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**CLEAN**

**Contract No. N62474-88-D-5086**

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**NAVAL STATION TREASURE ISLAND  
HUNTERS POINT ANNEX  
SAN FRANCISCO, CALIFORNIA**

**INVESTIGATION OF TRITIUM IN  
SURFACE SOILS AND PAVING MATERIALS  
SURROUNDING BUILDING 816**

**REVIEWED AND APPROVED BY:**



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TMA/Eberline**

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## **INVESTIGATION OF TRITIUM IN SURFACE SOILS AND PAVING MATERIALS SURROUNDING BUILDING 816 NAVAL STATION TREASURE ISLAND, HUNTERS POINT ANNEX SAN FRANCISCO, CALIFORNIA**

**May 17 - 18, 1993**

### **1.0 INTRODUCTION**

This report summarizes the results of sampling surface soils and paving materials surrounding Building 816 for tritium, at Naval Station Treasure Island, Hunters Point Annex (HPA), San Francisco, California, by Normandeau Associates, Inc. (NAI) and PRC Environmental Management, Inc. (PRC). The sampling was conducted over 2 days, May 18 and 19, 1993. As requested by PRC, 52 samples, individually consisting of soil, concrete, and asphaltic concrete (A/C), were collected at the site in accordance with the work plan for Building 816. The sampling strategy from the work plan is provided in Appendix A.

The purpose of the sampling was to determine if there was residual tritium contamination at the site due to previous Naval Radiological Defense Laboratory (NRDL) operations. In the past, the building contained a van de Graaff generator that was used as an electron and ion accelerator, and housed a radiochemistry laboratory. Mr. Filbert Fong, a former NRDL employee who now works for the State of California, Department of Health Services/ Environmental Management Branch (DHS), stated that there could be a potential for tritium contamination at the site due to previous operations involving tritium targets and tritiated thymidine.

In 1978, a survey was performed by Navy personnel from the Naval Ocean Systems Center, San Diego, and the Naval Nuclear Power Unit, Port Hueneme, California. In 1979, a smear (wipe) survey for tritium was performed by the Naval Nuclear Power Unit within Building 816. Results of the survey indicated that surface radiation levels for removable contamination did not exceed NRC Regulatory Guide 1.86 guidelines. Appendix B provides the results of the Navy's Monitoring and

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Decontamination of Building 364 and 816. The Navy recommended that the building and its environs be released for unrestricted occupancy and use. Although the environs of Building 816 were released by the Navy for unrestricted use, a survey for tritium was not performed outside the building.

Tritium contamination inside Building 816 was reported by DHS to have been removed by steam cleaning concrete surfaces. Contaminated wastewater that was generated during decontamination was believed to have been collected and disposed off site. As indicated in the work plan, a swipe survey performed in 1979 by the Naval Nuclear Power Unit indicated that no tritium contamination was present inside the building; however, a survey of tritium for the exterior soils and pavement was not performed. DHS is concerned that NRDL personnel may have walked through tritium-containing wastewater inside the building and transferred a small quantity of the contaminated water from their shoes to surfaces outside the building. The location of Building 816 is shown on Figure 1.

### **2.0 FIELD AND LABORATORY METHODS**

This section discusses the field and laboratory methods that were used to collect and analyze samples near Building 816.

