New World Technology

FINAL REPORT

WALTER REED ARMY INSTITUTE OF RESEARCH BUILDING 40 FORMER RESEARCH REACTOR ROOMS DECOMMISSIONING

Project No. USA 00-005

Revision 1
April 10, 2001

Prepared by:

New World Technology
448 Commerce Way
Livermore, CA 94550
(925) 443-7967

FINAL REPORT

WALTER REED ARMY INSTITUTE OF RESEARCH BUILDING 40 FORMER RESEARCH REACTOR ROOMS DECOMMISSIONING

Project No. USA 00-005

Revision 1
April 10, 2001

Prepared by: Dan Spicuzza _______ 4/10/01
Dan Spicuzza, NWT Project Manager — Date

Reviewed by: Thomas E Dias _______ 6/8/01
Thom Dias, NWT V.P. Operations — Date

Approved by: WC Haney _______ 6/8/01
Bill Haney, NWT Chief Operations Officer — Date

Prepared by:

New World Technology
448 Commerce Way
Livermore, CA 94550
(925) 443-7967

TABLE OF CONTENTS

Section **Page**

List of Figures, Tables, Appendixes and Attachments

ACRONYMS AND ABBREVIATIONS

AOC	Areas of concern
ALARA	As Low As Reasonably Achievable
Bkg	background
BRAC	Base Realignment and Closure
C-14	Carbon-14
cal	calibration
cm	centimeter
cm^2	square centimeter
Co-60	Cobalt-60
cpm	counts per minute
Cs-137	Cesium-137
DAC	Derived Air Concentration
DOT	Department Of Transportation
dpm	disintegrations per minute
dpm/$100cm^2$	disintegrations per minute per 100 square centimeters
eff	efficiency
Eu-152	Europium-152
Eu-154	Europium-154
FOP	Field Operating Procedures
g	gram
H-3	Tritium
inst	instrument
IAW	In Accordance with
LLD	Lower Level of Detection
LSC	Liquid Scintillation Counting
MARSSIM	Multi Agency Radiation Survey and Site Investigation Manual (NUREG-1575)
MDA	Minimum Detectable Activity
MDCR	Minimum Detectable Count Rate
NIST	National Institute of Standards and Technology
NRC	Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Commission Technical Report
NWT	New World Technology
OSC	U. S. Army Operations Support Command
OSHA	Occupational Safety and Health Administration
PACM	Presumed Asbestos Containing Material means thermal system insulation and surfacing material found in buildings constructed no later than 1980
PEL	Permissible Exposure Limit
pCi	pico Curie
PPM	parts per million
PM	Project Manager
PPE	Personal Protective Equipment
Sr-90	Strontium-90

SHASP	Site Health and Safety Plan
TBE	To Be Established
TSM	Tailgate Safety Meeting
USA	U. S. Army
μR/hr	microroentgen per hour
μCi	micro curie

1.0 BACKGROUND INFORMATION

New World Technology (NWT) was contracted by the U.S. Army, Operations Support Command (OSC) to decontaminate and decommission the former reactor facility gas recombination room and adjacent rooms, which are located in the basement of Building 40 at the Walter Reed Army Institute for Research, which is located at the Walter Reed Army Medical Center in Washington, D.C.

This decommissioning action involves the former reactor facility gas recombination room and irradiation room, which are located in the basement of Building 40. WRAIR personnel decommissioned the reactor there in the 1970s. Other historical documentation concerning the decommissioning activities is not available at this time. All information as to the status of the rooms and other radiological conditions has been obtained from a characterization survey performed by Aguirre Engineers, Inc. in August 1998 under contract to the Operations Support Command.

The decontamination action was based on a characterization survey performed by High Hazard Services in 1998 detailing locations of the contamination.

Building 40 housed a reactor used for research, irradiation of materials and the production of isotopes. Walter Reed personnel decommissioned the reactor in the 1970s. The Gas Recombiner room, and the adjacent specimen rooms (GR, R1, R2, and R3), were sealed by welding the steel access door closed at some point in the past. These areas were opened during the characterization survey for the first time in 25 years. Little additional information on the physical activities during the reactor removal or history of operation of the facility is available.

The licensee for this decommissioning action is the Walter Reed Army Medical Center NRC license. The radioactive materials license for the reactor facility has been terminated.

Work was performed under reciprocity with the Nuclear Regulatory Commission (NRC) under NWT's California Radioactive Materials License # 5363-01. A copy of NWT's Radioactive Material License and reciprocity documentation is presented in this report in Appendix Y.

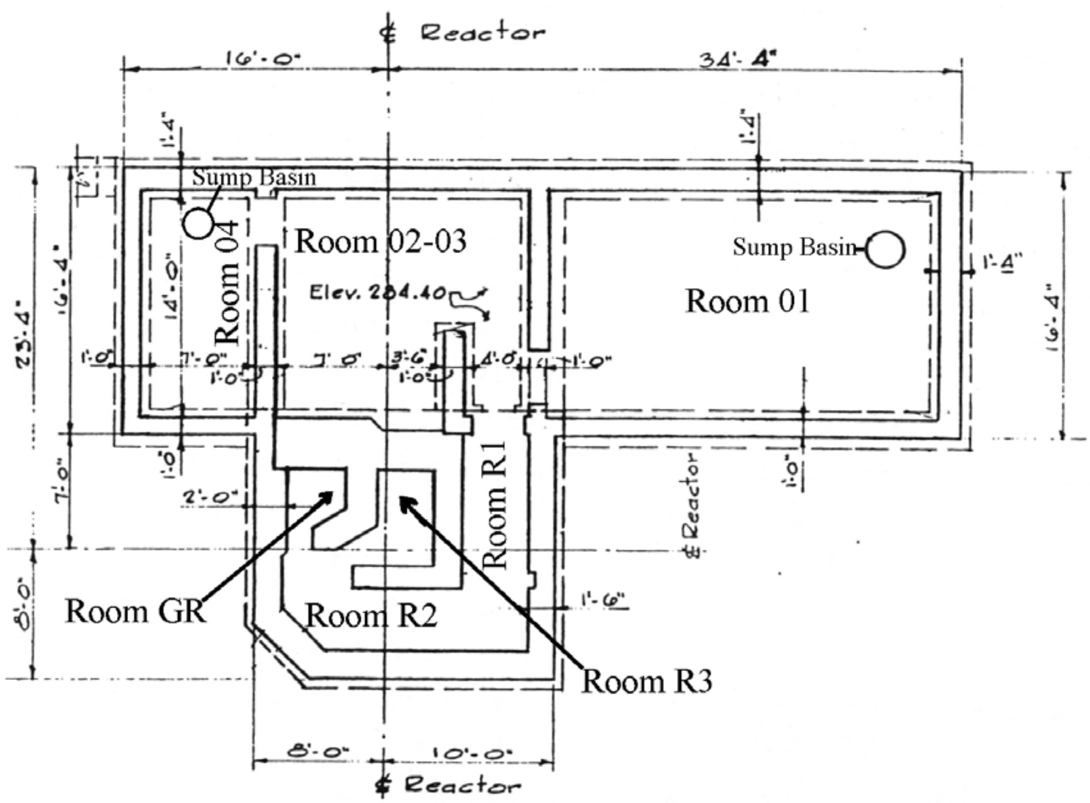
The objective of this action was release of the subject rooms for unrestricted reuse. Site specific release criteria and survey design were developed based on guidance detailed in the *Multi Agency Radiation Survey and Site Investigation Manual* (MARSSIM) (NUREG-1575). The specific Derived Concentration Guideline Limits (DCGL) are detailed in Section 4.2 of this report and are based on regulatory requirements that the Total Effective Dose Equivalent (TEDE) to an individual following remediation would not exceed 25 milli-Rem/year (mR/y). The statistically based release survey for the facility demonstrates compliance with the dose requirement.

