DETAILED REMEDIAL ACTION PLAN – ITEM 9B

1. Objective Of Decommissioning Action

The objective of the decommissioning of IRP Site No. RW-41 (Test Area C-74L) is to remediate the depleted uranium potentially present (buildings) to the extent that any residual radioactivity does not exceed the DCGLw. NRC residential screening values will be utilized as the DCGLw for the interior building surveys and typical equipment release criteria will be applied to the target area materials and external building surfaces (NRC Reg. Guide 1.86). The Eglin AFB DU soil DCGL will be utilized to screen soil/sediment samples that will be collected during scoping surveys of a drain and outfall area.

2. Critical Population

The critical population for IRP Site No. RW-41 is the range worker. The site is currently being used to test conventional munitions. Test Area C-74L is closed to all activities except testing of conventional munitions. Hunting and other recreational activities are not allowed. Range workers are present on site only during testing of munitions.

3. Activities and Tasks

A. MOBILIZATION AND TRAINING (1 day)

Mobilization includes procurement of necessary facilities, equipment, and materials to perform the surveys. Mobilization activities also include the assignment of personnel to the job site; personnel radiation safety and site-specific construction safety training; and regulatory permitting and notifications, as required.

Site-specific radiological and general hazard training will be provided, by the USACE Site Safety and Health Officer (SSHO), for all team members prior to the commencement of the survey. This will be further described in the SSHP (Site Safety and Health Plan) a component of the work plan.

B. SITE PREPARATION (1 day)

Site preparation will consist of an initial exposure rate survey of the buildings and target area.

1) Initial Radiological Survey

Prior to any field activities within the survey area, an initial walkover radiation survey will be conducted to determine additional safety considerations, if any. Measurements of gross alpha and beta levels for non-impacted construction materials, such as high on interior walls, will be obtained to determine count times and as an indication of whether background reference areas will be required.

2) Land Surveying by a Licensed Surveyor

The survey work will not require land surveying. Survey/sample locations will be identified on a scale drawing and by room dimension coordinates to be specified in the work plan.

3) Environmental Control Systems and Monitoring Program

a) Erosion and Sedimentation Controls
The survey work will not require erosion or sedimentation controls.

b) Dust Suppression
The survey work will not generate dust.

c) Airborne Contaminant Monitoring
The survey work will not generate airborne contaminants.

d) Environmentally Sensitive Areas
No threatened or endangered plants or animals have been observed at this site.

e) Decontamination
Decontamination techniques will be determined by the SSHO, specified in the Site Safety and Health Plan (SSHP), and the Eglin RSO will approve decontamination procedures.

C. SITE REMEDIATION OPERATIONS

Remedial actions, such as decontamination, are not expected to be required in the buildings or target areas.

1) Radiological Surveys and Laboratory Analysis

Radiological surveys of building and target surfaces will be conducted using alpha, beta, and gamma scintillation detectors. Detailed procedures will be given in the work plan. Results will be presented in dpm/100 cm^2 total uranium and μR/hr. Wipe samples for removable contamination will be collected to determine whether the use of the DCGL presented in NUREG/CR-5512 is appropriate. Wipe sample results should indicate that the average removable activity is less than ten percent (10%) of the DCGL. Soil/sediment samples collected from the drain area will be sent to an off-site laboratory for analysis of total uranium. Results will be presented in pCi/g. The laboratory used for survey purposes will be validated by the USACE – Omaha District.

2) Personnel Surveying

Prior to leaving the survey area, all personnel will be surveyed for contamination using hand held radiological meters. Surveys will be conducted in areas specified in the SSHP as will detailed frisking procedures.

3) Decontamination

Any contaminated waste generated by activities will be contained in bags and stored for transportation to the LLRM waste disposal site.

D. FINAL STATUS SURVEY (FSS) SAMPLING AND ANALYSIS (2 days)

A final status survey will be conducted for the impacted ballistics building interior surfaces, exterior surfaces of the ballistics building and the well house building. The final status surveys will be conducted using the guidance presented in the Multi-Agency Radiation Survey and Site Investigation Manual, NUREG-1575, Rev. 1, *Multi-Agency Radiation Survey and Site Investigation Manual* (NRC 2000a). The final status survey plan, a component of the work plan, will be provided to the USAF once completed.

E. CHARACTERIZATION SURVEYS

A characterization survey of the target area will be conducted using alpha, beta, and gamma scintillation detectors. Results will be presented in dpm/100 cm^2 total uranium and μR/hr. The objective of the characterization survey is to determine the extent of contamination on the catch box structure and augment scoping survey results that indicate only low levels of contamination. Should higher levels of contamination be identified during the characterization survey, the data will be used to select appropriate decontamination methods and plan remedial action. If contamination is found at a small fraction of the DCGL then decontamination may not be required and a final status survey could be performed. It is expected that the characterization survey will meet at least the needs of a Class 3 final status survey.

F. SUPPORTING OPERATIONS (Performed in Conjunction with Survey Activities)

1) Safety and Health, and Radiation Protection

The survey SSHP will be implemented to ensure both worker and public protection throughout the survey. These plans establish requirements in regard to medical surveillance, bioassays, PPE, air monitoring, stop-work authority, restricted work areas, hazardous and radiation work permits, training requirements, emergency response and notifications, and waste minimization and pollution prevention. The provisions of this plan are mandatory for all survey personnel.

2) Quality Control

The survey Quality Assurance Project Plan, a component of the work plan, will be implemented and monitored to ensure that all sampling, surveying, and construction quality objectives are met. Upon conclusion of work, a review will be completed to verify that all documentation is in order prior to close out and transfer of files to the USAF.

3) Decontamination and Release Operations

All equipment leaving a radiologically controlled area will be decontaminated and surveyed to demonstrate compliance with USACE EM-385-1-80, Table 6-4, the equivalent of NRC Regulatory Guide 1.86, Surface Contamination Guidelines. These procedures will be detailed in the work plan.

G. PERSONNEL, EQUIPMENT, AND FACILITIES DEMOBILIZATION (1 days)

At the conclusion of survey activities, the project team will demobilize from the site. All equipment will have been decontaminated and equipment tested and cleared through the Site RSO. Decontamination and testing details will be provided in the SSHP.

FINAL STATUS SURVEY PLAN – ITEM 10A

1. INTRODUCTION

A final status survey will be performed upon completion of the remediation activities of the RCA at IRP Site No. RW-41 Test Area C-74L Ballistics Measurement Facility in order to ensure compliance with the final cleanup criteria (Derived Concentration guideline Levels [DCGLs]) for the industrial scenario has been met. The DCGLs are determined from activity/dose relationships through various exposure pathway scenarios (RESRAD). The final status survey will be developed using the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (NUREG-1575).

DCGLs were determined for the industrial scenadrio, which is considered to be the most likely future land use designation. In addition, the construction scenario was evaluated to determine the DCGLs to be attained to allow future construction activities at the site. This allows Eglin flexibility in their future land use decisions. For comparative purposes, the residential scenario was evaluated to determine the most conservative DCGL value. Because of the potential presence of unexploded ordnance (UXO), it is unlikely that IRP Site No. RW-41 Test Area C-74L will be cleared for residential land use. Development of the DCGLs for IRP Site No. RW-41 allows the subsequent evaluation of appropriate future investigation/corrective measures at the site.

Remediation Description

The remediation of IRP Site No. RW-41 is based on the Characterization Study (CS)/Interim Corrective Action Report dated March 2000. During the study the Radioactive Material Controlled Area (RCA), Barrel Storage Area, Gun Corridor and the remaining land area of the site were characterized. The characterization study did not include the fire control building or the well house building. These two buildings were added based on the findings of the USAF Radioisotope Committee in May 2001, that the site was never removed from the NRC License SUBB 992, when DU testing was transferred from C-74L to Test Area C-64. This new information requires the site to be formally closed by development of a site decommissioning plan and issuance of a USAF Radioactive Material Permit for decommissioning of the site. Characterization and potential remediation of the control and well house buildings are included in Section 10B. Characterization and any remediation activities will be the responsibility of the US Army Corp of Engineers (USACE).

The recommendations of the CS report supported the remediation of the RCA. This included removal of the first six inches of soil below land surface. The other land areas including the barrel storage area and the gun corridor were remediated at the time of the CS FIDLER survey by removal of the areas of elevated activity individually. Documentation of the removal action and results is contained in the CS report. During this remediation all land areas will be 100 percent scanned with a FIDLER instrument in two directions. Performing the FIDLER survey in two directions will ensure a high confidence (95 percent) that all the DU fragments have been removed.

The remediation of the RCA will include collection of 30 surface soil samples a 100 percent FIDLER survey of the RCA after all DU penetrators or DU fragments aboveone half the $DCGL_w$ have been removed. The RCA is considered one survey unit for this remediation, however, the RCA has been broken down into five separate areas. The five areas will allow the final status soil samples to be collected and the 100 percent survey of each area as the next area is being remediated. This will reduce the total time required for the remediation and allow for each section to be cleared and clean soil to be placed over the remediation area. After the entire RCA has been remediated and removed soil replaced with clean soil an additional 100 percent FIDLER survey will be conducted in two directions to ensure the area has been properly remediated and meets the industrial criteria.

The remainder of the site's land area will be FIDLER surveyed. This includes the drum storage area, the gun corridor and the remainder of the land which was surveyed during the CS. Soil samples will not be taken in these areas. The CS report shows that the soil is not contaminated with DU but consists of areas of elevated activity due to DU fragments from the ground firing of DU munitions into various types of targets. This firing resulting in richochets of DU munitions into the ground surface. Bore hole sampling in the RCA and adjacent areas shows that the majority of the DU munitions impacted the soil and remained in the top six inches of soil. Removal of areas of elevated activity will take place during the final status FIDLER survey. Pre and post DU removal FIDLER scaler measurements will be taken after the DU fragment has been removed to show no elevated activity remains. This ensures that all areas of elevated activity have been removed.

Soil Sampling Strategy

The soil sampling conducted during this remediation is limited to only to the RCA. The number and location of samples is based on a modified MARSSIM protocol. At RW-41 Test Area C-74L the soil is not contaminated with DU, but contains fragments of DU penetrators ranging in size from several grams to whole penetrators of nearly 300 grams. Soil sampling in this case will not provide the reliability that the site has been remediated and meets the Industrial use criteria. Therefore the number of soil samples and location of soil samples may not meet the MARSSIM soil sample protocols. The soil sample locations will be biased in this case, soil samples will be taken in areas where elevated levels of activity have been found. A total of 30 soil samples will be taken in the RCA. Six soil samples in each of the five areas the RCA has been divided into. A 100 percent FIDLER survey will be conducted in two directions to ensure all areas of elevated activity have been located and removed. The site data will be evaluated by the LLRM Partnering

Team and the area determined to be remediated below the Industrial Use criteria prior to the addition of clean land fill.

Industrial Use Criteria

Remediation of buildings and land areas is normally completed to meet the requirements of a residential use. This is normally the criteria required by NRC and EPA. At this test area the LLRM Parterning Team recommended that the site be remediated to meet the industrial scenario. The DCGLs for the industrial scenario for soil samples is 600 pCi/gram and the DCGL for elevated areas of activity is 44 kcpm (utilized during scanning FIDLER surveys). The reasoning for utilizing the industrial scenario is the land is not being closed and turned over for public use. After radiological closure of the facility, the facility will still remain an active test area, with the testing of conventional munitions continuing. The current range use controls at the test area are based on explosive ordnance on the site. This is of more concern than any remaining radiological hazard. Existing land controls on the site require that additional study's be conducted if the land use of the test range changes. The radiological condition of the site, land and buildings will have to be accessed again if the land use changes. Currently there are no land use changes for seen which will result in closure of the test range. The present explosive ordnance hazard on the test area will prevent the land from changing to unrestricted land use. The explosive ordnance hazard will also prevent disturbance of the site soils of future building on the land.

