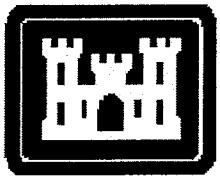
ADMINISTRATIVE RECORD
FOR THE MADISON SITE
MADISON, ILLINOIS

Remedial Investigation Documentation and Feasibility Study-

Final Remedial Investigation Report, and Feasibility Study for the Madison Site



US Army Corps
of Engineers
St. Louis District

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ABBREVIATIONS AND ACRONYMS

AEC	U.S. Atomic Energy Commission
CEDE	committed effective dose equivalent
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
cpm	counts per minute
$DCGL_w$	Derived Concentration Guideline Limit
DOE	U.S. Department of Energy
dpm	disintegrations per minute
DQO	Data Quality Objectives
EPA	Environmental Protection Agency
FUSRAP	Formerly Utilized Sites Remedial Action Program
H_0	Null Hypothesis
IAEA	International Atomic Energy Agency
IDNS	Illinois Department of Nuclear Safety
K-40	Potassium-40
LBGR	Lower Bound of Gray Region
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
NaI	sodium iodide
ORNL	Oak Ridge National Laboratory
pCi/g	picoCuries per gram
RPD	Relative Percent Difference
SAIC	Science Applications International Corporation
TEDE	total effective dose equivalent
Th-232	thorium-232
U-234	uranium-234
U-235	uranium-235
U-238	uranium-238
UCL	Upper Confidence Level
µrem/h	microrem/hour
USACE	U.S. Army Corps of Engineers
WRS	Wilcoxon Rank Sum

EXECUTIVE SUMMARY

During the late 1950s and early 1960s, uranium metal rods were extruded and straightened at the Dow Chemical Company facility located at College and Weaver Streets in Madison, Illinois, as part of a program conducted by the U.S. Atomic Energy Commission. A radiological scoping survey, performed in 1989, identified low concentrations of residual uranium contamination in Buildings 6 and 4 of the facility. The survey report concluded that the radioactive material did not pose a potential for significant radiation exposure to facility occupants, but further investigation to better define the extent of contamination was recommended.

A more detailed radiological survey of the facility was performed in 1998 under the U.S. Army Corps of Engineers' Formerly Utilized Sites Remedial Action Program (FUSRAP). The objective of this survey was to determine the current conditions of the site (presently occupied by Spectrulite Consortium, Inc.) and compare these conditions with appropriate radiological guidelines. The survey was designed using the guidance of the *Multi-Agency Radiation Survey and Site Investigation Manual* (MARSSIM). It consisted primarily of measurements in the vicinity of the extrusion press and straightening table to determine uranium activity levels on floors and walls; uranium levels on equipment surfaces; and concentrations of uranium in dust accumulated on overhead building surfaces. In addition, direct radiation levels and uranium contamination were measured at Building 6 and 4 exit and entrance locations, on the roof above the extrusion press, and on other surfaces in Buildings 6 and 4.

The survey identified detectable uranium in dust on overhead surfaces, with the highest concentrations occurring directly above the extrusion press. The evaluation of the concentrations detected demonstrated that the potential radiological risk to current production workers, posed by this residual uranium, is within the range recommended by the Environmental Protection Agency. However, based on assumed use, the evaluation found that utility workers conducting activities in close proximity to overhead structures could experience unacceptable exposure from the contaminated surfaces. In addition, average dust concentrations on overhead surfaces exceed the State of Illinois total activity surface contamination guideline of 1,000 dpm/100 cm^2 as well as the total activity guideline established in NRC Regulatory Guide 1.86 for natural uranium (i.e. 5,000 dpm/100 cm^2).

Although floor surfaces in the vicinity of the extrusion press did not have surface activity exceeding acceptable levels, collection of additional data will be necessary after remediation to fully quantify surface activity and perform statistical tests to demonstrate compliance with the Record of Decision for the site.

A small area of elevated direct radiation was noted outside one of the doors. Although the source of this radiation was not identified by the measurements conducted during the initial survey, a further investigation found that this contamination results from magnesium–thorium operations rather than from AEC operations. Details are given in the RI Addendum, Appendix C.

The remainder of the survey results indicate that radiological conditions are comparable to background. Final evaluations of these areas will be conducted after the Record of Decision is signed for this site.

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1. INTRODUCTION

1.1 SITE HISTORY

During the late 1950s and early 1960s, the Mallinckrodt Chemical Company (Mallinckrodt) contracted with Dow Chemical Company [currently Spectrulite Consortium, Inc.] [located at College and Weaver streets in Madison, Illinois] to perform extrusions of uranium metal and straightening of extruded uranium rods for the U.S. Atomic Energy Commission (AEC) at their plant. This plant comprises the Madison Site. The work was conducted on an extrusion press and straightening table located near the southwest end of Building 6. Records suggest that the total quantity of uranium involved in these operations was small and indicate that Mallinckrodt retained accountability for the uranium throughout the operations. Mallinckrodt was also responsible for removing unused uranium material and cleaning up facilities following operations. Detailed records describing the previous cleanup have not been located.

Prior to 1998, the U.S. Department of Energy (DOE), successor to the AEC, performed evaluations of previously used sites, where a potential for residual radioactive contamination was considered to exist. These evaluations were performed as part of the Formerly Utilized Sites Remedial Action Program (FUSRAP). Because of the lack of documentation to establish the radiological status of the Madison Site, a preliminary radiological survey was conducted in March 1989 by Oak Ridge National Laboratory (ORNL), under contract to DOE (ORNL 1990). That survey identified low concentrations of uranium in dust on overhead structures, but concluded that this residual radioactive material did not pose a potential for significant radiation exposure to current building occupants. The report recommended further investigations to better define the extent of uranium contamination.

On March 15, 1957, Mallinckrodt entered into a subcontract with Dow Chemical Company's (Dow) Madison Division Office in Madison, Illinois. This subcontract, issued under Mallinckrodt's primary contract W-14-108-eng-8, was for Dow to perform "certain research and development work in gamma phase extrusion of uranium metal" at the Madison, Illinois plant. The objective of this research was to determine factors in the extrusion of uranium metal that would affect the final selection and purchase of tools and auxiliary supplies for use with an extrusion press that was planned to be located at another AEC production facility.

These factors included the properties of various die metals, the contour of the die cavity, the nature of the lubricant to apply to the uranium metal, the composition of the "follower block" (the material placed between the uranium metal and the ram press), and the speed at which the metal could be extruded.

Under the terms of the subcontract (Subcontract No. 25034-M), Mallinckrodt designed (for approval by Dow) dust arresting and other protective equipment. Mallinckrodt was also responsible for arranging for the Health and Safety Laboratory of the AEC to perform periodic surveys of breathing zone air quality. Mallinckrodt also retained responsibility for the accountability of the uranium metal during the work cycle.

In the Designation Summary for the Former Dow Chemical Company Site in Madison, Illinois, the Department of Energy (DOE) a successor to the AEC indicated that Dow also supplied materials (chemicals, induction equipment, and magnesium metal products) and services under purchase orders issued by Mallinckrodt. In March 1960, the Uranium Division of the Mallinckrodt Chemical Works issued a purchase order for Dow to straighten Mallinckrodt-supplied uranium rods. Two rod straightening campaigns were identified in the purchase order. One was to be completed in December 21, 1959, the second in January 25, 1960. Each campaign also included a cost for the cleanup of the area after each campaign. The actual periods of performance for this work, and the actual quantity of uranium that was processed is unknown. However, the total value of the purchase order and the unit cost identified with the "lot size" indicate that the quantity of metal involved was most likely small.

DOE indicated that no other operation or period of involvement with the processing or handling of radioactive material at the former Dow Madison plant was performed.

Records located in the Nuclear Regulatory Commission (NRC) public document room indicate that Dow applied for an AEC license on December 12, 1956. Dow was granted an AEC license (number C-2782), effective January 1, 1958, to receive and possess thorium metal and thorium compounds, without limitation as to quantity. The thorium, under the terms of the license, was for use in the preparation of magnesium alloys at the Dow plants in Midland, Michigan; Bay City, Michigan; Madison, Illinois; and Freeport, Texas. In 1962 Dow applied for and was granted another AEC license (number STB-527).

Dow Chemical Corporation leased the Madison plant to Phelps Dodge Aluminum Corporation in 1969. Consolidated Aluminum Corporation (Consolidated) assumed the lease in 1973 and exercised an option to buy the plant in 1973. Consolidated applied for and received a license (number STB-1097) from the NRC in August 1982. Consolidated manufactured magnesium thorium alloys at the Madison Site.

Consolidated sold the Madison plant to Barnes Acquisition, Inc. (which appears to have been a subsidiary of Spectrulite) in September 1986. In August 1986, W. A. Barnes requested that the NRC license that Consolidated had "relating to the manufacturing of magnesium thorium alloys and the storage of same be transferred to the surviving company". Apparently, NRC denied this request and Spectrulite applied for and was granted an NRC license (number STB-1488) in October 1986. The Spectrulite license was for the manufacture of magnesium based thorium alloys and listed the byproduct, source, and special nuclear material covered under the license as thorium (solid metal), thorium (Mg-Th hardener), and thorium (magnesium sludge).

1.2 RESULTS OF PREVIOUS RADIOLOGICAL SURVEYS

The March 1989 survey by ORNL concluded that most of Building 6 was free of residual radioactive material attributable to former AEC- or DOE-sponsored activities (ORNL 1990). Above-background levels of uranium were identified in dust on overhead surfaces above the general vicinity of the extrusion press. The maximum concentration of uranium-238 (U-238)

measured in this dust was 310 picocurie per gram (pCi/g). This quantity of U-238 is equivalent to a total uranium concentration of approximately 635 pCi/g, based on the activity abundances of uranium isotopes present in natural uranium (see Section 3.0). The ORNL report also concluded that the uranium dust on the overhead surfaces corresponded to a total surface contamination level in excess of the DOE limit for unrestricted release (i.e., 5,000 disintegrations per minute per 100 square centimeters (dpm/100 cm^2)) applicable at the time. In addition to the residual uranium contamination on the overhead surfaces, the survey also identified the presence of smaller amounts of thorium-232 in the facility. This thorium is from separate, licensed processes conducted by the current facility owners in other areas of the plant and is not of AEC origin. The thorium process was located in a separate part of the plant. No thorium was processed in Building 6 and no uranium was processed in the building housing the thorium operations. The facility owners have ceased operations involving natural thorium metal and are in the process of license termination with the Illinois Department of Nuclear Safety (IDNS).

The ORNL report recommended further investigations to better define the extent of uranium contamination in Building 6 and the adjacent Building 4. As a result of the survey findings, the plant was designated for inclusion in FUSRAP.

1.3 RESPONSIBILITIES

U.S. Army Corps of Engineers (USACE) is implementing a program for the management and remediation of radioactive contamination at Madison Site. In 1974, the U.S. Congress authorized the U.S. Atomic Energy Commission (AEC) to institute the Formerly Utilized Sites Remedial Action Program (FUSRAP). FUSRAP was initiated to identify and cleanup or otherwise control sites where residual radioactivity remains either from activities conducted under contract to the Manhattan Engineer District (MED) and AEC during the early years of the nation's atomic energy program, or from commercial operations as directed by Congress. On October 13, 1997 the U.S. Congress transferred responsibility for FUSRAP from the U.S. Department of Energy (DOE) to the USACE as part of the 1998 Energy and Water Development Appropriations Act. Actions conducted by the USACE at the Madison Site are subject to the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA).

The major objectives of FUSRAP are to:

- Find and evaluate sites that supported MED/AEC nuclear work and determine whether they need additional cleanup or control;
- Remediate or manage these sites so they meet current guidelines;
- Dispose of or stabilize radioactive material in a way that is safe for the public and the environment;
- Perform all work in compliance with appropriate federal laws and regulations and comply with state and local environmental laws and land use requirements; and
- Certify sites for appropriate future use.

The U.S. Army Corps of Engineers (USACE) is responsible for addressing AEC-related contamination from uranium extrusion and rod straightening operations at the Madison Site. In

1998, USACE performed further evaluation of the current radiological conditions of the Madison Site to develop recommendations for further actions, if warranted. The survey was conducted during the periods of June 29 through July 3, 1998, and November 10 through 11, 1998. Work was conducted in accordance with the Radiological Survey Work Plan and Radiological Survey Work Plan Addendum (USACE 1998a,b). Additional sampling and surveying were performed in July 1999 as described in the RI Addendum (Appendix C).

2. SITE DESCRIPTION

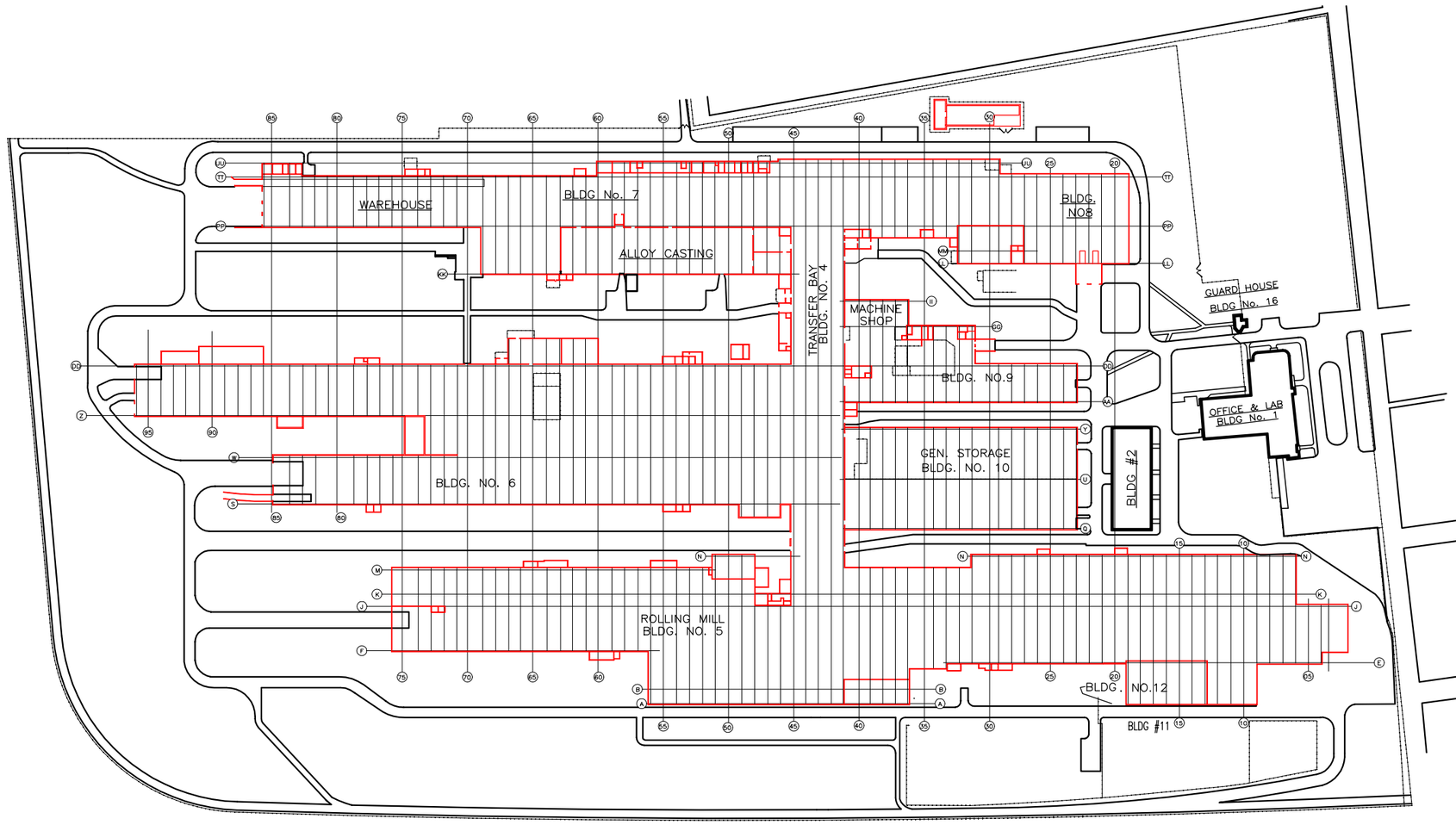
The Madison Site is located in an industrial area surrounded by residences, apartments, and other commercial enterprises. The site is near the Mississippi River, east of downtown St. Louis, in the state of Illinois.

The Madison Site consists of multiple large, interconnecting buildings (Figure 1). Building 6, where the uranium contamination was previously identified, is near the center of the complex. The southwest end of Building 6, where the uranium rod extrusion and straightening operations were performed, adjoins Building 4; there are no physical barriers between these two buildings. Building 6 is approximately 83 meters (275 ft) wide and 303 m (1,000 ft) in length. Ceilings of the main bays of Buildings 6 and 4 are approximately 14 m (46 ft) high with the ceilings reaching approximately 18 m (60 ft) at the highest point along the centerlines of the buildings. Basic structure support consists of steel columns on approximately 7.6-m [25-feet (ft)] centers, connected by large horizontal beams and multiple smaller vertical and horizontal cross members. Horizontal overhead surfaces are dust covered, thus limiting the sensitivity and accuracy of direct measurements of radioactivity on such surfaces.

Walls are concrete block with brick veneer. Floors are concrete; with rough, and pitted surfaces. Much of the floor in the vicinity of the extrusion press is covered with a thin layer of oily dirt and fine metal debris. There are no floor drains in Buildings 6 and 4, but there are multiple utility trenches, lubrication pits, equipment supports, and other penetrations into the floor. Machinery, feed materials, and product occupy a significant portion of the floor space.

Overhead cranes service most areas of Buildings 6 and 4 addressed by this survey. The main horizontal support beams are accessible from crane platforms in the center bay of Building 6. Limited access to higher structural surfaces is possible through windows in the upper ceiling areas. Access to other overhead surfaces, including the main horizontal beams in the two outer bays, requires a man-lift, but access above approximately 10 m is limited, due to the presence of support beams, smaller cross members, piping, electrical lines, cranes, and other obstructions. Also, positioning of a man-lift in the vicinity of the extrusion press is restricted by machinery and other materials in the area of interest in Buildings 6. Floor areas of Building 4, near the intersection with Building 6, are more accessible than the floor areas of Building 6.

The buildings have multiple access points for personnel and equipment. There are flat roof surfaces in the vicinity of the ceiling windows. These roofs have drains, which discharge into a plant-wide storm runoff drainage system.



LEGEND:

- ROAD
- WALLS
- FENCE
- WALKWAY
- BUILDINGS
- COLUMN LINE

0 125 250 500

SCALE: 1" = 250'

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Madison Site
Remedial Investigation
Madison, Illinois

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Figure 1. Plan View of the Madison Site

The facility was operating three shifts during the survey, and accessibility for sampling and measurements was restricted.

3. CONTAMINANT OF POTENTIAL CONCERN AND ASSOCIATED GUIDELINES

The only contaminant of potential concern is processed natural uranium (i.e., uranium that has been separated from the other naturally occurring members of the uranium and actinium decay series). In natural uranium, the U-234, U-235, and U-238 isotopes are present in their naturally occurring ratios. The naturally occurring activity ratios of U-234/U-235/U-238 are 1.0/0.05/1.0, respectively.

The uranium is due to past operation by Dow and Mallinckrodt in support of the Atomic Energy Commission (AEC). Any other hazardous substances or pollutants or contaminants that may be present on any part of the Madison Site are beyond the scope of FUSRAP and can not be addressed under this program. The facility has also processed thorium under U.S. Nuclear Regulatory Commission (NRC) and, more recently, State of Illinois licenses. Several items containing thorium and small amounts of thorium in dust samples were noted in Building 6 by the ORNL survey; however, the presence of thorium in such relatively small quantities was not expected to adversely impact this survey. This thorium is the result of processing of magnesium-thorium alloys and is beyond the scope of the FUSRAP program. Various standards and guidance values were used during the RI to assure that adequate data would be collected.

In the early 1980s, the State of Illinois became an agreement state pursuant to authority of the Atomic Energy Act. Illinois subsequently promulgated contamination standards. The State of Illinois surface contamination guideline (total activity) for alpha emitters is 1000 dpm/100 cm^2, average, over any one surface [known as the ($DCGL_W$) Derived Concentration Guideline Limit], and 5000 dpm/100 cm^2, maximum (IL 1995). The Illinois guideline for removable activity is 33 dpm/100 cm^2, average, over any one surface, and 100 dpm/ 100 cm^2, maximum. When compared to Nuclear Regulatory Commission Regulatory Guide 1.86, Table 1 – Acceptable Surface Contamination Levels, the State of Illinois guidelines are much more restrictive. The total activity guideline for natural uranium listed in Regulatory Guide 1.86 is 5,000 dpm/100 cm^2 averaged over one square meter and 15,000 dpm/100 cm^2 maximum, and the removable activity guideline for natural uranium is 1,000 dpm/100 cm^2 as a maximum value.

Because processed natural uranium emits both alpha and beta radiations at essentially the same rates, beta radiation measurements were primarily used in this survey to determine total surface activity levels of uranium. Alpha measurements were used to determine the removable activity levels because the detection sensitivity of the removable beta procedure (90 dpm/100 cm^2) was greater than the average guideline value.

The State of Illinois guidelines also state that the level of gamma radiation at 100 centimeters (cm) (1 m) from the surface shall not exceed background, the dose from alpha emitters shall not exceed 250 microrem/hr at 1 cm from the surface, and the dose from beta-gamma emitters shall not exceed 250 microrem/hr at 1 cm from the surface. Illinois has not

established a generic guideline concentration for processed natural uranium in soil or dust. The U.S. Environmental Protection Agency (EPA) CERCLA risk range of 10^{-4} to 10^{-6} was, therefore, selected for evaluation of uranium soil and dust concentrations.