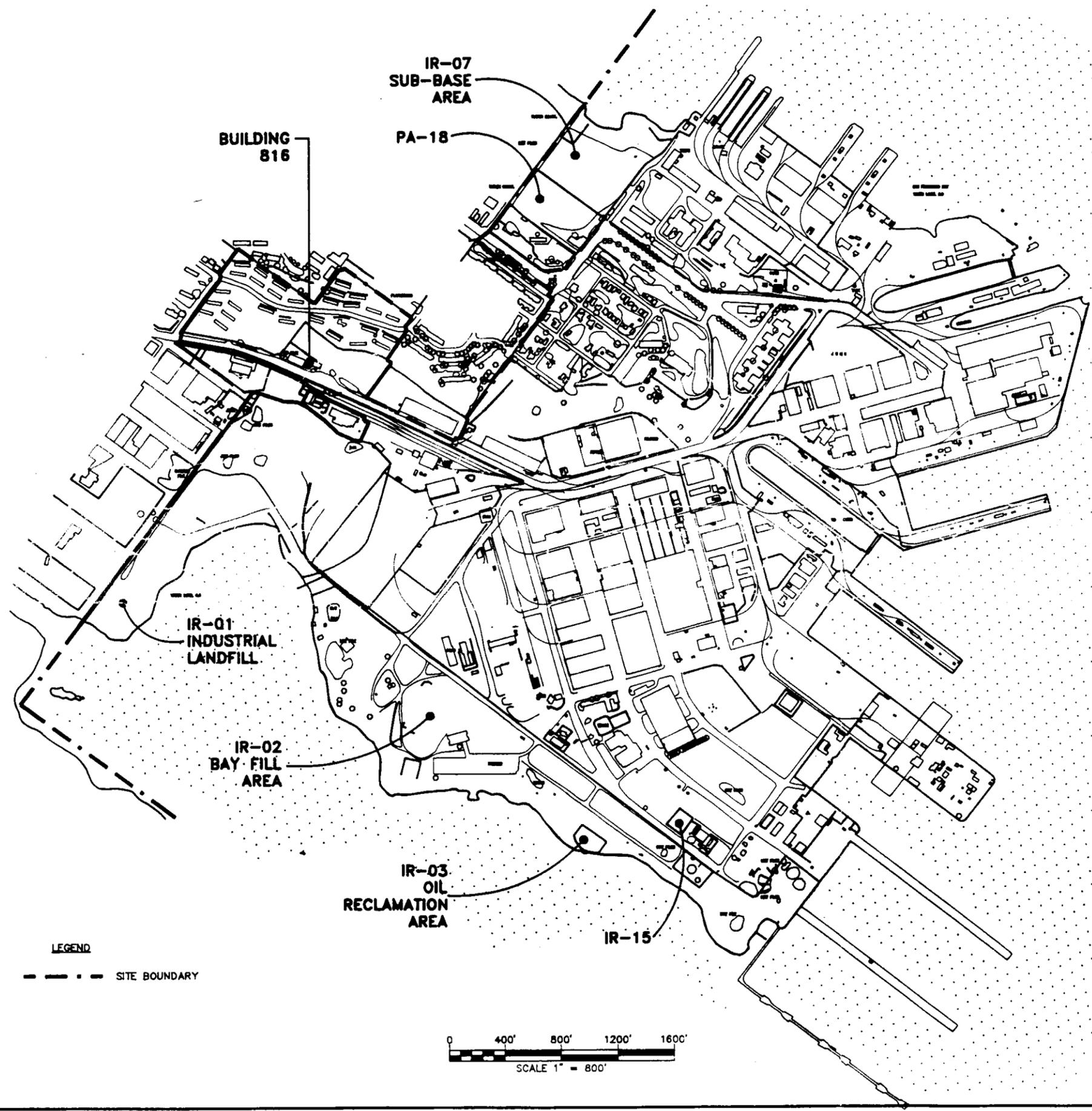
#### **2.1 FIELD METHODS**

##### **2.1.1 Sampling Approach**

Sample locations were selected based on site visits and discussions with Mr. Fong of DHS and representatives of the Navy Radiological Affairs Support Office (RASO). Mr. Fong reviewed and approved locations in the work plan for soil, concrete, and A/C sample collection.

Surface samples of soil, concrete, and A/C were collected around the perimeter of Building 816. Surface soils were defined as the first 6 inches of soil below ground surface. A surface sample for A/C and concrete was defined as the first one-half inch of material.

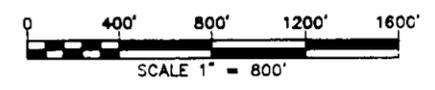
Concrete and A/C samples were taken immediately adjacent to the building foundation and were collected approximately 2 feet laterally from the building foundation. This sampling strategy



SAN FRANCISCO BAY

FIGURE 1  
SITE LOCATION MAP  
HUNTERS POINT ANNEX  
SAN FRANCISCO, CALIFORNIA

LEGEND  
- - - - - SITE BOUNDARY



(044-0155)HUNT-ALL.DWG - 04/15/82

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was based on the assumption that as people walk parallel to a wall on a narrow walkway, they will tend to be about 2 feet from the wall. Exposed soil was collected approximately 6 inches from walkways and from the building foundation. This 6 inch lateral sampling distance was selected for exposed soils around walkways and the building foundation based on the assumption that water runoff will flow into soil immediately downgradient and adjacent from a walkway or paved area.

Concrete and A/C samples were collected using a pneumatic chisel to scarify the surface for sample collection. Soils that were not covered with paving materials were collected from the first 6 inches of soil below ground surface. Soils below paving material were collected after breaking the pavement with a jack hammer to reach the underlying soil. All samples were individually placed into gasketed, polyethylene screw top jars, and stored at 4° C during transport to the laboratory. Figure 2 shows the locations where samples were collected.