Fidler Sensitivity and Surveyor Minimum Detectable Count Rate

A study was conducted to describe and quantify the sensitivity of the FIDLER (rate meter and probe) in detecting DU fragments in white sand and red clay found on Eglin AFB land ranges. The results of this study were also used to recommend a Derived Concentration Guideline Limit Elevated Measurement Comparison ($DCGL_{EMC}$) based on FIDLER measurements. The study condluded that the FIDLER is capable of detecting DU fragments at ten percent to fifty percent of the DCGL as recommended in the MARSSIM Manual. The FIDLER is capable of detecting average size fragments (greater than 50 grams) in the upper 12 inches of soil within 10 to 50% of the DCGL. The DU test also confirmed the Surveyor Minimum Detectable Count Rate ($MDCR_{surveyor}$) scan calculation value of 1078 cpm for use during field surveys. This value is the recommended scan MDCR that should be used during FIDLER surveys on Eglin Test Ranges.

After completion of the DU test, and completion of the MicroShield modeling program, the FIDLER was determined to be capable of detecting average size fragments (less than 50 grams) in 12 inches of soil within 10 to 50% of the DCGL. Therefore, it was recommended that a F"IDLER based $DCGL_{EMC}$ value of 44 kcpm be used for a 1 – meter area. This value is used as a direct comparison value for hot spot risk determination.

The MicroShield Modeling program was used to determine an average Scanning Minimum Detectable Concentration (scan MDC) for the FIDLER. A total of 72 models were calculated with DU fragments buried at various depths, geometry, and soil densities. The scan minimum detectable concentration (MDC) was determined to be equal to 14.17 pCi/g, which was the MDC for a 50 gram DU gragment buried in 12 inches of red clay.

This information was presented and discussed during the February 2000 LLRM Partnering Team Meeting. The LLRM Partnering Team reached consensus on a scan MDC of 14.17 pCi/g and a $DCGL_{EMC}$ of 44 kcpm. This $DCGL_{EMC}$ is considered the investigation level for all field work at Eglin LLRM DU sites and will be used as the action level during remediation at IRP Site No. RW-41.

Area Classifications and Survey Units

Area classification were established from data collected from the Site Characterization Study dated March 2000. Reference background soil samples were not taken because uranium is found in the soil at levels significantly lower than the $DCGL_W$. Background for use of gamma survey instruments, such as the FIDLER, were taken on areas of similar soil type, where no known DU contamination exits. Background for surface scans of structures will be taken on structures of similar construction where no known contamination exists.

Survey units are physical areas consisting of structures or land, which are divided into specified sizes where a separate decision will be made as to whether or not that area exceeds the release criterion. Survey units will be established that include only one classification of area, i.e. a survey unit cannot have Class 1 and Class 2 areas. Survey unit size will take into consideration the potential for small areas of elevated activities. Smaller survey units may be considered appropriate as determined by the site Certified Health Physicist in conjunction with the Project Manager, and Base Radiation Safety Officer. Currently survey unit size is not considered important. The scanning surveys and FIDLER scaler measurements give the confidence necessary to show the survey units do not contain areas of elevated activity above the $DCGL_{emc.}$ The $DCGL_{emc}$ and $DCGL_w$ are equivalent, therefore all land areas are ensured to be remediated to below the $DCGL_w$ if the goal is to remove all DU fragments below one half the $DCGL_{emc.}$

<u>Class 1 Areas</u>

Areas that have, or had prior to remediation, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiation surveys) above the $DCGL_w$.

<u>Class 2 Areas</u>

Areas that have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the $DCGL_w$.

Class 3 Areas

Any impacted areas that are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a small fraction of the $DCGL_W$.

Non-Impacted Areas

Areas that have no reasonable potential for residual contamination. These areas have no radiological impact from site operations and are typically identified early in decommissioning.

Reference Coordinate System - Land Area

A reference coordinate system will be established on all Class 1 areas to facilitate selection of measurement and sampling locations. The grid will consist of intersecting lines, referenced to a fixed site location or benchmark. The referenced grid size for Class 1 land areas is 10 feet. All Class 2 land areas will have a grid size of 30 feet.

Statistical Analysis

The MARSSIMs process normally requires using statistical tests and the elevated measurement comparison to be used to validate that the survey unit meets the release criterion. In the case of IRP Site No. RW-41 Test Area C-74L any measurements above the investigation level of one half the $DCGL_{emc}$ will be removed and post removal FIDLER measurements taken to verify the DU fragment has been completely removed. Therefore sine the radionuclide is not considered to be in the background and all of the final measurements are below the $DCGL_W$, then statistical tests do not need to be used. The number of scaler measurements taken will depend upon the number of areas of elevated activity are found. All areas of elevated activity will be investigated and all areas of elevated activity which exceed one half the $DCGL_{emc}$ will be remediated and post removal FIDLER scaler readings taken.

2. FINAL STATUS SURVEY (FSS)

The FSS of IRP Site No. RW-41 Test Area C-74L Ballistic Test Facility will consist of the land areas of Test Area C-74L, which are divided into the Radioactive Material Controlled Area (RCA), Barrel Storage Area, Gun Corridor and the remainder of the land surrounding the test area, which was surveyed and characterized during 1999. The characterization and Interim Corrective Measures report is found in Earth Tech, March 2000, *Characterization Survey/Interim Corrective Measure Report.* The Derived Concentration Guideline Levels for these areas are found in paragraph 5. Figure 1, at the end of this item, shows the location of each area.

a. Radioactive Material Controlled Area (RCA;Class 1 Area)

The RCA is a U-shaped area of land surrounding the site gun corridor. This is the only area which will be remediated by removal of contaminated soil and DU penetrator fragments (approximately 500 cubic yards of soil). After the remediation is complete approximately 30 soil samples will be taken throughout the RCA to a depth of six inches. After analysis shows the land has been successfully remediated the area will be backfilled with clean fill material. The RCA is located within a wire fence and is contaminated with DU fragments ranging in size from a few grams to whole penetrators (300 grams). The area has been broken down into five smaller areas for the remediation to allow any backfilling of the smaller areas while the remainder of the RCA is being remediated. The RCA is considered one survey unit and is being subdivided only to save time during the remediation. Approximately six samples will be taken from each of the five subdivisions. A land survey of the area will include the perimeter of the RCA, delineation of the five subdivisions and marking of the soil sampling locations.

b. Barrel Storage Area (Class 2 Area)

The barrel storage area is a fenced in area located in the northeast part of the site. This was a radioactive material controlled area when barrels full of DU were stored there in the early 1980s. Since removal of the DU barrels the area has not been used. The area still has a wire fence delineating its perimeter. This area has been characterized with a 100 percent FIDLER survey during 1999 and areas of elevated activity marked. During late 2000 the areas of elevated activity due to DU fragments were removed and no areas of elevated activity remained above one half the $DCGL_{EMC}$.

c. Gun Corridor (Class 2 Area)

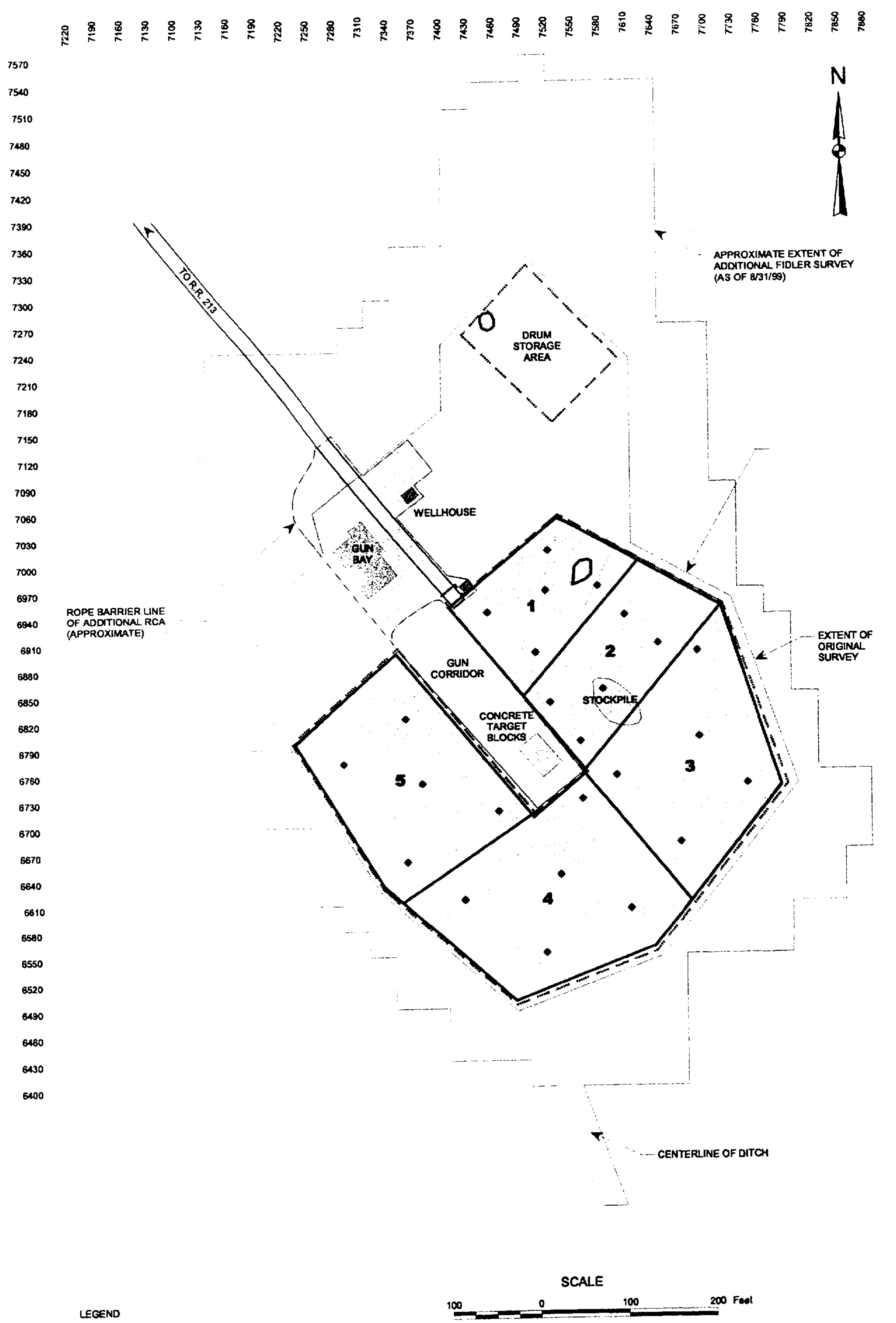
The gun corridor is a U shaped area located between the fire control building and the RCA. The RCA surrounds the gun corridor on three sides. The gun corridor was characterized during the March 1999 Site Characterization Survey and areas of elevated activity above the $DCGL_W$ were removed in late 2000. Soil samples taken within the area did not indicate any DU contamination above the DCGL (600 pCi/g). The gun corridor will be FIDLER surveyed again during the remediation effort to ensure conditions found during the CS conducted in 1999 have not changed.

d. Surrounding Land Area (Class 2 Area)

The area surrounding the fire control building, well housing building, barrel storage area, RCA, and gun corridor were 100 percent FIDLER surveyed in 1999, and areas of elevated activity removed. The surrounding area as shown in figure 1 will be 100 percent surveyed during the remediation activities of the RCA. Any areas of elevated activity exceeding one half the DCGL (DCGL equals 44 kcpm) will be removed and stored in the DU storage area located at Test Area C-64.

3. DERIVED CONCENTRATION GUIDELINE LEVELS (DCGLs)

The depleted uranium (DU) DCGLs for soil contamination and areas of elevated activity have been previously determined for all land areas on Eglin land test ranges. The $DCGL_w$ for soil contamination is 600 pCi/g and the $DCGL_{emc}$ for areas of elevated activity is 44 thousand counts per minute (kcpm). Background measurements for the FIDLER survey have been previously determined to be 5 kcpm for sand and 8 kcpm for red clay. The $DCGL_w$ and $DCGL_{emc}$ are equivalent to each other, which eliminates the possibility of small areas of elevated activity above the $DCGL_w$ and the need for statistical analysis of the results.