2.0 SITE INFORMATION

2.1 Site Description

The general construction of the rooms is of concrete, steel and lead. The subject rooms were no longer utilized for research or any work involving radioactive materials. The original components and equipment were removed and residual contamination (Cs^{137} and Sr^{90}), both fixed and loose, on room surfaces, utilities and various wall penetrations existed. The rooms were given designation numbers of 01 through 04, R1, R2, R3 and GR (see Figure 1 below).

Figure 1, Room Locations and Identification



2.2 Site Conditions at Time of Final Status Survey

The wall, floor, and ceiling surfaces of Rooms R1, R2, R3, and GR were decontaminated using sandblasting and grinding methods. All surfaces were free of dust and dirt following the vacuuming of the surfaces with a HEPA filtered vacuum cleaner.

3.0 REMEDIATION/DECONTAMINATION ACTIVITIES

3.1 Prerequisites

Once office spaces, instrumentation, and equipment and was mobilized and set up, dosimetry (TLD's) was issued to on site personnel to monitor external whole body radiation exposure. Entry bioassay samples were collected from all on site personnel to provide baseline data for the monitoring of internal radiation exposure.

Radiological scoping surveys were performed in the rooms to provide data to generate a Radiation Work Permit which specified the activities to be performed and all radiological safety requirements for the work. The RWP also designated personal protective equipment (PPE) requirements for the specific tasks to be performed. All personnel assigned to the site work were required to read and understand the requirements prior to beginning work. A copy of the RWP is presented in this report in Appendix A.

A Decommissioning Plan (January, 2000) was developed by NWT for the work to be performed based upon the results of the Characterization Survey performed by Aguirre Engineers, Inc. in August 1998 under contract to the Operations Support Command.

3.2 Selection Of Personnel

The selection of supervisory personnel directing the survey was based upon their experience and familiarity with survey procedures and processes. Health Physics technicians performing the surveys were selected based upon their experience and qualifications such as survey experience in accordance with NUREG-1575 (MARSSIM, 2000).

Supervisory and survey personnel are required to have widespread experience with NWT's procedures as well as the requirements of NUREG-1575. Supervisory personnel must have prior experience with the radionuclide(s) of concern and the instrumentation used to detect the radionuclides present on the site. Survey personnel are required to have working knowledge of the use of all survey equipment used on the project.

A copy of the site personnel's resumes and qualifications are archived the NWT corporate office in Livermore, CA and are available for review upon request.

3.3 Daily Tailgate Safety Meetings

Daily tailgate safety meetings were held prior to work each day to discuss the planned activities for that day and address any safety/radiological concerns involved in that work. All NWT personnel attended these meetings. Copies of the safety meetings are archived at the NWT corporate office in Livermore, CA and are available for review upon request.

3.4 Asbestos Floor Tile Removal

The work performed involved the collection of wet floor tiles that were suspected of containing asbestos, Presumed Asbestos Containing Material (pre-1980 construction, therefore PACM).

The definition of PACM is: "Thermal system insulation and surfacing material found in buildings constructed no later than 1980", as defined in 29 CFR Part 1926.1101.

Radiologically contaminated PACM floor tiles and radiologically clean PACM floor tiles were present on the floors of Room R-1 (radiologically contaminated) and Room 02-03 (radiologically clean). Due to recent flooding of the rooms the floor tiles were easily removable by hand and placed into a double lined 55-gallon drum for disposal. The total square footage of tiles removed was approximately 75 square feet.

The work did not involve abatement, disturbance or removal activities as defined in 29 CFR 1910.1101; the governing regulation based upon the definitions of the work as defined in 29 CFR 1926.12.

The work conditions, wet materials, use of protective clothing, and the operations involving the PACM did not meet the definition of an abatement project. The act of picking up wet, whole floor tiles does not qualify as an abatement project especially due to the limited amount of tiles removed.

Additionally, the level of protection afforded the workers as it related to the other hazards of concern exceeded the requirements of a Class IV ACM project.

3.5 Vacuuming Of Surfaces

The surfaces of Rooms R1, R2, R3, and GR were then vacuumed with a HEPA filtered vacuum cleaner to remove the loose rust and loose surface contamination inside of the rooms.

3.6 Water Tight Hatch Removal/Decontamination

There were three watertight hatches that were removed and downsized for disposal. One leading to Room 04 from Room 02-03, one leading to Room R1 from Room 02-03, and one leading to Room R2 from Room R1. The hatches were downsized using a cutting torch. Respiratory protection and local HEPA filtered ventilation was used during cutting operations due to the presence of radioactive contamination and lead paint on the surfaces of the hatches. Once downsized, the pieces were placed into lined 55-gallon drums for disposal.