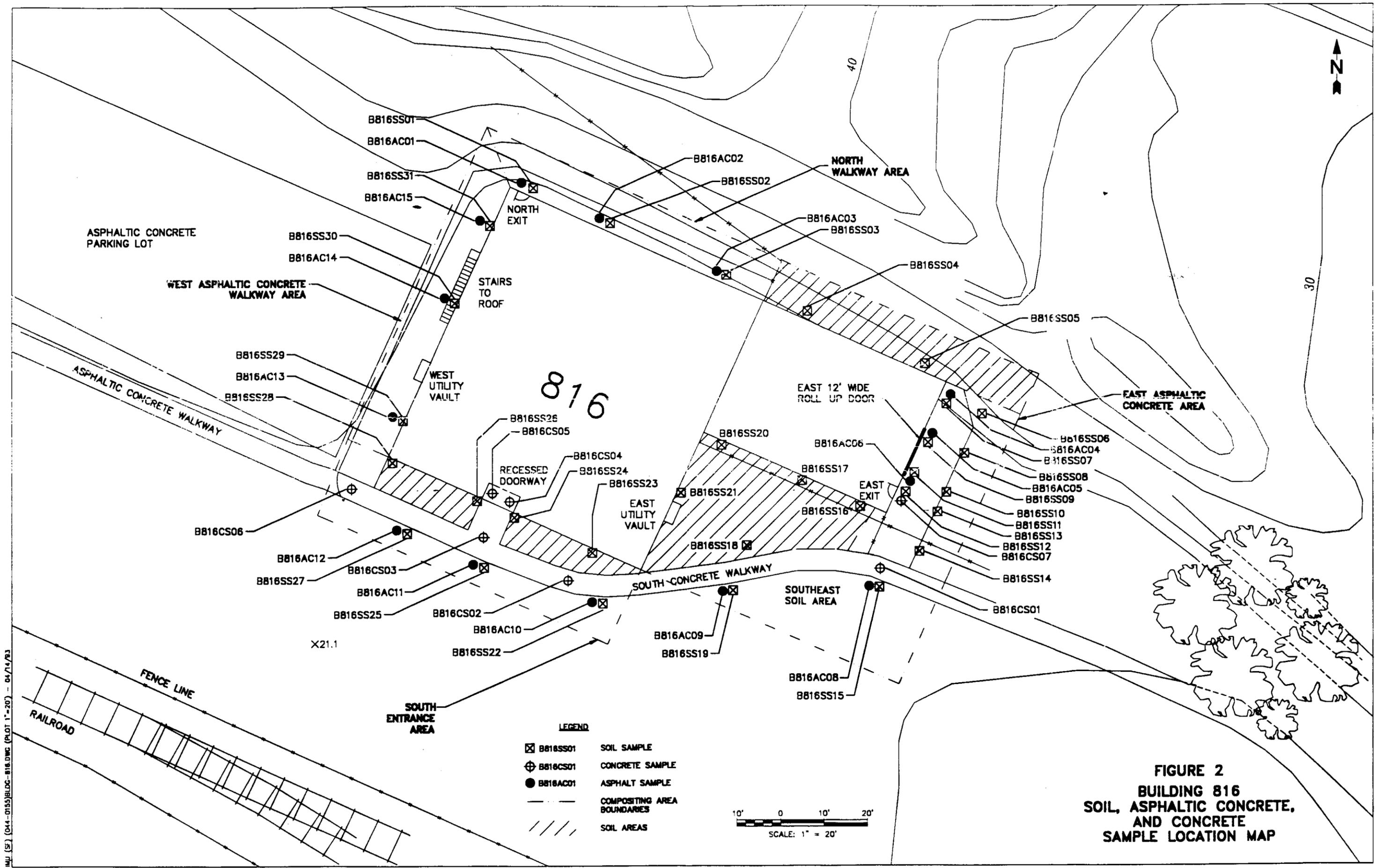
After decontaminating the shovel and digging bar, which were used to collect the samples, the tools were rinsed with deionized water. The water was collected and analyzed for tritium. A field blank for the deionized water was also submitted for tritium analysis.

### **2.1.2 Field Modifications of Work Plan**

The original number of concrete and A/C samples to be collected, as provided in the work plan, were six and fifteen, respectively. After field operations had begun, field personnel determined that the location from which A/C sample ID number B816AC07 was to be collected in the East A/C Area was covered with concrete. The sample ID number was changed to B816CS07. Therefore, the total number of concrete samples collected was increased to seven, and the number of A/C samples was decreased to fourteen.