LEGEND

- FIDLER READINGS 0 TO 10 KCPM
- FIDLER READINGS 10 TO 20 KCPM
- ALIQUOT SOIL SAMPLE LOCATIONS TAKEN TO COMPOSITE FOR SECTION PROFILE SAMPLE
- EXTENT OF PROFILE SOIL SAMPLING
- EXTENT OF ORIGINAL SURVEY
- APPROXIMATE EXTENT OF ADDITIONAL FIDLER SURVEY

EARTH TECH
A tyco INTERNATIONAL LTD. COMPANY

FIGURE 1
IRP SITE NO. RW-41
TEST AREA C-74L
EGLIN AIR FORCE BASE, FLORIDA
PROJECT NO. 44971

STRUCTURE FINAL STATUS SURVEY PLAN – ITEM 10B

1. INTRODUCTION

Test Area C-74L Gunnery Ballistics Facility (Building No. 9372) is an active facility comprised of office work areas, two gun bays, and a target area used to test the damage potential and terminal ballistics of various ammunitions (Becker and others, 1994). The test area has been in operation since at least 1963 as a gunnery ballistics facility. From late 1974 to 1978, Test Area C-74L was used for pre-production testing of the GAU-8/A gun system, which uses depleted uranium (DU) in the ammunition. In late 1978, all testing involving DU was transferred to Test Area C-64, and the mission at C-74L was changed to include only the firing of high incendiary explosives.

The ballistics building was not used to store DU munitions. DU munitions were brought to the site at the start of the test and any remaining rounds were taken back to the normal storage area at the end of the test. If present because of range worker deposition, DU contamination inside the ballistics building would likely be found on floors, lower walls below 2 meters, and possibly air handling systems of the building. These areas of the ballistics building are considered impacted as defined in MARSSIM.

In addition to the historical information that indicates a low potential for residual radioactivity, a scoping survey, conducted during October 2001, supports the classification of the ballistics building interior as a Class 3 area. It is unlikely that remedial efforts of the building will be required.

The NRC screening values presented in Volume 3 of NUREG/CR-5512 will be used as DCGLs for the survey of the ballistics building interior. Since DU is comprised of U-238, U-235, and U-234, a DCGL that accounts for each isotope is developed. Given the stated 90th percentile individual DCGLs (NRC 2001) and the activity percentage of these isotopes in DU (AEPI 1995), a DCGL of 99 dpm/100 cm^2 total uranium above background is established. The Oak Ridge Institute for Science and Education (ORISE) computer code COMPASS® was utilized to develop the DCGL and was verified by hand calculations. See the calculation CE-Eglin-001 attached in this Section.

The external surfaces of the ballistics building and the well house building (Building No. 9373), which is a non-occupied structure built after DU munitions work had ceased, are considered impacted due to windblown contamination and will be classified as Class 3 areas. The DCGL for target areas (discussed below) will be used to determine the radiological status of the building exteriors.

Though measurable contamination was identified during the October 2001 scoping survey, the levels are not anticipated to exceed an appropriate DCGL. Since the target areas, just as building exteriors, are not habitable, they are considered equipment for development of the DCGL. A DCGL of 5,000 dpm/100 cm^2 (USACE EM-385-1-80, Table 6-4, the equivalent of NRC Regulatory Guide 1.86, Surface Contamination Guidelines) will be used to ensure proper instrument selection and count times. The characterization survey of the target area will be designed to meet, at a minimum, the needs of a Class 3 final status survey for the structure surfaces.

The final status surveys will be conducted using the guidance presented in the Multi-Agency Radiation Survey and Site Investigation Manual, NUREG-1575, Rev. 1 (NRC 2000a). The final status survey plans, a component of the work plan, will be provided to the USAF once completed.

2. DATA QUALITY OBJECTIVES

Data quality objectives are developed following the process outlined in MARSSIM chapter 3 and Appendix D.

The DQOs for the building and target area surveys are summarized below and will be provided to the USAF in the survey work plan once completed. Specifically, the 7 step process to developing DQOs is followed.

1. **State the Problem**: The problem is the potential presence of residual radioactive material on Test Area C-74L building surfaces and target area structures, from former operations involving depleted uranium. The objective of the surveys is to obtain data of sufficient quality and quantity to support an unrestricted release of the building and target area materials.

    - Planning team consists of the Eglin LLRM Partnering team, USAF, and USACE.
    - The primary decision maker for the buildings and target area survey is the USAF.
    - USACE and the USAF have sufficient resources to complete the surveys.

2. **Identify the decision**:

    - The principal study question is: Do DU concentrations inside/outside buildings or on equipment exceed background by more than the appropriate derived concentration levels (DCGL)?
    - The following decision statements should be evaluated sequentially. If concentrations do not exceed the DCGLs, the release criterion is satisfied.
      1. Determine whether the survey unit DU surface concentrations (dpm/100cm^2) exceed background by more than the appropriate DCGL.
      2. If survey unit concentrations exceed background by more than the DCGL should remedial alternatives be considered.
      3. Recommend what survey units or areas should be remediated.

3. **Identify Inputs to the decision**: Several site characteristics must be determined to resolve the decision statements.

    - Concentrations of DU in the survey units or on equipment. This information will allow determination as to whether or not a survey unit is likely to be suitable for release. Obtaining this data will facilitate cost effective decision-making.

    - Concentrations of DU in non-impacted materials the survey units or on equipment. This information will be needed if there is an indication that background levels of alpha/beta emissions from a particular building material are a significant relative to the DCGL.

    - External exposure and count rates. This information will be used to qualitatively determine if further investigations or remediation may be required.

4. **Define study boundaries**:

    - The population of interest is the areal concentration of depleted uranium on building and equipment surfaces. This will be subdivided geographically by the use of survey units.
    - The spatial boundaries of the surveys are limited to: the ballistics building floors, walls below 2 meters, and air handling system; the exterior surfaces of the ballistics and well house buildings; and the target area catch box structure surfaces.
    - The decision applies to the time of the survey and for the future as long as the facility does not utilize radioactive materials. Data collection should be conducted based on project schedule and in coordination with the installation support of testing.
    - Decisions will be made for the building survey units and the target area equipment.
    - Constraints on data collection include site operations (testing), weather in target areas (outdoors, wet materials will impact readings), target material surfaces (extremely rough or jagged surfaces may limit data collection).

5. **Develop a Decision Rule:**

    - Parameters of interest are the mean, median, and standard deviation of data collected. Based on distribution characteristics from data collection, the parameters may be transformed to allow for statistical testing (log-normal, parametric, non-parametric).
    - Decisions are based on the DCGL presented for each area (99 dpm/100cm2 total uranium for interior building surfaces and 5,000 dpm/100cm2 total uranium for equipment).

- Decisions are made based on the piece of equipment, the surveys units, the building and the combination of the building and target area materials. In cases where contamination above the criteria are clearly indicated, decisions on remediation, reclassification, survey unit subdivision, etc., may be taken as appropriate.
- Inputs to the decisions are:
    - Survey unit dimensions
    - Surface alpha, beta, and gamma scans (pipes, equipment, slots, survey units)
    - Integrated surface alpha measurements
    - Wipe samples
- Decision rules
    - Survey unit dimensions: If the measured dimensions exceed the MARSSIM recommended size the boundaries will be adjusted accordingly.
    - Scans: Areas that exhibit elevated scan readings will be marked for further investigation (integrated readings and wipe samples), consideration, or remediation (pipes). These data are evaluated qualitatively.
    - Integrated surface alpha measurements;
        - If all measurements are less than the DCGL the survey unit is deemed to meet the release criteria.
        - If a measurement on a building surface exceeds the DCGL, it will be investigated and could require survey unit reclassification, subdivision, and/or remediation.
        - If any measurement on an equipment surface exceeds the DCGL, it will be investigated and could require survey unit reclassification, subdivision, and/or remediation.
    - Wipe samples: If results of wipe samples indicate greater than 10% of the building DCGL or 20% of the equipment DCGL is removable an evaluation of the DCGL is required.

6. **Specify limits on decision errors**: Statistical acceptability decisions are always subject to error. Two error types are associated with the decisions.

    - Type 1: The null hypothesis is rejected when true. This error could result in potential doses to the critical receptors greater than prescribed by the DCGL. The maximum Type I error rate is set at 0.05.
    - Type 2: Null hypothesis is not rejected when it is false. This error results in increased costs due to re-surveying, remediation, etc., when it is not necessary. The maximum Type II error rate is set at 0.05.

7. **Optimize the Design:** The survey design will continually be optimized as the plans are reviewed and edited.

3. FINAL STATUS SURVEY (FSS)

The Derived Concentration Guideline Levels for the buildings and target area are found in paragraph 4.

A. Ballistic Building Interior (Class 3 Area)

Given the historical use of DU munitions in the two gun bays at the ballistics building, and the design of the depleted uranium munitions, it is unlikely that contamination exists within the building at greater than background levels. The GAU-8/A 30 mm DU rounds produced by Aerojet and Honeywell all use an aluminum wind screen which, when combined with other components, effectively encapsulates the DU until the round strikes a target. Under normal handling and storage of these munitions, contamination is unlikely. Accidents or malfunctions of the munitions could be a potential source of contamination in the gun bay areas; however, this was not documented in the site history. Because firing was conducted over several years and the surface soils near the target were likely contaminated, it is possible that range workers carried DU contamination back into the gun bays and other work areas on their shoes or other clothing. If present, DU contamination inside the ballistics building would likely be found on floors, lower walls below 2 meters, and possibly air handling systems of the building. These areas of the ballistics building are considered impacted as defined in MARSSIM.

In addition to the historical information that indicates a low potential for residual radioactivity, a scoping survey, conducted during October 2001, supports the classification of the ballistics building interior as a Class 3 area. It is unlikely that remedial efforts of the building will be required.

B. Ballistics Building and Well House Building Exteriors (Class 3 Area)

The exterior surfaces of the ballistics building and the well house building will be surveyed as Class 3 areas. Since firing was conducted and contamination exists in soils, it is possible that building exteriors became contaminated through windblown contamination or in the case of the ballistics building, ricochet fragments.

C. Target Area (potential Class 3 survey)

The concrete blocks that supported the targets used in DU munitions testing at the range were disposed of as contaminated items during site remediation efforts in the 1980s. The original catch box, a concrete and metal structure behind the targets, remains at the site. DU contamination is likely to be present and the structure is considered an impacted area. It is possible that small fragments of DU penetrators may be lodged in the concrete of the catch box or finer particles may be disbursed on the surfaces. Though measurable contamination was identified during the October 2001 scoping survey, the levels are not anticipated to exceed the appropriate DCGL. The characterization survey of the target area will be designed to meet, at a minimum, the needs of a Class 3 final status survey for the structure surfaces.

4. DERIVED CONCENTRATION GUIDELINE LEVELS (DCGLs)

A. Buildings (internal): Given the Historical Site Assessment (HSA) contamination in the site buildings is not expected to exceed background by more than a small fraction of the DU DCGL. Therefore, it is expected that all building impacted areas will be classified as class 3 areas per MARSSIM. Wipe samples for removable contamination will be collected to determine whether the use of the screening DCGLs presented in NUREG/CR-5512 is appropriate. Wipe sample results should indicate that the average removable activity is less than ten percent (10%) of the DCGL.

TABLE 10B. Building DCGL and Potentially Contaminated Building Media

Contaminant	DCGL dpm/100 cm^2	Media	Measurement Method
DU	99[1]	Concrete Block	Building Surfaces only
		Steel and Aluminum	
		Concrete	Gross alpha[2] (plastic and zinc sulfide detector coupled to a scaler with an alpha and beta discriminator)
		Wood	
		Dry Wall	

[1] Accounts for U-238, U-235, and U-234 fractions and differing DCGLs, See Calculation CE-Eglin-001

[2] DQO for Gross Alpha and Beta activity will be developed, however, due to typical beta background it is expected that the gross beta may not be a suitable measurement technique on all media. A demonstration that the measurement method should be capable of detecting residual radioactivity is included as calculation CE-Eglin-002.