4. SURVEY OBJECTIVES

4.1 GENERAL

The purpose of the survey was to evaluate the current radiological conditions of the Madison Site, attributable to previous AEC operations. Various standards and guidance values were used during the RI to assure that adequate data would be collected. With the exception of reference area surveys, measurements and sampling were limited to Buildings 4 and 6 of the site. Interior structure surfaces in the vicinity of the extrusion press were the primary focus of the survey. The survey also addressed other interior surfaces in Buildings 4 and 6, the exterior Building 6 roof, ground areas immediately outside doors of Buildings 4 and 6, surfaces of the extrusion press and adjacent equipment, and floor penetrations (pits, trenches, etc.) in the immediate vicinity of the extrusion press.

Guidance provided in the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (DOD 1997) was followed in the design, implementation, and data interpretation of this survey. The survey was designed to provide data for purposes of both scoping/characterization and, where appropriate, guideline comparison (also referred to as Final Status Survey). The survey design was based on information available at the time of its preparation. It was recognized that additional historic information on site operations, conditions encountered at the time of the survey implementation, and findings as the survey progressed would likely require modifications to the Survey Work Plan. Minor modifications were determined necessary and implemented with appropriate justification and documentation.

4.2 USE OF THE MARSSIM PROCESS

The MARSSIM process was developed collaboratively by the NRC, EPA, DOE, and U.S. Department of Defense, for use in designing, implementing, and evaluating radiological surveys. This process emphasizes the use of EPA's Data Quality Objectives (DQOs) and Data Quality Assessment processes, along with a sound quality assurance/quality control program. The "graded approach" concept is also used to ensure that survey efforts are maximized in those areas where there is the highest probability for residual contamination or greatest potential for adverse impacts of residual contamination. Implementation of this graded approach is incorporated throughout the MARSSIM process. Examples include the following:

- classification of survey units by contamination potential,
- limitation of survey unit size on the basis of contamination potential,
- fractional area coverage by scanning,
- data point selection method (systematic vs. judgmental), and

- provisions for increased sampling and measurement frequency when scan sensitivity is insufficient to detect areas of elevated activity.

The MARSSIM process also embodies flexibility to allow adaptation of guidance to be responsive to specific site situations and agency objectives and requirements.

The primary focus of MARSSIM is to demonstrate compliance of a site or facility with regulatory agency criteria for future use without radiological restrictions. This type of survey is known as a Final Status Survey, and MARSSIM provides highly prescriptive guidance for designing and conducting such a survey. On the basis of the previous ORNL survey, residual uranium in excess of the regulatory unrestricted release criteria was anticipated in portions of this facility. Because the potential for such residual activity in most areas of the plant were minimal, a Final Status Survey of the entire facility was not considered appropriate at this time. The category of survey for many portions of the Madison Site is most aptly described as a combination of scoping and characterization. MARSSIM provides limited specific direction for such a survey, instead emphasizing that the design of such a survey should be largely judgmental, and based on knowledge of the facility history and conditions and the intended use of the survey findings.

Some aspects of MARSSIM are intended for application with dose-based guideline levels of residual contamination, implemented by averaging over an entire "survey unit." This dose-based guideline approach also incorporates provisions for establishing additional limitations for small isolated areas of elevated concentrations of the contaminant. Current State of Illinois guideline limits for residual radioactive contamination in structures are not dose-based, thus limiting applications of MARSSIM to this survey. The survey design provided a level of thoroughness and technical soundness that equals or exceeds that of MARSSIM. Specific MARSSIM guidance used in the survey design and implementation includes the following:

- establishing DQOs,
- selecting decision error limits,
- determining data requirements,
- selecting survey techniques,
- identifying and investigating areas of elevated activity,
- reviewing and assessing data,
- selecting and conducting statistical tests, and
- evaluating overall performance relative to DQOs.

In addition to the guidance of MARSSIM, this survey embodied selected recommendations of draft NUREG/CR-5849, "Manual for Conducting Radiological Surveys in Support of License Termination" (NRC 1992). Specifically, the method and equation for determining sample data needs was used for the overhead beam area.

4.3 SCOPING/CHARACTERIZATION ACTIVITIES

One primary purpose of this survey was to scope and characterize the radiological status of the Madison Site. In accordance with MARSSIM guidance, the scoping/characterization was designed to

- identify those portions of the site that had uranium concentrations distinguishable from background and caused by previous AEC operations;
- determine the levels of residual uranium concentrations distinguishable from background and caused by previous AEC operations on surfaces and in affected media; and
- identify site conditions and potential pathways to enable evaluation of dose/risk to current and future site occupants and to evaluate alternatives for further actions, if appropriate.

In addition, the survey was designed to determine which areas of the site are not radiologically contaminated as a result of previous AEC operations and provide data to evaluate radiological conditions relative to the State of Illinois guidelines for unrestricted release.

Field survey activities associated with the scoping/characterization efforts consisted of the following:

- surface alpha* and beta scans to identify potential locations of uranium,
- surface gamma scans to identify potential locations of uranium,
- measurements of total alpha* and beta surface activity,
- measurements of removable alpha and beta activity,
- measurements of gamma exposure rates at 1 m above the surface,
- sampling of surface dust from overhead surfaces, and
- sampling of residues from the floor and floor penetrations.

*The initial survey design included alpha scans and measurements on some surfaces. Due to poor surface conditions, which would have prevented accurate alpha measurements, alpha scans and measurements were deleted from the scope of the survey. Beta and gamma scan rates were slowed, as appropriate, to ensure greater sensitivity and to compensate for poor surface conditions. Section 14 of the Work Plan describes the estimated detection sensitivities based on surface conditions.

In addition, information regarding the nature of the uranium deposits on overhead surfaces was obtained for assessing the potential health risk from this material.

4.4 FINAL STATUS SURVEY ACTIVITIES

The other primary purpose of the final status survey was to (1) determine the radiological conditions of those areas of the site, which, based on results of the earlier ORNL survey and the scoping/ characterization efforts of this survey (see Section 4.2), have not been affected by

previous AEC operations; and (2) to compare the levels in those areas with State of Illinois guidelines for unrestricted release. Portions of the site addressed by the Final Status Survey included interior structure surfaces other than those in the vicinity of the extrusion press, equipment surfaces, floor pits and penetrations, exterior roofs, and entrances/exits of Buildings 6 and 4. Field survey activities for Final Status Survey efforts consisted of the following:

- surface gamma scans to identify locations of elevated radiation levels,
- surface alpha* and beta scans to identify potential locations of uranium,
- measurements of gamma radiation levels at 1 m above the surface,
- measurements of total alpha* and beta surface activity,
- measurements of removable alpha and beta surface activity, and
- sampling of surface dust, soil, and residue.

*Initial survey design included alpha scans and measurements on some surfaces. Due to poor surface conditions, which would have prevented accurate alpha evaluations, alpha scans and measurements were not performed.

In addition, data from areas determined by the scoping/characterization survey as not significantly contaminated would be evaluated as part of the Final Status Survey activities.

4.5 STATISTICAL TESTS

For final status surveys where the contaminant is present in background, such as uranium, the Wilcoxon Rank Sum (WRS) test is used. The WRS test assumes the reference area and survey unit data distributions are similar except for a possible shift in the median values. When the data are severely skewed, the value for the mean difference may be above the $DCGL_W$, while the median difference is below the $DCGL_W$. In such cases, the survey unit does *not* meet the release criterion regardless of the result of the statistical test. On the other hand, if the difference between the largest survey unit measurement and the smallest reference area measurement is less than the $DCGL_W$, the WRS test will always show that the survey unit meets the release criterion.

As part of the DQO process, the null hypothesis (H_0) for demonstrating compliance of data with guidelines must be stated. The H_0 is that residual contamination exceeds the acceptance criterion (guideline). If the H_0 is accepted, the conditions of the area surveyed do not satisfy the guideline, and further evaluation and/or remediation is necessary. If the H_0 is rejected, the alternative hypothesis must be accepted and the finding of the evaluation is that the site satisfies the guideline.

4.6 RADIOLOGICAL SURVEY PARAMETERS

Natural uranium emits alpha, beta, and gamma radiations. Each of these radiation types can be used to estimate quantities of residual uranium activity; however, because of relative low abundances of gamma radiations and adverse conditions of many surfaces, beta radiation is expected to be the most reliable indicator of the actual total surface activity. Determination of total surface activity was, therefore, based on beta radiation measurements. Removable surface

activity was based on measurements of alpha and beta radiation. Dose rates were obtained for comparison with the IDNS standard for gamma radiation at 1 meter from the surface. Exposure rates [in microRad per hour (μR/hr)] were determined by converting NaI detector gross counts to exposure rate. These were used to identify elevated areas of gamma radiation at each elevated location when readings were taken. Survey methods and instrumentation are described in greater detail in Section 5.0.

Surface beta scans were recorded in count rate [counts/minute (cpm)]. Surface gamma scans were recorded in units of exposure rate (μR/h), based on conversion of instrument count rate to exposure rate, using calibration factors for the specific instrument. The primary purpose of the scans was to identify locations where direct radiation levels were elevated, thus suggesting possible radiological contamination. Relative instrument count rate was continuously monitored during scanning through use of the audible response signal from the instrument.

Total surface activity measurements were recorded in counts, integrated over a period of 1 minute, and converted to units of dpm/100 cm^2, using factors appropriate for the particular detector used. Removable surface activity (alpha and beta) was determined by analyzing smear samples, collected over a surface area of 100 cm^2, and converted to units of dpm/100 cm^2.

Dose rates were measured in units of μrem/h using instruments calibrated for dose-rate response. Exposure rates were recorded in units of exposure rate (μR/h), based on conversion of instrument count rate to exposure rate, using calibration factors for the specific instrument.

Concentrations of uranium in dust, residue, and other volumetric samples were measured by chemical separation of uranium, followed by alpha spectrometry. Concentrations of the U-238, U-234, and U-235 isotopes were determined and reported in units of pCi/g. Gamma spectrometry was also performed to identify and quantify radionuclides other than uranium.

4.7 DATA QUALITY OBJECTIVES

The primary survey objective was to identify and characterize those portions of the site that contain residual uranium attributable to previous AEC operations. Concentrations of uranium in dust samples and measurements of surface activity were used to identify affected locations and determine the levels and extent of contamination. In addition, the quantity and quality of data collected was intended to be adequate for use in demonstrating that the areas of the site that were not expected to contain residual radioactivity satisfy the guidelines for unrestricted release. To enable testing of data relative to guidelines, the USACE established acceptable decision errors for this project. The Type I (alpha) decision error used in data testing was 0.05. This provides a confidence level of 95% that the statistical tests will not incorrectly determine that a surveyed area satisfies criteria when, in fact, it does not. The Type II (beta) decision error was also 0.05. This provides a confidence level of 95% that the statistical tests will not incorrectly determine that a surveyed area does not satisfy criteria when, in fact, it does.

Data quality indicators for precision, accuracy, representativeness, completeness, and comparability were established. For this investigation these were established for the one analyte (total uranium) and the one method (alpha-spec).

- Precision is determined by comparison of replicate values from field measurements and sample analysis. The objective is a relative percent difference of 30% or less at 50% of the criterion value.

- Accuracy is the degree of agreement with the true or known. The objective for this parameter is +/- 30% at 50% of the criterion value.

- Representativeness and comparability do not have numeric values. Performance for these indicators was ensured through the selection and proper implementation of systematic sampling and measurement techniques.

- Completeness refers to the portion of the data that meet acceptance criteria and are, therefore, useable for statistical testing. The objective was 90% for this project.

In addition, radiological survey data were expected to meet industry standards for documentation.

5. SURVEY METHODS AND TECHNIQUES

5.1 CLASSIFICATION OF AREAS BY CONTAMINATION POTENTIAL

For the purposes of guiding the degree and nature of final status survey coverage, MARSSIM identifies three classifications of areas, according to contamination potential. Class 1 areas have a potential for contamination that exceeds guidelines; Class 2 areas have a potential for contamination, but it is unlikely that the contamination level exceeds the average $DCGL_w$; and Class 3 areas are not expected to contain residual activity in excess of background.

The ORNL report indicated that 310 pCi/g of U-238 in beam dust is equivalent to a surface activity level 13.6 times the DOE limit of 5,000 dpm/100 cm^2. Based on this relationship, the State of Illinois guideline of 1,000 dpm/100 cm^2 would be equivalent to approximately 4.5 pCi/g of U-238. The ORNL survey identified uranium in excess of 4.5 pCi/g on overhead surfaces of Building 6, between vertical support columns 45 and 54. Since this area exceeds established guidelines, a characterization survey was considered appropriate in lieu of a final status survey for the overhead horizontal beam surfaces from support column 57 of Building 6 and extending 50 ft into Building 4. For this reason, the overhead horizontal beam surfaces were not classified according to MARSSIM guidance.

The ORNL survey did not identify significant contamination in other areas of the facility. The maximum level reported was 15% of State of Illinois guidelines and the majority of measurements were below the procedure detection sensitivity. Based on this information, other

areas of the facility were classified using MARSSIM final status survey guidance. The classifications established for this survey are shown in Table 1.

Table 1. Area Classification for the Madison Site Survey

Area	Surface	Class	Comments
From Building 6, support column 57 extending 50 ft into Building 4	Surfaces above 25 ft	N/A	A classification of contamination potential is not applicable for the characterization survey
Building 6, between support column 45 and 54	Surfaces below 25 ft	2	Includes floors, equipment, and walls below 25 ft
Remaining areas of Buildings 6 and 4	All surfaces	3	None

The Class 2 area was divided into 3 survey units, ranging from 22.9 m (75 ft) to 30.5 m (100 ft) wide and extending from column 45 to column 54 (see Figure 2). These survey units, identified as Unit 1, Unit 2, and Unit 3, included floor surface areas of 2,090 m^2 (22,500 ft^2), 1,568 m^2 (16,875 ft^2), and 2,264 m^2 (24,375 ft^2), respectively. Although the total area of each of these units is greater than the MARSSIM-recommended limit for Class 2 areas of 1,000 m^2 (10,760 ft^2), the presence of equipment reduces the actual floor area to approximately the area recommended by MARSSIM. In addition, the floor and equipment in these areas were each treated as a survey unit (10 samples were collected for each). If any locations of contamination in excess of the guidelines were identified in that area, the survey of that area would be considered a characterization for the affected survey unit. Floor, wall, and equipment surfaces in the remainder of the facility were considered Class 3 and surveyed as a single survey unit. Grid blocks were referenced with the north east coordinate.

Outside areas in the vicinity of doors and exterior roof surfaces were treated as Class 3 survey units.

5.2 DETERMINATION OF DATA REQUIREMENTS

5.2.1 Beam Dust Samples

Dust samples were obtained from overhead structures to determine the extent of the affected area and levels of uranium concentrations. Seventeen beam dust samples from the ORNL survey contained concentrations of U-238 exceeding the surface activity guideline equivalent of 4.5 pCi/g (ORNL 1990). The average and standard deviation of concentrations in these samples was 86.8 +/- 85.3 pCi/g. Using the method for determining sample data needs, described in draft NUREG/CR-5849 (equation 8-22), a minimum of 48 samples was calculated as necessary to achieve an estimate within ± 30% of the true average concentration.

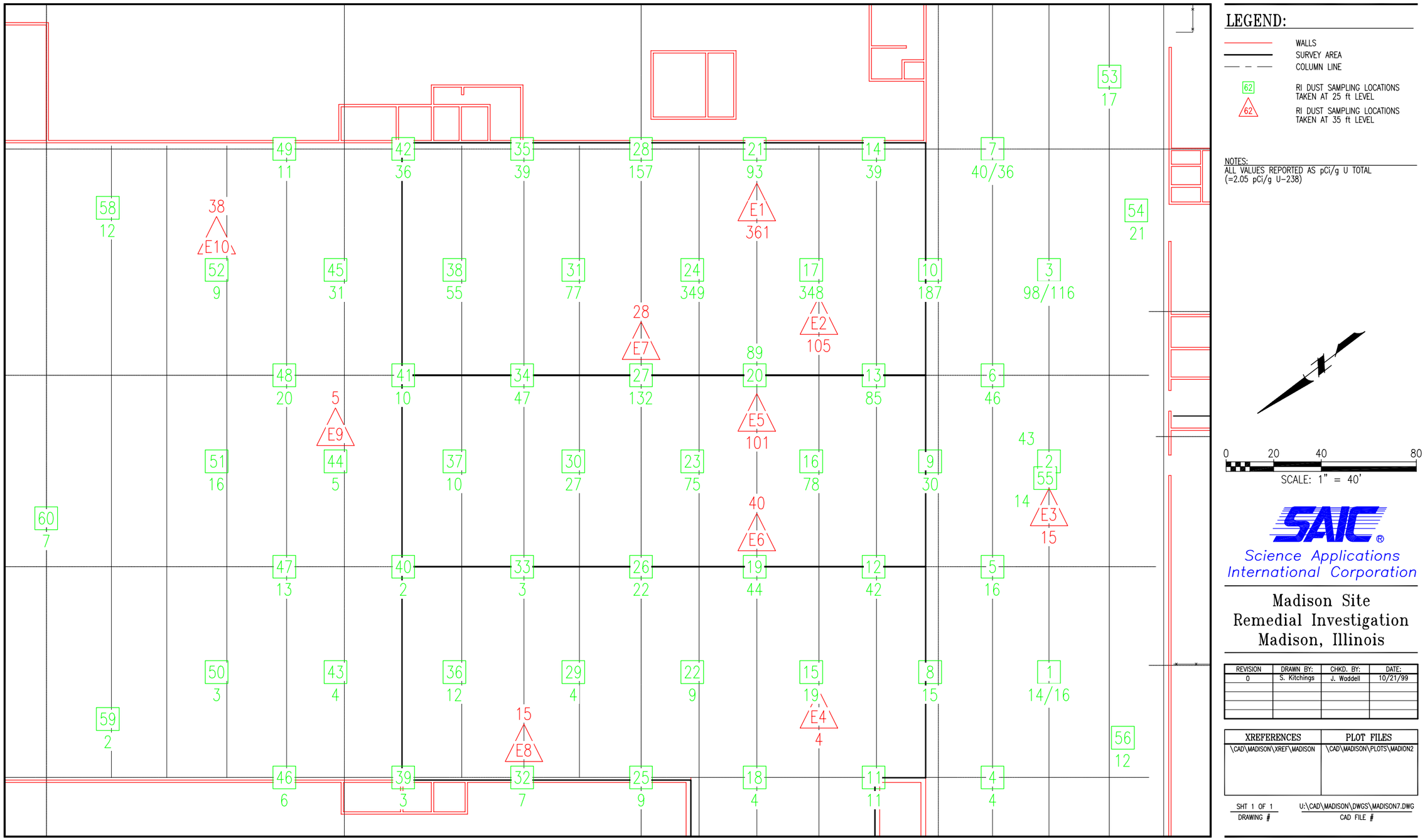


Figure 2. Diagram of Southwest portion of Building 6, indicating locations of dust sampling from major overhead beams in the vicinity of the Extrusion Press.

5.2.2 Measurements to Demonstrate Compliance with Guidelines

Data needs for Wilcoxon Rank Sum (WRS) statistical tests applicable to Class 2 and 3 areas were determined as follows:

1) Calculate the Relative Shift (Δ/σ)

   Δ = $DCGL_w$ – Lower Bound of the Gray Region (LBGR)

   a. Determine the $DCGL_w$

   b. Determine the LBGR

      The LBGR was selected to be 0 above background.

Sigma should be produced empirically for both the reference area (σ_r) and the survey area (σ_s). However, these data were not yet available. As a first approximation, values of sigma, typical of background measurements, were used to calculate data needs.

   c. The relative shift is calculated to be Δ/σ.

      MARSSIM recommends a range of 1 to 3 for Δ/σ. A value of 3 was assumed for initial planning purposes; if the value calculated was above approximately 3, the LBGR would be adjusted to provide a relative shift in that recommended range.

2) Determine Pr

   This was taken directly from MARSSIM (page 5-28, Table 5.1).

3) Determine the Decision Error Percentiles

   The null hypothesis (H_0) for each survey unit is that the residual radioactivity exceeds the $DCGL_W$. Acceptance decision errors for testing the hypothesis were set at 0.05 for both Type I and Type II errors.

$$N = \frac{(Z_{1-a} + Z_{1-b})^2}{3(P_r - 0.5)^2}$$

4) Calculate the Number of Data Points by:

where

N is the number of samples required for a given level of confidence;

$Z_{1-\alpha}$ and $Z_{1-\beta}$ are standard statistical values that vary with the level of confidence required and are obtained from tables; and

P_r was obtained in Step 2.

An additional 20% was added to allow for potential sample loss and quality control. The number of data points required is 20, to be split evenly between the reference area and the survey area.