## **2.2 LABORATORY METHODS**

Thermo Analytical/Norcal (TMA/Norcal) of Richmond, California, used an in-house analytical method of tritium analysis for the samples. The laboratory used method EP-211, which consists of an azeotropic distillation of water and toluene, to extract the tritium. The distillate was placed into a screw-top, scintillation cuvette, mixed with an organic scintillation cocktail, and loaded into a Packard Tri-Carb®, Model 2560 TR/XL, liquid scintillation analyzer. The sample was counted



**FIGURE 2**  
**BUILDING 816**  
**SOIL, ASPHALTIC CONCRETE,**  
**AND CONCRETE**  
**SAMPLE LOCATION MAP**

MJL (SF) (044-0155) BLDG-816.DWG (PLOT 1'-20') - 04/14/83

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for beta emissions from tritium for a minimum of 25 minutes. The beta emissions from each sample were counted long enough to achieve the requested minimum detectable activity (MDA) value of 0.5 picoCuries per gram (0.5 pCi/g). The results of analysis were reported in pCi/g of sample. The analytical method used by TMA/Norcal to analyze the sample, and the statistics used to calculate MDA, are provided in Appendix C. Appendix E, not included, but can be provided upon request, contains the raw data obtained from liquid scintillation analysis of the samples.

### **3.0 SUMMARY OF RESULTS**

The 55 samples collected for tritium analysis were comprised of 31 soil samples, 7 concrete samples, 14 A/C samples, and 3 equipment blank water samples. Table 1 provides results of tritium analysis. Sample matrices are identified by Sample ID numbers in the fifth and sixth characters of each number. For example, a soil sample collected in the North Walkway Area received a sample ID number, B816SS01. Concrete samples were identified by using "B816" followed by CS and the sequence number. Asphaltic concrete samples were identified by using "B816" followed by AC and the sequence number. Equipment blanks used EB in the fifth and sixth position. Of the 55 samples analyzed for tritium, none of the solid samples exceeded the requested MDA of 0.5 pCi/g. Reported MDAs ranged from 0.13 to 0.44 pCi/g.

### **4.0 HEALTH AND SAFETY**

Because pneumatic impact equipment (jack hammer and pneumatic chisel) was used to sample the A/C and concrete, a meeting was held before sampling began to discuss the unique hazards associated with the work. The field work plan and site-specific radiological safety measures were discussed to identify potential hazards that may occur during the work. Field personnel were required to wear Tyvek® coveralls, leather work gloves over nitrile gloves, steel-toed boots, hearing protection, and respirators fitted with particulate filters for dusts.

A cursory radiation survey was performed at Building 816 in 1991, during the Surface Confirmation Radiation Survey at HPA. No alpha, beta, or gamma activity above normally expected background was detected. Due to the extremely low potential for a radiological hazard at the site due to tritium, additional radiation screening was not deemed necessary.

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**TABLE 1**

**RESULTS OF TRITIUM ANALYSES**

(PAGE 1 OF 3)

SAMPLING AREA/ MATRIX	SAMPLE ID NUMBER	TRITIUM (pCi/g)	COUNTING ERROR (2 SIGMA)	MDA (pCi/g)
NO. WALKWAY AREA				
SOILS	B816SS01	- 0.14 <sup>a</sup>	0.13	0.23
	B816SS02	- 0.071	0.13	0.23
	B816SS03	- 0.11	0.12	0.22
	B816SS04	- 0.16 <sup>a</sup>	0.15	0.27
	B816SS05	- 0.19 <sup>a</sup>	0.14	0.26
ASPHALTIC CONCRETE	B816AC01	0.022	0.079	0.13
	B816AC02	0.014	0.080	0.13
	B816AC03	0.12	0.090	0.13
EAST A/C AREA				
SOIL	B816SS06	- 0.082	0.14	0.23
	B816SS07	0.064	0.12	0.19
	B816SS08	- 0.14	0.13	0.23
	B816SS09	- 0.081	0.12	0.21
	B816SS10	- 0.071	0.13	0.23
	B816SS11	- 0.052	0.13	0.22
	B816SS12	- 0.077	0.15	0.25
	B816SS13	- 0.085	0.13	0.23
	B816SS14	- 0.013	0.14	0.23
CONCRETE	B816CS07	- 0.022	0.10	0.17
ASPHALTIC CONCRETE	B816AC04	0.070	0.10	0.16
	B816AC05	0.038	0.086	0.14
	B816AC06	0.026	0.086	0.13

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**TABLE 1**  
**RESULTS OF TRITIUM ANALYSES**  
(PAGE 2 OF 3)