B. Target Area and external building structures: Since the target areas are not habitable and may be considered equipment, an initial DCGLw of 5,000 dpm/100 cm^2 (USACE EM-385-1-80, Table 6-4, the equivalent of NRC Regulatory Guide 1.86., Surface Contamination Guidelines) will be assumed for insuring proper instrument selection and count times.

Given the low occupancy frequency of personnel outside of buildings the exterior of buildings will be surveyed to the initial DCGLw of 5,000 dpm/100 cm^2.

CALCULATION SHEET

CLIENT & PROJECT **USAF** **US Army Corps of Engineers – Omaha District** **Building Survey and Decommissioning, Eglin Air Force Base, FL.**	PAGE 1 OF 5
CALCULATION TITLE **CE-Eglin-001, Derivation of Depleted Uranium DCGL from NRC Screening Values**	QA CATEGORY (✓) I NUCLEAR SAFETY RELATED ☐ II ☐ III ☐ (other)

CALCULATION IDENTIFICATION NUMBER				
JOB ORDER NO.	DISCIPLINE	CURRENT CALC NO	OPTIONAL TASK CODE	OPTIONAL WORK PACKAGE NO.
NA	**HP-FSS**	**CE-Eglin-001**		N/A

APPROVALS – SIGNATURE & DATE			REV. NO. OR NEW CALC NO.	SUPERSEDES CALC NO. OR REV NO.	CONFIRMATION REQUIRED ☑	
PREPARER(S)/DATE(S)	REVIEWER(S)/DATES(S)	INDEPENDENT REVIEWER(S)/DATE(S)			YES	NO
David C. Hays, USACE	Hans Honerlah, USACE	Brian Hearty CHP, USACE	0	N/A		
	Julie Peterson, USACE		0	N/A		

DISTRIBUTION

GROUP	NAME & LOCATION	COPY SENT (✓)	GROUP	NAME & LOCATION	COPY SENT (✓)
Project Manager Design Task Manager			Hans Honerlah Julie Peterson Brian Hearty	USACE, Baltimore USACE, HTRW CX USACE, HTRW CX	

CALCULATION IDENTIFICATION NUMBER				Page 8 of 5
J.O. OR W.O. NO.	DIVISION & GROUP	CALCULATION NO.	OPTIONAL TASK CODE	
NA	HP-FSS	CE-Eglin-001		

RECORD OF REVISIONS

Revision 0 - Original Issue

1.0 INTRODUCTION

NUREG/CR-5512 (NRC 2001) provides screening values for Derived Concentration Guidelines (DCGL) based on calculations using the NRC D&D computer code. The DCGLs are provided for building surfaces in dpm/100 cm^2 by isotope. Since depleted uranium is comprised of U-235, U-238, and U-234 either a DCGL that accounts for the three is developed or each isotope must be measured independently.

2.0 OBJECTIVE

The objectives of this calculation are to:

- Compare the calculated DCGL from the approaches used to determine a DU DCGL, and
- Determine a DCGL for DU.

3.0 ASSUMPTIONS

This calculation is based on NUREG/CR-5512 screening values and the isotopic abundance of U-238, U-235, and U-234 in DOD DU as reported by the Army Environmental Policy Institute (AEPI 1995) or the assumptions incorporated in the ORISE computer code COMPASS.

4.0 METHODOLOGY

One method to determine a total DCGL would be to set the total DCGL as the lowest reported isotopic DCGL. Another method would be to establish a total DCGL based on the relative activity percentage of each isotope to the total. The latter method is chosen for this calculation. Two approaches to developing the total (DU) DCGL are presented here. First the Oak Ridge Institute for Science and Education (ORISE) computer code COMPASS was utilized to develop the DCGL, then the DCGL was hand calculated for a comparison.

The NUREG 5512, 90th percentile screening DCGL used in this calculation are:

- U-238 = 101 dpm/100 cm^2
- U-235 = 98 dpm/100 cm^2
- U-234 = 91 dpm/100 cm^2

Lowest DCGL = 91 dpm/100 cm^2

COMPASS REPORT
DU DCGLw = 99 dpm/100 cm^2

Site Report

Depleted Uranium Summary

NOTE: Surface soil DCGLw units are pCi/g.
Building surface DCGLw units are dpm/100 cm².

Selected Method: Enter U-235 Weight Percent

U-235 Enrichment (weight %): 0.2

Radionuclide	Concentration (pCi/g)
U-234	21
U-235	1
U-238	77.6

	Surface Soil	Building Surface
U-234 DCGLw	N/A	91
U-235 DCGLw	N/A	98
U-238 DCGLw	N/A	101
Modified U-238 DCGLw	N/A	N/A
Total U DCGLw	N/A	99

Series Summary

Series	Radionuclide	Emission	Avg. Beta Energy (keV)	Yield
DU	U-234	Alpha	N/A	0.2108
	U-235	Alpha	N/A	0.01
	Th-231	Beta	76.4	0.01
	U-238	Alpha	N/A	0.7791
	Th-234	Beta	43.5	0.7791
	Pa-234m	Beta	819	0.7791

Uranium is a naturally occurring radioactive isotope. DU is uranium that has been separated from the other naturally occurring members of the uranium and actinium decay series and depleted of U-234 and U-235. In natural uranium, the U-234, U-235, and U-238 isotopes are present in their naturally occurring ratios, while this ratio has been altered in DU. The naturally occurring activity ratios of U-234/U-235/U-238 to U-238 are 1.0/0.047/1.0, respectively (AEPI 1995). The DU activity ratios of U-234/U-235/U-238 to U-238 are 0.18/0.013/1.0, respectively (AEPI 1995). The activity of a gram (g) of DU is approximately 0.4 micro-curies (uCi) (AEPI 1995). Thus the activity in 1g of DU is comprised of 0.052 uCi of U-234; 0.0052 uCi of U-235 and 0.348 uCi of U-238.

Since DU is comprised of approximately 99.8% U-238, 0.001% U-234, and 0.2% U-235 by weight and the NUREG/CR-5512 values are isotope specific, the DU value is derived from the activity ratios of U-238, U-235, and U-234 in DU of 83.2%, 15.7%, and 1.1% respectively.

$$\text{DU DCGLw} = (\text{DCGL}_{\text{U-238}})(\%\ \text{U-238}_{\text{DU}}) + (\text{DCGL}_{\text{U-235}})(\%\ \text{U-235}_{\text{DU}}) + (\text{DCGL}_{\text{U-234}})(\%\ \text{U-234}_{\text{DU}})$$

$$\text{DU DCGLw} = (101\ \text{dpm/100 cm}^2)(.832) + (91\ \text{dpm/100 cm}^2)(.157) + (98\ \text{dpm/100 cm}^2)(.011)$$

$$\text{DU DCGLw} = 99.4\ \text{dpm/100 cm}^2$$

Hand Calculation Method DU DCGLw = 100 dpm/100 cm^2 (rounding to 99 may be more appropriate)

of method using Sum of Ratios and a total DU result of 100 dpm/100 cm^2

$$\text{SOR} = \frac{(\text{DU dpm/100 cm}^2)(\%\ \text{U-238}_{\text{DU}})}{(\text{DCGL}_{\text{U-238}})} + \frac{(\text{DU dpm/100 cm}^2)(\%\ \text{U-235}_{\text{DU}})}{(\text{DCGL}_{\text{U-235}})} + \frac{(\text{DU dpm/100 cm}^2)(\%\ \text{U-234}_{\text{DU}})}{(\text{DCGL}_{\text{U-234}})}$$

$$\text{SOR} = \frac{(100\ \text{dpm/100 cm}^2)(0.83)}{(101\ \text{dpm/100 cm}^2)} + \frac{(100\ \text{dpm/100 cm}^2)(0.01)}{(98\ \text{dpm/100 cm}^2)} + \frac{(100\ \text{dpm/100 cm}^2)(0.15)}{(91\ \text{dpm/100 cm}^2)}$$

SOR = 1

5.0 SUMMARY OF RESULTS

There is no significant difference in calculated DU DCGLw from either the COMPASS software or the hand calculated method. The conservative approach of choosing the lowest value results in a DCGLw for DU approximately 10% less than the other methods.

The DU DCGLw best utilized for the Eglin building surveys is 99 dpm/100 cm^2

6.0 REFERENCES

(AEPI 1995) *Health and Environmental Consequences of Depleted Uranium Use in the U.S. Army: Technical Report*, U.S. Army Environmental Policy Institute, June, 1995

(NRC 2000a) NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual* (MARSSIM), U.S. Nuclear Regulatory Commission, dated August 2000

(NRC 2000b) NUREG-1727, *NMSS Decommissioning Standard Review Plan*, U.S. Nuclear Regulatory Commission, dated September 2000

(NRC 2001) NUREG/CR-5512 Vol. 3, SAND96-XXXX, *Residual Radioactive Contamination From Decommissioning*, Parameter Analysis, 2001

(ORISE 2001) *COMPASS version 1.0*, Computer Code, Oak Ridge Institute for Science and Education, 2001.

7.0 ATTACHMENTS

None

USACE – SWT- SO HEALTH PHYSICS
CALCULATION SHEET

<table>
<tr><td colspan="5">CLIENT & PROJECT
USAF
US Army Corps of Engineers – Omaha District
Building Survey and Decommissioning, Eglin Air Force Base, FL.</td><td colspan="4">PAGE 1 OF 4</td></tr>
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CE-Eglin-002, Demonstration of MDC for Eglin Building Survey</td><td colspan="4">QA CATEGORY (✓)
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☐ II
☐ III ☐ (other)</td></tr>
<tr><td colspan="5">CALCULATION IDENTIFICATION NUMBER</td><td colspan="4"></td></tr>
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<tr><td>David C. Hays, USACE</td><td>Hans Honerlah, USACE

Julie Peterson, USACE</td><td colspan="2">Brian Hearty CHP, USACE</td><td>0

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Design Task Manager</td><td></td><td></td><td>Hans Honerlah
Julie Peterson
Brian Hearty</td><td>USACE, Baltimore
USACE, HTRW CX
USACE, HTRW CX</td><td></td></tr>
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CALCULATION IDENTIFICATION NUMBER				Page 13 of 4
J.O. OR W.O. NO.	DIVISION & GROUP	CALCULATION NO.	OPTIONAL TASK CODE	
NA	HP-FSS	CE-Eglin-002		

RECORD OF REVISIONS

Revision 0 - Original Issue

1.0 INTRODUCTION

NUREG 1727, The Standard Review Plan for decommissioning plans requires that the NRC staff review the proposed survey methods to determine if the method is appropriate based on determination of the Minimal Detectable Concentration (MDC) being less than the DCGL. The licensee is expected to provide sufficient information in the plan to demonstrate this.

4.0 OBJECTIVE

The objective of this calculation is to estimate the priori MDC and demonstrate the adequacy of the expected survey method and instrumentation to be used during the Eglin AFB Ballistics building surveys.

5.0 ASSUMPTIONS

This calculation is based on the following assumptions:

1) Of the building construction materials, concrete is the most prevalent and provides a typical gross alpha background of 0 to 4 cpm with an average of 2 cpm. These value fit well with the typical ZnS background on various materials (steel, drywall, wood) presented in NUREG 1507, except for ceramics.

2) The instrumentation to be used for gross alpha measurements is a Ludlum 2360 coupled to a Ludlum model 43-89 alpha/beta scintillation detector. The instruments alpha intrinsic efficiency is 0.4 (Ludlum 2001). Typical alpha background is less than 3 cpm and less than 1% beta to alpha cross talk (Ludlum 1998).

3) The ISO7503-1 alpha source efficiency of 0.25 (NRC 1998) will be used for calculations. NUREG 1507 lists typical source efficiency for distributed sources as 0.22 on steel, 0.54 on wood, and 0.43 on concrete, therefore 0.25 is assumed to be a conservative estimate.

4) The isotopic abundance of U-235, U-238, and U-234 result in a combined 1 alpha emission per dpm of depleted uranium (ORISE 2001).

4.0 METHODOLOGY

The approach used in this calculation is explained in MARSSIM chapter 6 (NRC 2000).