5) Data Point Needs for Areas of Elevated Activity

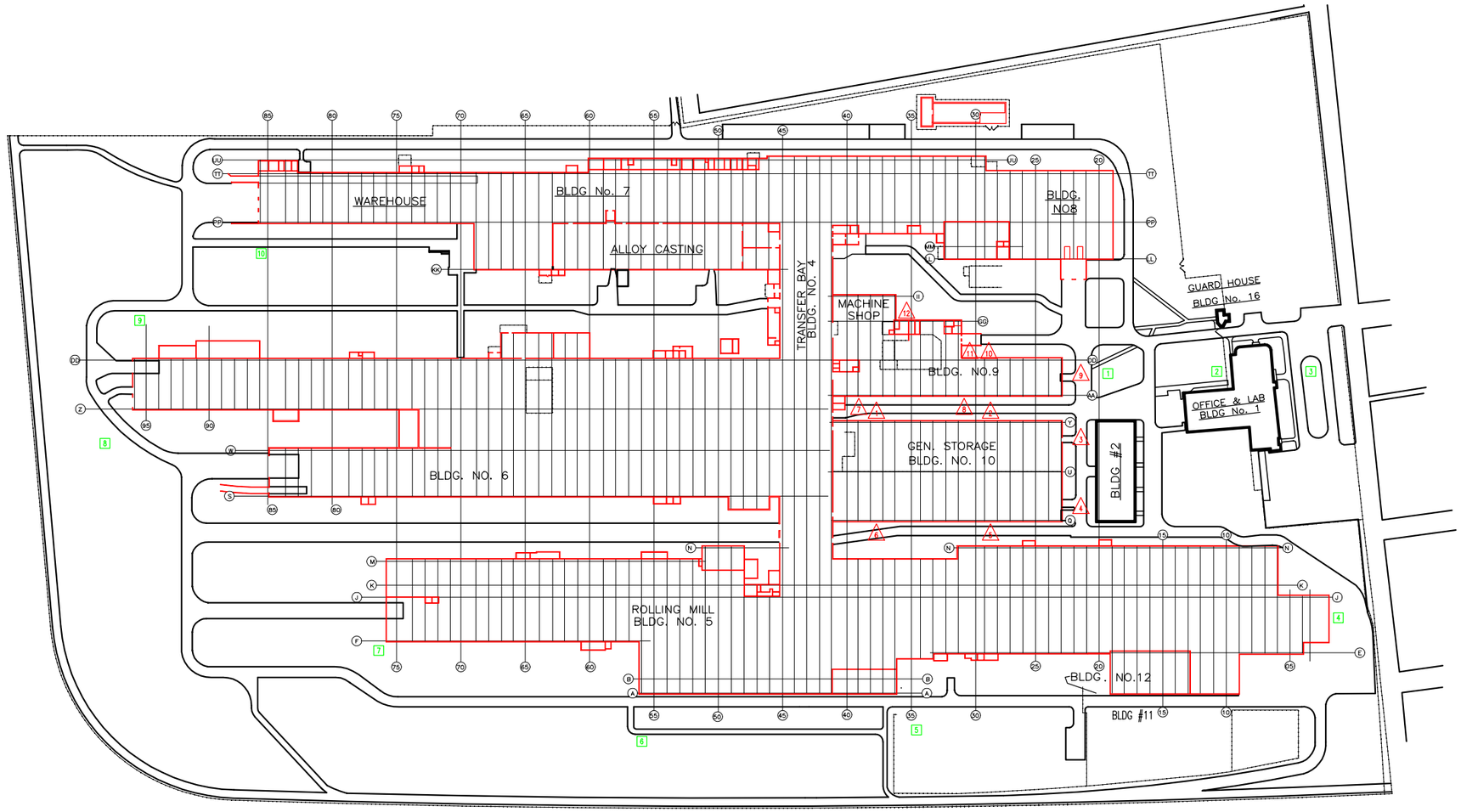
With exception of the gamma scintillation scans for uranium in soil and alpha scans of rough or covered surfaces, sensitivities of proposed instruments and techniques are such that $DCGL_w$ concentrations can be identified by scans (see Section 5.3 on Instrumentation). Because no Class I survey units were anticipated, there was not a requirement for determining data needs to satisfy elevated area provisions.

5.3 BACKGROUND (REFERENCE AREA) DETERMINATIONS

In accordance with the calculated data requirements (Section 5.2.2), 10 reference level measurements of gross beta activity and dose rate were performed in Buildings 9 and 10 (facilities without a history of radioactive materials use) over surfaces similar in nature to those to be surveyed in Buildings 6 and 4. In addition, 12 reference level measurements of exposure rate (converted to dose rate) were performed in Buildings 9 and 10 (facilities without a history of radioactive materials use) at exits similar in nature to those surveyed in Buildings 6 and 4. Ten soil background samples were obtained from soil areas away from doors to Buildings 6 and 4. The approximate locations of the exit point reference measurements and the background soil samples are shown in Figure 3.

5.4 INSTRUMENTATION

The radiological instruments used for this survey are listed in Table 2. For survey design purposes, detection sensitivities were estimated on the basis of the description of surface conditions in the ORNL report and the methodologies of NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments with Various Contaminants and Field Conditions" (NRC 1997). Nominal values for instrument response and background and literature values for survey instrument capabilities were used for these determinations. Refinements to these detection sensitivity estimates were made on the basis of actual instrument response and background data gathered during site survey activities. The resulting detection sensitivities for the beta surface activity measurements ranged from approximately 200 to 300 dpm/100 cm^2. Uncertainties associated with the beta surface activity measurements range from 10% to 30%.



LEGEND:

- ROAD
- WALLS
- FENCE
- WALKWAY
- BUILDINGS
- COLUMN LINE
- 17 SOIL REFERENCE LOCATIONS
- 59 GAMMA BASE RATE REFERENCE LOCATIONS

0 125 250 500

SCALE: 1" = 250'

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Madison Site
Remedial Investigation
Madison, Illinois

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Figure 3. Madison Site Reference Locations

Table 2. Instruments Used in the Madison Site Survey

Instrument	Application
Ludlum Model 2350 coupled with Ludlum Model 43-37 (floor monitor)	Beta Surface Scans
Ludlum Model 2350 coupled with Ludlum Model 44-2	Gamma Surface Scans & Exposure Rate Measurements
Bicron microrem (Tissue Equivalent Scintillation Detector)	Dose Equivalent Rate Measurements
Eberline BC-4 Beta Counter (Shielded GM Detector)	Removable Surface Activity
Eberline SAC-4 Alpha Counter (ZnS Scintillation Detector)	Removable Surface Activity
Ludlum Model 2350-1 coupled with Ludlum Model 44-2	Gamma Surface Scans & Exposure Rate Measurements
Ludlum Model 2350-1 coupled with Ludlum Model 43-68 (Gas proportional detector – 126 cm^2 effective area)	Beta Surface Scans

All instruments had been calibrated (within 12 months prior to use). Daily performance checks were conducted in accordance with individual instrument use procedures. These performance checks were conducted prior to and following daily field activities and at any time the instrument response appeared questionable. Only data obtained using instruments that satisfied the performance requirements were accepted for use in the evaluation.

5.5 OVERHEAD SURFACE SURVEYS

Dust samples were obtained from 52 systematic locations on the main horizontal overhead beams (25-ft level) of Buildings 6 and 4, in the vicinity of the extrusion press. Ten samples were also obtained from the second level horizontal support beams (36-ft level) in this general area. Ten additional dust samples from the main horizontal beams were obtained from other areas of Buildings 6 and 4. Sampling locations are indicated on Figures 2 and 4. Samples were collected by removing all loose material from a beam surface area of approximately 200 cm^2. The entire sample from each location was packaged and uniquely identified. Total beta surface activity measurements were performed before and after the sampling. A description of the methods for surface activity measurement is provided in Section 5.9.

5.6 GAMMA SCANNING

Gamma scans were performed using Ludlum Model 44-2 NaI scintillation detectors with Ludlum Model 2350 scaler/ratemeters. Audible response of the instruments was monitored while moving the detector in a serpentine pattern approximately 1 m wide and advancing at a speed of approximately 0.5 m/s. Locations of elevated audible response were noted for further investigation. The response range was recorded for each 25 ft $\times$ 25 ft area surveyed. Coverage was 100 % of accessible surfaces in the area beneath the potentially affected overhead structure surfaces (Survey Units 1, 2, and 3) and 5 to 10 % in other areas of Buildings 6 and 4. Gamma scans were also performed at entrances/exits to Buildings 6 and 4.

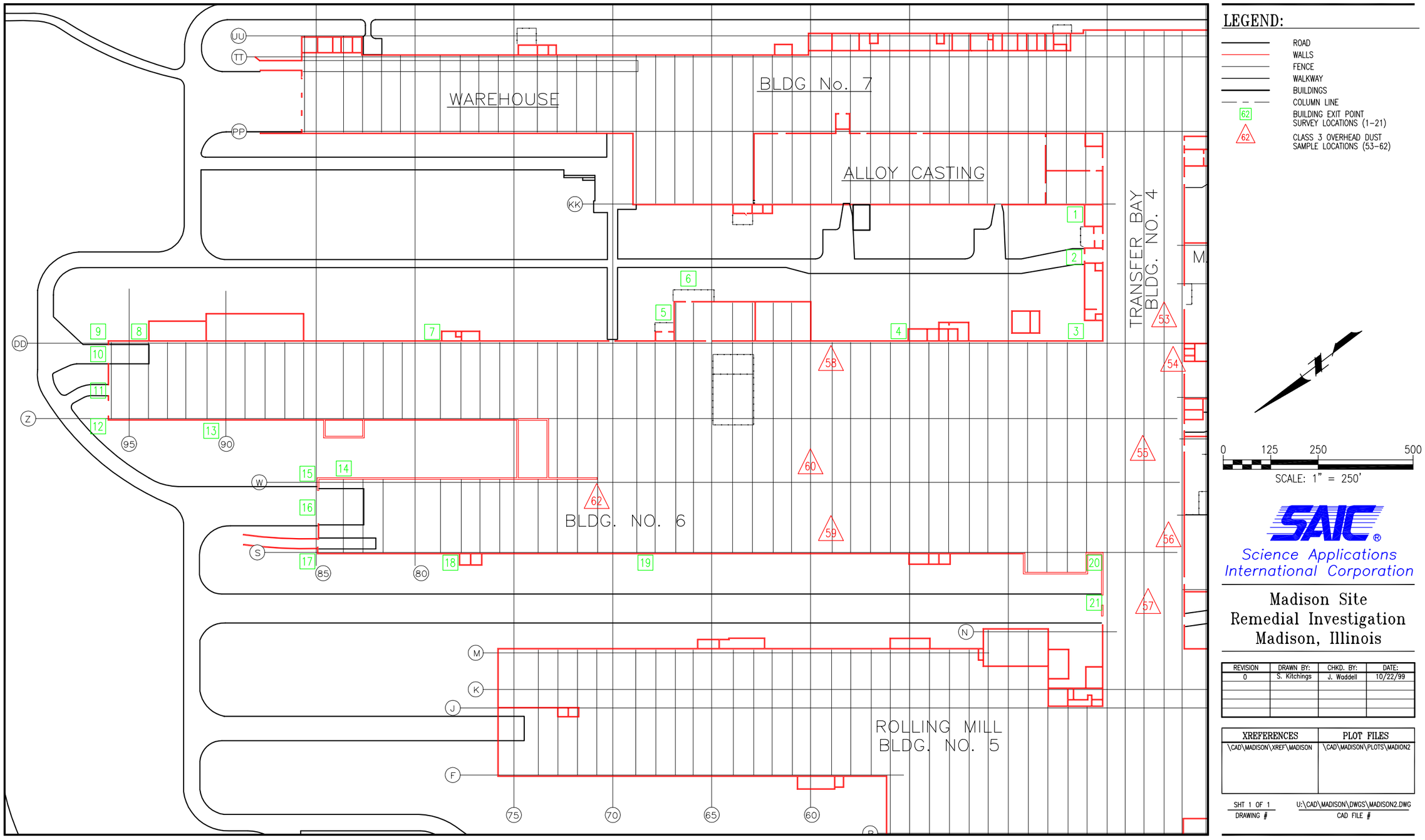

Figure 4. Building 6 and 4 (Partial)

5.7 BETA SCANNING

Beta scans of accessible floor surfaces were performed using Ludlum Model 239-1F floor monitors with large area (Model 43-37) gas-proportional detectors and Ludlum Model 2350 scaler/ratemeters. Floor monitor scan speed was approximately 0.5 m/s. Beta scans of floor surfaces, equipment surfaces, wall surfaces, roof surfaces, and other locations inaccessible by the floor monitor were performed using Ludlum Model 43-68 gas proportional detectors with Ludlum Model 2350 scaler/ratemeters. Scan speed with these smaller detectors was approximately 0.1 m/s. Audible response of the instruments was monitored, and locations of elevated audible response were identified for further investigation. The response range was recorded for each 25 ft $\times$ 25 ft floor area and for other locations surveyed. Coverage was 100 % of accessible surfaces in the area beneath the potentially affected overhead structure surfaces (Survey Units 1, 2, and 3) and 5 to 10 % in other areas.

5.8 SURFACE ACTIVITY MEASUREMENTS

Measurements of total beta surface activity were performed using Ludlum Model 43-68 gas proportional detectors with Ludlum Model 2350 scaler/ratemeters. Measurements were integrated for a counting period of 1 minute. Removable activity was measured by smearing an area of approximately 100 cm^2 with a dry filter paper; alpha and beta activity on the smear was then measured. Ten measurement locations each on floor, lower walls, and equipment surfaces were uniformly spaced throughout each survey unit.

5.9 SOIL SAMPLING

Soil samples of approximately 1 kilogram (kg) were obtained to a depth of 15 cm at 10 doorway locations. Samples were packaged and uniquely identified.

5.10 MISCELLANEOUS SAMPLING AND MEASUREMENTS

Samples of residues were collected from pits and trenches in Survey Units 1, 2, and 3, beneath the potentially affected overhead structures. Scrapings of floor residue were obtained from this same area; these sampling locations included locations of elevated beta scanning response. Five sludge samples and 10 floor scraping samples were collected. The sludge samples were collected from a large subsurface utility trench that runs from columns 47 to 59, north-south; and from Z to DD, east-west.

5.11 DOSE EQUIVALENT RATE MEASUREMENTS

Dose rate measurements were performed at 1 m (3.3 ft) above the surface, using Bicron microRem meters. Dose equivalent rate measurements were performed at 10 locations in each

Class 2 survey unit, at 17 locations in the Class 3 area. Seven of the Class 3 measurement locations were near entrances/exits to Buildings 6 and 4.

5.12 EXPOSURE RATE MEASUREMENTS

Exposure rate measurements were performed at 1 m (3.3 ft) above the surface, using a Ludlum Model 2350 coupled to a Ludlum Model 44-2 NaI detector. Exposure rate measurements were performed at each of the 23 exits from Buildings 4 and 6.

5.13 ANALYTICAL METHODS

Smear samples were analyzed for gross alpha and gross beta activity by using an Eberline Model SAC-4 alpha scintillation counter and an Eberline Model BC-4 beta G-M counter, respectively. Soil and miscellaneous samples were transferred to the FUSRAP radioanalytical laboratory at Hazelwood, Missouri, where they were dried, homogenized, analyzed by alpha spectrometry for isotopic uranium. High resolution gamma spectrometry analyses were also conducted to determine concentrations of K-40 and members of the natural uranium, thorium, and actinium decay series.

5.14 QUALITY CONTROL ACTIVITIES

Operational checks, including source response and background, were performed for all instruments at the beginning and completion of daily measurements. Results of these checks were compared to established performance criteria to determine acceptance of quantitative direct measurement data. Duplicate samples were obtained at 5% of the sampling locations for comparison of overall measurement variability. Duplicate dose equivalent rate measurements were indicated in the Survey Work Plan but were not performed.

6. SURVEY RESULTS

6.1 REFERENCE VALUE DETERMINATIONS

The results of the reference value determinations are summarized in Table 3, and the complete sets of data are provided in the tables in Appendix A. The values are typical of values for these types of measurements and sample results.

Table 3. Summary of Reference Levels

Measurement Type	Range of Values	Table in Appendix
Beta surface activity	291–650 dpm/100 cm^2	A-1
Dose equivalent rate	2-4 μrem/h	A-2
Exposure rate (exits)	6.7-12.2 μR/h	A-3
Uranium in Soil	0.9-2.7 pCi/g	A-4

6.2 OVERHEAD SURVEYS

The results of the surface scans, direct measurements on the overhead beams, and analytical results for dust samples are provided in Appendix A in Tables A-5 (scans and measurements) and A-6 (analytical results). Surface scans for beta activity identified generally elevated direct radiation levels throughout the area above the extrusion press. Direct measurements before sampling were typically above background. After sampling, the surfaces from which the samples had been collected were well within the guideline level. These results indicate that the source of the elevated level is the dust and residue on the beams. Analytical results for dust samples are presented in Appendix A in Table A-6. Samples 1 through 52, collected from the 25-ft level surfaces in the vicinity of the extrusion press, contained total uranium concentrations ranging from 2.3 to 348.7 pCi/g. Samples E-1 through E-10 from the 36-ft level contained total uranium concentrations ranging from 3.5 to 360.8 pCi/g.

Uranium concentrations in dust were highest directly above the extrusion press. The pattern of contamination is similar to that observed by the 1989 ORNL survey. An evaluation of this uranium contamination in dust is provided in Appendix B.

6.3 SURFACE SCANS

The results of beta and gamma surface scans of Class 2 and Class 3 building surfaces and equipment are summarized in Table 4. Complete results are provided in Appendix A in Tables A-7 and A-8, for Class 2 and 3, respectively. Class 2, Survey Units 2 and 3 and one of the exits (Exit 1) had areas identified with direct radiation levels from 1 to 2 times background. Beta scan data were normalized to adjust for effective area and response of different detectors used for this aspect of the survey. Based on this normalization, several locations exhibited beta activity levels in excess of 1000 dpm/100 cm^2 (see Table 4).

Table 4. Summary of Surface Scan Results

Class	Unit No.	Surface	No. of Blocks or Locations	Elevated Radiation
2	1	Floor	0	N/A
2	2	Floor	2 Blocks (Y-50, Z-49)	Beta
2	3	Floor	4 Blocks (V-45, W-47, W-48 and W-51)	Beta
2	3	Walls	3 Blocks (S-46, S-47, S-49)	Beta
3[1]	N/A	Floor and equipment	1 (floor and equipment in block K-44)	Beta and gamma
3	N/A	Areas near exits	1 location	Gamma

[1] Three blocks in the Class 3 area had elevated direct gamma radiation levels attributed to mag-thorium. These were R-42, R-44, and KK-41.

6.4 SURFACE ACTIVITY MEASUREMENTS

The results of measurements for total and removable surface activity are included in Appendix A in Tables A-9 and A-10, for Class 2 and 3 areas, respectively. The results are summarized in Table 5.

Table 5. Summary of Surface Activity Measurements

Class	Unit No.	Surface	Range of Removable Alpha Activity[1] (dpm/100 cm^2)	Range of Total Beta Activity[2] (dpm/100 cm^2)	Number of Measurements Greater than the $DCGL_w$
2	1	Equipment	-0.9 to 8.5	-73 to 357	0
2	1	Floor	-0.9 to 2.2	93-998	0
2	1	Walls	-0.9 to 2.2	302-550	0
2	2	Equipment	-0.9 to 2.2	-3 to 412	0
2	2	Floor	-0.9 to 2.2	204 to 641	0
2	3	Equipment	-0.9 to 2.2	-7 to 492	0
2	3	Floor	-0.9 to 2.2	392 to 806	0
2	3	Walls	-0.9 to 2.2	316 to 1130	3
3	N/A	Floor	-0.9 to 2.2	25 to 622	0
3	N/A	Roof	N/A	-53 to 1601	1
Floor Reference Level				**291 to 650**	
$DCGL_W$			**33**	**1000**	

[1] The detection sensitivities for removable alpha is 15 dpm/100 cm^2.
[2] The detection sensitivity for total beta activity is approximately 300 dpm/100 cm^2.

All of the measurements for removable alpha and beta activity were less than the detection sensitivity of the measurement procedures used. All but four of the total beta activity measurements were less the $DCGL_W$. Total beta activity at these four locations ranged from 1,031 dpm/100 cm^2 to 1,601 dpm/100 cm^2. All of these are less than the maximum surface contamination guideline. Follow-up measurements on the outside of the walls at grid blocks S-46, S-47, and S-48 indicated a beta activity range of 1,039 to 1,432 dpm/100 cm^2. Based on this finding, it is concluded that the three wall measurements above 1,000 dpm/100 cm^2 are due to natural content of the wall construction material. The source of the single elevated measurement on the roof is likely associated with naturally occurring material in the recently applied roofing material in this location. The data set for the roof included one direct measurement result of 1,601 dpm/100 cm^2. This value is above the average $DCGL_W$ of 1,000 dpm/100 cm^2, but is less than the maximum allowable level of 5,000 dpm/100 cm^2. The average for the roof measurements is approximately 500 dpm/100 cm^2 – also well below the average $DCGL_W$ of 1,000 dpm/100 cm^2. Reference area measurements were not performed for roof surfaces, and, therefore, these data cannot be tested using the WRS test.

6.5 SOIL SAMPLING

Total uranium concentrations in the soil samples collected near the exits range from 0.8 to 3.8 pCi/g, total uranium. A sample collected near doorway 1, which had an elevated gamma level of 32 µR/h, and contained 1.6 pCi/g total uranium and 3.0 pCi/g thorium-232. These concentrations of radionuclides are insufficient to account for the elevated direct gamma (32 µR/h) radiation level measured at this location. The data is provided in Appendix A, Table A-11. Further investigations were conducted with the in-situ gamma spec system as described in the RI Addendum (Appendix C) to identify the source of the higher measurements. This investigation revealed that the source of the elevated gamma readings is the result of Th-232. The Th-232 is used in other licensed processes conducted at the facility and is not associated with the AEC process.