SAMPLING AREA/ MATRIX	SAMPLE ID NUMBER	TRITIUM (pCi/g)	COUNTING ERROR (2 SIGMA)	MDA (pCi/g)
<b>SOUTHEAST SOIL AREA</b>				
SOIL	B816SS15	- 0.082	0.12	0.21
	B816SS16	- 0.11	0.14	0.24
	B816SS17	0.16	0.14	0.22
	B816SS18	- 0.089	0.13	0.23
	B816SS19	0.036	0.14	0.23
	B816SS20	0.062	0.25	0.40
	B816SS21	- 0.21	0.25	0.44
CONCRETE	B816CS01	- 0.044	0.93	0.16
ASPHALTIC CONCRETE	B816AC08	- 0.020	0.079	0.13
	B816AC09	0.025	0.081	0.13
<b>SOUTH ENTRANCE AREA</b>				
SOIL	B816SS22	0.10	0.15	0.23
	B816SS23	0.057	0.15	0.24
	B816SS24	- 0.099	0.14	0.24
	B816SS25	0.011	0.17	0.28
	B816SS26	0.091	0.18	0.29
	B816SS27	0.064	0.14	0.22
	B816SS28	- 0.021	0.11	0.18
CONCRETE	B816CS02	- 0.014	0.10	0.17
	B816CS03	0	0.10	0.17
	B816CS04	- 0.031	0.11	0.18
	B816CS05	0	0.11	0.17
	B816CS06	0.007	0.11	0.17

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**TABLE 1**

**RESULTS OF TRITIUM ANALYSES**

(PAGE 3 OF 3)

SAMPLING AREA/ MATRIX	SAMPLE ID NUMBER	TRITIUM (pCi/g)	COUNTING ERROR (2 SIGMA)	MDA (pCi/g)
ASPHALTIC CONCRETE	B816AC10	0.083	0.084	0.13
	B816AC11	0.003	0.078	0.13
	B816AC12	0.014	0.078	0.13
WEST A/C WALKWAY				
SOIL	B816SS29	- 0.083	0.12	0.20
	B816SS30	0.047	0.13	0.20
	B816SS31	- 0.077	0.11	0.19
ASPHALTIC CONCRETE	B816AC13	0.099	0.11	0.16
	B816AC14	- 0.049	0.10	0.16
	B816AC15	- 0.024	0.094	0.16
EQUIPMENT BLANKS	SAMPLE ID NUMBER	TRITIUM (pCi/L)	COUNTING ERROR (2 SIGMA)	MDA (pCi/L)
WATER	B816EB01 <sup>b</sup>	30	130	220
	B816EB02 <sup>c</sup>	77	130	210
	B816EB03 <sup>d</sup>	-37	130	220

<sup>a</sup> -Reported value is less than the negative of its 2 sigma counting error.

<sup>b</sup> -Equipment blank - shovel

<sup>c</sup> -Equipment blank - digging bar

<sup>d</sup> -Blank - deionized water

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### **5.0 RELEVANT REGULATORY REQUIREMENTS**

#### **5.1 HEALTH HAZARDS OF TRITIUM**

Tritium ( $^3\text{H}$ ) is a radioactive, beta-emitting isotope of hydrogen ( $^1\text{H}$ ). This radioisotope has a physical half-life of approximately 12.3 years. Although tritium is radioactive, it reacts chemically as if it were hydrogen. If ingested, tritium combines with free hydroxyl groups within the body to form tritiated water. In this form, tritium is chemically identical to normal water, is distributed uniformly throughout the body, and is readily excreted in sweat, urine, and feces. The biological half-life of tritium is considered to be approximately 10 days. The biological half-life of a radioisotope is the amount of time that it takes for one-half of an ingested radionuclide to be excreted from the body, and does not relate to the radioactive decay or physical half-life of an isotope. Beta particles emitted from tritium are of low energy and are not highly ionizing. Additionally, because tritium is water soluble and is uniformly distributed to all soft tissues within the body once it is ingested, the body water becomes the critical organ resulting in a total body burden.

Due to these biological characteristics, the Annual Limit on Intake (ALI) for ingestion of tritium is approximately 80,000 microCuries ( $\mu\text{Ci}$ ) per year. The ALI is the amount of radioactive material taken into the body of an adult worker by inhalation or ingestion in a year, which is the smaller value of intake of a given radionuclide in a year by the reference man (ICRP 1975) that would result in a committed effective dose equivalent of 5 rems (stochastic effects) or a committed dose equivalent of 50 rems (nonstochastic effects) to any individual organ or tissue. Since Building 816 may be released for public use, the applicable committed effective dose is 100 mrem. The most probable route of entry into the body at this site is by ingestion of soils or vegetation; therefore the effective committed dose equivalent for tritium is 0.064 mrem/ $\mu\text{Ci}$ .

It is also important to note that the ALI quoted is for tritiated water. ALIs are not recommended for organic compounds of tritium. Radioactive thymidine that is ingested clears the body at a different biological half-life. For radioactive thymidine, the ALI may be 10 to 50 times lower (ICRP 30).