3.7 Room Isolation

Rooms R1, R2, R3, and GR were isolated from the rest of the rooms by sealing the doorway from Room 02-03 to Room R1 with a double layer of 6-mil plastic sheeting. Also, the doorway leading from Room R1 to Room R2 was sealed in the same manner. This created a double airlock to prevent migration of dust to the other rooms during decontamination operations.

3.8 HEPA Ventilation

500-CFM and 1900-CFM HEPA filtration units were set up to provide negative airflow inside of the rooms to be decontaminated. The 500-CFM HEPA unit was set up and connected to a penetration leading to Room GR from Room 02-03. The 1900-CFM HEPA unit was setup on the floor above Room R3 through the vent opening in the ceiling of Room R3. Access to the vent opening was made by removing the concrete in the floor of the room above Room R3 using a jackhammer. During this activity contaminated concrete was discovered above the vent opening. The contaminated concrete was placed into a 55-gallon drum for disposal. A sample of the contaminated concrete was taken and sent to an independent off site laboratory (Barringer Labs) for gamma spectroscopy and Sr-90 analysis. The results indicated the presence of Europium (Eu-152) at 12 pCi/g and Cobalt (Co-60) at 1.2 pCi/g, which are common radionuclides found in irradiated concrete and metal. The results of the sample are included in this report in Table 4 and Appendix F. Further investigations of the contaminated concrete were performed and are discussed in this report in Section 6.15.

3.9 Decontamination of Rooms R1, R2, R3, and GR Surfaces

Room surfaces were decontaminated using grinders and a sandblaster. Personnel inside of the rooms used Level C PPE ensemble with full-face negative pressure respiratory protection during grinding and sandblasting operations. Initially, each one hundred-pound bag of sandblast grit was recycled and reused to minimize the amount of waste generated.

Following gross decontamination of the room surfaces using the sandblast technique, small-localized areas that were found to be contaminated above the DCGL were decontaminated using electric angle grinders. The sandblast media was then vacuumed up with a HEPA filtered vacuum cleaner and then placed into a 55-gallon drum for disposal.

3.10 Decontamination of Room 02-03 Valve Boxes

Decontamination of the valve boxes on the west wall of Room 02-03 was performed by removal of the boxes using a hammer and chisel. The ends of the contaminated tubing were removed by drilling out the tubing with an electric high-speed drill.

3.11 Decontamination of Wall Penetrations

Wall penetrations were decontaminated using the sandblast techniques wherever practical. Where sandblasting was not practical the wall penetrations were decontaminated using gun-cleaning wire brushes attached to a high-speed electric drill.

3.12 Decontamination of Sump Well In Room R3

A sump well was present in the floor of Room R3. The well was approximately 7 1/2 feet in depth and 12 inches in diameter. Figure 2 of this report presents a diagram of the sump well. A contaminated metal liner was removed from the sump well. The sump wells surfaces were then decontaminated using an angle grinder attached to a long handle pole. Following decontamination, the well was vacuumed with a HEPA filtered vacuum cleaner and then wiped down with damp raps to remove the remaining dust. Following decontamination the sump well was then surveyed using a large area gas proportional detector and a thin windowed GM detector attached to a long handle to reach the surfaces to be surveyed. The results of the survey indicated that the sump well was decontaminated to below the DCGL. Copies of the surveys of the sump well are presented in this report in Appendix V.

3.13 Areas of Elevated Activity Above the DCGL

Two small areas of elevated activity were found in exposed concrete sections of the floors in Rooms GR and R3. The areas were approximately .5 m^2 in each room. Approximately three inches of concrete was removed from the surfaces of these areas in an attempt to decontaminate the areas below the DCGL unsuccessfully.

Samples of the concrete were obtained and sent to an off-site laboratory (Barringer Labs) for gamma spectroscopy analysis and Sr-90 analysis. The results of the samples indicated the presence of Cs-137 and Sr-90. Table A below summarizes the remaining contamination compared to the DCGL, sample results from the concrete and the Total Effective Dose Equivalent (TEDE) for the two rooms.

Table A
Elevated Activity Summary

	Room GR		**Room R3**	
	Cs-137	Sr-90	Cs-137	Sr-90
DCGL (dpm/100 cm^2) Maximum	9500	2900	9500	2900
Remaining Surface Activity (dpm/100 cm^2)	3,000		11,000	
Activity Concentration (pCi/g)	22	6.1	20	5.2
TEDE* (mR/y)	2.2	5.5 E-3	2.0	4.8 E-3

**TEDE is calculated based on the concrete sample activity, which produces a TEDE more restrictive than surface activity. Remaining surface activity is fixed.*

These areas of elevated activity above the DCGL were dose equivalent modeled using the *"RESRAD-BUILD Version 3.0 Modeling Code: A Computer Model for Analyzing the Radiological Doses Resulting from the Remediation and Occupancy of Buildings Contaminated with Radioactive Material"*, ANL/EAD/LD-3, Argonne National Laboratory, August 2000.

The purpose of the modeling was be to demonstrate that the site met the limiting dose criteria as defined for unrestricted use as required in 10 CFR 20.1403(b). RESRAD-BUILD was performed using default scenarios and modeling assumptions. These parameters assume more restrictive conditions (i.e., direct continuous exposure) than would be expected in the projected use of the facility.

The results of the RESRAD-BUILD analysis demonstrate that the TEDE from the remaining contamination is well below the dose restrictions for residual contamination for an unrestricted reuse of the areas.

The complete results of the RESRAD-BUILD data for Rooms GR and R3 are presented in this report in Attachments 1 and 2 respectively.

Results of the concrete samples are summarized at Table 4 with full results presented in this report in Appendix X.

3.14 Air Sampling

A low volume air sampler was positioned in Room 02-03 during decontamination activities.

The airborne activity action level for the project was 2.0 E-10 μCi/ml, which is 10% of the Derived Air Concentration (DAC) value for Sr-90 (most restrictive radionuclide). The DAC value for Sr-90 is presented in 10 CFR Part 20, Appendix B, Table 1, Column 3.

The air sample filters were counted on a Ludlum Model-2929 dual channel scaler for gross alpha-beta activity.

Air sample results were compared to the action level to verify that no release of radioactive materials to the outside rooms occurred during decontamination efforts. This data was also used to make decisions and evaluations for personnel protective equipment (PPE) requirements at the site.