6.6 MISCELLANEOUS SAMPLING

Total uranium concentrations in samples of sludge taken from the pits and trenches in the Class 2 survey units 1, 2, and 3 range from 0.5 to 2.4 pCi/g (Appendix A, Table A-12). The uranium concentrations in the sludge samples are within the range observed in the reference area samples. Total uranium concentrations in floor scrapings taken from Class 2, Survey Units 1, 2, and 3 range from 0.9 to 4.3 pCi/g (Appendix A, Table A-13). There is some evidence of slightly elevated uranium concentrations in some of the floor scraping samples. Further investigation/quantification of these areas will be necessary to determine the nature and extent of the residual uranium.

6.7 DOSE EQUIVALENT RATE MEASUREMENTS

The results of the dose equivalent measurements are summarized in Table 6 and are provided in their entirety in Appendix A, Tables A-14 and A-15, for Class 2 and 3, respectively.

Table 6. Summary of Dose Equivalent Measurement Results

Survey Class	Location	Dose Equivalent Rate (mrem/h)
2	Floor	2 to 6
3	Floor	2 to 4
Reference Area	various (see Table A-2)	2 to 4

In general, the dose equivalent rates are within the range of values observed in the reference area. One measurement, 6 µrem/h, in Survey Class 2 was greater than 4 µrem/h.

6.8 EXPOSURE RATE MEASUREMENTS

The results of the exposure rate measurements at exit points are summarized in Table 7 and are provided in their entirety in Appendix A, Table A-16.

Table 7. Summary of Exposure Rate Measurement Results

Survey Class	Location	Exposure Rate (μR/h)
3	Exits	6.1 to 18.2
Reference Area	various (see Table A-3)	6.7 to 12.2

Excluding the 18.2 μR/h measurement result from Exit 1, the exposure rates are within the range of values observed in the reference area.

7. DATA ASSESSMENT AND EVALUATION

7.1 COMPARISON WITH PROJECT OBJECTIVES

Data quality indicators for precision, accuracy, representativeness, completeness, and comparability were established and are specified in Section 4.7.

Precision is the degree of reproducibility of measurements under a given set of conditions. Field duplicate samples were collected to ascertain the contribution to variability (i.e., precision) due to the combination of environmental media, sampling consistency, and analytical precision. Field duplicate samples were collected from the same spatial and temporal conditions as the primary environmental sample. The precision of duplicate analyses was calculated as the Relative Percent Difference (RPD) using the following formula:

$$RPD = \text{Absolute Value } \{[(\text{measurement} - \text{duplicate})/(\text{measurement} + \text{duplicate})]\} \times 200$$

The results of these calculations are shown in Table 8 below.

Table 8. Evaluation of Duplicate Samples

Sample Location	Sample #	Sample Type	Total Uranium (pCi/g)	RPD
OH-03	M4D104	dust – grab	97.9	-16.9
OH-03	M4D204	dust – field duplicate	116.0	
OH-07	M4D106	dust – grab	39.6	10.6
OH-07	M4D206	dust – field duplicate	35.6	
OH-01	M4D109	dust – grab	13.7	-14.1
OH-01	M4D209	dust – field duplicate	15.8	
6S-102	M6S102	soil – grab	3.3	-9.7
6S-102	M6S202	soil – field duplicate	3.7	
SL-3	M6L103	sludge – grab	0.4	-76.9
SL-3	M6L203	sludge – field duplicate	1.0	
CC50	M6F104	floor scrapings – grab	1.5	23.4
CC50	M6F204	floor scrapings – field duplicate	1.2	

The stated objective for precision was a RPD of 30% or less at 50% of the guideline value. The only RPD value outside of the acceptable range is for Sample M6L103 which reported the lowest total uranium concentration, and is well below any realistic guideline value. The measurement and duplicate values for Sample M6L103 which gave an RPD of 76.9 are at or near the detection limits for the method. The values agree within the measurement uncertainty. A high RPD for measurements near the detection limit does not necessarily indicate a lack of agreement between measurements.

Analytical accuracy is expressed as the percent recovery of an analyte that has been added to a blank sample or environmental sample at a known concentration prior to analysis. This is generally determined through the use of Laboratory Control Samples (LCSs) and/or a Matrix Spike (MS) analysis. The laboratory did not perform matrix spike analyses; however, radionuclide LCS recoveries were well within the control limits of ± 30%, at 50% of the guideline value.

Representativeness and comparability is ensured through the selection and proper implementation of sampling and measurement techniques. The survey was performed by personnel trained in the procedures consistent with industry standards and appropriate for the type and levels of radioactive material present.

The quantity of data of acceptable quality satisfies the stated objective for completeness of 90%.

A thorough review of field survey documentation identified discrepancies and deficiencies, which required resolution, prior to final evaluation of the data. Most of these were of a minor nature and were adequately resolved through discussions with survey personnel or by collection of additional data and information. With exception of one exit location, for which the source of an isolated area of elevated direct ground-surface radiation has not been determined, there were no unresolved issues that prevented evaluation and recommendation for release of Class 3 areas. Subsequent surveys were conducted at the exit location as described in the RI Addendum, Appendix C. There are several unresolved deficiencies or discrepancies in data from Class 2 areas. These areas will likely require further data collection and evaluation before unrestricted release, and unresolved issues will be addressed at that time.

7.2 OVERHEAD SURFACE SURVEYS

Evaluation of uranium in dust on overhead surfaces and an assessment of the risk to plant workers is provided in Appendix B. The only significant levels of residual uranium from AEC operations are in the dust accumulated on overhead horizontal surfaces above the extrusion press, including structure-support beams, cross members, and window ledges. The affected surfaces were broken down into three categories of surface that total approximately 2,300 m^2 (25,000 ft^2) in area. The evaluation concludes that an acceptable risk to the production workers, less than 10^{-4} as specified in EPA CERCLA risk criteria, would be achieved at an average uranium

concentration of less than 186 pCi/g, for the described scenario and assumptions. The average concentrations measured by this survey are 48.6 pCi/g at the 25-ft level and 70.9 pCi/g at the 36-ft level. Similar levels were assumed for the difficult to access areas in the high bays. Only 3 of the 62 individual sampling locations in the vicinity of the extrusion press had concentrations above 186 pCi/g; these were in the area of columns BB to CC and 47 to 49, directly above the press. However, utility workers pulling cables and changing light bulbs are in closer proximity to the overhead surfaces and may receive unacceptable exposure from the contaminated surfaces. Using conservative assumptions, the evaluation found that the dose to utility workers could be as high as 210 mrem/yr with an associated excess lifetime cancer risk of 5.3 E-4, exceeding the upper end of the CERCLA risk range of 10^{-4}.

7.3 CLASS 2 AREAS

The beta surface scans identified floor blocks in Unit 2 and three in Unit 3 that were slightly elevated. However, all of the Class 2 floor and equipment surface activity measurements were less than the guideline level of 1,000 dpm/100 cm^2. Beta surfaces scans and direct measurements of wall grid blocks S-46, S-47, and S-48 indicated surface beta levels ranging from 1,031 to 1,130 dpm/100 cm^2. The surface of all of these blocks was brick. Measurements on the outside surface of the same brick wall produced values ranging from 1,039 to 1,432 dpm/ 100 cm^2. These data indicate that the results are consistent with this specific type of construction material (i.e., brick) and are not indicative of residual contamination.

Dose equivalent rates in the Class 2 areas compare favorably with the reference area results. The mean and standard deviation for the Class 2 and Reference areas are 3.7 ± 1.2 μrem/hr and 2.7 ± 0.8 μrem/hr, respectively.

The floor scraping sampling results indicate that residual uranium is present at low concentrations in Blocks BB-45 and CC-48. Further investigation will be required to determine the nature and extent of the residual uranium. The uranium concentrations in the remaining floor scraping and sludge sample results are within the range of the reference concentrations.

A requirement for Class 2 areas is that no locations of activity greater than the guideline level are present. Identification of several individual measurements above the $DCGL_W$ on surfaces in the vicinity of the press indicates that portions of the Class 2 area may have been incorrectly classified. The data are, therefore, not appropriate for evaluation under the MARSSIM approach. These areas will be reclassified and resurveyed in accordance with MARSSIM after remediation. Further investigation, likely including additional surveys of portions of the floor in this area, will be required to enable final status determination, relative to release criteria. If any removal of dust from overhead surfaces is performed, further surveys should be conducted after such activities.

The exposure rates for the exits and the Class 2 areas also are slightly higher than those in the reference area, but these data pass the WRS test, if the $DCGL_w$ is set to 2 μR/hr and Exit 1 is excluded from the group. Similar to the dose equivalent measurements, this difference between the Class 2 survey area and the reference area can not be attributed to the presence of residual

uranium. It does suggest that the range of values in the Building 9 and 10 reference areas may not be representative of the true range of exposure rate values throughout other portions of the facility. Fluctuations of this small increment (2 μR/h) are well within those observed in natural background. At Exit 1, the surface gamma scan result and the exposure rate at 1 m were both elevated; the results were 32 μR/h and 18 μR/h, respectively. Further investigation presented in Appendix C determined the source of the elevated gamma level at this exit is from Non-AEC thorium activities at the site.

7.4 CLASS 3 AREAS

In the Class 3 areas, except for the roof, each beta surface activity measurement is below the average guideline level or $DCGL_W$. Therefore, statistical testing of the data is not required to demonstrate compliance with the guidelines.

The data set for the roof included one direct measurement result of 1,601 dpm/100 cm^2. This value is above the average $DCGL_W$ of 1,000 dpm/100 cm^2, but is less than the maximum allowable level of 5,000 dpm/100 cm^2. The average for the roof measurements is approximately 500 dpm/100 cm^2, also well below the average $DCGL_W$ of 1,000 dpm/100 cm^2. Reference area measurements were not performed for roof surfaces, and, therefore, these data cannot be tested using the WRS test. However, based on the general conformance of the activity levels with the guidelines, the isolated nature of the area of elevated activity, and the likelihood that the elevated activity is associated with naturally occurring material in the recently applied roofing, the roof is considered to satisfy release criteria.

The dose equivalent measurements in the Class 3 area are slightly higher than those in the reference area, but these data pass the WRS test, if the $DCGL_w$ is set to 2 μR/hr. Based on the negative scanning results, this difference between the Class 2 survey area and the reference area can not be attributed to the presence of residual uranium. It does suggest that the range of values in the Building 10 reference area may not be representative of the true range of dose equivalent values throughout other portions of the facility. Fluctuations of this small increment (2 μrem/hr) are well within those observed in natural background.

A guideline level has not been established for uranium in soil for this project. Concentrations in soil samples from exit locations pass the WRS test, if the $DCGL_W$ is set equal to 2 pCi/g. Such a concentration is well within the fluctuations observed in natural background. Typically, uranium guidelines in soil have been in the range of 30 to 100 pCi/g for other decommissioning actions.

8. CONCLUSIONS AND RECOMMENDATIONS

The survey of the Madison Site, conducted in June, July, and November of 1998, identified residual uranium contamination in dust on overhead structures near the extrusion press.

An evaluation of the potential for plant production worker exposure to this dust demonstrates that the risk to these workers is below the 10^{-4} risk level. However, the risk evaluation found that utility workers operating in close proximity to the overhead structures may receive an exposure exceeding the CERCLA risk range due to the contaminated surfaces.

Surface activity on floors and equipment in the vicinity of the extrusion press (Class 2 areas) averages less than guideline levels and does not pose any potential risk to building occupants. However, because of indications of isolated locations of surface activity slightly above the guideline level, additional measurements are needed to perform an evaluation, adequate to justify unrestricted release of the facility. If the remedial action includes removal of dust from the overhead structures, additional measurements should be conducted on the floors and equipment following such actions.

An isolated area of elevated direct gamma radiation was identified outside exit door 1. The source of this radiation level must be resolved to perform final evaluation of this location, relative to unrestricted release.

Measurements and samples from the remaining areas of Buildings 4 and 6 (Class 3 areas) indicate that radiological conditions are below the various standards or guidance values for unrestricted release or are indistinguishable from normal background. No further radiological evaluation of those areas are considered necessary.

Based on the conclusions of this evaluation, it is recommended that the appropriate environmental documentation be completed to conduct limited remedial action and site closure consistent with the CERCLA process.

9. REFERENCES

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APPENDIX A

TABLES OF RADIOLOGICAL MEASUREMENT AND ANALYTICAL DATA

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Table A-1. Reference Area Beta Surface Activity Measurements

Measurement Location[1]	Beta dpm/100 cm^2
Building 9 beta R1	438
Building 9 beta R2	306
Building 9 beta R3	650
Building 9 beta R4	272
Building 9 beta R5	601
Building 10 beta R1	291
Building 10 beta R2	488
Building 10 beta R3	537
Building 10 beta R4	729
Building 10 beta R5	522

[1] Measurements taken on concrete

Table A-2. Reference Area Dose Equivalent Rate Measurements

Measurement Location[1]	Dose Rate (µrem/h)
Building 10 dose rate R1	3
Building 10 dose rate R2	2
Building 10 dose rate R3	2
Building 10 dose rate R4	3
Building 10 dose rate R5	2
Building 10 dose rate R6	2
Building 10 dose rate R7	4
Building 10 dose rate R8	3
Building 10 dose rate R9	2
Building 10 dose rate R10	4

[1] Measurements taken on concrete

Table A-3. Reference Area Exposure Rate Measurements for Exits

Measurement Location	Exposure Rate (µR/h)
Buildings 9 & 10 exposure rate R1	6.7
Buildings 9 & 10 exposure rate R2	7.5
Buildings 9 & 10 exposure rate R3	8.4
Buildings 9 & 10 exposure rate R4	8.1
Buildings 9 & 10 exposure rate R5	10.1
Buildings 9 & 10 exposure rate R6	9.7
Buildings 9 & 10 exposure rate R7	8.7
Buildings 9 & 10 exposure rate R8	7.8
Buildings 9 & 10 exposure rate R9	8.9
Buildings 9 & 10 exposure rate R10	12.2
Buildings 9 & 10 exposure rate R11	11.7
Buildings 9 & 10 exposure rate R12	8.9

Table A-4. Reference Soil Samples – Total Uranium Concentrations*

Sample Location	Sample #	Total Uranium (pCi/g)	Error (pCi/g)
SS-101	MBS101	2.3	0.8
SS-102	MBS102	1.7	0.6
SS-103	MBS103	1.9	0.7
SS-104	MBS104	2.1	0.7
SS-105	MBS105	1.5	0.6
SS-106	MBS106	2.3	0.7
SS-107	MBS107	2.4	0.8
SS-108	MBS108	1.8	0.7
SS-109	MBS109	0.8	0.4
SS-110	MBS110	1.6	0.6

* The error and detection limits for these samples are shown in Table A-17.

Table A-5. Beam Beta Scans, Sample Counts, and Smear Results

Sample and Measurement Location (see map)				Total Beta Activity			
				Pre-Sample		Post-Sample	
Measurement Location	Grid I.D.	Building	Scan Range (cpm)	gross cpm	dpm/100 cm^2	gross cpm	dpm/100 cm^2
1	U-43	4	400-650	575	888	432	386
2	Y-43	4	350-480	443	536	299	35
3	CC-43	4	550-800	800	1779	442	533
4	T-44	4	450-700	399	270	399	270
5	W-44	4	500-800	621	1050	467	509
6	AA-44	4	300-400	363	258	147	-494
7	DD-44	4	400-600	707	1455	438	519
8	U-45	4	400-600	548	794	387	228
9	Y-45	4	300-420	443	536	368	275
10	CC-45	4	700-1000	950	2301	317	97
11	T-46	6	350-450	455	385	374	111
12	W-46	6	500-600	618	1039	332	35
13	AA-46	6	500-700	696	1313	572	878
14	DD-46	6	500-650	436	400	422	351
15	U-47	6	350-400	386	152	316	-84
16	Y-47	6	700-800	845	1837	424	358
17	CC-47	6	1100-1300	1264	3117	332	-30
18	T-48	6	500-600	553	716	538	665
19	W-48	6	450-500	590	941	436	400
20	AA-48	6	350-420	444	428	329	25
21	DD-48	6	400-500	495	520	444	348
22	U-49	6	300-350	370	98	341	0
23	Y-49	6	350-400	508	1106	242	172
24	CC-49	6	1200-1400	1374	3489	445	351
25	T-50	6	300-350	404	213	320	-71
26	W-50	6	350-400	409	230	347	20
27	AA-50	6	550-650	674	1689	250	200
28	DD-50	6	900-1100	962	2097	324	-57
29	U-51	6	350-400	388	159	347	20
30	Y-51	6	350-450	414	776	196	11
31	CC-51	6	450-550	541	675	315	-88
32	T-52	6	350-400	343	7	343	7
33	W-52	6	350-400	365	81	296	-152
34	AA-52	6	250-300	242	172	167	-91
35	DD-52	6	350-450	383	142	301	-135
36	U-53	6	400-450	415	250	337	-14
37	Y-53	6	200-250	236	151	200	25
38	CC-53	6	450-600	660	1640	278	298
39	T-54	6	300-350	366	84	343	7
40	W-54	6	350-400	357	54	302	-132
41	AA-54	6	200-250	241	169	207	49
42	DD-54	6	450-500	560	740	397	189
43	U-55	6	300-350	334	-24	324	-57
44	Y-55	6	200-250	259	232	193	0
45	CC-55	6	450-500	533	648	272	-233
46	T-56	6	350-400	383	142	370	98
47	W-56	6	400-450	424	280	305	-122

Table A-5. Beam Beta Scans, Sample Counts, and Smear Results (continued)

Sample and Measurement Location (see map)				Total Beta Activity			
				Pre-Sample		Post-Sample	
Measurement Location	Grid I.D.	Building	Scan Range (cpm)	gross cpm	dpm/100 cm^2	gross cpm	dpm/100 cm^2
48	AA-56	6	400-450	418	260	283	-196
49	DD-56	6	300-350	387	155	302	-132
50	U-57	6	350-400	352	37	303	-128
51	Y-57	6	250-300	307	400	170	-81
52	CC-57	6	350-400	534	652	345	14
53	EE-42	4	300-350	313	421	183	-35
54	DD-41	4	400-450	393	702	192	-4
55	X-43	4	250-300	262	242	166	-95
56	T-42	4	250-300	283	316	180	-46
57	Q-43	4	250-300	281	309	208	53
58	CC-59	6	200-250	262	242	200	25
59	U-60	6	250-300	244	179	174	-67
60	Y-61	6	200-250	253	211	182	-39
61	CC-64	6	250-300	298	369	207	49
62	V-71	6	200-250	207	49	208	53
E-1	CC-48	6	1200-1400	1563	4127	386	152
E-2	BB-47	6	800-900	903	1898	547	696
E-3	Y-41	6	450-600	404	400	NA	NA
E-4	U-47	6	300-350	348	24	368	91
E-5	Z-48	6	400-550	809	2163	320	446
E-6	X-48	6	400-550	580	906	358	126
E-7	AA-50	6	300-350	347	541	304	390
E-8	T-52	6	400-450	487	493	370	98
E-9	Z-55	6	300-350	229	126	238	158
E-10	CC-57	6	450-500	490	503	399	196

NA - No post sample due to instrument malfunction

Table A-6. Uranium Concentrations in Samples of Dust from Overhead Surfaces*

Sample Location[1]	Sample #	Total Uranium Concentration[2] (pCi/g)	Error (pCi/g)
Overhead beams – 25-ft level			
01	M4D109	13.7	2.6
01[3]	M4D209	15.8	2.8
02	M4D101	43.1	6.6
03	M4D104	97.9	30.8
03[3]	M4D204	116.0	20.3
04	M4D111	3.3	0.9
05	M4D108	16.0	3.0
06	M4D103	45.9	7.0
07	M4D106	39.6	6.8
07[3]	M4D206	35.6	5.9
08	M4D110	15.2	3.1
09	M4D102	29.7	5.8
10	M4D105	146.7	28.9
11	M6D124	11.1	2.3
12	M6D105	42.3	8.6
13	M6D102	85.3	12.7
14	M6D101	38.6	5.8
15	M6D122	19.0	3.6
16	M6D103	78.2	12.0
17	M6D108	348.7	54.9
18	M6D128	3.7	0.9
19	M6D106	43.9	8.3
20	M6D104	89.2	13.4
21	M6D109	92.6	14.4
22	M6D125	8.6	1.9
23	M6D146	75.0	11.8
24	M6D110	348.7	57.5
25	M6D127	8.9	2.2
26	M6D126	22.0	3.8
27	M6D152	131.8	20.0
28	M6D114	157.2	24.3
29	M6D129	3.7	0.9
30	M6D147	26.9	5.4
31	M6D115	77.2	11.6
32	M6D130	7.0	1.4
33	M6D131	2.5	0.7
34	M6D151	47.3	8.5
35	M6D113	39.3	6.3
36	M6D132	11.6	2.2
37	M6D145	10.0	2.5
38	M6D156	54.6	9.1
39	M6D135	3.2	0.9
40	M6D134	2.3	0.7
41	M6D150	9.8	2.2
42	M6D121	35.6	6.2
43	M6D136	4.2	1.0
44	M6D148	5.2	1.2

Table A-6. Uranium Concentrations in Samples of Dust from Overhead Surfaces* (continued)

Sample Location[1]	Sample #	Total Uranium Concentration[2] (pCi/g)	Error (pCi/g)
45	M6D116	31.2	7.8
46	M6D138	6.1	1.4
47	M6D137	12.4	2.8
48	M6D117	19.4	3.7
49	M6D120	10.7	2.0
50	M6D139	2.5	0.7
51	M6D149	15.9	3.2
52	M6D119	8.4	2.4
Overhead Beams – 36-ft level			
E1	M6D111	360.8	60.2
E2	M6D112	104.6	16.5
E3	M4D107	15.2	2.7
E4	M6D123	3.2	0.9
E5	M6D154	100.9	15.5
E6	M6D107	39.5	6.6
E7	M6D155	27.6	4.5
E8	M6D133	14.7	3.1
E9	M6D153	4.6	1.2
E10	M6D118	37.7	6.5
Overhead Beams – 25-ft level (other building areas)			
53	M4D114	17.2	2.8
54	M4D112	20.9	3.9
55	M4D113	13.9	2.7
56	M4D115	11.9	2.4
57	M4D116	7.3	1.8
58	M6D144	11.7	2.5
59	M6D140	1.6	0.6
60	M6D142	6.9	1.5
61	M6D143	5.0	1.2
62	M6D141	2.7	0.9

* The error and detection limits for these samples are shown in Table A-17.