Critical Level Calculation

MARSSIM equation 6.6: $Lc = 2.33\sqrt{B}$

Where, Lc = critical level

B = Background counts expected in measurement

CALCULATION IDENTIFICATION NUMBER				Page 14 of 4
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NA	HP-FSS	CE-Eglin-002		

1 min count time

$Lc = 2.33\sqrt{B}$

$Lc = 2.33\ \sqrt{2}$

Lc = 3.3 counts in 1 min

2 min count time

$Lc = 2.33\sqrt{B}$

$Lc = 2.33\ \sqrt{4}$

Lc = 4.66 counts in 2 min

Detection Limit Calculation

MARSSIM Equation 6-6. $Ld = 3 + 4.65\sqrt{B}$
Where, Ld = detection limit

1 min count time

$Ld = 3 + 4.65\sqrt{B}$

$Ld = 3 + 4.65\sqrt{2}$

Ld = 10 counts

2 min count time

$Ld = 3 + 4.65\sqrt{B}$

$Ld = 3 + 4.65\sqrt{4}$

Ld = 12 counts

MDC Calculation

MARSSIM Equation 6-7. MDC = C x Ld
Where C is used to convert from counts to concentration.

$$C = \frac{1}{(Ieff)(Seff)(A)(t)}$$

Given:
Ieff = Intrinsic efficiency = 0.4
Seff = Source efficiency = 0.25
A = Detector area /100 cm^2 = 1.25
t = count time in minutes

1 min count time

$$C = \frac{1}{(Ieff)(Seff)(A)(t)}$$

$$C = \frac{1}{(.4)(.25)(1.25)(1)}$$

C = 8 dpm/100 cm^2/count

2 min count time

$$C = \frac{1}{(Ieff)(Seff)(A)(t)}$$

$$C = \frac{1}{(.4)(.25)(1.25)(2)}$$

C = 4 dpm/100 cm^2/ count

<u>1 min count time</u>
MDC = C x Ld

MDC = (8 dpm/100 cm^2/count)(10counts)

MDC = 80 dpm/100 cm^2

<u>2 min count time</u>
MDC = C x Ld

MDC = (4 dpm/100 cm^2/count)(12counts)

MDC = 48 dpm/100 cm^2

Calculation Check

The ORISE computer code COMPASS was used to verify the hand calculations presented here. The output from COMPASS for DU and the same assumptions as in this calculation resulted in gross alpha MDCs of 79 and 51 dpm/100 cm^2 for count times of 1 and 2 minutes respectively.

5.0 SUMMARY OF RESULTS

The estimated MDC for gross alpha measurements and a 2 minute count time at Eglin AFB Ballistics building is 48 dpm/100 cm^2. This value is less than the expected DCGLw of 99 dpm/100 cm^2 and meets the suggested requirements of MARSSIM (50% of the DCGLw). Actual count time should be determined based on MDC calculations using actual field measurements of background materials.

6.0 REFERENCES

(AEPI 1995) *Health and Environmental Consequences of Depleted Uranium Use in the U.S. Army: Technical Report*, U.S. Army Environmental Policy Institute, June, 1995

(Ludlum 1998) Instruction Manual, *Ludlum Model 43-89,43-90, and 44-116 Alpha/Beta Scintillators*, December 1998

(Ludlum 2001) Calibration Paperwork, *Bench Test Data for Detector Ludlum Model 43-89 SN 14546B*, dated 4 May 2000

(NRC 2000) NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual* (MARSSIM), U.S. Nuclear Regulatory Commission, dated August 2000.

(NRC 2001) NUREG/CR-5512 Vol. 3, SAND96-XXXX, *Residual Radioactive Contamination From Decommissioning*, Parameter Analysis, U.S. Nuclear Regulatory Commission, dated 2001

(NRC 1998) NUREG 1507, *Minimal Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions,* U.S. Nuclear Regulatory Commission, dated June 1998.

(ORISE 2001) *COMPASS version 1.0*, Computer Code, Oak Ridge Institute for Science and Education, 2001.

7.0 ATTACHMENTS None

WASTE MANAGEMENT – ITEM 11

HEALTH AND SAFETY PLAN – ITEM 11A

Note: All references to Section Numbers and Tables in Item 11 refer to the Site Radiation Protection, which will be completed and approved by the Eglin LLRM Partnering Team prior to the beginning of remediation at the site.

1. RADIATION SAFETY TRAINING

A. TRAINING AND OTHER RADIATION WORKER QUALIFICATIONS

Training is one of the most important elements in an effective radiation and ALARA program. The amount and type of training depends upon the individual's work assignment.

A fully trained and experienced Site RSO, responsible to the Earth Tech Regional Health and Safety Specialist (RHSS), will be continually on site to implement and enforce the health and safety procedures outlined in this document during all site operations. The Earth Tech RHSS, experienced in hazardous and radioactive waste site operations, will be responsible for the implementation and oversight of the project health and safety program. Before work area entry, all site personnel and visitors must attend a (general and site-specific) safety and health briefing session, to be conducted by the Site RSO or a qualified HPT. The briefing will cover potential site hazards and all aspects of the Site Remedial Action Work Plan.

B. REGULATORY REQUIREMENTS

All radiation workers will be required to submit documentation of introductory (40-hour), supervisory (if applicable), and refresher (8-hour) training in accordance with OSHA, 29 CFR 1910.120, 29 CFR 1926.65, and USACE EM 385-1-1 (USACE, September 1996) prior to site work. Certificates of training will be maintained at the project site for the duration of the project. Copies of the certificates or other official documentation will be used to fulfill this requirement. Visitors will be required to register with the Site RSO and sign in on a daily basis. No visitors will be allowed in the Radioactive Material Controlled Area (RCA)/work zone.

In addition to the OSHA training requirements listed in the preceding section, all personnel will be required to successfully complete 8 hours of Radiation Worker Training before performing activities inside of the project RCA.

This radioactive training will be designed to meet the requirements of NRC, USACE, and the IRP contractor. The NRC, in 10 CFR 19.10, requires that all individuals who in the course of their employment are likely to receive an occupational dose in excess of 100 millirem per year (mrem/year) be instructed in the health protection issues associated with exposure to radioactive materials or radiation (NRC, *Regulatory Guide 8.29*, February 1996). Although Earth Tech and USACE do not anticipate that workers will be exposed to over 100 mrem/year, the suggested topics as presented in the Appendix of NRC (February 1996) and as summarized below will be included in the radioactive training. The topics listed below also encompass USACE's training requirements as presented in USACE EM-385-1-80 (*Safety Radiation Protection Manual,* USACE, May 1997).

The sample coordinator shall also have sufficient experience to meet the training requirements contained in 49 CFR Part 172 Subpart H.

C. IMPLEMENTATION

The radiological worker training will entail eight hours and include the following topics:

- The general theory and physics of radiation, including radioactive decay, types of nuclear radiation, units of measurement, radiation's interaction with matter, and the mathematics necessary to understand the above subjects.
- The biological effects of radiation, including early and delayed effects.
- Chronic and acute radiation doses.
- Health risks, including a discussion on what the risks numbers mean and how the radiation risks are estimated.
- Radiation dose limits and exposure control, including a discussion on internal versus external exposures and the ALARA principle.
- Radiological dosimetry and radiation control practices.
- The instrumentation necessary to detect, monitor, and survey radiation, and the use of such instrumentation, including practical hands-on experience using radiation instrumentation and procedures.
- Radiation safety techniques and procedures, including the use of time, distance, shielding, engineering controls, and PPE to reduce exposure to radiation.
- Decontamination.
- Instruction in the employee's rights and responsibilities under the applicable NRC licenses and permits.
- Sources of radiation exposure.
- Emergency procedures for several events, including personnel injury, fire, site evacuation, and project emergency procedures.
- Notification of incidents, all workers will be briefed on incident reporting prior to commencement of work. At a minimum, incidents (e.g., falls, over exposure to elements) occurring at the work site will be immediately reported to the Eglin RSO and Site RSO. In the event that any member of the field crew has an incident, experience adverse effects or symptoms of exposure while at the work site, the entire field crew shall immediately halt work and act according to the instructions provided by the SSHP. The Site RSO will

reevaluate the situation at the site, take measures to correct the situation, and reevaluate the level of personal protection equipment required to complete site work. In addition, the Site RSO will complete an incident report and submit it to the Site PM.

Even though some of the topics presented in the Appendix of USNRC (February 1996) are not explicitly included in the list above all of these questions will be addressed during the radiation worker training. Training will be performed by qualified Earth Tech personnel. The training will consist of both classroom and practical factors training and will be documented by the successful completion of written exam.

A discussion on counseling to occupationally exposed women for the control of radiation exposure to embryo and fetus is presented in Appendix B of this document.

D. TRAINING DOCUMENTATION

All workers shall be tested on their comprehension of the training instruction and shall sign a statement acknowledging that they have received the training and shall comply with the radiation protection rules and requirements before being allowed to enter a radiation area.

E. OTHER RADIATION WORKER QUALIFICATIONS

All individuals requiring access to a Restricted Area or a RCA (Radiation Area, Contaminated Area or Airborne Radioactivity Area) must be issued personnel dosimetry.

All individuals requiring access to RCAs must complete a History of Occupational Radiation Exposure (USNRC Form 4, USAF Form 1527 or equivalent), and have received an approved Radiation Worker physical examination within the past year (including pulmonary function tests).

Individuals requiring access to RCA must have received or have record of in the Radiation Safety Office, a baseline bioassay urinalysis.

Upon receipt of the bioassay urinalysis, each individual who works at the Eglin site for a period greater than one year shall receive a periodic bioassay urinalysis in accordance with the bioassay program requirements.

Bioassay results will be recorded on a USAF Form 2753, Radiological Sampling Data, or equivalent format and be retained by the Site RSO for the duration of the project (see Records Retention for ultimate disposition).

2. FACILITIES AND EQUIPMENT

A. FACILITIES

Site support will require two air conditioned trailers of sufficient size to support the remediation efforts. This will include an administrative trailer which will provide desks or office space for the Project Manager, Base RSO, and additional space to conduct site training and daily safety briefings. The second trailer will be the Quality Assurance Trailer, which will provide office space for the Quality Assurance Officer, Site Health Physicist, and space for the health physics support personnel to meet. This trailer will be used to store health physics equipment and radiological samples awaiting shipment off site. The trailer will have sufficient space for calibration and maintenance of radiological equipment. A third smaller trailer will be available on site to store decontamination supplies and other support equipment.

Bottled water will be supplied in the trailers and drinking water will be available inside the CRZ for workers to drink. Electricity will be provided from electrical access at the site. A portable generator will be available to provide backup electrical power during critical operations. Cell phones will be provided for communications at the site.

B. EQUIPMENT

Several pieces of heavy equipment will be required to conduct excavation activities. A brief description of each of these pieces of equipment and their intended use is provided below:

- Front-end loader – A front-end loader will be utilized in situations where a large volume of soil may need to be moved in one bucket. The front-end loader will be wheel-mounted. A wheel-mounted front-end loader can travel quickly between the stockpiled soils and the location containers are loaded. A small bobcat front-end loader will be used.
- Portable Water Tank – A portable water tank will be required on-site to assist in the control of dust during the remedial activities. The portable water tank will be required to control dust within the excavation area during excavation and during backfill operations. The water to be used for dust control will be obtained from the water well located at IRP Site No. RW-41 (Section 1.1.13).
- Forklift – A heavy-duty forklift could be used to move the filled containers from the excavation area to the load-out area. The forklift can also be used for loading the containers onto transport trucks for delivery to the C-64 DU waste storage area.
- Flat Bed Truck – A flat bed truck will be used in moving containers to and from the loading pad and the rail facility. The trucks will be carrying multiple filled disposal containers.

3. RADIATION MONITORING INSTRUMENTS

- FIDLERs will be maintained on site to perform walk over surveys of the test area. The minimum detectable scan limit is 1708 cpm.
- GM Meters/pancake probe – a minimum of 3 meters will be available on site. These instruments will be used to frisk personnel and equipment out of the controlled areas.
- Sodium Iodide 2" – a 2" sodium iodide detector will be maintained on site to perform down hole logging if the site health physicist determines the need to take core samples greater than six inches in depth.