The following formula was used when counting air samples:

$$\text{Air Sample Activity in uCi/ml} = \frac{\text{Gross CPM} - \text{Bkrd CPM}}{2.22\text{ E+6} * E * V * T * F * 2.832\text{E+4}}$$

Where:

V= Sample Volume in Milliliters
T= Sample Count Time in Minutes
E= Instrument Efficiency Expressed as a Decimal
F= Filter Efficiency
2.834E+4 = Conversion from Cubic Feet Milliliters if Necessary

The following formula was used to calculate the MDA of the air sample:

$$\text{MDA in uCi/ml} = \frac{2.71 + 4.65\ \text{SQRT}(R_B/T)}{2.22\text{ E+6} * E * V}$$

Where:

V= Sample Volume in Milliliters
T= Sample Count Time in Minutes
E= Instrument Efficiency Expressed as a Decimal
R_B = Background Count Rate in CPM

The range of the MDA for the air samples counted using the above formula during the project was:

~ 1.0 E-12 μCi/ml to ~ 2.4 E-12 μCi/ml

In order to account for the presence of radon daughters the half-life of the air sample activity was calculated using the formula below:

$$t_{1/2} = \frac{-.693(t)}{\ln \dfrac{\text{Final Activity in } \mu \text{Ci / ml}}{\text{Initial Activity in } \mu \text{Ci / ml}}}$$

Where:

$T_{1/2}$= Sample Half Life in Minutes
(t) = Time in Minutes Between Initial Count and Final Count
ln = Natural Logarithm

The results of the air samples indicated that there was no airborne radioactivity above the 10 % DAC value generated outside of the rooms being decontaminated. Air sample data is presented in this report in Appendix C.

3.15 Bioassay Program

Entry/Pre-job/baseline, and exit bioassay urine samples were obtained from personnel on the project. The samples were sent to the New World Technology laboratory in Livermore, California for analysis by Liquid Scintillation Counting (LSC) for the presence of beta emitting radiation. The results of the analysis are included in this report in Appendix D. No sample indicated results exceeding background concentrations.

3.16 Waste Packaging and Disposal

The waste volume of 81 cubic feet of material was packaged into eleven 55-gallon drums. The drums were surveyed, weighed, labeled and manifested in accordance with DOT regulations. A NWT OSC approved waste broker shipped five of the drums on 9/6/2000 to the Chem Nuclear Consolidation Facility (CNCF) located in Barnwell, SC for processing and disposal. OSC personnel at a later time shipped the other six drums to the CNCF. Copies of the waste manifests of the five drums shipped by the NWT OSC approved waste broker are presented in this report in Appendix T. A summary of the waste generated during the project is presented in this report in Table 5.

4.0 FINAL STATUS SURVEY/SAMPLING OBJECTIVE

4.1 Survey/Sampling Objective

The purpose of the Final Status Survey and sampling effort was to provide data and documentation to demonstrate that the site meets the established release criteria for unrestricted use.

4.2 Derived Concentration Guideline Limits (DCGL's)

Cesium (Cs^{137}) and Strontium (Sr^{90}) were the sole identified radioactive contaminants at the site.

The DCGL's as defined in the Decommissioning Plan (New World Technology, Dated January 2000) for Cs-137 and Sr-90 for use during this action are defined below:

4.2.1 Building Surfaces and Structures

a) 1,500 dpm/100 cm^2, averaged over 1 m^2 βγ
b) 2,900 dpm/100 cm^2, maximum βγ
c) 200 dpm/100 cm^2 removable βγ [1]

1 *As defined in: "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Materials (NRC 1987), Office of Nuclear Material Safety and Safeguards (NMSS)."*

As a comparison, the dose based screening values as defined in the Federal Register/Vol 63, Number 222 dated Wednesday, Nov 18, 1998 for Cs-137 and Sr-90 are presented below:

a) 28,000 dpm/100 cm^2, Cs-137 [1,2]
b) 8,700 dpm/100cm^2, Sr-90 [1,2]

1 Screening levels are based on the assumption that the fraction of removable surface contamination is equal to 0.1. For cases when the fraction of removable contamination is undetermined or higher than 0.1, users may assume, for screening purposes, that 100% of surface contamination is removable, and therefore the screening levels should be decreased by a factor of 10.

2 Units are disintegrations per minute per 100 square centimeters. 1 dpm is equivalent to 0.0167 becquerel (Bq). The screening values represent surface concentrations of individual

radionuclides that would be deemed in compliance with the 0.25 mSv/yr (25 mrem/yr) unrestricted release dose limit in 10 CFR 20.1402.

5.0 RADIOLOGICAL SURVEY METHODS AND INSTRUMENTATION

5.1 Radiological Survey Methods

5.1.1 Scanning

Prior to conducting any fixed measurements, surfaces were scanned to identify the presence of elevated direct radiation that might indicate residual gross activity or hot spots. The preferred distance between the detector and the surface measured is less than 3/8". The exception to this general rule is for alpha scanning, for which the distance should be less than 1 cm. Scanning rates did not exceed 1/2 detector width per second and audible indicators were used to detect changes in instrument count rate. The intensity of the coverage of the scanning survey depended on the classification of the survey unit.

5.1.2 Scanning Minimum Detectable Concentration (MDC)

For scanning the building surfaces for beta and gamma emitters (Sr-90/Y-90 and Cs-137), the scanning minimum detectable concentration (MDC_{scan}) was determined. The MDC_{scan} was determined in the field using site specific background values. The scanning technique initially considers the sensitivity of the technique and background. The initial value is calculated using the following equation (MARSSIM 2000):

$$S_i = d'\sqrt{b_i}$$

Where:

s_i = minimum detectable number of net source counts in the interval;

d' = recommended value of sensitivity of 3.28 , that is for 95% detection of a concentration equal to MDC_{scan} with a 5% false-positive rate; and

b_i = number of background counts in counting interval.

The s_i is then used to calculate the minimum detectable count rate (MDCR) using the following equation (MARSSIM 2000):

$$MDCR = S_i(60/i)$$

Where:

MDCR	=	minimum detectable count rate;
S_i	=	minimum detectable number of net source counts in the interval; and
i	=	counting interval (scan interval).