[1] Refer to Figure 2 and 3 for sample locations.

[2] Sum of U-234, U-235, and U-238 concentrations. Refer to Table A-17 of Appendix A for isotopic analytical data.

[3] Duplicate sample.

Table A-7. Class 2 Equipment, Floor, Floor Sump Area and Walls Beta and Gamma Scan Results

Unit	Surface[1, 2]	Measurement Location	Beta Scan Range		Gamma Scan Range (μR/hr)	Locations with Elevated Direct Radiation
			(cpm)	(dpm/100 cm^2)		
1	Equipment	BB-47	metal 210-300	194 to 506	NA	no
1	Equipment	BB-48	metal 250-320	333 to 575	NA	no
1	Equipment	BB-50	metal 240-270	298 to 402	NA	no
1	Equipment	BB-51	metal 210-300	194 to 506	NA	no
1	Equipment	CC-46	metal 150-170	-14 to 55	NA	no
1	Equipment	CC-47	metal 160-320	21 to 575	NA	no
1	Equipment	CC-48	metal 250-320	333 to 575	NA	no
1	Equipment	CC-51	not performed	not performed	2.8-3.3	no
1	Equipment	CC-52	not performed	not performed	3.8-6	no
1	Equipment	CC-53	not performed	not performed	3.4-3.8	no
1	Floor	AA-45	800-1200	164 to 648	3.8-4.1	no
1	Floor	AA-46	800-1300	242 to 855	3.7-4.9	no
1	Floor	AA-47	800-1200	242 to 733	2.5-3.6	no
1	Floor	AA-48	500-1100	-126 to 610	3-3.5	no
1	Floor	AA-49	990-1100	456 to 591	4.3-4.7	no
1	Floor	AA-50	980-1050	443 to 530	4.5-5	no
1	Floor	AA-51	990-1170	456 to 678	3.4-4.5	no
1	Floor	AA-52	1000-1170	468 to 678	4.2-5.3	no
1	Floor	AA-53	950-1040	406 to 517	5.2-6.1	no
1	Floor	BB-45	800-1800	164 to 648	3.8-4.7	no
1	Floor	BB-46	900-1200	364 to 733	4-4.9	no
1	Floor	BB-47	900-1200	364 to 733	2.1-3.2	no
1	Floor	BB-48	400-800 metal; 800-1200 concrete	-249 to 733	2.1-4.2	no
1	Floor	BB-49	920-1050	369 to 530	4.3-4.5	no
1	Floor	BB-50	890-1004	332 to 473	4.2-4.5	no
1	Floor	BB-51	990-1060	456 to 542	7	no
1	Floor	BB-52	840-1040	270 to 517	7.3	no
1	Floor	BB-53	not performed	not performed	5.7-6.2	no
1	Floor	CC-45	900-1300	285 to 770	3.3-4.1	no
1	Floor	CC-46	900-1200	364 to 733	3.9-5	no
1	Floor	CC-47	900-1100	364 to 610	2.7-3.3	no
1	Floor	CC-48	400-900 metal; 800-1200 concrete	-249 to 733	2.5-3.3	no
1	Floor	CC-49	900-1020	344 to 493	4.2-4.5	no
1	Floor	CC-50	910-2000	357 to 554	3.9-4.2	no
1	Floor	CC-51	830-890	258 to 332	3.7-6.3	no
1	Floor	CC-52	710-790	110 to 209	4.1-5.7	no
1	Floor	CC-53	880-920	320 to 369	3.5-4.8	no
1	Floor	DD-45	900-1200	285 to 648	3-4.7	no
1	Floor	DD-46	800-1100	242 to 610	3.4-5	no
1	Floor	DD-47	900-1200	364 to 733	4.1-4.8	no
1	Floor	DD-48	600-1200	-4 to 733	3-4.2	no
1	Floor	DD-49	900-1200	364 to 733	3.8-4.9	no
1	Floor	DD-50	600-1300	-4 to 855	3.9-5	no
1	Floor	DD-51	850-1000	283 to 468	3.5-5.2	no
1	Floor	DD-52	750-1030	159 to 505	3.6-5	no

Table A-7. Class 2 Equipment, Floor, Floor Sump Area and Walls Beta and Gamma Scan Results (continued)

Unit	Surface[1, 2]	Measurement Location	Beta Scan Range		Gamma Scan Range (μR/hr)	Locations with Elevated Direct Radiation
			(cpm)	(dpm/100 cm^2)		
1	Floor	DD-53	990-1130	456 to 628	4.2-6.5	no
1	Floor Sump Area	AA-51	not performed	not performed	6.5-7.4	no
1	Floor Sump Area	AA-53	not performed	not performed	5.2-6.8	no
1	Floor Sump Area	CC-47	not performed	not performed	4.7-5.4	no
1	Floor Sump Area	CC-49	not performed	not performed	4.4-5.4	no
1	Floor Sump Area	CC-49	not performed	not performed	4.2	no
1	Floor Sump Area	CC-51	not performed	not performed	4.8	no
1	Floor Sump Area	CC-53	not performed	not performed	5.2-6.8	no
1	Floor Sump Area	CC-56	not performed	not performed	5.5	no
1	Floor Sump Area	DD-50	not performed	not performed	5.2-6.8	no
1	Floor Sump Area	DD-57	not performed	not performed	5.2-6.8	no
1	Floor Sump Area	DD-59	not performed	not performed	7	no
1	Floor Sump Area	EE-52	not performed	not performed	6.4	no
1	Floor Sump Area	EE-53	not performed	not performed	5.2-6.8	no
1	Walls	DD-46	270-310	186 to 323	NA	no
1	Walls	DD-47	300-350	289 to 460	NA	no
1	Walls	DD-48	280-370	220 to 529	NA	no
1	Walls	DD-49	250-340	117 to 426	NA	no
1	Walls	DD-50	260-360	151 to 495	NA	no
1	Walls	DD-51	270-310	186 to 323	NA	no
1	Walls	DD-52	280-300	220 to 289	NA	no
1	Walls	DD-53	290-400	254 to 632	NA	no
1	Walls	DD-45	310-390	323 to 598	NA	no
2	Equipment	X-48	metal 250-320	333 to 575	NA	no
2	Equipment	X-49	metal 300-350	506 to 679	NA	no
2	Equipment	X-51	metal 290-360	471 to 714	NA	no
2	Equipment	X-52	metal 290-360	471 to 714	NA	no
2	Equipment	Y-48	metal 170-200	55 to 159	NA	no
2	Equipment	Y-49	not performed	not performed	1.8-3	no
2	Equipment	Y-51	metal 290-360	471 to 714	NA	no
2	Equipment	Y-52	metal 290-360	471 to 714	NA	no
2	Equipment	Z-48	metal 250-320	333 to 575	NA	no
2	Equipment	Z-49	metal 300-350	506 to 679	NA	no
2	Floor	X-45	1260-1600	321 to 728	3.5-4.2	no
2	Floor	X-46	1300-1720	369 to 872	3.5-4.2	no
2	Floor	X-47	1300-1800	369 to 968	3.6-4.2	no
2	Floor	X-48	not performed	not performed	NA	no
2	Floor	X-49	1270-1600	333 to 728	2.9-3.8	no
2	Floor	X-50	1250-1800	309 to 968	3.7-4.5	no
2	Floor	X-51	1200-1450	249 to 549	3.8-4.5	no
2	Floor	X-52	not performed	not performed	3.8-4.2	no
2	Floor	X-53	not performed	not performed	3.8-4.2	no
2	Floor	Y-45	1390-1800	477 to 968	4-4.5	no
2	Floor	Y-46	1330-1750	405 to 908	3.4-4.3	no
2	Floor	Y-47	1400-1700	489 to 848	3.8-4.7	no
2	Floor	Y-48	not performed	not performed	NA	no
2	Floor	Y-49	1220-1420	273 to 513	3.1-3.7	no
2	Floor	Y-50	1200-1840	249 to 1016	4.6-5.2	yes

Table A-7. Class 2 Equipment, Floor, Floor Sump Area and Walls Beta and Gamma Scan Results (continued)

Unit	Surface[1, 2]	Measurement Location	Beta Scan Range		Gamma Scan Range (mR/hr)	Locations with Elevated Direct Radiation
			(cpm)	(dpm/100 cm^2)		
2	Floor	Y-51	1130-1400	165 to 489	3.8-4.3	no
2	Floor	Y-52	not performed	not performed	4.3-5	no
2	Floor	Y-53	not performed	not performed	3.7-4.5	no
2	Floor	Z-45	1450-1800	549 to 968	3-4.9	no
2	Floor	Z-46	1225-1650	279 to 788	3.8-4.8	no
2	Floor	Z-47	1480-1650	584 to 788	3.3-4.2	no
2	Floor	Z-48	not performed	not performed	NA	
2	Floor	Z-49	1120-1850	153 to 1028	3.2-3.7	yes
2	Floor	Z-50	1300-1810	369 to 980	3.3-4.6	no
2	Floor	Z-51	1100-1350	129 to 429	4-4.5	no
2	Floor	Z-52	1200-1650	249 to 788	4.7-4.8	no
2	Floor	Z-53	1110-1400	141 to 489	3.6-4.8	no
3	Equipment	S-47	metal 310-375	541 to 766	NA	no
3	Equipment	T-48	metal 290-305	471 to 523	NA	no
3	Equipment	V-45	metal 210-290	194 to 471	NA	no
3	Equipment	V-48	metal 170-250	55 to 333	NA	no
3	Equipment	V-50	metal 275-300	419 to 506	NA	no
3	Equipment	V-51	not performed	not performed	2.8-4	no
3	Equipment	W-46	metal 170-250	55 to 333	NA	no
3	Equipment	W-47	metal 170-250	55 to 333	NA	no
3	Equipment	W-48	metal 250-320	333 to 575	NA	no
3	Equipment	W-49	not performed	not performed	1.7-2.8	no
3	Floor	T-45	970-1050	431 to 530	4.2-4.9	no
3	Floor	T-46	900-1020	344 to 493	4.5-5	no
3	Floor	T-47	790-1200	209 to 715	4.5-4.7	no
3	Floor	T-48	980-1030	443 to 505	4.8-5.4	no
3	Floor	T-49	890-1100	332 to 591	4.6-5.8	no
3	Floor	T-50	930-1100	381 to 591	3.9-4.2	no
3	Floor	T-51	950-1110	406 to 604	4-4.8	no
3	Floor	T-52	940-1200	394 to 715	4.7-5.3	no
3	Floor	T-53	980-1260	443 to 789	4.9-7.1	no
3	Floor	U-45	940-1100	394 to 591	4-4.6	no
3	Floor	U-46	900-1010	344 to 480	4.7-5.2	no
3	Floor	U-47	890-1160	332 to 665	4.7-4.9	no
3	Floor	U-48	900-1190	344 to 702	4.4-4.9	no
3	Floor	U-49	930-1130	381 to 628	4.3-5.4	no
3	Floor	U-50	950-1100	406 to 591	4.5-5.2	no
3	Floor	U-51	940-1100	394 to 591	4.4-5.1	no
3	Floor	U-52	980-1100	443 to 591	4.6-5.1	no
3	Floor	U-53	980-1100	443 to 591	4.5-5.2	no
3	Floor	V-45	1270-1830	370 to 1007	4-4.8	no
3	Floor	V-46	1390-1820	507 to 995	4.4-4.7	no
3	Floor	V-47	1230-1760	325 to 927	3.9-5.1	no
3	Floor	V-48	1330-1700	439 to 589	4.5-5.3	no
3	Floor	V-49	1420-1800	541 to 973	4.1-5.0	no
3	Floor	V-50	1440-1750	564 to 916	4.4-5.1	no
3	Floor	V-51	1370-1680	484 to 836	4.3-4.7	no
3	Floor	V-52	1410-1760	530 to 927	4.1-5.1	no

Table A-7. Class 2 Equipment, Floor, Floor Sump Area and Walls Beta and Gamma Scan Results (continued)

Unit	Surface[1, 2]	Measurement Location	Beta Scan Range		Gamma Scan Range (mR/hr)	Locations with Elevated Direct Radiation
			(cpm)	(dpm/100 cm^2)		
3	Floor	V-53	1450-1780	575 to 950	4.1-4.8	no
3	Floor	W-45	1110-1710	189 to 870	3.3-4.2	no
3	Floor	W-46	1150-1760	234 to 927	3.6-4.6	no
3	Floor	W-47	1170-2000	257 to 1200	3.6-4.4	yes
3	Floor	W-48	1280-1850	382 to 1030	3.9-4.8	yes
3	Floor	W-49	1120-1690	200 to 848	3.5-4.1	no
3	Floor	W-50	1500-1740	632 to 905	3.7-4.3	no
3	Floor	W-51	1250-1960	348 to 1155	3.7-4.7	yes
3	Floor	W-52	1190-1700	280 to 859	4-4.4	no
3	Floor	W-53	1390-1740	507 to 905	3.9-4.2	no
3	Walls	S-45	290-350	254 to 460	NA	no
3	Walls	S-46	410-570[3]	667 to 1216	NA	no
3	Walls	S-47	400-540[3]	632 to 1113	NA	no
3	Walls	S-48	500-570[3]	976 to 1216	NA	no
3	Walls	S-49	300-390	289 to 598	NA	no
3	Walls	S-50	290-350	254 to 460	NA	no
3	Walls	S-51	270-330	186 to 392	NA	no
3	Walls	S-52	300-350	289 to 460	NA	no
3	Walls	S-53	270-305	186 to 306	NA	no

[1] The surface is concrete unless otherwise specified.
[2] Scans of equipment and wall surfaces were performed using a smaller area probe.
[3] Measurement taken on brick surface.

Table A-8. Class 3 Doorway, Equipment, and Floor Beta and Gamma Scan Results

Surface	Measurement Location	Beta Scan Range		Gamma Scan Range (mR/hr)	Locations with Elevated Direct Radiation
		(cpm)	(dpm/100 cm^2)		
Door 1	not performed	not performed	not performed	11.0-32.0	yes
Door 2	not performed	not performed	not performed	8.5-9.0	no
Door 3	not performed	not performed	not performed	6.4-6.7	no
Door 4	not performed	not performed	not performed	6.4-9.0	no
Door 5	not performed	not performed	not performed	5.7-6.1	no
Door 6	not performed	not performed	not performed	5.6-5.9	no
Door 7	not performed	not performed	not performed	4.4-4.6	no
Door 8	not performed	not performed	not performed	5.3-5.5	no
Door 9	not performed	not performed	not performed	6.2-7.0	no
Door 10	not performed	not performed	not performed	6.5-6.6	no
Door 11	not performed	not performed	not performed	5.9-6.2	no
Door 12	not performed	not performed	not performed	5.9-6.1	no
Door 13	not performed	not performed	not performed	5.4-5.5	no
Door 14	not performed	not performed	not performed	5.0-5.4	no
Door 15	not performed	not performed	not performed	5.7-6.0	no
Door 16	not performed	not performed	not performed	5.7-6.0	no
Door 17	not performed	not performed	not performed	7.0-7.5	no
Door 18	not performed	not performed	not performed	9.0-9.5	no
Door 19	not performed	not performed	not performed	5.1-5.3	no
Door 20	not performed	not performed	not performed	6.2-6.6	no
Door 21	not performed	not performed	not performed	5.8-6.2	no
Door 22	not performed	not performed	not performed	5.8-6.3	no
Door 23	not performed	not performed	not performed	4.8-4.9	no
Equipment	A-55E	not performed	not performed	2.9-3.1	no
Equipment	BB-44E	not performed	not performed	2.3-2.7	no
Equipment	BB-58E	not performed	not performed	1.8-2.5	no
Equipment	CC-68E	not performed	not performed	1.9-2.5	no
Equipment	CC-74E	not performed	not performed	1.8-2.7	no
Equipment	DD-41E	not performed	not performed	4-5.8	no
Equipment	K-44E	not performed	not performed	6.6-8.8	yes
Equipment	KK-41E	not performed	not performed	3.8-4.8	no
Equipment	P-42E	not performed	not performed	3.8-5.2	no
Equipment	T-60E	not performed	not performed	2.1-3.5	no
Equipment	U-41E	not performed	not performed	2.8-4.2	no
Equipment	W-58E	not performed	not performed	2.9-3.1	no
Equipment	X-44E	not performed	not performed	2.5-3.3	no
Equipment	Z-43E	not performed	not performed	1.8-2.6	no
Floor	BB-44	269-373	-3 to 327	not performed	no
Floor	BB-44	not performed	not performed	3.2-4	no
Floor	BB-55	380-431	349 to 511	not performed	no
Floor	BB-58	not performed	not performed	2.8-3.5	no
Floor	CC-59	329-410	187 to 444	not performed	no
Floor	CC-61	318-439	152 to 537	not performed	no
Floor	CC-66	not performed	not performed	3.5-3.8	no
Floor	CC-74	not performed	not performed	3.4-4.1	no
Floor	CC-80	240-310	-95 to 127	not performed	no
Floor	DD-41	not performed	not performed	3.8-5.8	no
Floor	DD-44	250-327	-63 to 181	not performed	no

Table A-8. Class 3 Doorway, Equipment, and Floor Beta and Gamma Scan Results (continued)

Surface	Measurement Location	Beta Scan Range		Gamma Scan Range (mR/hr)	Locations with Elevated Direct Radiation
		(cpm)	(dpm/100 cm^2)		
Floor	DD-54	390-431	381 to 511	not performed	no
Floor	DD-55	not performed	not performed	3.4-4.5	no
Floor	II-43	350-466	254 to 622	not performed	no
Floor	K-44	not performed	not performed	6.0-10	yes
Floor	KK-41	not performed	not performed	4.7-11 [1]	no
Floor	P-42	not performed	not performed	5.2-6	no
Floor	Q-42	332-450	197 to 571	not performed	no
Floor	R-42	not performed	not performed	13 [1]	yes
Floor	R-44	not performed	not performed	87 [1]	yes
Floor	S-44	260-420	-32 to 476	not performed	no
Floor	T-60	not performed	not performed	3.9-5.6	no
Floor	T-69	not performed	not performed	4-4.7	no
Floor	T-75	not performed	not performed	4.7-5.5	no
Floor	U-41	not performed	not performed	3.5-4.7	no
Floor	U-54	350-450	data not provided	not performed	no
Floor	U-57	not performed	not performed	4.5-5.1	no
Floor	U-58	330-490	data not provided	not performed	no
Floor	U-66	286-446	data not provided	not performed	no
Floor	U-80	not performed	not performed	4.6-5.9	no
Floor	V-42	310-404	data not provided	not performed	no
Floor	V-72	217-328	data not provided	not performed	no
Floor	V-72	not performed	not performed	3.9-4.7	no
Floor	W-58	not performed	not performed	4.1-4.3	no
Floor	W-62	not performed	not performed	4-4.7	no
Floor	W-66	not performed	not performed	4.7-5.3	no
Floor	W-77	not performed	not performed	4.4-5.5	no
Floor	X-44	not performed	not performed	2.7-4	no
Floor	X-64	not performed	not performed	3.6-3.9	no
Floor	Y-41	290-415	data not provided	not performed	no
Floor	Y-54	not performed	not performed	3.9-5	no
Floor	Y-56	380-430	data not provided	not performed	no
Floor	Y-60	not performed	not performed	4.3-5	no
Floor	Y-62	269-387	data not provided	not performed	no
Floor	Y-70	not performed	not performed	4.1-4.6	no
Floor	Z-43	not performed	not performed	2.9-5.6	no
Floor	Z-44	260-426	data not provided	not performed	no
Floor	Z-55	not performed	not performed	4.2-5.6	no

[1] Elevated gamma levels attributed to mag-thorium

Table A-9. Class 2 Equipment, Floor, and Walls Total and Removable Activity Levels