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### **5.2 ENVIRONMENTAL SOIL/SOLID CONCENTRATIONS**

Currently, there are no standards or limits for tritium activity in solids such as soil, concrete, and A/C. Release criteria must be developed on a project basis. Guideline limits on other radionuclides in soil have been evaluated by the Nuclear Regulatory Commission (NRC) for gamma-emitting, cobalt-60 ( $^{60}\text{Co}$ ) and the beta/gamma emitter, cesium-137 ( $^{137}\text{Cs}$ ). These radioisotopes were considered to present a primary exposure pathway to humans through direct irradiation; this assumption led to the establishment of an allowable soil concentration criteria guidance for a Department of the Army site at Fort McClellan, Alabama, establishing soil concentration limits for strontium-90, radium-226, and radium-228. The NRC memorandum containing the guidance is provided in Appendix D.

Direct irradiation may be an appropriate exposure pathway for certain sites, but it contrasts with the exposure scenario that might be expected if tritium soil contamination had been present at Building 816, since tritium is a low-energy beta emitter. At Building 816, it was determined that ingestion of tritium in soils and paving materials would be the primary exposure pathway. The only criteria that is appropriate for Building 816 is tritium activity resulting in a committed effective dose of 0.064 mrem/ $\mu\text{Ci}$ .

The International Atomic Energy Agency (Hisamatsu) has reported mean retention times for tritium in various plants and soils. Although influenced by climatic conditions and route of exposure, the following times were reported: 10 days for many food grains and vegetables, up to 300 days for blueberries, 87 days for maize in Mexico, and times in the middle of this range for trees. Mean retention times for soils were reported to be 91 days to 5 years.

Based on concentrations found in soils, an internal dosimetry assessment could be made to ensure that levels are below 0.1 rem/year, which is the exposure rate deemed applicable for the public (NRC, 1992).

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## 5.3 FIXED AND REMOVABLE CONTAMINATION LIMITS

Surface contamination limits are outlined in Naval, Department of Energy, and NRC standards. NRC Reg. Guide 1.86, Termination of Operating Licenses for Nuclear Reactors, although not directly applicable to this investigation, lists acceptable surface contamination levels for specific radioisotopes and non-specific radioactivity. The acceptable surface contamination activity levels, in disintegrations per minute (dpm) over a 100 square centimeter (100/cm<sup>2</sup>) area, are provided in NRC Reg. Guide 1.86 for three categories: average, maximum, and removable activity. Since tritium is a beta emitter, the limits for beta-gamma emitters are used. These limits from the guide are as follows.

Excerpted from NRC Reg. Guide 1.86, Table 1,  
Acceptable Surface Contamination Levels

Nuclide	Average (dpm/100 cm <sup>2</sup> )	Maximum (dpm/100 cm <sup>2</sup> )	Removable (dpm/100 cm <sup>2</sup> )
Tritium	5,000	15,000	1,000

## 6.0 DISCUSSION OF RESULTS

Soil, concrete, and A/C surrounding Building 816 were sampled to determine if residual tritium from former NRDL operations remained in these materials. The samples were collected to evaluate the potential for human exposure to tritium through contact, inhalation, or ingestion of tritium from soils and paving materials. All of the samples of soil, concrete, and A/C that were collected at the site provided laboratory results for tritium that were below the requested MDA of 0.5 pCi/g.

## 7.0 RECOMMENDATIONS

The sampling protocol was approved by the State to enable PRC to determine if tritium may be present on soil, concrete or A/C surrounding Building 816. This report summarizes the data collected based on the sampling protocol. There is no release criteria for tritium in soil, however, all sample results show no detectable levels of tritium.

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**APPENDIX A**

**EXCERPT FROM PROJECT WORK PLAN:  
INVESTIGATION OF TRITIUM IN SURFACE SOILS  
AND PAVING MATERIALS SURROUNDING  
BUILDING 816, SECTION 2 - TECHNICAL APPROACH**

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## **2.0 TECHNICAL APPROACH**

### **2.1 TASK DESCRIPTIONS**

This section provides a description of the tasks that will be performed during the investigation of tritium in soils and paving materials that surround Building 816.

#### **2.1.1 Sampling Strategy**

The sampling strategy was developed based upon the theory that tritium contamination originating inside Building 816 would be primarily found around doors and exits outside the building. Further, if tritium contamination had migrated outside, it may have been transferred by foot traffic to other locations around the exterior of the building. Therefore, areas around doors and exits, soil and paving materials immediately adjacent to walkways surrounding Building 816 were selected for sampling.

Surface samples of soil, concrete, and A/C will be collected around the perimeter of Building 816. Surface soils are defined as the first 6 inches of soil depth. A surface sample for A/C and concrete is defined as the first one-half inch of material. Thirty-one soil samples, six concrete samples, and fifteen A/C samples will be collected for tritium analysis.

Concrete and A/C samples that are taken immediately adjacent to the building foundation will be collected approximately 2 feet away from the building foundation. This sampling strategy is based upon the assumption that as people walk parallel to a wall on a narrow walkway, they tend to be about 2 feet from the wall. Exposed soil will be collected approximately 6 inches from walkways and from the building foundation. This 6-inch sampling distance was selected for exposed soils around walkways and the building foundation based upon the assumption that water runoff will penetrate soil surfaces immediately adjacent from a walkway or paved areas.

