.

Occupational Health and Safety Technologist Course (40 hours), 1996.

40-Hour OSHA HAZWOPER (29 CFR 1910.120) Training (2001) and eight-hour OSHA Annual Refresher (29 CFR 1910.120), current through 2009.

Asbestos Abatement Contractors/Supervisor Training (40 hours), 2002.

Hazardous and Radioactive Material/Waste Transportation Certification Training, 2008

Registrations/Certifications

Registered Radiation Protection Technologist (RRPT), National Registry of Radiation Protection Technologists.

Authorized User - Maryland Department of the Environment Radioactive Materials License No. MD-31-281-01.

Department of Energy Radiation Worker II (2001) with requalification through 2007.

ANSI-Qualified 3.1 Senior Health Physics Technician (continual since 1989)

Department of Energy Radiological Control Technician Qualification, 2003-2007.

Hazardous Material/Waste Transportation Training, current through 2008

Radioactive Material Transportation Training, current through 2008.

Experience and Background

2008-Present *Project Manager and Health Physics Technician, Integrated Environmental Management, Inc., Knoxville, Tennessee* - Duties include decontamination work plans and Final Status Survey Plan development and performance, with particular emphasis on MARSSIM style surveys, radioactive waste packaging and transportation, radiation safety program instruction and audits, surveillance activities, site characterization and risk assessment, report preparation, cost/schedule assessment, research/analysis, and general health physics duties. Mr. Kimbro serves as the Manager of IEM's instrumentation program. He is also qualified as a Health Physics Technician pursuant to Radiation Safety Procedure No. RSP-006, "Training and Qualification of Radiation Protection Personnel".

2006-2007 *Senior Health Physics Technician, Various DOE, FUSRAP, Commercial Power Facilities, and University Sites* - Performed HP support activities in varying capacities. Projects included final status surveys, decontamination and decommissioning, and power reactor refueling/maintenance.

2006 *Remediation Field Coordinator, Key West Naval Base Remediation Project, Key West, Florida* - Scheduled and produced daily activity/man-power reports for the heating, ventilation, air conditioning (HVAC), and plumbing remediation during the Hurricane Wilma remediation of over 500 properties at the Key West Naval Base.

2002-2005 *Corporate Health Physics Specialist, Safety and Ecology Corporation, Knoxville, Tennessee* - Provided corporate Health Physics oversight, radiological engineering, and project development on numerous remediation projects nationwide. Corporate lead for radiologically related training projects. Involved in all aspects of D&D projects from proposal stage, to planning, performance, and final reports. Served on various project management teams as radiological issues representative. Authored numerous plans, procedures, work instructions and technical basis documents for corporate interest and clients. Served as primary Emergency Responder concerning radiological issues.

2001-2002 *Senior Radiological Controls Technician, British Nuclear Fuels Ltd., Oak Ridge, Tennessee* - Provided operational Health Physics and Industrial Hygiene support during the Three Building D&D Project (K-29, K-31, and K-33) at the East Tennessee Technology Park.

1993-2001 *Radiation Safety Specialist/Technical Support, Battelle Memorial Institute, Columbus, Ohio* - Provided operational Health Physics, ALARA, and technical support services for active Research and Development (R&D) projects as well as support to D&D activities at the BMI King Avenue and West Jefferson sites.

1987-1993 *Health Physics Technician, Various Commercial Nuclear Power Facilities* - Provided operational Health Physics coverage in varying capacities at 10 nuclear power facilities during 19 refueling and/or maintenance outages throughout the United States.

Awards

Safety and Ecology Corporation, Professional Services Employee of the Year, 2004

Example Project Descriptions

Project Manager for the radiological characterization, decontamination, and Final Status Survey of a facility that manufactured thorium fluoride for use as an optical surfacing product. Conducted radiation and contamination surveys to determine the extent and the magnitude of the radiological contamination. Prepared the state approved decontamination work plan and contributed to the state approved Final Status Survey Plan. Served as field Health Physicist during the decontamination and Final Status Survey. Coordinated the disposal of all waste generated during decontamination. Prepared the Final Status Survey Report in support license termination activities.

Project Manager/ Certified Shipper for numerous disposal activities. Responsible for DOT/IATA compliance issues regarding the transportation of radiative waste and sources.