Unit	Surface	Measurement Location	Removable Activity				Total Beta Activity	
			Beta		Alpha			
			net cpm	dpm/100 cm^2	net cpm	dpm/100 cm^2	gross cpm	dpm/100 cm^2
1	Equipment	AA-51	2	9	-0.3	-0.9	192	132
1	Equipment	BB-47	0	< 90	2.7	8.5	166	42
1	Equipment	BB-48	9	42	0.7	2.2	133	-73
1	Equipment	BB-50	11	52	-0.3	-0.9	223	239
1	Equipment	BB-50	15	71	-0.3	-0.9	235	281
1	Equipment	BB-51	6	28	0.7	2.2	257	357
1	Equipment	CC-46	9	42	1.7	5.3	140	-49
1	Equipment	CC-47	3	14	0.7	2.2	156	7
1	Equipment	CC-47	1	5	0.7	2.2	161	24
1	Equipment	CC-47	-2	-9	1.7	5.3	160	21
1	Floor	AA-49	12	57	-0.3	-0.9	488	485
1	Floor	AA-50	8	38	0.7	2.2	556	689
1	Floor	AA-53	6	28	-0.3	-0.9	502	527
1	Floor	BB-45	11	52	-0.3	-0.9	501	696
1	Floor	BB-47	-1	-5	-0.3	-0.9	357	93
1	Floor	BB-48	1	5	-0.3	-0.9	385	177
1	Floor	CC-46	6	28	-0.3	-0.9	460	401
1	Floor	CC-48	6	28	-0.3	-0.9	400	222
1	Floor	CC-50	2	9	-0.3	-0.9	552	998
1	Floor	DD-53	8	38	0.7	2.2	508	545
1	Walls	DD-45	5	24	0.7	2.2	376	550
1	Walls	DD-46	4	19	-0.3	-0.9	333	402
1	Walls	DD-47	0	< 90	-0.3	-0.9	331	395
1	Walls	DD-48	12	57	0.7	2.2	355	478
1	Walls	DD-48	8	38	-0.3	-0.9	304	302
1	Walls	DD-49	7	33	0.7	2.2	344	440
1	Walls	DD-50	5	24	0.7	2.2	326	378
1	Walls	DD-51	2	9	-0.3	-0.9	312	330
1	Walls	DD-52	4	19	-0.3	-0.9	282	227
2	Equipment	X-48	-4	-19	-0.3	-0.9	208	187
2	Equipment	X-49	-4	-19	-0.3	-0.9	212	201
2	Equipment	X-51	12	57	-0.3	-0.9	205	177
2	Equipment	X-52	5	24	0.7	2.2	257	357
2	Equipment	Y-48	10	47	-0.3	-0.9	190	125
2	Equipment	Y-49	7	33	0.7	2.2	202	166
2	Equipment	Y-50	11	52	-0.3	-0.9	153	-3
2	Equipment	Y-51	7	33	0.7	2.2	207	184
2	Equipment	Y-52	6	28	0.7	2.2	188	118
2	Equipment	Z-48	10	47	-0.3	-0.9	273	412
2	Floor	X-46	10	47	-0.3	-0.9	517	572
2	Floor	X-51	10	47	-0.3	-0.9	424	294
2	Floor	Y-48	4	19	-0.3	-0.9	473	440
2	Floor	Y-50	3	14	-0.3	-0.9	540	641
2	Floor	Y-52	4	19	-0.3	-0.9	394	204
2	Floor	Z-45	2	9	-0.3	-0.9	487	482
2	Floor	Z-47	-1	-5	-0.3	-0.9	523	590
2	Floor	Z-48	-1	-5	-0.3	-0.9	484	473
2	Floor	Z-49	4	19	0.7	2.2	434	323

Table A-9. Class 2 Equipment, Floor, and Walls Total and Removable Activity Levels (continued)

Unit	Surface	Measurement Location	Removable Activity				Total Beta Activity	
			Beta		Alpha			
			net cpm	dpm/100 cm^2	net cpm	dpm/100 cm^2	gross cpm	dpm/100 cm^2
2	Floor	Z-53	9	42	-0.3	-0.9	464	413
3	Equipment	S-47	8	38	-0.3	-0.9	219	225
3	Equipment	T-48	-2	-9	0.7	2.2	296	492
3	Equipment	U-45	1	5	-0.3	-0.9	179	87
3	Equipment	V-48	-4	-19	0.7	2.2	158	14
3	Equipment	V-50	7	33	-0.3	-0.9	153	-3
3	Equipment	W-45	5	24	-0.3	-0.9	206	180
3	Equipment	W-46	-4	-19	-0.3	-0.9	262	374
3	Equipment	W-48	-6	-28	-0.3	-0.9	152	-7
3	Equipment	W-51	-1	-5	-0.3	-0.9	163	31
3	Equipment	W-52	5	24	-0.3	-0.9	164	35
3	Floor	T-48	0	< 90	0.7	2.2	488	485
3	Floor	T-50	6	28	-0.3	-0.9	497	512
3	Floor	T-53	4	19	-0.3	-0.9	514	563
3	Floor	U-45	1	5	-0.3	-0.9	508	545
3	Floor	U-49	3	14	-0.3	-0.9	529	608
3	Floor	U-52	12	57	-0.3	-0.9	526	599
3	Floor	V-50	1	5	-0.3	-0.9	461	404
3	Floor	W-47	5	24	-0.3	-0.9	595	806
3	Floor	W-49	7	33	0.7	2.2	501	524
3	Floor	W-53	9	42	-0.3	-0.9	457	392
3	Walls	DD-53	8	38	-0.3	-0.9	382	570
3	Walls	S-46	6	28	-0.3	-0.9	545	1130 [1]
3	Walls	S-47	11	52	0.7	2.2	516	1031 [1]
3	Walls	S-48	10	47	0.7	2.2	523	1055 [1]
3	Walls	T-45	8	38	-0.3	-0.9	315	340
3	Walls	T-49	10	47	-0.3	-0.9	352	467
3	Walls	T-50	10	47	0.7	2.2	324	371
3	Walls	T-50	9	42	-0.3	-0.9	314	337
3	Walls	T-51	3	14	0.7	2.2	334	405
3	Walls	T-52	1	5	-0.3	-0.9	308	316
3	Walls	T-53	-2	-9	-0.3	-0.9	406	653

[1] Brick surface

Table A-10. Class 3 Floor and Roof Total and Removable Activity Levels

Surface	Measurement Location	Removable Activity				Total Beta Activity	
		Beta		Alpha			
		net cpm	dpm/100 cm^2	net cpm	dpm/100 cm^2	gross cpm	dpm/100 cm^2
Floor	BB-55	2	9	-0.3	-0.9	431	511
Floor	DD-44	8	38	-0.3	-0.9	313	137
Floor	DD-54	14	66	-0.3	-0.9	431	511
Floor	II-44	9	42	-0.3	-0.9	466	622
Floor	S-44	11	52	-0.3	-0.9	393	359
Floor	U-54	1	5	0.7	2.2	357	276
Floor	V-42	1	5	-0.3	-0.9	404	425
Floor	X-71	5	24	-0.3	-0.9	278	25
Floor	Y-56	7	33	0.7	2.2	444	552
Floor	Y-62	6	28	-0.3	-0.9	349	251
Roof	RF101	NA	NA	NA	NA	649	1601
Roof	RF102	NA	NA	NA	NA	279	302
Roof	RF103	NA	NA	NA	NA	350	551
Roof	RF104	NA	NA	NA	NA	305	393
Roof	RF105	NA	NA	NA	NA	366	608
Roof	RF106	NA	NA	NA	NA	421	801
Roof	RF107	NA	NA	NA	NA	270	270
Roof	RF108	NA	NA	NA	NA	291	344
Roof	RF109	NA	NA	NA	NA	235	147
Roof	RF110	NA	NA	NA	NA	178	-53

Table A-11. Exit Samples – Total Uranium Concentrations*

Sample Location	Sample #	Total Uranium (pCi/g)	Error (pCi/g)
Exit 1	M6S108	1.7	0.7
Exit 1	M6S110	1.5	0.6
Exit 3	M6S109	2.5	0.9
Exit 4	M6S106	1.8	0.8
Exit 4	M6S107	1.5	0.6
Exit 5	M6S105	0.7	0.4
Exit 18	M6S104	1.6	0.6
Exit 20	M6S103	1.3	0.5
Exit 21	M6S101	3.5	1.0
Exit 21	M6S102	3.3	1.1

* The error and detection limits for these samples are shown in table A-17.

Table A-12. Sludge – Total Uranium Concentrations*

Sample Location	Sample #	Total Uranium (pCi/g)	Error (pCi/g)
SL-1	M6L101	0.9	0.4
SL-2	M6L102	0.4	0.3
SL-3	M6L103	0.4	0.3
SL-4	M6L104	2.0	0.8

* The error and detection limits for these samples are shown in table A-17.

Table A-13. Floor Scrapings – Total Uranium Concentrations*

Sample Location	Sample #	Total Uranium (pCi/g)	Error (pCi/g)
BB-45	M6F101	4.0	1.1
CC-48	M6F102	2.6	1.2
BB-48	M6F103	1.7	0.8
CC-50	M6F104	1.5	0.7
CC-50	M6F204	1.2	0.6
BB-52	M6F105	1.9	0.7
X-46	M6F106	1.5	0.6
Y-50	M6F107	1.0	0.5
T-48	M6F108	1.6	0.7
U-52	M6F109	0.7	0.4
W-49	M6F110	0.9	0.5

* The error and detection limits for these samples are shown in table A-17.

Table A-14. Class 2 Dose Equivalent Rates

Measurement Location	Dose Equivalent Rate (mrem/hr)
CC-48	4
CC-50	4
BB-45	2
BB-48	3
BB-52	4
Y-50	6
X-46	3
W-47	3
U-52	5
T-48	3

Table A-15. Class 3 Dose Equivalent Rates

Measurement Location	Dose Equivalent Rate (urem/hr)
DD-79	3.5
U-80	3.5
U-70	2
Y-65	3
U-62	4
DD-61	3.5
P-43	4
U-42	3
DD-42	2
HH-42	3

Table A-16. Class 3 Exposure Rates At Exits

Measurement Location [1]	Exposure Rate (mR/h)
Exit 1	18.2
Exit 2	11.6
Exit 3	8.8
Exit 4	12.2
Exit 5	8.5
Exit 6	6.5
Exit 7	6.1
Exit 8	7.5
Exit 9	9.4
Exit 10	9.5
Exit 11	8.1
Exit 12	9.3
Exit 13	7
Exit 14	6.7
Exit 15	9.4
Exit 16	8.2
Exit 17	10.6
Exit 18	12.7
Exit 19	6.9
Exit 20	9.4
Exit 21	7.8
Exit 22	9
Exit 23	7

[1] Refer to Figure 1.

Table A-17. Analytical Data
(all results in pCi/g)

Sample Location	Sample #	Sample Type	U-234			U-235			U-238		
			Result	Error	Detection Limit	Result	Error	Detection Limit	Result	Error	Detection Limit
Dust											
OH-01	M4D209	Field duplicate	8.1	2.0	0.2	0.2	0.2	0.2	7.5	1.9	0.1
OH-03	M4D204	Field duplicate	56.1	14.3	0.4	3.5	1.4	0.2	56.5	14.4	0.2
OH-07	M4D206	Field duplicate	17.9	4.3	0.1	1.8	0.8	0.3	15.8	3.9	0.1
OH-01	M4D109	Grab	6.7	1.8	0.3	0.2	0.2	0.3	6.8	1.8	0.2
OH-02	M4D101	Grab	20.3	4.5	0.2	0.8	0.5	0.3	22.0	4.8	0.2
OH-03	M4D104	Grab	44.7	10.8	0.3	2.3	1.0	0.4	50.9	28.8	1.0
OH-04	M4D111	Grab	1.3	0.6	0.3	0.1	0.1	0.3	1.9	0.7	0.3
OH-05	M4D108	Grab	7.3	2.0	0.3	0.4	0.4	0.3	8.2	2.2	0.1
OH-06	M4D103	Grab	21.8	4.8	0.3	1.1	0.6	0.3	22.9	5.0	0.1
OH-07	M4D106	Grab	19.6	4.9	0.2	1.4	0.7	0.2	18.6	4.6	0.2
OH-08	M4D110	Grab	7.8	2.3	0.3	0.3	0.3	0.2	7.1	2.1	0.2
OH-09	M4D102	Grab	14.3	4.0	0.4	0.3	0.4	0.4	15.1	4.2	0.2
OH-10	M4D105	Grab	69.3	19.9	0.6	4.4	1.9	0.3	73.1	20.9	0.5
OH-11	M6D124	Grab	5.9	1.7	0.3	0.2	0.2	0.2	5.0	1.5	0.3
OH-12	M6D105	Grab	20.1	5.9	0.4	1.1	0.8	0.3	21.1	6.2	0.2
OH-13	M6D102	Grab	42.1	9.0	0.3	1.4	0.6	0.2	41.7	8.9	0.1
OH-14	M6D101	Grab	18.4	4.0	0.3	0.9	0.5	0.1	19.3	4.2	0.1
OH-15	M6D122	Grab	9.1	2.5	0.2	0.4	0.3	0.3	9.5	2.5	0.2
OH-16	M6D103	Grab	37.1	8.2	0.3	1.8	0.8	0.3	39.4	8.7	0.1
OH-17	M6D108	Grab	170.0	38.5	0.3	6.4	2.0	0.4	172.3	39.1	0.2
OH-18	M6D128	Grab	1.8	0.6	0.2	0.1	0.1	0.1	1.9	0.7	0.2
OH-19	M6D106	Grab	22.6	6.1	0.4	1.3	0.7	0.2	20.1	5.5	0.2
OH-20	M6D104	Grab	42.8	9.3	0.3	2.0	0.8	0.3	44.5	9.6	0.1
OH-21	M6D109	Grab	44.4	10.0	0.4	2.6	1.0	0.2	45.6	10.3	0.1
OH-22	M6D125	Grab	4.3	1.4	0.3	0.3	0.3	0.2	4.1	1.3	0.1
OH-23	M6D146	Grab	34.4	7.9	0.1	2.0	0.8	0.2	38.6	8.8	0.3
OH-24	M6D110	Grab	171.8	40.8	0.4	6.7	2.1	0.4	170.2	40.4	0.2
OH-25	M6D127	Grab	4.3	1.5	0.2	0.1	0.2	0.4	4.5	1.6	0.2
OH-26	M6D126	Grab	10.3	2.6	0.1	0.4	0.3	0.2	11.4	2.8	0.2
OH-27	M6D152	Grab	62.9	14.0	0.3	4.4	1.4	0.2	64.5	14.3	0.1
OH-28	M6D114	Grab	76.5	17.1	0.3	3.7	1.3	0.3	77.0	17.2	0.3

Table A-17. Analytical Data (continued)

Sample Location	Sample #	Sample Type	U-234			U-235			U-238		
			Result	Error	Detection Limit	Result	Error	Detection Limit	Result	Error	Detection Limit
OH-29	M6D129	Grab	1.7	0.6	0.2	0.1	0.2	0.1	1.9	0.6	0.2
OH-30	M6D147	Grab	12.5	3.7	0.4	0.9	0.6	0.4	13.4	3.9	0.4
OH-31	M6D115	Grab	38.3	8.3	0.1	1.5	0.7	0.2	37.4	8.1	0.1
OH-32	M6D130	Grab	3.5	1.0	0.1	0.1	0.2	0.2	3.3	1.0	0.1
OH-33	M6D131	Grab	1.3	0.5	0.1	0.0	0.0	0.2	1.2	0.5	0.1
OH-34	M6D151	Grab	21.7	5.7	0.6	1.3	0.9	0.4	24.3	6.3	0.6
OH-35	M6D113	Grab	20.1	4.7	0.3	1.5	0.7	0.4	17.7	4.2	0.3
OH-36	M6D132	Grab	5.4	1.5	0.2	0.2	0.2	0.1	6.0	1.6	0.2
OH-37	M6D145	Grab	4.3	1.6	0.3	0.2	0.2	0.2	5.5	1.9	0.2
OH-38	M6D156	Grab	26.2	6.2	0.3	0.5	0.4	0.2	27.8	6.6	0.3
OH-39	M6D135	Grab	1.3	0.5	0.1	0.0	0.0	0.1	1.9	0.7	0.1
OH-40	M6D134	Grab	1.1	0.5	0.1	0.2	0.2	0.1	1.0	0.5	0.2
OH-41	M6D150	Grab	4.9	1.6	0.2	0.0	0.1	0.5	5.0	1.6	0.2
OH-42	M6D121	Grab	17.1	4.3	0.3	0.9	0.5	0.2	17.6	4.4	0.3
OH-43	M6D136	Grab	1.7	0.7	0.1	0.3	0.3	0.1	2.1	0.7	0.1
OH-44	M6D148	Grab	2.3	0.8	0.2	0.2	0.2	0.1	2.7	0.9	0.2
OH-45	M6D116	Grab	15.0	5.4	0.3	0.9	0.8	0.8	15.3	5.5	0.3
OH-46	M6D138	Grab	2.7	0.9	0.3	0.3	0.3	0.3	3.2	1.0	0.2
OH-47	M6D137	Grab	5.4	1.8	0.4	0.3	0.3	0.4	6.7	2.1	0.2
OH-48	M6D117	Grab	9.5	2.6	0.2	0.3	0.3	0.4	9.7	2.6	0.2
OH-49	M6D120	Grab	5.1	1.4	0.1	0.2	0.2	0.2	5.3	1.4	0.1
OH-50	M6D139	Grab	1.4	0.6	0.2	0.0	0.0	0.1	1.2	0.5	0.1
OH-51	M6D149	Grab	7.8	2.2	0.2	0.4	0.4	0.4	7.7	2.2	0.2
OH-52	M6D119	Grab	4.7	1.9	0.5	0.2	0.3	0.6	3.5	1.5	0.5
OH-53	M4D114	Grab	8.1	1.9	0.2	0.5	0.3	0.2	8.7	2.0	0.1
OH-54	M4D112	Grab	8.8	2.5	0.3	1.2	0.7	0.2	10.8	2.9	0.3
OH-55	M4D113	Grab	7.2	2.0	0.3	0.2	0.2	0.3	6.5	1.8	0.1
OH-56	M4D115	Grab	5.7	1.6	0.4	0.1	0.2	0.2	6.1	1.7	0.1
OH-57	M4D116	Grab	3.4	1.2	0.4	0.3	0.3	0.2	3.6	1.3	0.2
OH-58	M6D144	Grab	5.6	1.7	0.3	0.1	0.2	0.2	6.0	1.8	0.3
OH-59	M6D140	Grab	0.7	0.4	0.3	0.1	0.2	0.3	0.7	0.4	0.1
OH-60	M6D142	Grab	2.9	1.0	0.3	1.0	0.5	0.3	3.1	1.0	0.1
OH-61	M6D143	Grab	2.4	0.8	0.3	0.2	0.2	0.3	2.3	0.8	0.2
OH-62	M6D141	Grab	1.4	0.6	0.1	0.1	0.1	0.2	1.2	0.6	0.1

Table A-17. Analytical Data (continued)

Sample Location	Sample #	Sample Type	U-234			U-235			U-238		
			Result	Error	Detection Limit	Result	Error	Detection Limit	Result	Error	Detection Limit
OH-E1	M6D111	Grab	171.6	41.4	0.2	8.2	2.6	0.2	181.0	43.6	0.2
OH-E10	M6D118	Grab	18.8	4.7	0.3	0.7	0.5	0.2	18.2	4.6	0.3
OH-E2	M6D112	Grab	52.2	11.8	0.3	2.1	0.9	0.3	50.3	11.4	0.1
OH-E3	M4D107	Grab	7.8	2.0	0.2	0.3	0.3	0.3	7.0	1.8	0.2
OH-E4	M6D123	Grab	1.7	0.7	0.1	0.0	0.1	0.4	1.5	0.6	0.1
OH-E5	M6D154	Grab	48.8	10.8	0.3	2.0	0.8	0.3	50.1	11.1	0.3
OH-E6	M6D107	Grab	18.8	4.5	0.3	0.8	0.5	0.3	20.0	4.8	0.1
OH-E7	M6D155	Grab	14.0	3.3	0.1	0.4	0.3	0.2	13.3	3.1	0.1
OH-E8	M6D133	Grab	8.0	2.4	0.3	0.3	0.3	0.2	6.4	2.0	0.2
OH-E9	M6D153	Grab	2.1	0.8	0.2	0.1	0.2	0.2	2.4	0.9	0.3
Soil											
6S-101	M6S101	Grab	1.6	0.7	0.4	0.1	0.2	0.2	1.7	0.7	0.2
6S-102	M6S102	Grab	1.7	0.8	0.4	0.3	0.3	0.2	1.3	0.7	0.2
6S-102	M6S202	Field duplicate	1.7	0.9	0.2	0.2	0.3	0.3	1.8	0.9	0.2
6S-103	M6S103	Grab	0.6	0.4	0.3	0.0	0.1	0.3	0.7	0.4	0.1
6S-104	M6S104	Grab	0.9	0.5	0.3	0.0	0.0	0.4	0.7	0.4	0.2
6S-105	M6S105	Grab	0.3	0.2	0.2	0.1	0.1	0.1	0.3	0.2	0.2
6S-106	M6S106	Grab	0.9	0.5	0.2	0.0	0.0	0.4	1.0	0.6	0.2
6S-107	M6S107	Grab	0.8	0.4	0.1	0.1	0.2	0.4	0.7	0.4	0.1
6S-108	M6S108	Grab	1.0	0.5	0.3	0.0	0.0	0.2	0.7	0.4	0.3
6S-109	M6S109	Grab	1.2	0.6	0.3	0.0	0.0	0.2	1.3	0.6	0.2
6S-110	M6S110	Grab	0.7	0.4	0.1	0.1	0.1	0.2	0.8	0.4	0.2
SS-101	MBS101	Grab	1.1	0.5	0.3	0.1	0.1	0.2	1.2	0.6	0.1
SS-102	MBS102	Grab	1.0	0.5	0.2	0.0	0.0	0.1	0.7	0.4	0.2
SS-103	MBS103	Grab	1.2	0.6	0.2	0.0	0.1	0.4	0.8	0.4	0.2
SS-104	MBS104	Grab	0.9	0.5	0.1	0.0	0.0	0.3	1.1	0.5	0.1
SS-105	MBS105	Grab	0.7	0.4	0.2	0.0	0.0	0.2	0.8	0.4	0.2
SS-106	MBS106	Grab	1.1	0.5	0.1	0.0	0.0	0.3	1.2	0.5	0.1
SS-107	MBS107	Grab	1.4	0.6	0.2	0.1	0.2	0.4	0.9	0.5	0.2
SS-108	MBS108	Grab	1.0	0.5	0.3	0.0	0.0	0.2	0.8	0.5	0.3
SS-109	MBS109	Grab	0.4	0.3	0.2	0.0	0.0	0.1	0.4	0.2	0.1
SS-110	MBS110	Grab	0.7	0.4	0.1	0.1	0.1	0.2	0.8	0.4	0.2
Sludge											
SL-3	M6L203	Field duplicate	0.7	0.6	0.5	0.0	0.0	0.4	0.3	0.4	0.5

Table A-17. Analytical Data (continued)

Sample Location	Sample #	Sample Type	U-234			U-235			U-238		
			Result	Error	Detection Limit	Result	Error	Detection Limit	Result	Error	Detection Limit
SL-1	M6L101	Grab	0.5	0.3	0.2	0.0	0.0	0.2	0.4	0.2	0.2
SL-2	M6L102	Grab	0.3	0.2	0.1	0.1	0.1	0.2	0.0	0.1	0.1
SL-3	M6L103	Grab	0.3	0.2	0.2	0.0	0.0	0.1	0.2	0.2	0.1
SL-4	M6L104	Grab	0.8	0.5	0.2	0.0	0.2	0.4	1.2	0.6	0.2
Floor Scrapings											
CC50	M6F204	Field duplicate	0.5	0.4	0.4	0.0	0.2	0.5	0.6	0.4	0.4
BB45	M6F101	Grab	1.8	0.7	0.3	0.0	0.0	0.3	2.2	0.8	0.1
BB48	M6F103	Grab	0.7	0.5	0.6	0.0	0.0	0.3	1.0	0.6	0.2
BB52	M6F105	Grab	0.9	0.5	0.2	0.2	0.2	0.2	0.9	0.5	0.1
CC48	M6F102	Grab	1.5	0.9	0.6	0.1	0.3	0.7	1.0	0.7	0.3
CC50	M6F104	Grab	0.8	0.5	0.4	0.0	0.0	0.4	0.7	0.5	0.2
T48	M6F108	Grab	0.7	0.4	0.4	0.1	0.1	0.2	0.8	0.5	0.2
U-52	M6F109	Grab	0.3	0.3	0.3	0.0	0.0	0.2	0.4	0.3	0.1
W-49	M6F110	Grab	0.5	0.3	0.2	0.1	0.1	0.2	0.4	0.3	0.1
X-46	M6F106	Grab	0.6	0.4	0.3	0.0	0.0	0.3	0.8	0.4	0.1
Y-50	M6F107	Grab	0.5	0.3	0.3	0.0	0.0	0.3	0.5	0.3	0.1

APPENDIX B

EVALUATION OF RISKS AND DOSES DUE TO URANIUM IN DUST ON OVERHEAD STRUCTURES

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APPENDIX B

EVALUATION OF URANIUM IN DUST ON OVERHEAD STRUCTURES

ESTIMATE OF HORIZONTAL SURFACE AREA ON OVERHEAD BEAMS

Introduction

The only significant levels of residual uranium from the AEC operations identified in the at the Madison Site, are in the dust accumulated on overhead horizontal surfaces. For purposes of determining the total quantity of residual uranium on these surfaces, the areas of upper surfaces of horizontal structure-support beams and other overhead horizontal surfaces were estimated as part of the field survey activities.