The MDCR is then used to calculate the MDC_{scan} using the following equation (MARSSIM 2000):

$$MDC_{Scan} = \frac{MDCR}{\sqrt{p}\, e_i e_s \frac{\text{probe area in } cm^2}{100 cm^2}}$$

Where:

MDC_{scan}	=	scanning MDC (pm/100 cm^2);
MDCR	=	minimum detectable count rate;
p	=	surveyor efficiency
e_i	=	instrument efficiency for the emitted radiation; and
e_s	=	source efficiency in emissions per disintegration

The results of the scan surveys of the rooms are presented in this report in Appendix K through Appendix Q.

5.1.3 Direct Measurements

One minute integrated counts using a large area gas proportional detector (100 to 500 cm^2) were taken at calculated intervals (see Section 5.3) for each survey unit.

The minimum detectable concentration (MDC) was determined for each instrument and technique used during the Final Status Survey. The MDC is the concentration that a specific instrument and technique can be expected to detect 95% of the time under field use conditions. The MDC was calculated using the following equation (MARSSIM 2000):

$$MDC = C \times (3 + 4.65\sqrt{B})$$

Where:

C = conversion factor from CPM to appropriate concentration units

The results of the direct measurement surveys of the rooms are presented in this report in Appendix K through Appendix Q.

5.1.4 Exposure Rate Measurements

Exposure rate measurements were obtained as part of the Final Status Survey. Readings were taken at a distance of 1 m from the surface of concern.

The results of the exposure rate surveys of the rooms are presented in this report in Appendix K through Appendix Q.

5.1.5 Removable Contamination Measurements

Smears were used to obtain measurements of removable contamination. Smears for removable surface activity were obtained by wiping an area of approximately 100 cm^2.

Loose surface contamination surveys of alpha and beta emitters were performed using cloth smears. The smears were analyzed on site for gross beta-alpha analysis using a Ludlum Model-2929 Dual Channel Scaler.

Loose surface contamination surveys for H-3 and C-14 was performed using polyfoam smears. The smears were analyzed by Liquid Scintillation Counting (LSC) analysis at NWT's laboratory located in Livermore, CA.

5.2 Instrumentation

5.2.1 Surface Scans/Direct Measurements

Large area gas proportional detectors coupled to count rate/scaler meters were used for the surface scans and direct measurements performed during the Final Status Survey.

5.2.2 Exposure Rate Measurements

Exposure rate surveys were performed using Ludlum Instruments Model-19 micro-R meters (1" by 1" NaI detector) or equivalent.

5.2.3 Removable Contamination Measurements

Loose surface contamination surveys of alpha and beta emitters were performed using cloth smears and analyzed with a Ludlum Model-2929 Dual Channel Scaler phoswich detector.

Loose surface contamination surveys for H-3 and C-14 was performed using polyfoam smears and analyzed by a liquid scintillation counter (LSC).

All instruments were calibrated a minimum of once every 12 months, using NIST-traceable standards. Performance and background checks were performed and documented at least once per shift on instrument use.

Copies of the instrument calibration documentation and daily performance checks are presented in this report in Appendix E.

Table B below lists the instrumentation that was used during the final status survey and the Minimum Detectable Activity (MDA) or Minimum Detectable Count Rate (MDCR) for each instrument.

TABLE B
INSTRUMENTATION FOR RADIOLOGICAL SURVEYS

Type of Measurement	Instrumentation Detector	 Meter	Background	Eff. %	Detection Sensitivity
Surface Scans-Alpha/Beta/Gamma (Equipment)	Large Area Gas Proportional Detector., Ludlum Model 43-68 (125 cm^2)	Count Rate Meter Ludlum Model-2224 Scaler/Ratemeter Ludlum Model-2350-1 Data Logger	100-300 CPM βγ 0-5 CPM α	~10 βγ ~10 α	~1,000- 2,000 dpm/100cm^2 βγ 150-250 dpm/100cm^2 α
Exposure Rates	NaI Scintillation Micro R Meter Ludlum Model-19	(Same as detector)	8-10 µR/hr	N/A	1µR/hr
Gross Alpha/Beta/Gamma On Smears	ZnS(A_g)Scintillation Detector Ludlum Model 43-10-1	Ludlum Model-2929 Dual Channel Scaler	40-65 CPM βγ 0.1-0.5 CPM α	~25 βγ ~14 α	80-100 dpm/100cm^2 βγ 10-15 dpm/100cm^2 α
Direct Measurement Static Reading (1 minute)	Large Area Gas Proportional Detector, Ludlum Model 43-68 (125 cm^2)	Count Rate Meter Ludlum Model 2224 Scaler/Ratemeter Ludlum Model-2350-1 Data Logger	100-300 CPM βγ 0-5 CPM α	~10 βγ ~10 α	550-700 dpm/100cm^2 βγ 60–110 dpm/100cm^2 α
Surface Scans-Beta/Gamma (Equipment)	Thin Window GM Pancake Detector (20 cm^2) Ludlum Model 44-9	Count Rate Meter Ludlum Model-3 Ludlum Model-12	50-100 CPM βγ	~15 βγ	~3,000- 4,5h00 dpm/100cm^2 βγ

5.3 Area Classifications

Aguirre Engineers, Inc. under contract to the U.S. Army Operations Support Command conducted a Scoping Survey in August 1998. The information collected from the Scoping Survey was used to identify the general extent of contamination present in the impacted areas. Class 1, 2, and 3 areas have been identified within the impacted areas. The classification of the area determined the type of final status survey that was performed. Surveys are defined in MARSSIM as:

Class 1 Survey: A type of final status survey that applies to areas with the highest potential for contamination, and meet the following criteria: (1) impacted; (2) potential for delivering a dose above the release criterion; (3) potential for small areas of elevated activity; and (4) insufficient evidence to support reclassification as Class 2 or Class 3

Class 2 Survey: A type of final status survey that applies to areas that meet the following criteria: (1) impacted; (2) low potential for delivering a dose above the release criterion; and (3) little or no potential for small areas of elevated activity.

Class 3 Survey: A type of final status survey that applies to areas that meet the following criteria: (1) impacted; (2) little or no potential for delivering a dose above the release criterion; and (3) little or no potential for small areas of elevated activity.