At the time when tritium contamination may have occurred inside Building 816, the area surrounding the building may not have been paved. Currently, there is not sufficient information to establish when alleged episodes of tritium contamination occurred or when paving materials were placed around Building 816. Since there is a possibility that tritium may have been introduced into

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exposed soil and unpaved areas, soil that lies below concrete and A/C will also be tested for tritium. The paving material will be removed at each A/C and concrete sampling location to expose underlying soil for sampling. If several layers of paving material are encountered, each layer will also be sampled. All samples will be stored and shipped in accordance with the information provided in Appendix A.

As shown on Figure 2, the area surrounding Building 816 will be divided into five general areas from which samples will be collected. On Figure 2, these areas are listed clockwise from the north corner of Building 816: the North Walkway Area, the East Asphaltic Concrete Area, the Southeast Soil Area, the South Entrance Area, and the West Asphaltic Concrete Walkway Area. Table 1 lists the type of sample matrix and number of samples to be collected from each area and the number of replicate analyses for each matrix. For each matrix, Table 2 shows the analysis to be performed, the number and type of sample containers to be used, sample preservation, and holding time information.

### **2.1.2 Soil Surface Sampling**

Soil samples will be collected in accordance with the PRC's CLEAN Draft Ionizing Radiation Protection Program April 18, 1993 (in review), Sections 17.1.1 through 17.1.1.6. The applicable soil sampling protocol and sample handling procedures are provided in Appendix A. The samples will be placed in 200-ml screw top plastic jars and stored at 4°C prior to shipment to the laboratory.

### **2.1.3 Concrete Surface Sampling**

Concrete will be sampled by chipping the surface with a rotohammer and scarifying tool to a depth of approximately one-half inch over a 3.5-inch diameter area. This will provide approximately 100 grams of concrete for laboratory analysis. The samples will be placed in 200-ml screw top plastic jars and stored at 4°C prior to shipment to the laboratory.

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### **2.1.4 Asphaltic Concrete Surface Sampling**

Asphaltic concrete will be sampled by chipping the surface with a rotohammer to a depth of approximately one-half inch over a 4-inch diameter area. This will provide approximately 100 grams of A/C for laboratory analysis. The samples will be placed in 200-ml screw top plastic jars and stored at 4°C prior to shipment to the laboratory. Table 3 shows the sample ID numbers of the samples to be collected at each location for each matrix.

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**APPENDIX B**

**REPORT OF MONITORING AND DECONTAMINATION OF BUILDINGS 364 AND 816**

SUPERVISOR OF SHIPBUILDING, CONVERSION AND REPAIR  
SAN FRANCISCO, CALIFORNIA

REPORT OF MONITORING AND DECONTAMINATION  
OF BUILDINGS 364 AND 816  
HUNTER'S POINT SHIPYARD,  
SAN FRANCISCO, CALIFORNIA

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### APPENDIX:

A. NRL Counting Data for H-3 and C-14 Smears,  
Building 816, Hunter's Point Shipyard,  
San Francisco, California

## I. INTRODUCTION

A cursory radiation survey of potentially contaminated facilities formerly occupied by the Naval Radiological Defense Laboratory was made by personnel from the Naval Nuclear Power Unit, Port Hueneme, California during the period 30 September through 1 October 1978 and reported in Naval Nuclear Power Unit letter 40:RHS:sdr 3256 Ser 1235 of 12 October 1979.

Buildings surveyed showed less than minimum detectable activities except for Building 364, where levels in a number of locations exceeded established NRC and BUMED limits. Accordingly, decontamination and resurvey efforts were recommended for Building 364.

Building 816 surveys indicated no measurable contamination with the instrumentation used. The building had a history of tritium usage, which the instrumentation was not capable of measuring. Accordingly, a resurvey by appropriate detection methods was recommended.

## II. DISCUSSION

The cursory survey at Building 364 and environs showed contamination levels up to 63,000 dpm/100cm<sup>2</sup> in room 108, 800,000 dpm/100cm<sup>2</sup> in the Hot Cell and 32,000 dpm/100cm<sup>2</sup> in the pipe trench and waste tank east of Building 364. All other rooms and areas of the building showed no activity above minimum detectable (MDA).