ALARA Specialist/Technical Support for the decommissioning project of the hot cell facility, the sub-critical assembly building, and the research reactor building at Battelle Memorial Institute's West Jefferson North Site. These buildings, in particular, the fourteen hot cells were contaminated with an estimated 4,000 curies of mixed fission and activation products, as well as fuel residues and transuranics. Contributed to work plans and processes involving the off-load of numerous hot cells. Duties included pre and post-job reviews of activities, internal and external dose assessments, and shielding calculations for dose reduction.. Development of lessons learned documentation, pre-job exposure estimates, and exposure trending/ALARA goal reports. Prepared RWPs, including ALARA considerations for work packages.

Radiation Safety Services Technician for active Research and Development at Battelle Memorial Institute's Columbus, OH campus. Provided radiologically related technical support in the development of research study protocols including briefing and training research staff in specific radiation protection controls for each study. Client confidentiality limits study descriptions. Laboratory isotopes used include, but not limited to C-14, H-3, I-131, I-125, P-32, Ni-63, Tc-99m and Re-188. Provided routine radiological surveillance and surveys, as well as providing coverage for active studies, including radiolabeling of solutions and pharmaceutical. Performed 100's of radioactive source leak tests on sealed sources and laboratory equipment.

Radiological Specialist/Sample Coordinator for the Excess Material Project at the East Tennessee Technology Park (formerly the K-25 site), Oak Ridge, TN. Served as the Health Physics liaison between employer and client, Bechtel Jacobs Company. Authored radiological project plans and compliance documents. Provided radiological/ALARA engineering, as well as oversight for wasted handling and loading operations. Served as the sample coordinator for the radiological characterization of material. This encompassed over 4,000 samples and/or radiological surveys and associated data reports. Additionally, served as project QA specialist responsible for project assessments and audits, as well as trending and implementation of corrective actions of deficiencies.

Health Physicist/Project Manager for the radiological characterization, decontamination, and Final Status Survey of a research facility contaminated with Germanium-68. Conducted contamination surveys to determine the extent and the magnitude of the radiological contamination.. Served as field Health Physicist during the decontamination and Final Status Survey. Coordinated the disposal of all waste generated during decontamination. Prepared the Final Status Survey Report in support license termination activities.

Field Health Physicist for the risk assessment survey of warehouse facility with elevated levels of Naturally Occurring Radioactive Materials (NORM). Performed radiation and contamination surveys of the warehouse facility including the collection of sample media for radioactive analysis.

Radiological Engineer/ALARA Specialist at for the New Hydrofracture Facility D&D at the Oak Ridge National Laboratory. The facility was contaminated with an abundance of isotopes including Cs-137, Sr-90, and transuranics. Contributed to the work plans and processes used for the safe dismantlement of the facility. Developed ALARA goals and exposure reduction methods. Provided radiological oversight as well as personnel and day to day activity management as a part of the project management team.

Project Manager for Radiation Worker training for corporate employer. Responsible for the development of lesson plans, test and answer development, grading, records management, ensuring the proper maintenance and integrity of examination test banks and quality assurance of all documentation. Instructor for over thirty classroom sessions.

Health Physics Specialist at the abandoned Gulf Nuclear radioactive source manufacturing facility in Webster, TX. Acted as liaison/corporate representative between employer and the client, Shaw/US Army Corps of Engineers during the health physics support transition phase from one subcontractor to another. Performed interviews with management and operations personnel to establish project status and to assist in the operations planning phase. Additionally, performed procedural audits and instrumentation/ source inventory and training,

Health Physics Lead for the MARSSIM type final status survey of a facility machining Magnesium/Thorium alloys at Sermatech Power Solutions, Inc. (a.k.a. Airfoils Technologies Florida, Inc) in Boynton Beach, FL. Served as primary interface with the client and state regulators on the performance of survey activities. Compiled all data and authored the Final Status Survey Report for license termination.

Health Physics Lead for the characterization and MARSSIM final status survey of laboratory facilities contaminated with Sr-90 and Am-241 at the Oak Ridge Institute for Science and Education. Provided health physics operational support for decontamination activities, as well as serving as primary client interface.

Senior Health Physics Technician at the DOE Hanford Site K-Reactor Basin Closure Project. Provide operational HP coverage for the removal of debris (including fuel handling equipment) and sludge from the reactor basins (fuel pools) as part of bulk containerization activities.

Health Physics Technician at the University of Washington (Seattle) Research Reactor. Performed MARSSIM type Final Status Survey of the reactor building and associated buildings in support of license termination.

Procedure writer for Knoxville, TN engineering firm, S&ME. Reviewed firm's laboratory and radioactive source user program and authored complete radiological procedures compliant with the Tennessee Bureau of Environmental Health Services, Division of Radiological Health.