The data presented in the *Project Final Report, Radiological Scoping Survey of Walter Reed Army Institute for Research* provided the basis for classification of impacted areas for the Final Status Survey.

Table 1 below provides the impacted room total surface area, MARSSIM classification, number of direct measurements and minimum required distance between direct measurements:

Table 1 – Impacted Rooms MARSSIM Classification ,Total Surface Area, Minimum Number of Direct Measurements and Minimum Distance Required Between Direct Measurements

-

Room Number	MARSSIM Class	Total Surface Area (Square Meters)	Minimum Number of Direct Measurements (N)	Minimum Required Distance Between Direct Measurements (Meters)
O2-03	2	97	15	1.0
R1	1	30	14	1.0
R2	1	50	14	1.0
R3	1	19	53	0.5
GR	1	11	15	0.33
01	3	118	14	1.0
04	3	62	14	1.0

5.4 Final Status Surveys

Surface scan surveys, direct measurement surveys, and loose surface contamination surveys were performed in accordance with the schedule presented in Table 2 below:

Table 2 – Impacted Room Survey Schedule

Type of Survey	Area Classification	Room Number(s)	Area Surveyed	% Area Surveyed	Distance Between Direct Measurements or Swipe Samples (Meters)
Beta-Gamma Surface Scan	1	GR, R1, R2, R3	Floor, walls, ceiling	100	N/A
Beta-Gamma Surface Scan	2	02-03	Floor, walls	100	N/A
Beta-Gamma Surface Scan	3	01, 04	Floor, walls	100	N/A
Alpha Surface Scan	1	GR, R1, R2, R3	Floor	100	N/A
Alpha Surface Scan	2	02-03	Floor	100	N/A
Alpha Surface Scan	3	01, 04	Floor	100	N/A
Beta-Gamma Direct Measurements	1	GR	Floor, walls, ceiling	N/A	0.33
Beta-Gamma Direct Measurements	1	R3	Floor, walls, ceiling	N/A	0.5
Beta-Gamma Direct Measurements	1	R1, R2	Floor, walls, ceiling	N/A	1.0
Beta-Gamma Direct Measurements	2	02-03	Floor, walls	N/A	1.0
Beta-Gamma Direct Measurements	3	01, 04	Floor, walls	N/A	1.0
Alpha Direct Measurements	1	GR	Floor	N/A	0.33
Alpha Direct Measurements	1	R3	Floor	N/A	0.5
Alpha Direct Measurements	1	R1, R2	Floor	N/A	1.0
Alpha Direct Measurements	2	02-03	Floor	N/A	1.0
Alpha Direct Measurements	3	01, 04	Floor	N/A	1.0
Loose Surface Contamination Gross Alpha-Beta Activity	1	R1, R2	Floor, walls, ceiling	N/A	1.0
Loose Surface Contamination Gross Alpha-Beta Activity	1	GR	Floor, walls, ceiling	N/A	0.6
Loose Surface Contamination Gross Alpha-Beta Activity	1	R3	Floor, walls, ceiling	N/A	1.0
Loose Surface Contamination Gross Alpha-Beta Activity	2	02-03	Floor	N/A	1.0
Loose Surface Contamination Gross Alpha-Beta Activity	3	01, 04	Floor	N/A	1.0
Loose Surface Contamination Low Energy Beta Activity	1	R1, R2	Floor, walls, ceiling	N/A	1.0
Loose Surface Contamination Low Energy Beta Activity	1	GR	Floor, walls, ceiling	N/A	0.33
Loose Surface Contamination Low Energy Beta Activity	1	R3	Floor, walls, ceiling	N/A	1.5

Loose Surface Contamination Low Energy Beta Activity	2	02-03	Floor	N/A	1.0
Loose Surface Contamination Low Energy Beta Activity	3	01, 04	Floor	N/A	1.0

6.0 SUMMARY OF SURVEY FINDINGS

6.1 Statistical Considerations

6.1.1 Demonstration of Compliance

When determining compliance with remediation goals, the entire site (survey unit) is examined. One measurement does not determine compliance. Rather, the site data are examined statistically. The three compliance tests are summarized in Table 3 below. They include:

- Compare the largest site measurement to the DCGL.
- Compare the average site measurement to the DCGL.
- Use the Sign Test (MARSSIM, Chapter 8) to determine if the site data exceed the DCGL.

Table 3.
Statistical Comparisons with DCGL

Survey Result	**Conclusion**
All measurements less than the $DCGL_w$.	Survey unit meets release criterion.
Average greater than the $DCGL_w$.	Survey unit does not meet release criterion.
Any measurement greater than $DCGL_w$ and the average less than $DCGL_w$.	Conduct Sign Test and elevated measurement comparison.

6.1.2 Null Hypothesis

Using the MARSSIM methodology, the null hypothesis is stated as "the residual activity in the survey unit exceeds the release criteria" (Rev. 1, August 2000). Thus, in order to pass the survey unit (that is, release the area), the null hypothesis must be rejected. The Sign Test will be used on the survey data to test the null hypothesis.

6.1.3 Statistical Sign Test

The one-sample Sign Test is used if the contaminant is not present in background. The test is designed to demonstrate compliance with the release criterion when the radionuclides of interest are not present in background and the distribution of the data is not symmetric (NUREG-1505). In this case, Cesium-137 and Sr-90 are not present in background. The Sign Test is appropriate for this condition also, according to MARSSIM.

The Sign Test is used to compare site date to the DCGL. Significance is measured by confidence levels. To conduct the test, each of the site measurements is subtracted from the DCGL. If the result is positive, it is given a value of +1. If negative, it is given a value of -1. All the positive (+1) results are summed, and the total is compared to the Critical Value. If the sum exceeds the Critical Value, then the DCGL is met.

The Critical Value for the each survey unit is presented in the survey data for each survey unit.

An important factor in performing this test is the number of measurements taken. A minimum number must be taken. The minimum is calculated based on confidence limits, the general distribution of the contaminant at the site, the DCGL and other factors.

The number of 1-minute direct measurements to be obtained for the one sample Sign Test is calculated using the following formula:

$$N = \frac{\left(Z_{1-a} + Z_{1-b}\right)^2}{4\left(Sign\,P - 0.5\right)^2}$$

where,

N	=	number of samples;
$Z_{1-\alpha}$	=	percentile represented by selected value of $\alpha = 0.05$;
$Z_{1-\beta}$	=	percentile represented by selected value of $\beta = 0.05$; and
Sign P	=	the estimated probability that a random measurement from the survey unit will be less than the DCGL when the survey unit median is actually at the LBGR.