There are three distinct categories of overhead horizontal surfaces in that portion of the facility above the extrusion press. One of these categories is the horizontal support beams at approximately 25 ft above floor level. These beams run the entire 83.8 m (275 ft) width of the building. The main beams and cross-members at this level were determined to have a total surface area of approximately 53.1 m^2 in each 25-ft section between each major vertical support column. The combined surface area for the 14 total sections in the vicinity of the press is 743 m^2.

The second category of surfaces is beams and cross-members at approximately 36 ft above the floor level. These beams also run the entire width of the building. The beams and cross-members at this level were determined to have a total surface area of approximately 35.4 m^2 in each 25-ft section between each major vertical support column. The combined surface area for the 14 total sections is 496 m^2.

The third category of surfaces is supports, cross-members, and window ledges in the high-bay areas extending from the 45 ft to the 60-ft levels. There are 5 of these high-bay areas, extending the length of this portion of Building 6. The surface in each high-bay area was determined to be approximately 15.1 m^2 for a total surface of 75.5 m^2 in each 25-ft section between each major vertical support column. The combined surface area for the 14 total sections is 1,057 m^2.

The combined total area for all three categories is 2,296 m^2.

Dust Quantity

Survey records describe the dust on overhead surfaces as ranging from “dry to oily layers” except above the extrusion press where the dust was a “hard cake type material.” Dust thickness is reported to range from 0.64 to 0.95 cm. Total sample weight was not determined for the dust samples collected from 200 cm^2 areas on the overhead surfaces. Weight of material from this surface area was estimated, assuming a material density of 1.5 g/cm^3 and an average thickness of 0.8 cm. Based on these values, the quantity of dust on a 200 cm^2 area is estimated at 240 g.

URANIUM CONCENTRATIONS IN DUST FROM OVERHEAD STRUCTURES

Analytical results for dust samples are presented in Appendix A Table A-6. Samples 1 through 52, collected from the 25-ft level surfaces in the vicinity of the extrusion press, contained total uranium concentrations ranging from 2.3 to 348.7 pCi/g. Samples E-1 through E-10 from the 36-ft level contained uranium concentrations ranging from 3.5 to 360.8 pCi/g. Concentrations of U-234 and U-235 were compared with U-238 concentrations (see Figures B-1 and B-2). Based on the average ratios noted, the contaminant is confirmed to be natural uranium.

Uranium concentrations in dust were highest directly above the extrusion press. The distribution of uranium determined by this survey and the 1989 ORNL survey are generally similar; however, uranium concentrations determined by this recent survey were approximately 40 to 50 % lower than those from the ORNL survey.

The average total uranium concentration of the 52 samples on the 25-ft level surfaces is 48.6 ± 70.4 (1σ) pCi/g. The 95 % upper confidence level (UCL) for the mean uranium concentration in these samples is 65.1 pCi/g.

The average total uranium concentration of the 10 samples on the 36-ft level surfaces is 70.9 ± 108.1 (1σ) pCi/g. The 95 % upper confidence level (UCL) for the mean uranium concentration in these samples is 133.4 pCi/g.

Concentrations of U-238 in dust samples obtained from 10 additional locations in Buildings 6 and 4 are also presented in Appendix A Table A-6. Total uranium concentrations in these samples ranged from 1.8 to 20.9 pCi/g with an average concentration of 10.0 ± 6.3 (1σ) pCi/g.

EXPOSURE SCENARIOS

Exposure scenarios considered here assume that no remedial actions are taken to reduce, contain, or remove the contamination in the building, and no worker controls are implemented to reduce exposure to the contaminated dusts.

Two types of workers are considered for the dose and risk assessment: a facility worker on the floor level who is exposed daily for 8 hours, 250 days per year, and 25 years; and a utility worker in closer proximity to the contaminated overhead surfaces, pulling cables and changing light bulbs, for an estimated 20 hours per year. A crane operator's exposure and risk are assumed to be conservatively estimated by the utility worker scenario.

Inhalation of uranium-contaminated dust is assumed to be the major pathway for exposure and accounts for greater than 99% of the exposure. The airborne concentration the facility worker would be exposed to is modeled by assuming the entire volume of contaminated dust is released into the air over 25 years, with some general assumptions about ventilation removal. The airborne concentration for the utility worker is based on an assumed surface activity resuspension factor for work that is performed fairly close to the contaminated surface. An ingestion pathway is also

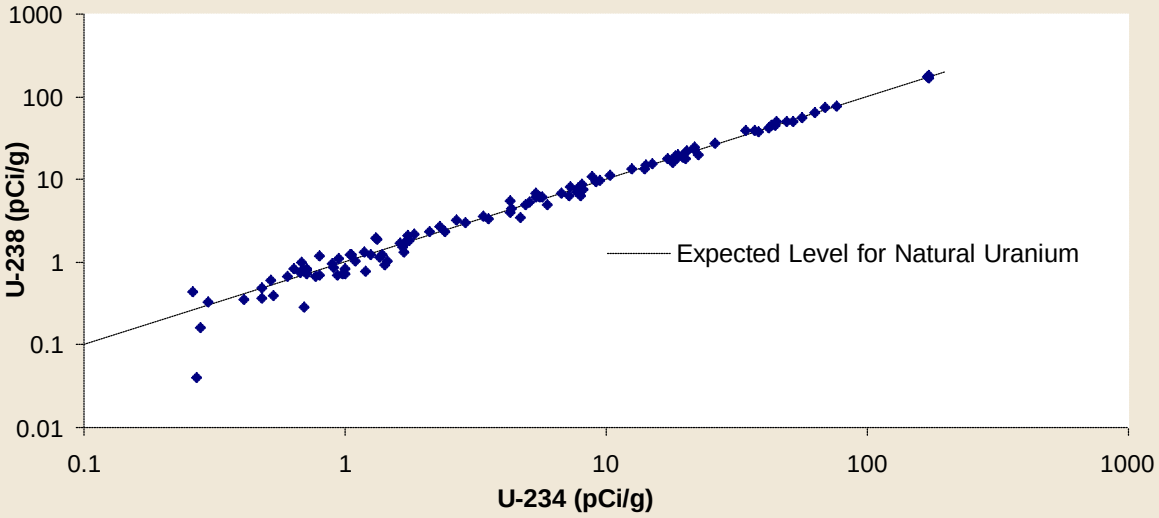

Figure B-1
Concentrations of U-234 and U-238
in Dust Samples from Overhead Beams

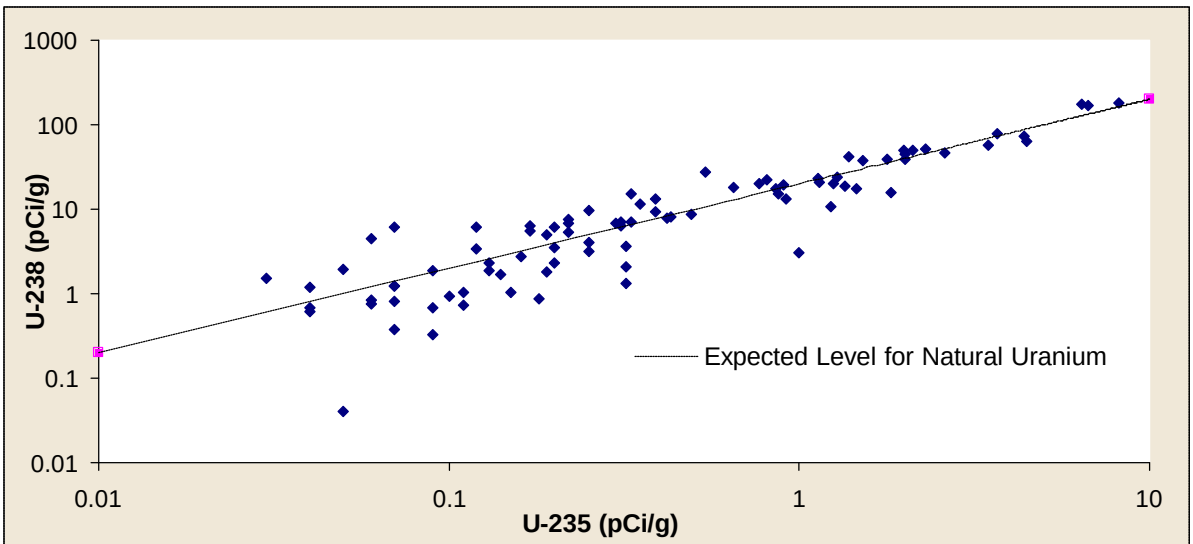

Figure B-2
Concentrations of U-235 and U-238
in Dust Samples from Overhead Beams

evaluated for the utility worker due to the workers contact with the surface contamination and assumed hot conditions which would cause the worker to touch his face and mouth thereby transferring contamination.

RISK ASSESSMENT FOR FACILITY WORKERS

Exposure parameters for the facility worker scenario are listed in Table B-1.

Table B-1. Facility Work Exposure Parameters

Exposure Parameter	Value	Source/Comments
Inhalation rate (m^3/day)	10	The 1997 Exposure Factor Handbook lists the mean hourly rate for adults as 1.0 m^3/hr for light activities and 1.6 m^3/hr for moderate activities. A mix of activities was used to represent the facility worker, activities for an 8 hour shift spent mostly in the contaminated zone. The value used was 10 m^3/day or 1.25 m^3/hr.
Exposure frequency (days/yr)	250	EPA (1991) Working 5 days per week for 50 weeks per year.
Exposure duration (yrs)	25	EPA (1991) Exposure duration for commercial/industrial use.
Inhalation class	Y	Chemical form inhalation class refers to the clearance half time from the pulmonary region of the lungs. Class Y is the most conservative uranium class.
Risk coefficients, Inhalation scope factors (risk/pCi)	U-234 = 1.4 E-8 U-235 = 1.3 E-8 U-238 = 1.24 E-8	EPA HEAST (1995) tabulated values. An average value of 1.3 E-8 will be used as the average for the natural uranium isotopic mixture at the Spectrulite facility.
Exposure-to-dose conversion factor for inhalation (Sv/Bq)	U-234 = 3.58 E-5 U-235 = 3.32 E-5 U-238 = 3.2 E-5	EPA (1988) values from Table 2.1 for Total Effective Dose Equivalent for class Y uranium isotopes. The dose conversion factor for U-234 will be conservatively used for the dose assessment.

Other assumptions used for this assessment include the following:

- Average ventilation rate of 3 air changes /hr.
- Material dispersion is limited to the volume area under the affected beams.
- The total contaminated surface area is 2300 m^2.
- All the contaminated surfaces are modeled at 40 feet above the floor; this represents an average height for the contaminated beams.
- The volume of the work area beneath the affected beams is 1.09 E5 m^3 [350 ft (14 column sections) $\times$ 275 ft $\times$ 40 ft]/[35.3 ft^3/m^3].
- All surfaces are assumed to be uniformly contaminated at the 36-ft level measured average 70.9 pCi/g. A thickness of 0.8 cm and a density of 1.5 g/cm^3 is also assumed.
- All uranium enters the air over a working lifetime of 25 years.

Calculations

Total Removable Activity

The total removable activity available for dispersion to the air is

$70.9\ pCi/g \times 1.5\ g/cm^3 \times 0.8\ cm \times 10^4\ cm^2/m^2 \times 2300\ m^2 = 2.0\ E\ 9\ pCi.$

Uranium Activity Dispersion to Air

All activity is assumed to be dispersed to the air over the 25 year working lifetime. For the purpose of this assessment, the activity is removed in equal quantities and the average hourly removal rate from the surface is calculated:

$[2.0\ E9\ pCi]/[25\ yr \times 365.25\ d/yr \times 24\ hr/d] = 9.1\ E3\ pCi/hr.$

Effect of Ventilation Removal and Calculation of Air Concentration

The concentration in air follows the linear first order kinetics equation

$C(t) = S\ [1 - exp\ (-rt)]/[rV]$

Where:
- $C(t)$ is the concentration at any time, t (pCi/m^3),
- S is the emission rate to the air = the removal rate from the surface (9.1 E3 pCi/hr),
- r is the ventilation removal rate (3/hr), and
- V is the volume of the air to which the contamination will be dispersed (1.09 E5 m^3).

Since we are interested in the steady state concentration (which is quickly reached), the equation reduces to

$C(steady\ state) = S/[rV] = 9.1\ E3\ pCi/hr/[3/hr \times 1.09\ E5\ m^3] = 2.8E\text{-}2 pCi/m^3.$

Calculation of Inhalation Intake of Activity

The total activity the facility worker will intake over the 25 year exposure duration is

$2.8\ E\text{-}2\ pCi/m^3 \times 10\ m^3/d \times 250\ d/yr \times 25\ yrs = 1.7\ E3\ pCi.$

Calculation of Excess Cancer Risk and TEDE

The radionuclide slope factor provides a lifetime cancer incidence (fatal and nonfatal) risk per unit inhalation. The excess cancer risk is calculated by

$1.7\ E3\ pCi \times 1.32\ E\text{-}8\ risk/pCi = 2.2\ E\text{-}5$ *lifetime attributable cancer risk.*

Since greater than 99% of the total effective dose equivalent, TEDE, is assumed to be due to inhalation; the committed effective dose equivalent (CEDE) obtained using the exposure-to-dose conversion factor from the Federal Guidance Report No. 11 (EPA 1988) is assumed to be equivalent to the TEDE

1.7 E3 pCi × 0.037 Bq/pCi × 3.58 E-5 Sv/Bq × 1 E5 mrem/Sv = 225 mrem/25 yrs.

RISK ASSESSMENT TO UTILITY WORKER FROM URANIUM IN DUST

Exposure parameters for the utility worker who will potentially work very close to the contaminated surfaces, are listed in Table B-2. Inhalation and ingestion are considered for risk and exposure estimates. Ingestion is added to this scenario because of the utility worker's contact with contaminated surfaces.

Table B-2. Utility Worker Exposure Parameters

Exposure Parameter	Value	Source/Comments
Inhalation rate (m^3/hr)	1.875	The 1997 Exposure Factors Handbook lists the mean hourly rate for adults as 1.0 m^3/hr for light activities, 1.6 m^3/hr for moderate activities, and 3.2 m^3/hr for heavy activities. Activities for utility workers are typically moderate activities, but the value was increased to account for brief periods of heavy activities. The value used was 1.875 m^3/hr.
Exposure frequency (hours/yr)	20	20 hours is an estimate for someone pulling utility cables or changing light bulbs.
Exposure duration (yrs)	25	EPA (1991) Exposure duration for the commercial/industrial use.
Inhalation class	Y	Chemical form inhalation class refers to the clearance half time from the pulmonary region of the lungs. Class Y is the most conservative uranium class.
Resuspension factor (m^{-1})	5 E-5	NRC (1998) The resuspension factor is noted to vary by 6 orders of magnitude depending on the conditions. The value of 5 E-5 is the value cited by the IAEA for operating nuclear facilities.
Transfer rate for ingestion of removable surface contamination (m^2/hr)	1 E-4	NRC (1998) This factor represents a plausible ingestion fraction.
Risk coefficients, inhalation slope factors (risk/pCi)	U-234 = 1.4 E-8 U-235 = 1.3 E-8 U-238 = 1.24 E-8	EPA HEAST (1995) tabulated values. An average value of 1.3 E-8 will be used as the average for the natural uranium isotopic mixture at the Spectrulite facility.
Risk coefficients, ingestion slope factors (risk/pCi)	U-234 = 4.4 E-11 U-235 = 4.7 E-11 U-238 = 6.2 E-11	EPA HEAST (1995) tabulated values. An average value of 5 E-11 will be used as the average for the natural uranium isotopic mixture at the Spectrulite facility.
Exposure-to-dose conversion factor for inhalation (Sv/Bq)	U-234 = 3.58 E-5 U-235 = 3.32 E-5 U-238 = 3.2 E-5	EPA (1988) values from Table 2.1 for TEDE for class Y uranium isotopes. The dose conversation factor for U-234 will be conservatively used for the dose assessment.
Exposure-to-dose conversion factor for ingestion (Sv/Bq)	U-234 = 7.66 E-8 U-235 = 7.19 E-8 U-238 = 6.88 E-8	EPA (1988) values from Table 2.2 for TEDE for uranium isotopes. The dose conversation factor for U-234 will be conservatively used for the dose assessment.

The resuspension factor is the ratio of the concentration in inhaled air to surface contamination concentration (pCi/m^3 per pCi/m^2). A draft letter report reviewing the parameter data for the NUREG/CR-5512 Building Occupancy Scenario describes the various resuspension factors found in literature. Some of the conclusions within this report follow:

- The range of resuspension factors is from 2 E-8 to 4 E-2 m^{-1}. The lower end represents low air flow conditions, with no mechanical stress. The upper end of the range represents vigorous work activities such as vigorous sweeping.

- NRC recommends using a value of 5 E-5 m^{-1} reported by the International Atomic Energy Agency (NRC 1998).

Physical factors affecting the indoor resuspension factor include how tightly the particles are bound to the surfaces; mechanical forces that will remove the particles from the surface such as walking, sweeping, scraping, etc; and strong air movements.

For the utility worker in the overhead regions at the Spectrulite facility, the forces that will increase the resuspension of uranium-contaminated dusts include brushing across the surfaces during work and mechanical vibrations from the crane. The contaminated dust particles in the overhead regions do not appear to be tightly bound. Review of the surface activity measurements before and after dust sampling reveals that the method for obtaining the dust samples frequently removed 100% of the activity and, on average, removed approximately 70% of the activity. This data is provided in Table A-5 of this report. The loose nature of the contamination increases the resuspension to air.