Senior Health Physics technician at the US Army Corps of Engineers St. Louis Airport Project Site, the Hazelwood Interim Storage Site, and the Latty Avenue Vicinity Properties. Identified areas requiring remediation by use of gamma walk-over surveys using Trimble Global Positioning Systems and collection of environmental media.. Guided excavation activities based on these results.

Radiological Emergency Responder at the Norfolk Southern Rail Yard in Elkhart, IN. Responded to unknown condition identified radiation by detection system alarm. Identified the cause of the alarm, identified the contaminant and magnitude, and remediated the effected area.

Industrial Hygiene Technician at the Environmental Management Waste Management Facility in Oak Ridge, TN. Performed Beryllium sampling and packaged samples for lab analysis.

Radiological Emergency First Responder at multiple nuclear sites.

Senior Radiological Controls Technician at the East Tennessee Technology Park's Three Building Project. Provided operational HP and Industrial Hygiene support for the BNFL Super Compactor and other D&D operations. Served as HP representative during scheduling/planning of Super Compactor maintenance shut-down.

Senior Health Physics Technician for the characterization/scoping of the Ford Nuclear Reactor at the University of Michigan

ANSI Qualified 3.1 Senior Health Physics Technician at numerous commercial power facilities. Provided operational Health Physics coverage for most of any number of tasks common to commercial reactor refueling and maintenance. These tasks include, but not limited to refueling floor operations such reactor head removal and replacement, refueling/fuel movement, reactor head inspection, cavity decontamination. Additionally, provided coverage for steam generator inspections and tube plugging, valve and piping replacements, reactor coolant pump repair and/or replacement. Provided coverage for balance of plant operations including waste processing, transportation, auxiliary building activities, and turbine deck operations.

Developed numerous business proposals for nuclear decommissioning and decontamination projects including job walk downs, cost estimation, scheduling, and technical content of proposals.

Steven J. Baker - Environmental Manager

Professional Qualifications

Mr. Baker is a Senior Project Manager with over 20 years of professional experience in the areas of environmental site investigations, waste removal actions, and site remediation tasks. Skilled leader with a proven record for completing tasks on-time, on-budget, and exceeding client expectations for project quality and completeness. Particular expertise in orchestrating complicated logistical tasks, developing detailed project documentation, and interfacing with clients and Federal/State/Local regulatory authorities. Experienced in evaluating regulatory requirements, developing working relationships with clients and regulators and achieving/negotiating compromise between opposing points of view. Strong oral and written communications skills.

Education

Bachelor of Arts (BA), Geography, University of Southern California, Los Angeles, CA.

Special Training

40-hour Hazardous Waste Operations and Emergency Response (HAZWOPER) training (OSHA 29 CFR 1910.120), 1991.

Annual 8-hour HAZWOPER Refresher Training (OSHA 29 CFR 1910.120), 1992 - 2007.

Confined Space Training (OSHA 29 CFR 1910.146), 1997.

Environmental Regulatory Audit Training, 1995.

Resource Conservation and Recovery Act (RCRA)/Superfund Industry Assistance Hotline Training (six-week training course), 1990.

Experience and Background

1994-Present *Senior Program/Project Manager, BMT Designers & Planners, Arlington, VA.* - Manage all administrative and field aspects of large ($500,000 to $2 million) environmental projects at a key client site in Long Island, New York; Responsible for all administrative and technical issue resolution measures necessary to maintain project schedules and operations.; Select and train project staff for various in-house and field assignments; Develop Statements of Work (SOWs), review subcontractor technical and cost proposals, and administer contracts for various vendor or subcontractor services; Monitor project costs, develop billings, and write detailed monthly reports to the client to document project progress; Write detailed technical reports to document investigations and/or remedial actions for review and comment by the client and regulatory stakeholders; Organize and host meetings and conference calls to present technical findings, data, and recommendations for subsequent actions; Maintain detailed project files in accordance with corporate Standard Operating Procedures; Participate fully in all physical and administrative aspects of field assignments; Act as the field Health and Safety Officer (HSO) to ensure that appropriate OSHA and corporate safety requirements are followed and maintained; Operate, on an occasional basis, a variety of heavy equipment (e.g., loaders, backhoes); Maintain personal OSHA 1910.120 Hazardous Waste Site Worker Training certification (since 1991); and Other Corporate, non-Program/Project duties.

1990 to 1994 *Halliburton NUS Corporation/Brown and Root Environmental, Gaithersburg, MD.* - Served as a field technician and junior/mid-level environmental scientist on a variety of environmental investigations at Federal DOD and DOE sites around the nation; Maintained a "living library" of Federal environmental regulations for DOE's Environmental Headquarters office in Washington D.C.; Contributed to the development of large operating permits and/or plans for various Federal Facilities that required munitions destruction permits, oil spill prevention plans, Annual Reports to Congress or NEPA impact statements.