The calculated sample number using the Sign Test method for each Class 1, Class 2, and Class 3 survey unit is presented with the survey data for each survey unit.

The lower bound of the gray region (LBGR) value was selected as one half the DCGL (1,500 dpm/100cm^2) for Sr-90 and (2,900 dpm/100cm^2)for Cs-137.

The initial step in determining the number of data points in the one-sample case is to calculate the relative shift, Δ/σ = (DCGL-LBGR)/σ, from the DCGL value, the lower bound of the gray region (LBGR), and the standard deviation of the contaminant in the survey unit, σ. Values of the relative shift that are less than one will result in a large number of measurements needed to demonstrate compliance.

The calculated value of the relative shift for each survey unit is presented with the survey data for each survey unit.

Sign P is the estimated probability that a random measurement from the survey unit will be less than the DCGL when the survey unit median is actually at the LBGR.

The Sign P is used to calculate the minimum number of data points necessary for the survey to meet the DQOs. The value of the relative shift calculated above is used to obtain the corresponding value of Sign P from Table 5.4 in Chapter 5 of MARSSIM.

The value of Sign P is presented with the survey data for each survey unit.

The percentiles, $Z_{1-\alpha}$ and $Z_{1-\beta}$, represented by the selected decision error levels, α and ß, respectively are calculated from Table 5.2 in Chapter 5 of MARSSIM.

The value of the percentiles, $Z_{1-\alpha}$ and $Z_{1-\beta}$ is 1.645.

The calculated standard deviation for each survey unit is presented with the survey data for each survey unit.

6.2 Room Grid Maps

Room grid maps are presented in this report in Appendix H and preceding the survey data for each survey unit in the corresponding appendix.

6.3 Loose Surface Contamination Surveys

6.3.1 Alpha-Beta Activity

The results of the loose surface contamination survey for all of the rooms are presented in this report in Appendix I. The results of the survey indicate that all the smear samples were below the DCGL.

6.3.2 Low Energy Beta Activity

The results of the loose surface contamination survey for all of the rooms are presented in this report in Appendix J. The results of the survey indicate that all the smear samples were below the DCGL.

6.4 Room 01

The results of the scan survey and direct measurement survey for Room 01 along with the statistical Sign-Test is presented in this report in Appendix K. The results of the survey indicate that the null hypothesis is rejected and that the survey unit meets the DCGL.

6.5 Room 04

The results of the scan survey and direct measurement survey for Room 04 along with the statistical Sign-Test is presented in this report in Appendix L. The results of the survey indicate that the null hypothesis is rejected and that the survey unit meets the DCGL.

6.6 Room 02-03

The results of the scan survey and direct measurement survey for Room 02-03 along with the statistical Sign-Test is presented in this report in Appendix M. The results of the survey indicate that the null hypothesis is rejected and that the survey unit meets the DCGL.

6.7 Room R1

The results of the scan survey and direct measurement survey for Room R1 along with the statistical Sign-Test is presented in this report in Appendix N. The results of the survey indicate that the null hypothesis is rejected and that the survey unit meets the DCGL.

6.8 Room R2

The results of the scan survey and direct measurement survey for Room R2 along with the statistical Sign-Test is presented in this report in Appendix O. The results of the survey indicate that the null hypothesis is rejected and that the survey unit meets the DCGL.

6.9 Room R3

The results of the scan survey and direct measurement survey for Room R3 along with the statistical Sign-Test is presented in this report in Appendix P. The results of the survey indicate that the null hypothesis is rejected and that the survey unit meets the DCGL.

6.10 Room GR

The results of the scan survey and direct measurement survey for Room GR along with the statistical Sign-Test is presented in this report in Appendix Q. The results of the survey indicate that the null hypothesis is rejected and that the survey unit meets the DCGL.

6.11 Ventilation Duct Sample

A sample of the rust inside of the ventilation duct in Room 04 was taken and sent to an independent laboratory (Barringer Labs) for gamma spectroscopy and Sr-90 analysis. No radioactivity distinguishable from background was detected in the sample. The results of this sample are presented in this report in Table 4 and Appendix G.

Chain of custody forms were maintained and used for each sample taken for tracking and QC purposes. Copies of the chain of custodies are included in this report in Appendix S.

6.12 Sump Samples

Samples of the sludge in the bottom of the sumps located in Rooms 01 and 04 were taken and sent to an independent laboratory (Barringer Labs) for gamma spectroscopy and Sr-90 analysis. No radioactivity distinguishable from background was detected in the samples. The results of these samples are presented in this report in Table 4 and Appendix G.

Chain of custody forms were maintained and used for each sample taken for tracking and QC purposes. Copies of the chain of custodies are included in this report in Appendix S.

6.13 Waste Profile Sample

A waste profile sample was obtained early in the project from the material vacuumed from the surfaces of Rooms R1, R2, R3, and GR. This sample was sent to an independent laboratory (Barringer Labs) for gamma spectroscopy and Sr-90 analysis. The results of this sample are presented in this report in Table 4 and Appendix B.

Chain of custody forms were maintained and used for each sample taken for tracking and QC purposes. Copies of the chain of custodies are included in this report in Appendix S.

6.14 Room 01 Sink Survey

Detailed survey of the sink located in Room 01 was performed. The survey included investigation of the sink drain and drain trap underneath the sink. The results of the survey indicate that the sink area meets the DCGL.

Copies of the sink survey are included in this report in Appendix U.

6.15 Concrete Core Samples

Twelve 2-1/4" diameter, 0-13" depth concrete core samples were taken from the floor above the reactor rooms around the vent hole in the ceiling of Room R3 to define the extent of the contaminated concrete due to irradiation caused by the former reactor. The samples were sent to an off-site laboratory (Barringer Labs) for gamma spectroscopy analysis. The results of the samples indicated no evidence of irradiated concrete in the areas that were core sampled, thus defining the area of irradiated concrete to an area immediately adjacent to the vent hold in the ceiling of Room R3. According to blueprints, the vent hole in the ceiling of Room 3 corresponds (within 2 feet) to the centerline of the former reactor.