- Micro-R Meter – a minimum of 2 meters will be maintained on site during remediation operations. The meters will be used to determine dose levels within the EZ as required by the Site Health Physicist.
- Portable Gamma Spec – a portable gamma spec will be utilized on site to characterize profile samples. The portable gamma spec will also be used to verifiy the absence of other radionuclides in the waste soil. The gamma spec will be operated by the site health physicist or other trained health physics personnel.

4. RADIATION SAFETY PROGRAM AND ALARA

A. ALARA POLICY

During radiological work at Eglin, adherence will be given to the ALARA principle. This principal requires that the Site RSO take all reasonable actions to reduce personnel exposures to as low a level as possible, given the existing technology, cost and operational requirements. The ALARA program will be implemented through the use of the following:

- Administrative Exposure Control Levels that are well below the regulatory limits of 10 CFR 20;
- Training of employees and subcontractor personnel in appropriate radiation protection practices, work procedures, and RWP use;
- Good housekeeping practices;
- Engineering controls; and
- Use of personal protective equipment (PPE), as necessary.

B. ADMINISTRATIVE EXPOSURE CONTROL LEVELS

In order to maintain external radiation exposure levels ALARA, administrative exposure control limits shall be established for projects that have the potential for worker exposure to radiation. These limits shall be established based on the site activities to be performed and the radiation levels expected during the performance of site operations.

Whole Body

The annual administrative whole body dose limit (combined external deep dose and internal committed effective dose equivalent) is 500 millirem (mrem) total effective dose equivalent (TEDE). The sum of the deep dose equivalent and the internal committed dose equivalent to any individual organ or tissue other than the lens of the eye is 5 rem. Based on the nature of the radioactive material known to be present (depleted uranium [DU]), this dose limit should not be exceeded under any circumstances.

Skin and Extremities

The annual administrative dose limit for the skin or any extremity is 5 rem (5000 mrem) shallow dose equivalent.

Lens of the Eye

The annual administrative dose limit to the lens of the eye is 1.5 rem (1500 mrem).

Declared Pregnant Worker

The Administrative Dose Limit to a Declared Pregnant Worker is 500 mrem TEDE per year. This administrative limit shall limit the dose to the fetus to less than 300 mrem (accounting for the duration of the pregnancy and the additional shielding provided to the fetus by the mother's abdomen). Upon declaration of pregnancy, the Site RSO shall:

- Perform a retrospective review of the dose received to date by the declared pregnant worker. If the dose during gestation is approaching 500 mrem, the declared pregnant worker shall be removed from any further work involving radioactive sources or other radiation exposure for the remainder of her pregnancy.
- Provide increased radiation safety surveillance of the declared pregnant worker to ensure that her radiation exposure is maintained ALARA; that she does not exceed the prescribed Administrative Dose Limit; and to ensure that the Administrative Dose Limit is evenly applied during the period of her pregnancy (e.g., dose should be limited to 125 mrem per calendar quarter).

C. ALARA PROCEDURES

One of the basic tenants of radiation protection is that external radiation dose may be reduced by paying careful attention to time, distance and shielding. That is, minimizing the time spent in a radiation area, maximizing the distance between the worker and the source of the radiation, and employing shielding between the worker and the source of the radiation. In addition to these tenants, there are also basic practices that can be used to limit internal exposure to radioactive material. The following sections detail these principles.

Methods of Minimizing Time

Work done in a radiation area should be carefully planned to minimize the time spent in the area by workers. Examples of such planning are as follows:

- Review of scope of work before entering the radiation area. Careful review of the work should eliminate delays caused by incorrect tools, misidentification of item to be worked on, the nature of the work to be performed or using inappropriate personnel to perform the work. This is partly the reason why RWPs are used at the site (see Section 6.1 for more information on RWPs).
- Pre-staging tools, equipment and supplies that will be necessary to accomplish the scope of work.

- Simulation of complex activities. In rare cases, an operation may be complex enough to require setting up a simulation in a non-radiation area. Another example of simulation might be taking background samples for practice before taking samples in a radiation area. Thus, when starting a new task it is usually ALARA to start the task in a low-radiation area before proceeding to perform the task in an area of higher radiation exposure.

Methods for Maximizing Distance from Radiation Sources

Methods may be used to increase the distance between the worker and the source of radiation. Increasing the distance is an excellent radiation protection technique since the radiation dose rate decreases with the square of the distance from the source. Methods of increasing the distance include the following:

- Use long-handled tools whenever possible.
- Rather than handling radioactive sources directly, use tongs or forceps.
- Remove unnecessary workers from the RCA, or move them to a low-radiation area of the RCA when a worker's presence is not immediately needed.
- Store radioactive material in an isolated location.

Proper Uses of Shielding

DU is commonly considered to present a low external radiation hazard because of its predominantly alpha radiation. However, shielding is appropriate when dealing with DU, because it also emits beta and gamma radiation (or, more accurately, the progeny of DU emit these radiations). In fact, the contact beta dose rate from a DU source may approach 200 millirad per hour (mrad/hr). Gamma dose rates are lower, although concentration of large numbers of DU sources can produce gamma dose rates of at least moderate radiation protection concerns (from an ALARA prospective). Shielding practices may include:

- Using low-Z materials such as plastic or aluminum to reduce beta radiation exposure.
- Use of high-Z material such as lead, iron or even soil to reduce gamma radiation exposure.

For combined beta-gamma sources, always shield for the beta radiation first (low-Z material), then the gamma radiation second (high-Z material). Use of high-Z material to shield beta radiation may increase the dose rate due to X-ray production in the high-Z material from the interaction with the beta radiation.

Normal PPE also acts as a significant shielding material against alpha and beta radiation, but not for gamma radiation.

D. OCCUPATIONAL DOSE

Methods to Reduce Internal Radiation Dose

Internal dose reduction is actually internal dose prevention. Once radioactive material enters the body, it is usually difficult to remove, hence the dose is said to be committed once an intake occurs. Therefore, internal radiation dose reduction methods include methods to reduce ingestion and inhalation of radioactive material, as well as contamination control practices (once an object is contaminated, the possibility of re-suspension and ingestion or inhalation of the radioactive material exists).

- *Ingestion*—To prevent ingestion of radioactive material, no eating, drinking, smoking or chewing (gum or tobacco) is permitted in the RCA. Also, care should be taken by workers not to touch their mouth or face with their hands while in the RCA.
- *Inhalation*—In addition to the above ingestion control measures, workers must be careful not to increase the levels of dust or generate airborne aerosols. Measures to prevent this include: dust control techniques such as wetting surfaces, using barriers to prevent suspension of dust in the air, avoiding rapid actions which may stir up dust, and use of electrically or hydraulically powered tools or hand tools rather than pneumatically operated tools. In cases where dust generation is unavoidable, isolating the source and providing HEPA exhaust ventilation to the area will capture most of the dust generated. If none of these methods are totally successful respiratory protection (respirators) will be used to prevent inhalation of airborne radioactive material.
- *Contamination Control*—Finally, the role of contamination control in preventing ingestion or inhalation of radioactive material is essential. Contaminated skin, clothing or equipment are not hazards in their own right, however, the contamination may be transferred to food, drink or eating utensils and be ingested. Also, the contamination on skin, clothing or equipment may be re-suspended and inhaled by workers who are outside of the RCA. The techniques of using PPE, bagging or wrapping equipment and tools, and performing contamination surveys of all personnel, equipment, tools or any other items leaving the RCA is essential for contamination control.

5. SURVEYS AND MONITORING

A. SURFACE CONTAMINATION SURVEYS

This section describes several different surface contamination survey techniques that may be used during the remediation of IRP Site No. RW-41.

INDIRECT SURVEY METHODS

These methods measure removable contamination. The indirect survey techniques (smear and wipe) are as follows:

- *Smear Surveys*—A smear is obtained by using an absorbent filter disk to wipe with moderate pressure, across the area/item to be surveyed. The smear is to cover an area of approximately 100 square centimeter (cm^2). The smear is then counted by using either laboratory counting equipment or a ratemeter with an appropriate detector probe, using reproducible geometry.

- *Wipe Surveys*—A wipe (also called a large area wipe or smear) is obtained by wiping an absorbent pad (e.g., Maislin) or paper towel over a larger area or the entire surface, if practical. The wipe is then counted using a ratemeter with an appropriate detector.

DIRECT SURVEY METHODS

These methods measure both fixed and removable levels of surface contamination. The direct frisk is performed by scanning the survey location using an instrument, such as a ratemeter, equipped with an appropriate alpha, beta or gamma-sensitive detector.

Surface contamination surveys are performed by the Radiation Safety Technicians to assess surface contamination to aid in controlling the spread of radioactive contamination to unrestricted or less contaminated surfaces.

PRECAUTIONS/LIMITATIONS

Survey techniques used to monitor personnel contamination are presented in Section 15.0. of the site Radiation Protection Plan.

Documentation and record keeping requirements for surface contamination surveys are discussed in Section 19.0. of the Site Radiation Protection Plan.

Acceptable surface contamination survey limits for the release of area, items, or materials are presented in Table 12.1. of the Site Radiation Protection Plan.

GENERAL GUIDELINES AND REQUIREMENTS

Personnel performing surface contamination surveys shall take necessary precautions to minimize the possibility of cross contamination (e.g., changing gloves after handling highly contaminated surface, wrapping instruments in plastic sheeting or other suitable material).

Detector windows of α or β instrumentation must remain uncovered.

Before entering an area to perform a surface contamination survey, personnel must be aware of anticipated contamination levels. A review of previous surveys and operations performed in the area since the last survey should be made to determine the expected radiation and surface contamination levels. When high levels of surface contamination are expected, start the survey at the periphery of the area and proceed toward the point suspected of having high levels of contamination. To minimize the spread of high levels, change shoe covers before leaving the highly contaminated area, if practical.

When low background, sufficient sensitivity, accessibility, surface geometry permit, a direct scan using a AC-3-7 probe with an appropriate compatible count rate meter for alpha surveys should be performed in accordance with Section 12.5. Any portable count rate instrument used for contamination surveys should have the capability of providing an audible response for the observed count rate.

If background levels, surface geometry, large area to be surveyed (e.g., floors or walls) do not permit a direct frisk, a smear survey in accordance with Section 12.6 or a wipe survey in accordance with Section 12.7 should be performed. When background radiation levels permit, smears/wipes may be counted on the spot with the rate meter for alpha surveys. Where background levels do not permit on-the-spot counting, the smears/wipes shall be taken to a low background area within a radiological control area for counting. Care must be taken to ensure that the smear/wipe are counted such that no spread of loose surface contamination takes place as a result of the counting process.

Any surface contamination found in non-contaminated areas shall have the area immediately secured and further surveys made in the vicinity to determine the extent of the activity. The survey data shall be reported to the Site RSO for evaluation.

Major changes in loose surface contamination in known contaminated areas should be reported to the Site RSO.

Smears/wipes in known highly contaminated areas that do not serve a specific purpose should not be taken.

B. PERFORMANCE OF DIRECT FRISKS

Direct frisks shall be performed as follows:

- Observe general requirements addressed in Section 12.4.
- Taking precautions to ensure the instrument/probe does not come in contact with the surveyed surface, hold the probe within 1/8-inch of the surface.
- Scan the entire surface of the survey point or item at a slow rate. (Approximately two inches per second.)
- If the direct scan indicates levels greater than the allowable limits shown on the table in Section 12.3, perform a smear or wipe survey as indicated in Section 12.6 or 12.7 as applicable. Control the area as a contamination area if the smear/wipes indicate levels greater than the allowable limits shown in Table 12.1.
- Determine activity levels in accordance with Section 12.8.
- Document results in accordance with Section 19.0.

C. SMEAR SURVEY TECHNIQUE (INDIRECT METHOD)

Where background levels or surface geometry prohibit direct scans for surface contamination, perform a smear survey as follows:

- Observe the general requirements addressed in Section 12.4.
- Obtain a survey form to map the locations of the smear samples.
- Using moderate pressure, wipe an area approximately 100 cm^2 with the smear.