1990 *Geo/Resource Consultants, Inc., Washington, DC* - Served as an Information Specialist on a Federal "hotline" that provided directions, explanations, and interpretations for a board range of Federal RCRA and CERCLA environmental regulatory statutes, regulations, and policies.

1986 to 1990 *Greenhorne & O'Mara, Greenbelt, MD* - Served as an Imagery Analyst for classified national defense topics of interest to the Defense Mapping Agency.

1980 to 1986 *The Bionetics Corporation, Warrenton, VA* - Served as an Imagery Analyst/Photo Interpretation Specialist conducting historical site evaluations for the U.S. EPA's CERCLA "Superfund" Program.

Special Training

Confined Space Entry Training (OSHA 29 CFR 1910.146), 2008
Health and Safety Training — Supervisor (OSHA 29 CFR 1910.120), 2008
American Red Cross First Aid and CPR Training, 2005 and 2008
Delaware Valley Safety Council Safety Training, 2005-2006
Job Safety Analysis Training, 2005 and 2006
Fire Extinguisher Training, 2005
Interstate Technology Regulatory Council, Phyto-Technologies Training, 2004
University of Richmond, Geographic Information Systems, 2003
Health and Safety Training (OSHA 29 CFR 1910.120), 2003, w/ annual refreshers

Patrick T. Phillips - Site Safety and Health Officer

Professional Qualifications

Mr. Phillips has six years of experience in site assessment field programs as an Environmental Scientist and Field Operations Manager providing technical, analytical, and management services related to characterization of hazardous and non-hazardous waste sites. He has extensive experience coordinating and executing all facets of environmental site investigations at commercial, industrial, and Government facilities. He is able to utilize innovative site characterization methods, including passive diffusion bag samplers, temporary groundwater monitoring points, and small diameter continuous multi-channel tubing wells to increase the efficiency and quality of site investigations. Mr. Phillips has provided management and oversight of large scale site investigations at facilities including petroleum refineries, bulk petroleum storage facilities, landfills, and research facilities. He integrates GIS technologies into environmental site investigations for database development, reproducibility of sample locations, and precision. He also provides design and implements alternative remedial strategies including in situ remediation of contaminated soil and groundwater, the use of phyto-technologies for soil and groundwater treatment, and the development, assessment, and implementation of alternative landfill cover designs. For four years he has also serving as Site Health and Safety Officer, ensuring safe work practices and health and safety risk mitigation for environmental site investigations at commercial and industrial facilities.

Additional experience includes: Two years experience as Corporate Health and Safety Officer. Manage the Health and Safety program for employees conducting environmental characterization and remediation at hazardous waste sites in accordance with OSHA regulations. Conduct periodic Health and Safety Audits to ensure that health and safety policies are being implemented and that employees are engaging in safe and environmentally responsible work practices. Routinely review job safety hazards with employees and verify that all precautions have been made to mitigate identified hazards. Responsibilities also include corporate reporting requirements, development of corporate and site-specific Health and Safety plans, administration of medical monitoring program, and coordinating and/or administering health and safety training.

Education

B.S., Biology and Chemistry, University of Richmond, 2003

Coursework toward M.S., Environmental Sciences and Policy, Johns Hopkins University, 2008 - Present

Experience and Background

2007 - Present - BMT Designers and Planners

2004-2007 - B&B Diversified Enterprises

2003-2004 - Sovereign Consulting

Special Training

Confined Space Entry Training (OSHA 29 CFR 1910.146), 2008
Health and Safety Training - Supervisor (OSHA 29 CFR 1910.120), 2008
American Red Cross First Aid and CPR Training, 2005 and 2008
Delaware Valley Safety Council Safety Training, 2005-2006
Job Safety Analysis Training, 2005 and 2006
Fire Extinguisher Training, 2005
Interstate Technology Regulatory Council, Phyto-Technologies Training, 2004
University of Richmond, Geographic Information Systems, 2003
Health and Safety Training (OSHA 29 CFR 1910.120), 2003, with annual refreshers

Kenneth C. Duvall - Quality Assurance Manager

Professional Qualifications

Mr. Duvall has over 30 years of oversight, management and technical support experience as a physicist and health physicist. He has had progressively responsible positions at the National Institute of Standards and Technology, the U. S. Department of Energy, general environmental firms and as a health physics consultant. He has participated in environmental policy development and planning, conducted environmental regulatory analyses, participated in rulemaking activities and the development of agency-based orders, responded to congressional inquiries, and provided facility oversight and safety services. Of particular note is that he was one of the developers of MARSSIM, a multi-agency survey manual, and was a peer reviewer of MARSAME, a companion analytical manual. Mr. Duvall has also participated in the development of nuclear threat scenarios and related risk assessments for government agencies.