A map showing the locations of the concrete sample and concrete core samples is presented in this report in Appendix R. Results are presented in this report in Table 4 and Appendix R.

Chain of custody forms were maintained and used for each sample taken for tracking and QC purposes. Copies of the chain of custodies are included in this report in Appendix S.

The area of irradiated concrete above the reactor rooms was dose equivalent modeled using the "RESRAD-BUILD Version 3.0 Modeling Code: A Computer Model for Analyzing the Radiological Doses Resulting from the

Remediation and Occupancy of Buildings Contaminated with Radioactive Material", ANL/EAD/LD-3, Argonne National Laboratory, August 2000.

The purpose of the modeling is to demonstrate that the site met the limiting dose criteria as defined for unrestricted use as required in 10 CFR 20.1403(b). RESRAD-BUILD was performed using default scenarios and modeling assumptions.

The results of the RESRAD-BUILD data is presented in this report in Attachment #3.

7.0 SUMMARY

Based upon the survey data and dose equivalent modeling performed in Attachments #1, #2, and #3 of this report the reactor rooms meet the radiological requirements for unrestricted use as defined in 10 CFR 20.1403(b).

TABLE 4
SAMPLE SUMMARY TABLE

Sample ID#	Sample Location	Ac-228 pCi/g	Bi-212 pCi/g	Bi-214 pCi/g	Cs-137 pCi/g	Co-60 pCi/g	Eu-152 pCi/g	Eu-154 pCi/g	Pb-212 pCi/g	Pb-214 pCi/g	K-40 pCi/g	Tl-208 pCi/g	Sr-90 pCi/g
WR-01SA	Room 01 Sump	1.1	N/A	0.91	0.27	N/A	N/A	N/A	1.2	0.89	9.5	0.35	-0.31
WR-01SB	Room 01 Sump	N/A	N/A	N/A	N/A	N/A	N/A	N/A	1.2	N/A	N/A	0.37	-0.39
WR-04S	Room 04 Sump	N/A	N/A	N/A	ND	N/A	N/A	N/A	N/A	N/A	N/A	N/A	0.23
WR-04VD	Room 04 Vent Duct	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	0.93	N/A	-0.41
WR-R3C	Concrete Above Flange in Ceiling Room R3	0.81	N/A	0.29	0.11	1.2	12	0.81	0.72	0.24	2.6	0.3	-0.26
WR-GR-1	Concrete In Floor of Room GR	0.87	0.91	0.39	22	N/A	N/A	N/A	0.86	0.39	3.8	0.28	6.1
WR-R3-1	Concrete in Floor of Room R3	1.2	1.3	0.40	20	N/A	N/A	N/A	1.2	0.48	3.8	0.39	5.2
WR-CC-1	5' West of Vent Hole 0-12" Depth	1.0	1.1	0.34	N/A	N/A	N/A	N/A	1.0	0.34	1.6	0.36	N/A
WR-CC-2	5' North of Vent Hole 0-13" Depth	0.83	0.93	0.27	N/A	N/A	N/A	N/A	0.88	0.28	2.2	0.26	N/A
WR-CC-3	5' East of Vent Hole 15-21" Depth	0.97	0.87	0.28	N/A	N/A	N/A	N/A	0.92	0.26	N/A	0.31	N/A
WR-CC-4	5' Southeast of Vent Hole 0-13" Depth	1.1	1.1	0.28	ND	N/A	N/A	N/A	1.2	0.34	N/A	0.44	N/A
WR-CC-5	5' South of Vent Hole 0-10" Depth	0.83	0.75	0.22	N/A	N/A	N/A	N/A	0.81	0.29	N/A	0.28	N/A
WR-CC-6	5' Southwest of Vent Hole 0-13" Depth	0.95	0.99	0.43	N/A	N/A	N/A	N/A	0.83	0.39	N/A	0.3	N/A
WR-CC-7	8' Southwest of Vent Hole 0-12" Depth	1.2	1.3	0.22	ND	N/A	N/A	N/A	1.1	0.29	N/A	0.41	N/A
WR-CC-8	8' Northwest of Vent Hole 0-11" Depth	N/A	N/A	0.21	ND	N/A	N/A	N/A	0.19	0.2	N/A	N/A	N/A
WR-CC-9	8' Northeast of Vent Hole 0-12" Depth	0.96	1.3	0.24	ND	N/A	N/A	N/A	1.0	0.34	1.4	0.35	N/A
WR-CC-10	8' East of Vent Hole 0-11" Depth	1.1	N/A	0.29	ND	N/A	N/A	N/A	0.99	0.32	2.4	0.3	N/A
WR-CC-11	8' Southeast of Vent Hole 0-11" Depth	1.1	1.0	0.24	ND	N/A	N/A	N/A	1.2	0.26	2.6	0.43	N/A
WR-CC-12	8' Southwest of Vent Hole 0-11" Depth	0.96	1.2	0.33	ND	N/A	N/A	N/A	1.1	0.26	1.4	0.36	N/A

TABLE 5
WASTE SUMMERY TABLE

Drum ID#	Drum Contents	Contact Dose Rate in microR/hr	1 Meter Dose Rate in microR/hr	Drum Gross Weight in Lbs	Cs-137 Activity in mCi	Sr-90 Activity in mCi
40-1	Scrap metal, air filters, PPE, sandblast grit	15	5	820	0.05	0.04
40-2	Air filters, PPE, sandblast grit	15	5	510	0.03	0.03
40-3	Asbestos floor tiles	5	5	340	0.02	0.02
40-4	Sandblast grit, PPE, air filters, misc. scrap metal	15	5	910	0.06	0.05
40-5	Scrap metal, PPE, sandblast grit	50	10	410	0.02	0.02
40-6	Bag rust, PPE, sandblast grit	15	5	690	0.04	0.03
40-7	Sandblast grit	15	5	880	0.06	0.05
40-8	Sandblast grit	15	5	720	0.05	0.04
40-9	Sandblast grit	15	5	780	0.05	0.04
40-10	Sandblast grit, concrete rubble, air filters	15	5	590	0.04	0.03
40-11	Air filters, HEPA hose, sandblast grit	15	5	470	0.03	0.02

Figure 2
Room R3 Sump Well Diagram