Calculations

Surface Activity

The mean surface activity is estimated using the mean concentration of 70.9 pCi/g, the assumed dust thickness of 0.8 cm and the assumed dust density of 1.5 g/cm3.

$$70.9 \text{ pCi/g} \times 1.5 \text{ g/cm}^3 \times 0.8 \text{ cm} \times 10^4 \text{ cm}^2/\text{m}^2 = 8.5 \text{ E5 pCi/m}^2$$

Airborne Concentration

The airborne concentration is calculated using the resuspension factor of 5 E-5 m^{-1}.

$$8.5 \text{ E5 pCi/m}^2 \times 5 \text{ E-5 m}^{-1} = 42.5 \text{ pCi/m}^3$$

Calculation of Inhalation and Ingestion Intake of Activity

The total activity the utility worker is assumed to intake through inhalation over the 20 hours of work per year and the 25-year exposure duration is calculated.

$$42.5 \text{ pCi/m}^3 \times 1.875 \text{ m}^3/\text{hr} \times 20 \text{ hours/yr} \times 25 \text{ yrs} = 4.0 \text{ E4 pCi } \textit{from inhalation}.$$

The intake of activity due to ingestion is similarly calculated using the transfer rate for ingestion.

$$8.5 \text{ E5 pCi/m}^2 \times 1 \text{ E-4 m}^2/\text{hr} \times 20 \text{ hours/yr} \times 25 \text{ yrs} = 4.3 \text{ E4 pCi } \textit{from ingestion}.$$

Calculation of Excess Cancer Risk and TEDE

The radionuclide slope factor provides a lifetime cancer incidence (fatal and nonfatal) risk per unit inhalation. The excess cancer risk is calculated by

4.0 E4 pCi × 1.32 E-8 risk/pCi = 5.3 E-4 *lifetime cancer risk from inhalation.*
4.3 E4 pCi × 5.0 E-11 risk/pCi = 2.1 E-6 *lifetime cancer risk from ingestion.*

The total *lifetime attributable cancer risk is, therefore, 5.3 E-4,* and exceeds the CERCLA risk range of 1 E-4 to 1 E-6.

TEDE from inhalation and ingestion is obtained using the exposure-to-dose conversion factors from Federal Guidance Report No. 11 (EPA 1988)

4.0 E4 pCi × 0.037 Bq/pCi × 3.58 E-5 Sv/Bq × 1 E5 mrem/Sv = 5.3 E3 mrem/25 yrs
4.3 E4 pCi × 0.037 Bq/pCi × 7.66 E-8 Sv/Bq × 1 E5 mrem/Sv = 1.2 E1 mrem/25 yrs

For comparison with the 25 mrem/yr annual TEDE criterion in 10 CFR 20 subpart E, the above equates to an annual TEDE of 210 mrem/yr, exceeding the dose criterion.

SUMMARY OF RESULTS AND CONCLUSIONS

Table B-3 below summarizes the results. The conclusion of this assessment is that the uranium present in the dust on overhead structures of Buildings 6 and 4, in the vicinity of the extrusion press, does not pose an unacceptable potential risk to facility workers. However, the utility worker scenario does result in unacceptable potential risks.

Table B-3. Summary of Risk and TEDE

	Comparison Criteria	**Scenario**	
		Facility Worker	**Utility Worker**
Excess lifetime cancer risk	CERCLA: 10^{-4} – 10^{-6}	2.2 E-5	5.3 E-4
TEDE, mrem/yr	25 mrem/yr	9	210

It should be noted that the calculated values contain some conservatism.

- The calculation for the facility worker assumed that the dust would be completely dispersed into the air over a period of 25 years. However, uranium is still present on these surfaces, over 35 years since it was deposited there. Assuming complete dispersal over a 25-year period overestimates the potential exposure.

- The calculation for the facility worker also assumes that the contamination would only be dispersed within the portion of the building immediately below the contaminated overhead surfaces. There are no physical boundaries separating this portion of the building from adjacent portions, and any material released into the air would be dispersed throughout a larger volume of air than was assumed for the calculation.

- In addition, if dust was easily dispersible, dust-sampling activities might have resulted in airborne contamination. Personal air samplers, worn during dust sampling, did not

indicate any positive airborne uranium concentrations (less than detection limit of 1.7×10^{-12} μCi/ml) right at the location of the sampling activities. Since sampling was conducted by personnel trained in contamination control practices, the low air samples may be due in part to the diligence of the sampling team.

- Finally, contamination on equipment and floor surfaces would be expected, if dust is being dislodged from the overhead surfaces. Neither the 1989 ORNL survey nor the survey described in this report identified evidence of significant uranium contamination on ground-level surfaces in this section of the building.

The utility worker scenario results are dependent on the surface activity resuspension and exposure duration. Dose and risk are directly proportional to these factors. Surface activity resuspension factors have recently received a great deal of discussion and are noted to vary by six orders of magnitude.

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APPENDIX C

RESULTS OF JULY 1999 INVESTIGATION OF HAZARDOUS CONSTITENTS IN DUST AND RADIOACTIVITY IN OUTSIDE SOILS

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APPENDIX C

RESULTS OF JULY 1999 INVESTIGATION OF HAZARDOUS CONSTITUENTS IN DUST AND RADIOACTIVITY IN OUTSIDE SOILS

PURPOSE AND SCOPE

This memo summarizes the data collection techniques and analytical results of waste characterization of dust and radiological characterization of soils outside the door at the corner of Buildings 4 and 7 at the Madison Site. This evaluation supplements previous investigations of the facility. Samples were taken in Building 6, primarily the interior beam surfaces in the vicinity of the extrusion press, where a previous survey identified residual uranium. Figure 1 shows a layout of the facility. The results of the previous assessment indicated that the dust on the beams will require remediation. Waste characterization data were collected to properly evaluate the disposition of this dust. Soil samples were collected to determine the source of elevated radiation readings identified in the previous investigation. The in-situ gamma spectroscopy measurements were collected to confirm or refute whether the AEC contaminant (uranium) was present in these soils.

SUMMARY OF FIELD ACTIVITIES

Five dust samples were collected from overhead beams inside of Building 6 on July 26, 1999 at the locations identified in Figure C-1. These sample locations were chosen because they represent areas where uranium was found in the dust on overhead beams. The sample locations are representative of the remediation waste stream if the decision in the ROD is to remove dust from the buildings. All of these samples were collected with a dedicated stainless steel spoons/scoops and placed directly into sample containers. Chemical analyses were performed in an offsite laboratory. The samples were packaged according to SAIC FTP 650 - Packing and Shipping of Environmental Samples.

Four of these samples, MAD00001, MAD00002, MAD00003, and MAD00004, were bias discrete samples and analyzed for toxic characteristic leaching procedure (TCLP) metals as specified in 40 CFR 261.24. The fifth sample consisted of a composite of the four samples and was analyzed for the full waste characterization parameters: TCLP, pH, ignitability, reactivity (including reactive sulfide and cyanide). In addition, this sample included volatile organic compound analysis.

Four measurements were taken near an underground sump using the ISOCS calibrated In-Situ Gamma Spectroscope system. The measurements were taken between Buildings 6 and 7, where Buildings 4 and 7 join. One measurement was taken at a 180° field-of-view encompassing approximately 100m². The remaining 3 measurements were taken at a 90° field-of-view encompassing approximately 1m² (all within the 180° field-of-view referenced previously) (See Figure 3). These are bias, discrete locations chosen because elevated radiation readings were identified in the previous investigations. Soil samples collected during the RI were not sufficient to account for the elevated exposure rate measurements identified at this location. These samples were analyzed for isotopic uranium (excluding U-234), isotopic thorium, isotopic radium, protactium-231, actinium-227, potassium -40, and cesium-137.

SUMMARY OF LABORATORY RESULTS

The sample results for the dust samples, MAD00001, MAD00002, MAD00003, MAD00004, MAD00005 are listed in Table 1. One hundred percent of these data have been validated in accordance with the data management plan and related procedures.

The results were compared to the requirements for determination of a hazardous waste and PCB waste in accordance with 40 CFR 261 Subpart C and 40 CFR 761, respectively. Based on this review, samples MAD00002 and MAD00004 exhibit a characteristic for lead. The TCLP limit for lead is 5 mg/L. No other hazardous constituents were identified. Samples MAD00001 and MAD00003 are not characteristic hazardous waste.

Sample MAD00005 is the composite sample. It was not analyzed for TCLP metals. Based on the analyses this sample is not characteristic hazardous waste or PCB waste. Soil measurements are shown in Table 2.

The radiological contaminant of concern for the Madison Site is pure natural uranium that was released during uranium metal forming operations (USACE 1998). None of the FUSRAP process related radionuclides exceeded the MDA, as seen in Table 2. Thorium-232 was the only radionuclide to exceed the MDA. The thorium is associated with other licensed activities at the facility and is not associated with AEC processes that FUSRAP is authorized to remediate. The AEC processes only involved uranium. The thorium is associated with the production of magnesium-thorium alloys and other products not associated with the AEC processes.

SUMMARY AND RECOMMENDATIONS

The waste characterization data indicate that the dust exhibits a characteristic hazardous waste for lead in two of the four samples analyzed. However, the composite sample did not exhibit the characteristic. The design should provide for proper treatment and disposition of this waste in accordance with 40 CFR 268 and appropriate state regulations and facility waste acceptance criteria. The radiological analyses show that none of the FUSRAP related radionuclides are present in the soils measured at this facility. However, this waste dust does have a radioactive component and must be disposed of appropriately.

REFERENCES

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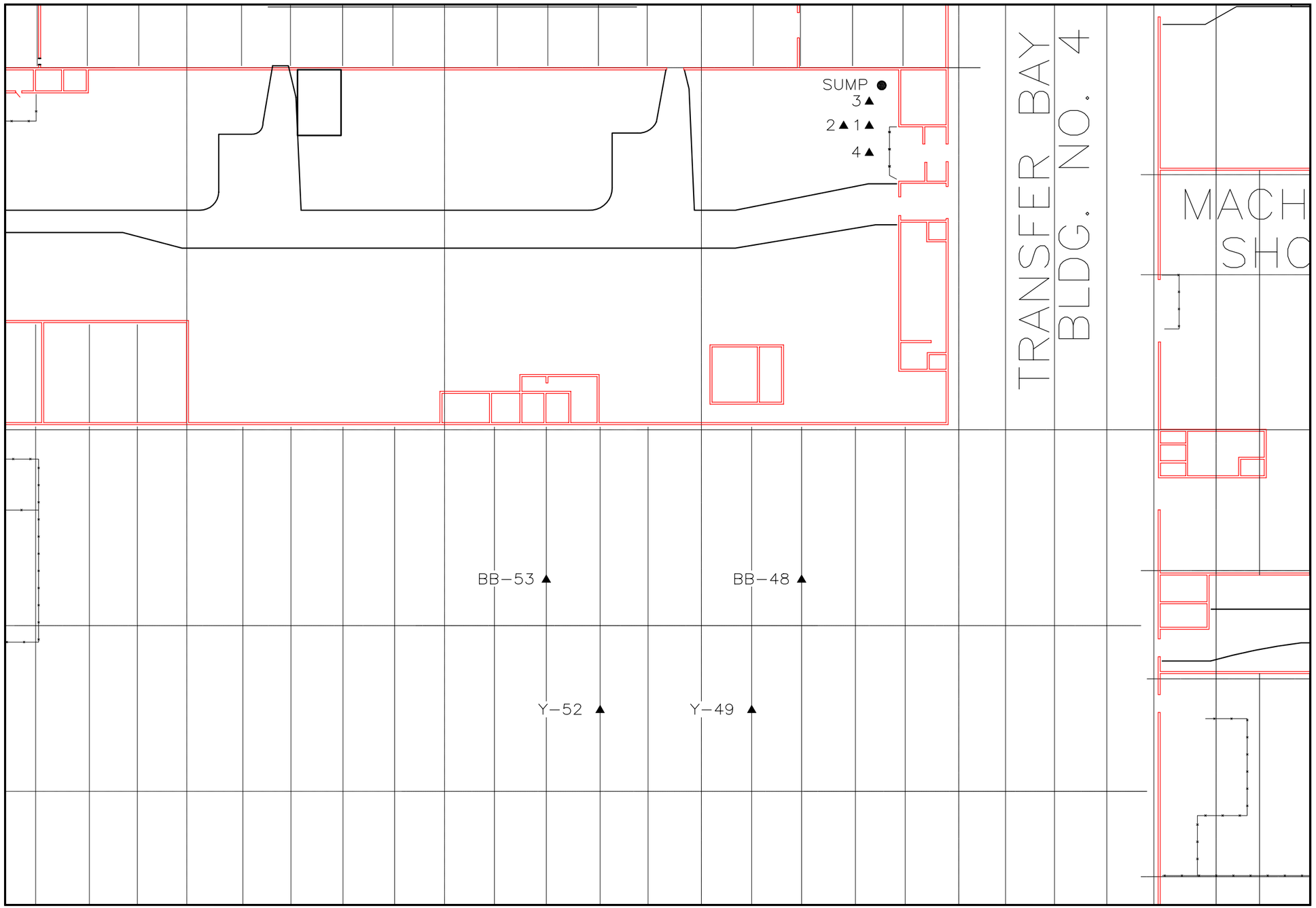

LEGEND:

- ROAD
- WALLS
- FENCE
- WALKWAY
- BUILDINGS
- COLUMN LINE
- SUMP PUMP
- SAMPLE LOCATION

0 125 250 500

SCALE: 1" = 250'

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Remedial Investigation
Madison, Illinois

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Figure C-1. Sample Locations

Table 1. Analytical Results for Dust Samples Collected July 25, 1999

Station	Sample ID	Chemical	Results	Units	Qualifier	Detection Limit
BB-53	MAD00002	Mercury	1.1	UG/L	=	0.8
BB-53	MAD00002	Arsenic	9.2	UG/L	U	1200
BB-53	MAD00002	Barium	72.2	UG/L	=	800
BB-53	MAD00002	Cadmium	324	UG/L	=	20
BB-53	MAD00002	Chromium	206	UG/L	=	40
BB-53	MAD00002	Copper	5450	UG/L	=	100
BB-53	MAD00002	Lead	23000	UG/L	=	400
BB-53	MAD00002	Selenium	15.6	UG/L	U	1000
BB-53	MAD00002	Silver	2.8	UG/L	U	40
BB-53	MAD00002	Zinc	79300	UG/L	=	800
BB-48	MAD00001	Mercury	0.85	UG/L	=	0.8
BB-48	MAD00001	Arsenic	7.6	UG/L	U	1200
BB-48	MAD00001	Barium	285	UG/L	=	800
BB-48	MAD00001	Cadmium	8.4	UG/L	U	20
BB-48	MAD00001	Chromium	13.7	UG/L	=	40
BB-48	MAD00001	Copper	705	UG/L	=	100
BB-48	MAD00001	Lead	256	UG/L	=	400
BB-48	MAD00001	Selenium	15.6	UG/L	U	1000
BB-48	MAD00001	Silver	2.8	UG/L	U	40
BB-48	MAD00001	Zinc	241	UG/L	=	80
Y-49	MAD00003	Mercury	0.56	UG/L	=	0.8
Y-49	MAD00003	Arsenic	7.6	UG/L	U	1200
Y-49	MAD00003	Barium	102	UG/L	=	800
Y-49	MAD00003	Cadmium	12.8	UG/L	=	20
Y-49	MAD00003	Chromium	9.6	UG/L	=	40
Y-49	MAD00003	Copper	744	UG/L	=	100
Y-49	MAD00003	Lead	179	UG/L	=	400
Y-49	MAD00003	Selenium	15.6	UG/L	U	1000
Y-49	MAD00003	Silver	2.8	UG/L	U	40
Y-49	MAD00003	Zinc	376	UG/L	=	80
Y-52	MAD00004	Mercury	0.61	UG/L	=	0.8
Y-52	MAD00004	Arsenic	7.6	UG/L	U	1200
Y-52	MAD00004	Barium	144	UG/L	=	800
Y-52	MAD00004	Cadmium	100	UG/L	=	20
Y-52	MAD00004	Chromium	916	UG/L	=	40
Y-52	MAD00004	Copper	2710	UG/L	=	100
Y-52	MAD00004	Lead	30400	UG/L	=	400
Y-52	MAD00004	Selenium	15.6	UG/L	U	1000
Y-52	MAD00004	Silver	2.8	UG/L	U	40
Y-52	MAD00004	Zinc	43000	UG/L	=	80
composite	MAD00005	Vinyl Chloride	100	UG/L	U	100
composite	MAD00005	1,1-Dichloroethene	50	UG/L	U	50
composite	MAD00005	Chloroform	50	UG/L	U	50
composite	MAD00005	1,2-Dichloroethane	50	UG/L	U	50
composite	MAD00005	2-Butanone	200	UG/L	U	200
composite	MAD00005	Carbon Tetrachloride	50	UG/L	U	50
composite	MAD00005	Trichloroethene	50	UG/L	U	50

Table 1. Analytical Results for Dust Samples Collected July 25, 1999 (Cont'd)

Station	Sample ID	Chemical	Results	Units	Qualifier	Detection Limit
composite	MAD00005	Benzene	50	UG/L	U	50
composite	MAD00005	Tetrachloroethene	50	UG/L	U	50
composite	MAD00005	Chlorobenzene	50	UG/L	U	50
composite	MAD00005	Pyridine	50	UG/L	U	50
composite	MAD00005	1,4-Dichlorobenzene	50	UG/L	U	50
composite	MAD00005	2-Methylphenol	50	UG/L	U	50
composite	MAD00005	m+p Methylphenol	50	UG/L	U	50
composite	MAD00005	Hexachloroethane	50	UG/L	U	50
composite	MAD00005	Nitrobenzene	50	UG/L	U	50
composite	MAD00005	Hexachlorobutadiene	50	UG/L	U	50
composite	MAD00005	2,4,6-Trichlorophenol	50	UG/L	U	50
composite	MAD00005	2,4,5-Trichlorophenol	50	UG/L	U	50
composite	MAD00005	2,4-Dinitrotoluene	50	UG/L	U	50
composite	MAD00005	Hexachlorobenzene	50	UG/L	U	50
composite	MAD00005	Pentachlorophenol	250	UG/L	U	250
composite	MAD00005	Lindane	0.5	UG/L	U	0.5
composite	MAD00005	Heptachlor	0.5	UG/L	U	0.5
composite	MAD00005	Heptachlor Epoxide	0.5	UG/L	U	0.5
composite	MAD00005	Endrin	0.5	UG/L	U	0.5
composite	MAD00005	Methoxychlor	1	UG/L	U	1
composite	MAD00005	Chlordane	5	UG/L	U	5
composite	MAD00005	Toxaphene	20	UG/L	U	20
composite	MAD00005	2,4-D	40	UG/L	U	40
composite	MAD00005	Silvex	10	UG/L	U	10
composite	MAD00005	PCB-1016	3400	UG/KG	U	3400
composite	MAD00005	PCB-1221	3400	UG/KG	U	3400
composite	MAD00005	PCB-1232	3400	UG/KG	U	3400
composite	MAD00005	PCB-1242	3400	UG/KG	U	3400
composite	MAD00005	PCB-1248	3400	UG/KG	U	3400
composite	MAD00005	PCB-1254	8100	UG/KG	=	3400
composite	MAD00005	PCB-1260	3400	UG/KG	U	3400
composite	MAD00005	Cyanide	3.8	MG/KG	=	0.48
composite	MAD00005	Ignitability (Flashpoint)	60	DEG C	=	
composite	MAD00005	Free Liquids	0		U	
composite	MAD00005	pH	8.88	STD UNIT	=	
composite	MAD00005	Cyanide, reactive	0.05	MG/KG	U	0.05
composite	MAD00005	Sulfide, reactive	22.2	MG/KG	U	22.2
composite	MAD00005	TOX	156	MG/KG	=	52.3

Table 2 Results of Gamma Spectrometry Measurements taken July 1999

Sample ID	Analyte	Result > MDA	Error	Units	MDA
madison1	K-40	9.68	0.63	pCi/g	1.24
madison1	Cs-137			pCi/g	0.25
madison1	Ra-226	0.73	0.07	pCi/g	0.36
madison1	Ac-227			pCi/g	2.42
madison1	Ra-228	9.04	0.24	pCi/g	0.57
madison1	Th-228	9.04	0.24	pCi/g	0.57
madison1	Th-232	9.04	0.24	pCi/g	0.57
madison1	Th-230			pCi/g	76.70
madison1	Pa-231			pCi/g	9.67
madison1	U-235			pCi/g	2.14
madison1	U-238			pCi/g	7.00
madison1	Am-241			pCi/g	0.76
madison2	K-40	13.25	0.90	pCi/g	2.03
madison2	Cs-137		0.06	pCi/g	0.19
madison2	Ra-226	0.85	0.09	pCi/g	0.34
madison2	Ac-227			pCi/g	2.40
madison2	Ra-228	13.91	0.33	pCi/g	0.54
madison2	Th-228	13.91	0.33	pCi/g	0.54
madison2	Th-232	13.91	0.33	pCi/g	0.54
madison2	Th-230			pCi/g	78.00
madison2	Pa-231	10.63	2.96	pCi/g	9.52
madison2	U-235			pCi/g	2.37
madison2	U-238			pCi/g	7.21
madison2	Am-241			pCi/g	0.77
madison3	K-40	11.65	1.23	pCi/g	2.55
madison3	Cs-137			pCi/g	0.43
madison3	Ra-226			pCi/g	0.85
madison3	Ac-227			pCi/g	2.87
madison3	Ra-228	4.12	0.26	pCi/g	0.84
madison3	Th-228	4.12	0.26	pCi/g	0.84
madison3	Th-232	4.12	0.26	pCi/g	0.84
madison3	Th-230			pCi/g	98.30
madison3	Pa-231			pCi/g	15.30
madison3	U-235			pCi/g	2.89
madison3	U-238			pCi/g	9.05
madison3	Am-241			pCi/g	0.99
madison4	K-40	14.02	1.18	pCi/g	1.79
madison4	Cs-137	0.33	0.08	pCi/g	0.25
madison4	Ra-226	0.69	0.12	pCi/g	0.46
madison4	Ac-227			pCi/g	2.72
madison4	Ra-228	5.16	0.29	pCi/g	0.75
madison4	Th-228	5.16	0.29	pCi/g	0.75
madison4	Th-232	5.16	0.29	pCi/g	0.75
madison4	Th-230			pCi/g	99.80
madison4	Pa-231			pCi/g	14.10
madison4	U-235			pCi/g	2.83
madison4	U-238			pCi/g	9.43
madison4	Am-241			pCi/g	1.00