- Count the smear samples in accordance with Section 12.8.
- Determine activity levels in accordance with Section 12.8.
- Document the results in accordance with Section 19.0.

D. WIPE SURVEY TECHNIQUE (INDIRECT METHOD)

Where surface geometry, etc., prohibit direct frisks of an area and/or it is desired to survey large areas, perform a wipe survey as follows:

- Observe the general requirements listed in Section 12.4.
- Wipe the surface to be surveyed with cloth or paper wipe. Maislin or other oil-impregnated materials should not be used.
- Count samples in accordance with Section 12.8.
- Determine the activity levels in accordance with Section 12.8.
- Document the results in accordance with Section 19.0.

E. COUNTING SMEAR/WIPE SAMPLES

Prior to handling smear/wipe samples, take the necessary precautions to prevent cross-contamination.

Scan the samples with a pancake G-M detector or equivalent prior to continuing.

Determine the β/γ activity level as follows:

$$\beta/\gamma \text{ dpm/smear} = \frac{\text{cpm}_{\text{gross}} - \text{cpm}_{\text{bkg}}}{\text{detector efficiency}}$$

NOTE: A beta efficiency of 10 percent shall be used when using a pancake G-M detector in the hand-held geometry.

Smears shall be counted in a laboratory alpha counting system for sufficient time to ensure a lower level of detection equal to 10 percent or less of the appropriate contamination standards.

Determine the alpha (α) activity level as follows:

$$\alpha \text{ dpm/smear} = \frac{\text{counts}_{\text{gross}} - \text{counts}_{\text{bkg}}}{E \cdot T \cdot SAF}$$

Where: E = detector efficiency

T = counting time (minutes)

SAF = α Self-Absorption Factor (a factor that takes into account the self-absorption of α radiation caused by dirt on the smear—the SAF must be determined experimentally for the site.

NOTE: The detector efficiency is determined during or following the periodic instrument calibration and is checked daily.

Record the results.

If further analysis is not desired, dispose of samples appropriately.

F. EQUIPMENT

The equipment needed to perform surface contamination surveys include the following:

- Survey forms.
- Envelopes or plastic bags.
- Smear papers and Maislin cloths.
- Appropriate portable survey instruments.
- Laboratory counters.

6. MATERIAL CONTROL, TRANSPORTATION AND WASTE DISPOSAL

A. INTRODUCTION

SCOPE

This document applies only to the waste management, transportation, and disposal of depleted uranium (DU) at IRP Site No. RW-41 Test Area C-74L. Disposal of other radioactive contaminants, mixed waste, RCRA waste and other unwanted wastes are not authorized without approval of the Eglin LLRM Partnering Team.

PURPOSE

The purpose of this document is to establish procedures for the safe on-site management, transportation and disposal of low level radioactive waste and other unwanted materials by Earth Tech and its subcontractors from IRP Site No. RW-41 Test Area C-74L.

APPLICABILITY

This document is applicable to all material and waste management activities conducted at IRP Site No. RW-41 Test Area C-74L.

B. MATERIAL AND WASTE SOURCES AND CLASSIFICATIONS

This section describes the sources and classifications of the materials and wastes that may be generated during the remedial activities to be performed at the remediation site.

1) SOURCES OF MATERIALS AND WASTES

Implementation of the remedial action activities at the site will result in the generation of materials and wastes that will require appropriate on-site and off-site management. These materials and wastes will be generated during the excavation of radiologically contaminated soil, and during the execution of associated support operations (for example, equipment decontamination and sampling). The anticipated materials and wastes to be generated from the implementation of the remedial action activities include, but are not limited to the following:

SOURCE	**POTENTIAL MATERIALS AND WASTE**
Contaminated Soil Excavation Activities	Radiologically (DU) contaminated soils, concrete, PPE, debris, and disposable equipment.
Decontamination activities	Decontamination water, used PPE, debris, disposable equipment
Other	Common trash and garbage, waste oil (from filters, equipment maintenance).

PPE = Personal Protective Equipment

2) MATERIAL AND WASTE CLASSIFICATIONS

The primary material to be handled during the remedial action activities will be Low Specific Activity (LSA) material, SCO and non-regulated material. The potential for encountering hazardous waste is essentially non-existent. Analysis results for material and waste management characterization will be recorded on the appropriate tracking log. The potential for encountering unexploded ordinance of some type does exist, since this is still an active munitions test range.

a) Radiological Materials and Waste

The primary contaminant is DU, and its associated decay products. The random deposition of the DU fragments in the soil makes it virtually impossible to determine how contaminated the removed soil is. Therefore all the soil removed from the excavation will be considered contaminated and designated as LSA soils. The possibility of encountering mixed wastes is extremely remote.

b) RCRA Hazardous Materials and Waste

There are no known RCRA Hazardous Materials or Waste located on IRP Site No. RW-41.

c) Mixed Materials and Waste

Mixed wastes are defined by the Low Level Radioactive Waste Policy Act, Public Law 96-573; this includes radioactive material not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or by-product material as defined by Section 11.e(2) of the Atomic Energy Act, and contains hazardous waste that is either listed as a hazardous waste in Subpart D of 40 CFR 261, or hazardous waste which also contains naturally occurring radioactive materials. Encountering mixed waste is not anticipated at the site although care should be taken not to create any mixed waste. Hazardous materials will not be taken into the RCA without approval of the Site RSO.

d) Unregulated Materials and Waste

In addition to the waste classifications identified above, remedial action activities may also result in the generation of waste materials that are not classified as radiological material or RCRA hazardous wastes but may contain hazardous substances requiring special management procedures (regulated wastes). Such regulated wastes are not anticipated at the site.

e) Other Materials and Wastes

A variety of non-hazardous materials will likely be generated during remedial action activities that will also require proper management. These materials may include the following:

1. Construction Debris. This category includes items that have not been contaminated by radiological materials or otherwise impacted by hazardous substances or wastes. Clean debris may not be used as on-site fill material and must be disposed of at a licensed off-site construction debris landfill.
2. Trash and Rubbish. This material includes spent packaging materials, equipment, and general garbage and trash that has not been impacted by radioactive or hazardous substances. Trash and rubbish will be stored on-site in appropriate containers and will be disposed of at a licensed, off-site municipal waste facility. A licensed local municipal waste hauler will transport this material to a municipal landfill.

C. REGULATORY REQUIREMENTS

The following prerequisites must be met prior to any individual shipping or assisting in the shipment of waste by any conveyance on the public highway, by vessel or rail, or by air.

Training Requirements

The person performing activities associated with the shipment of hazardous, radioactive, or mixed waste shall be properly trained in accordance with the requirements of 49 CFR 172, Subpart H and will meet all qualification requirements of U.S. Army Industrial Operations Command, "Shipping Procedures for Unwanted Radioactive Materials".

Broker Administrative Requirements

The Broker performing shipments from Test Area C-74L remediation project shall ensure that the following administrative requirements are addressed prior to any shipment of materials or wastes off-site:

- The primary contaminant is depleted uranium, and its associated decay products. The random deposition of the DU fragments in the soil makes it virtually impossible to determine how contaminated the removed soil is. Therefore, all the soil removed from the excavation will be considered contaminated and designated as LSA soils. The possibility of encountering mixed wastes is extremely remote. All materials being shipped from the project site must be identified by the most appropriate proper shipping name in accordance with the Hazardous Materials Tables of 49 CFR 172. Radioactive and mixed wastes being shipped for disposal or shipped to a collector or processor for eventual disposal must also be classified in accordance with 10 CFR 61 or a valid disposal site license, as appropriate.
- If hazardous or mixed waste materials are generated on-site, these waste materials must be identified by the most appropriate EPA Waste Code in accordance with 40 CFR.
- For any hazardous or mixed wastes generated on-site, all notifications and certifications for waste material subject to the land disposal restrictions must be completed in accordance with 40 CFR 268.

Disposal Facility Waste Disposal Criteria

The Broker shall ensure that all shipments of waste for treatment or disposal are prepared and shipped in compliance with the receiving facility's waste acceptance criteria and site radioactive material license.

Hazardous Material Shipment

Hazardous materials not shipped as waste shall be shipped in such a manner as to conform to all federal, state and local ordinances. Material safety data sheets for hazardous materials to be shipped shall be reviewed, if available, prior to any shipping related activities. Non-waste radioactive materials shall only be shipped to a facility upon provision of evidence, such as a valid USNRC license, that the material is acceptable at the receiving facility.

D. TOOLS, MATERIALS, AND EQUIPMENT

The Air Force Radioactive Material Broker will ensure that all tools, administrative forms (including manifests), survey instruments, labels, markings, and placards are available for each shipment of materials. Special care must be taken by the Broker to ensure that an adequate supply of such materials is maintained. Broker may not be on site during actual remediation project.

E. MATERIAL AND WASTE MANAGEMENT PROCEDURES

This section presents the specific guidelines and procedures that will be followed for the management of material and wastes handled during the remedial action activities at the site. These procedures are generally applicable to the management of wastes after they have been excavated and removed. Specific procedures for material and waste excavation and removal will be available and utilized by the remediation subcontractor. The procedures presented in this section are based on the project goals of minimizing threats to site workers, human health, and the environment during all material and waste handling activities. Specific procedures and guidelines for visually characterizing, segregating, handling, staging, storing, sampling, packaging, labeling, and transporting material and waste are presented in the following sections.

1) GENERAL

Material and waste handling activities will be performed in a manner that minimizes the threat of a release of potentially contaminated material to the environment and surrounding community, and protects worker health and safety. Care will be taken during operations and activities that will generate materials and wastes, such as excavation, demolition, and dewatering, to prevent releases of material, waste, and dust to the surrounding environment.

2) MATERIAL AND WASTE HANDLING PRECAUTIONARY MEASURES

The following steps may be implemented prior to or during remedial activities to ensure that there are no releases of material and/or waste to the environment and surrounding community, and to protect site workers.

- Engineering controls such as water sprays may be used during activities that could potentially generate dust (i.e., excavation and loading) to prevent the spread of contaminants via wind dispersion.
- Plastic sheeting may be placed under and around containers while they are being loaded. Any material that falls onto the plastic sheeting during loading will be collected and placed in the container.
- Site workers will wear PPE appropriate for the specific task being performed, in accordance with the Site Safety and Health Plan. Used PPE and contaminated disposable equipment and materials will be containerized and disposed of appropriately.

- Equipment used during construction and remediation activities in potentially contaminated areas will be properly decontaminated before moving through clean areas of the site or leaving the site.
- Loading areas will be surveyed for radiation and contamination periodically during excavation and at the end of the project to verify that these areas have not been contaminated during material and waste handling operations.

3) PRE-EXCAVATION SCREENING

All materials and wastes to be excavated will be screened using appropriate radiation detection equipment. In addition, a FIDLER walk-over survey will be conducted to define the perimeter of the excavation.

F. MATERIAL AND WASTE HANDLING, STAGING, AND STORAGE

The guidelines that will be used for handling, staging, and storage of waste materials generated during the remediation activities at the site are presented below:

1) DU CONTAMINATED SOILS

As DU contaminated soils are excavated, they will be transferred to the designated staging area. This location will be within the exclusion zone. DU contaminated soils may also be loaded directly into appropriate disposal containers at the excavation area. Filled containers can be transferred to the staging area using a forklift.

2) MATERIALS WITH FREE LIQUIDS

Materials transported for disposal must not contain any free liquids. Due to the sandy soils and location of the water table it is not anticipated that there will be any free liquid in the excavated soil. In the event that rainfall occurs during the remediation activities, field work will be stopped until the soil has dried sufficiently to prevent moisture from accumulating in the disposal container.

3) NON-DU CONTAMINATED SOIL STAGING ARE

In the event that non-DU contaminated soils are removed from the excavation area, clean soil staging areas will be constructed within the exclusion zone as close as possible to the areas where remedial activities will occur.

4) CONTAMINATED SOIL SCREENING/ANALYSIS PROCEDURE

DU contaminated soils shall be packaged in appropriate containers at the stockpile area. Filled containers will be loaded on to flat-bed trucks for transportation to the DU waste storage area at Test Area C-64..