Education

Howard University, Washington. DC - Bachelor of Science, Physics

University of Maryland, College Park - Graduate School

Awards

Two-time recipient of the federal government's "Hammer Award" for creative, distinguished efforts that make government more effective.

Experience and Background

2007 to Present - Consulting Health Physicist, Integrated Environmental Management, Inc. (Gaithersburg, MD) - Duties include technical support on client activities, peer review of technical documents and deliverables, applied health physics, field support and project management, quality assurance oversight, final status survey plan development and data review/validation/verification.

2000 - Present - Environmental Scientist, NE Research (Washington, DC) - EPA Science Advisory Board to peer review MARSAME, a guidance document on procedures for determining the environmental risk and disposition of Materials and Equipment from Decommissioned Nuclear Facilities. Developed decision framework for identifying and selecting decommissioning strategies and guidance on data quality. Reviewed and provided comments to the Department of Transportation on its Environmental Assessments (EAs) under NEPA for NAFTA; provided guidance to the Department of Energy on developing Risk-Based End States (RBES) for the cleanup and closure of sites; provided Comments to the Conference of Radiation Control Program Directors (CRCPD) on the Draft Guidance Document for the License Termination of Facilities where Radioactive Material was Used, and developed guidance on environmental cleanups. He is also involved in the development of Programmatic EISs for DOE Nuclear Weapons and Nuclear Energy projects.

2000 - 2000 - Environmental Consultant, Energetics, Inc. (Columbia, MD) - Technical Support Contract for US Department of Energy, Office of Industrial Technologies). Studied the environmental impacts and risks of recycling cleaned scrap metal from DOE operations and prepared reports.

1990 - 2000 - Environmental Scientist, U. S. Department of Energy, Office of Environment, Safety and Health (Washington, DC) - Provided radiation-related technical support and consulting to a variety of departments, inspected facility air emission monitoring equipment and conducted audits of data analysis procedures as part of an oversight responsibility for a variety of DOE laboratories and sites.

1975 - 1990 - Physicist, National Institute of Standards and Technology (NIST), Center for Radiation Research (Gaithersburg, MD) - Provided basic physics research, conducted workplace safety inspections at offices, labs and facilities which included occupational safety, chemical, radiation, and electrical hazards and served as special nuclear materials custodian for the Center, performing inventories and inspections of relevant labs and facilities.

Representative Projects

Participated in environmental policy development and planning, conducted environmental regulatory analyses, participated in rulemaking activities and the development of agency-based orders, and responded to congressional inquiries.

Provided technical assistance to program and field elements, provided oversight for multi-media environmental programs, conducted environmental audits and investigations, developed technical guidance documents, conducted environmental transport modeling and risk assessment, and participated in Emergency Preparedness and Response exercises.

Participated in the development of federal and industry-based environmental standards, participated on inter-agency committees, interacted with state and local government organizations, participated in public and industry forums, and represented the agency in public outreach activities and in mediating stakeholder concerns.

Experienced with Federal Environmental Regulations such as CAA, CWA, SDWA, RCRA, CERCLA, TSCA and Community Right to Know, Cultural Resources, Environmental Justice, and the requirements for the Handling and Transportation of Hazardous Materials.

Experienced with Environmental Management Systems under ISO 14000 series, Quality Management Systems under ISO 9000 series, and environmental assessments (EAs) and impacts (EISs) under the National Environmental Policy Act (NEPA).

Participated in the preparation and presentation of proposals, managed evaluation of contract bids, grant requests, and the procurement of technical services. Managed and monitored technical projects with multiple tasks, and conducted cost/benefit analyses and strategic planning.

Developed and managed training programs that provided advanced skills to over 1000 Federal and Contract employees.

Conducted document review and prepared detailed, specific comments. Prepared reports and summaries for the presentation of scientific findings and participated in collaborative research, technology development, and technology transfer.

Publications and Presentations

More than 25 publications in the scientific literature on measurement, methods and analyses.

12.6 - Instrument Records

12.7 - Beta Scan Maps

12.8 - Measurement Results (Spreadsheets)

12.9 - Collection Logs and Radiological Certificates of Analysis

12.10 - Concrete Core Scan Results