5) VEHICLE RELEASE - EXCLUSION ZONE

All trucks, excluding those dedicated to the excavation site, will remain on the "contaminant-free" haul road. All vehicles leaving the exclusion zone will be scanned, and, if leaving the site property, tested for removable contamination prior to release. If contamination is discovered, the vehicles will be decontaminated at the equipment decontamination facility and scanned again, prior to release.

6) SOIL MANAGEMENT GUIDELINE

All hazardous materials and wastes generated at the remediation site will be stored in compliance with applicable state and federal law based on suspected or known contaminants using the following general guidelines for management of soils:

- The stockpiles of DU contaminated soils will be sampled as required to meet the characterization data quality objectives.
- Chemically incompatible materials shall not be stored together.
- DU contaminated soil stockpiles awaiting packaging or transport will be stored in a "Radioactive Materials Area" will yellow and magenta rope barriers or other physical boundaries and postings as specified in 10 CFR.
- Hazardous or mixed wastes, if identified, will be properly labeled upon their accumulation within a storage or disposal container. Labeling and storage requirements will be as specified in 40 CFR for interim storage of hazardous wastes. No hazardous waste will remain in storage for greater than 90 days after accumulation begins. It is not anticipated that any hazardous or mixed waste will be generated at the remediation site.

G. MATERIAL AND WASTE SAMPLING AND ANALYSIS

Historical analytical results will be used to prepare a profile for the materials to be shipped to the disposal facility specified by the Air Force. Additional samples of excavated materials will be collected as needed during the remedial activities for disposal classifications. The procedures for sampling excavated materials will be determined by the Site RSO in conjunction with the Site Certified Broker, if available.

H. MATERIAL AND WASTE PACKAGING

All material and waste scheduled for off-site transportation and disposal will be properly packaged in accordance with all applicable local, State and Federal regulations, including DOT Hazardous Materials Regulations contained in 49 CFR Parts 171 through 180. Materials scheduled for shipment out of state will be packaged in containers suitable for shipment by rail. All other material and wastes to be shipped off-site will be packaged in appropriate containers such as lined roll-off boxes or 55-gallon drums. However, the packaging selected must be authorized for the specific material being shipped. The following sections describe the minimum packaging requirements for material to be shipped.

NON-BULK PACKAGING (CONTAINERS)

Non-bulk packaging must, at a minimum, meet the applicable requirements contained in 49 CFR 173.24, *General Requirements for Packaging and Packages*.

- Containers must be properly sealed to prevent load movement from "pumping" dust-laden air out of the container.
- Containers must be clean. They must not have any waste materials, or other material that could be mistaken for waste material, on the outer surface.
- Containers in a shipment must be properly loaded, blocked, and braced securely to prevent shifting and damage during transport. Shippers should examine the specific transport loading requirements contained in 49 CFR 174 for rail and 49 CFR 177 for highway.
- Over-packed containers will be used only when necessary to meet DOT requirements for shipment. Drums may be shipped on pallets with prior approval from the Site Transportation and Disposal Coordinator. However, the pallets must be strong enough to withstand collapse during transit. The drums should be securely banded to the pallet. Drums may not be stacked.
- Moving van trailers may not be used.
- Each container that requires marking must be properly marked in accordance with the requirements of 49 CFR 172 Subpart D and/or 49 CFR 173.421 and 425.
- Each container that requires labeling must be properly labeled in accordance with the requirements of 49 CFR 172 Subpart E.

3) PACKAGING AND PREPARATION OF MATERIALS

All packaging and preparation of materials for transport from the remediation site shall be in strict adherence to the requirements of 49 CFR, *Transportation*, and all other applicable federal, state, local and disposal site regulations.

4) PACKAGE INSPECTION

Materials shall be packaged and the packaging inspected in accordance with the requirements of 49 CFR 173 for the proper shipping name and DOT subtype of the material being offered for transport. Any required Type B and NRC-approved Type A packages shall be prepared in accordance with the applicable Certificate of Compliance.

5) SHIPMENT AS NON-HAZARDOUS MATERIALS

If the waste is not identified as contaminated with DU or no other DOT hazard is identified, the materials may be shipped without regard to the Hazardous Materials Regulations contained in 49 CFR.

6) CLASS 7 HAZARDOUS MATERIAL

DU contaminated waste will be shipped as Class 7 hazardous material. Based on preliminary analysis of these materials, the appropriate proper shipping name will be "Radioactive Material, Low Specific Activity" or "Radioactive Material, Surface Contaminated Object." These materials will be packaged in appropriate disposal containers such as collapsible, strong-tight intermodal containers.

7) OTHER HAZARDOUS OR MIXED WASTE

Any unwanted materials or wastes found to be a hazardous or mixed waste based on characterization data, will be packaged either as Class 7 (Radioactive-LSA or SCO) materials or Class 9 (Hazardous Waste, Solid) materials depending upon its radiological constituents. The materials will also be packaged in the same type container as mentioned above in paragraph 4.6.6.

8) PACKAGE MARKINGS AND LABELING

All packages offered for transport shall be properly marked and labeled in accordance with the requirements of 49 CFR 172 prior to shipment.

9) SHIPPING PAPER PREPARATION (INFORMATION ONLY)

Shipping papers will be prepared for shipments as follows:

- All hazardous materials (unless otherwise excepted) shall have DOT hazardous materials shipping papers prepared in accordance with 49 CFR 172.200 through 172.205.
- All hazardous and mixed waste shall, in addition to USDOT hazardous materials shipping papers, shall have a Uniform Hazardous Waste Manifest selected and prepared in accordance with 40 CFR 262.20.
- All radioactive waste shall have a NRC Uniform Radioactive Waste Manifest prepared in accordance with the requirements of 10 CFR 20.311.
- Additional forms shall be prepared as may be required by federal, state, and local ordinances, and by receiving site license or acceptance criteria.

I. PROCEDURES FOR MATERIAL LOADING

With the exception of common carrier shipments of hazardous materials (non-waste shipments), the following procedure shall be followed when loading material for transportation.

1) VEHICLE VISUAL INSPECTION

Conduct and document a visual inspection of the vehicle and ensure any discrepancies are repaired prior to loading. This inspection shall include all vehicle safety devices, tires, brakes, and trailer as applicable.

2) RADIATION AND CONTAMINATION SURVEY

For radioactive and mixed waste, a radiation and contamination survey of the vehicle will be performed and documented prior to loading. Compare the survey results to the requirements of 49 CFR. Vehicles with contamination levels greater than 10% of the DOT limits shall not be loaded.

3) BROKER INSPECTION (INFORMATION ONLY)

The Broker shall inspect all packages as they are loaded to ensure that the packages are in compliance with all the requirements set forth in this procedure. Incompatible materials shall be segregated as required by 49 CFR.

NOTE: Special care shall be taken to ensure that all strong tight containers used for radioactive material transport are completely sealed to the maximum extent practical. This includes the use of sealant on seams of metal boxes. Special care shall also be taken to ensure that all specification packages are properly prepared for transport and in good condition prior to transport.

4) LOAD VERIFICATION

Upon completion of loading, visually verify that all disposal containers are loaded.

J. POST LOADING REQUIREMENTS

1) SHIPPING PAPERS

Have the driver (or transporter's representative) and shipper (or shipper's agent) sign all required forms including the exclusive use instructions. Review all paperwork to ensure legibility. Copy and distribute paperwork in accordance with the Paperwork Distribution Checklist specified in U.S. Army Industrial Operations Command, *Shipping Procedures for Unwanted Radioactive Materials, Appendix H (need date).* Uniform Hazardous Waste Manifests shall be distributed in accordance with 40 CFR 262 and as required by the laws of the generating state.

2) SPECIAL INSTRUCTIONS

Verify that the driver (transporter's representative) understands all special instructions such as the maintenance of exclusive use and prior notification requirements. The shipment may now be released for transport.

3) SHIPMENT NOTIFICATIONS

Make any required prior notification of correction telephone calls. Mail copies of the Radioactive Shipment Manifest (RSM) cover sheets to the disposal site for radioactive waste shipments, if applicable.

K. MATERIAL AND WASTE LABELING AND DATING

Material and waste containers and packages will be marked in accordance with applicable local, state and federal requirements (49 CFR 172 Subpart D). In addition, a unique identification number will be assigned to each container used for material storage to allow for proper tracking of the material from the time of shipment through off-site disposal and receipt of a certificate of disposal, if applicable. Containers will also be labeled to indicate the type of material they contain, the date of shipment, and the area from which the material originated. The information will be recorded on a Material and Waste Container Management Data Sheet. The material and waste management database will be periodically reviewed to ensure that no materials are stored on-site while awaiting shipment for a period of time longer than allowed by applicable waste accumulation regulations (90 days for hazardous wastes).

L. MATERIAL AND WASTE TRANSPORTATION AND DISPOSAL

This section describes the steps for transportation and disposal of material and waste during remedial activities. Material and waste disposal summary will be contained in the appropriate tracking log.

1) RADIOACTIVE MATERIAL DISPOSAL

All DU contaminated materials excavated during the remedial activities will be transported to an appropriate, licensed and permitted facility. The radioactive wastes will be disposed of at the Low Level Radioactive Material Facility near Anderson, Texas.

2) RCRA HAZARDOUS WASTE DISPOSAL

If RCRA hazardous wastes are identified during the remedial activities, a licensed hazardous waste hauler will transport them to a permitted hazardous waste disposal facility. Each shipment of waste must be accompanied by a uniform hazardous waste manifest and must be labeled, marked, and placarded in accordance with DOT regulations.

3) REGULATED WASTE DISPOSAL

Regulated wastes, which are not classified as radioactive, RCRA hazardous, or mixed wastes, will be transported to an appropriate permitted/licensed disposal facility by a hauler that is licensed to carry the specific type of regulated waste. Materials must be packaged, marked, labeled, and placarded in accordance with DOT regulations. Approval by the Industrial Operations Command is required prior to disposal of these regulated materials and wastes.

4) NON-CONTAMINATED WASTE DISPOSAL

A licensed hauler will transport non-contaminated wastes to a permitted municipal waste or construction and demolition debris landfill.

M. RECORDS

1) BROKER RECORD REQUIREMENTS (INFORMATION ONLY)

The broker shall retain copies of records, forms, and shipping papers generated as a result of this procedure until written acknowledgment is received from the consignee for all waste shipments or telephone acknowledgment is received for all non-waste shipments. Notifications and reports shall be in accordance with US Army Industrial Operations Command, Shipping Procedures for Unwanted Radioactive Materials (need pub. date).

2) SHIPPING PAPER REQUIREMENTS (INFORMATION ONLY)

Copies of shipping papers shall be maintained by the site QAO and kept in the on-site project file for the duration of the project, and will be returned to Earth Tech's Fort Walton Beach office at the completion of the remediation activities.

N. AUDIT PROGRAM

The remediation operation planned at IRP Site No. 41 Test Area C-74L is not expected to require more than 90 days to complete from mobilization to demobilization and shipment of all waste containers to the disposal facility. An audit program to validate the site radiation protection plan is not required if the remediation activity will last less than one year.

HEALTH AND SAFETY PLAN – ITEM 11B

1. SITE SAFETY AND HEALTH PLAN (SSHP)

A SSHP will be developed as and provided to the USAF for comment and acceptance. The plan will include all pertinent aspects of USACE EM 385-1-80, EM 385-1-1, ER 385-1-92, and ER 385-1-80 and will address installation and AF specific safety issues.

SPECIFICALLY THE PLAN WILL ADDRESS THE FOLLOWING:

1) Site Description and Contaminants
2) All Hazard and Risk Analysis (UXO, Radiological, physical, etc.)
3) Staff Organization, Responsibilities, and Qualifications
4) Training
5) PPE
6) Medical Surveillance
7) Radiation Dosimetry
8) Exposure Monitoring
9) Heat and Cold Stress
10) SOPs and Work Practices
11) Site Control Measures
12) Personal Hygiene and Decontamination
13) Equipment Decontamination
14) Emergency Equipment and First Aid
15) Emergency Response
16) Accident Prevention
17) Record Keeping, Logs
18) Reporting