Beta absorption measurements were made at several locations at maximum activity to determine beta energies and to identify the contaminant. These measurements indicated a beta energy at approximately 0.65 MEV. Gamma spectra measurements identified Cs-137 as the major contaminant. Accordingly, the clearance limits for the class of radionuclides including Cs-137 were applied. As per NRC Regulatory Guide 1.86 these limits are:

- a. 5000 dpm/100cm<sup>2</sup> averaged over 1m<sup>2</sup> or less
- b. 15,000 dpm/100cm<sup>2</sup> maximum over 100cm<sup>2</sup> or less
- c. 1,000 dpm/100cm<sup>2</sup> removable
- d. 0.2 mrad/hr average radiation level
- e. 1.0 mrad/hr maximum radiation level

Although there was no history of usage of alpha emitters in Building 364 alpha survey measurements were made, with negative results.

### III. INSTRUMENTATION

The basic instrumentation used for building 364 survey was the Eberline Model E-140N count rate meter with a DT-304/PDR detector probe in conjunction with a Radiac Counter-Timer (Model 2) for timed digital measurement read-out. Instrument performance data was:

Average background - 32.4 cpm

Efficiency (CS-137 beta) - 19.6%

Probe Sensitive area - 20 cm<sup>2</sup>

Minimum Detectable Activity (MDA) - 17.1cpm net,  
50 cpm gross cpm or 435 dpm/100cm<sup>2</sup> for Cs-137 beta

The detection method for Building 816 survey was smearing with metrical filter paper and sending to the Naval Research Laboratory, Washington, D.C. for liquid scintillation counting.

#### IV. DECONTAMINATION AND FINAL SURVEY

Decontamination was accomplished by the Survey - Clean - Survey method. In general decontamination was done by paint removal and by concrete chipping.

Final post-decontamination survey was done by HMC G. M. Jones and HMI R. J. Franks of RASO.

Areas of known or suspected contamination were surveyed by counting for one minute at the centers and corners of a one-foot grid over the area. Smears were taken at each corner and in the center of a three-foot grid over the area.

Areas of low contamination potential were surveyed as above, using one-meter grids for direct reading and smear survey.

In Building 816 smear samples were taken throughout the building with an average of one smear for each 50 square feet of floor and wall space.

## V. SURVEY RESULTS AFTER DECONTAMINATION

In all spaces surveyed the maximum surface reading found was less than the limit for average levels of contamination.

Maximum and average (over the entire surface) are presented as:

AREA	MAXIMUM (dpm/100cm <sup>2</sup> )	AVERAGE (dmp/100cm <sup>2</sup> )
Room 108 (North Laboratory):		
East Wall	3500	460
South Wall, East Section	1600	MDA
South Wall, West Section	1450	MDA
North Wall, East Section	2300	MDA
North Wall, West Section	900	MDA
West Wall	MDA	MDA
Floor	MDA	MDA
Ceiling	1050	MDA
Hot Cell:		
West Wall	1400	500
North Wall	1500	600
East Wall	2200	750
South Wall	2150	750
Floor	4300	1550
Ceiling	MDA	MDA
Pit East Wall	4500	2800
Pit West Wall	2100	1200
Pit North Wall	2800	1650
Pit South Wall	4000	2200
Room 104 (SE Laboratory)	MDA	MDA
Room 105 (Center Laboratory)	MDA	MDA
Room 1 (SW Laboratory)	MDA	MDA
Entry Hall	MDA	MDA
Balcony	MDA	MDA
Pipe Trench	MDA	MDA
Tank Pit	MDA	MDA

Navy publication "Principles of Radiation and Contamination Control" NAVSHIPS 250-341-3, Volume:2, page 201, states that a surface beta contamination level of 35,000 dpm/100cm<sup>2</sup> will produce a surface dose rate of 1 mrad/hr. Based on this relationship the maximum radiation level encountered during the final survey did not exceed 0.13 mrad/hr.

Building 364 smear sample counting results indicated no removable contamination approaching the limit of 1000 dpm/100 cm<sup>2</sup>.

Building 816 smear sample counting results received from NRL were negative. These results are shown in Appendix A.

## V. CONCLUSIONS AND RECOMMENDATIONS

Survey results indicate no fixed or removable contamination or surface radiation level exceeds BUMED, (NAVMED P-5055) limits or NRC Regulatory Guide 1.86 guidelines. Accordingly, it is recommended that Buildings 364, 816 and environs be released for unrestricted occupancy and use.

NAVAL RESERACH LABORATORY  
COUNTING DATA FOR H-3 AND C-14 SMEARS  
BUILDING 816  
HUNTER'S POINT SHIPYARD  
SAN FRANCISCO, CALIFORNIA













UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
REGION V

1990 N. CALIFORNIA BOULEVARD  
SUITE 202, WALNUT CREEK PLAZA  
WALNUT CREEK, CALIFORNIA 94596

January 23, 1980

Department of the Navy  
Naval Sea Systems Command  
Washington, D.C. 20362

Attention: James J. Rush

Gentlemen:

This is in response to your letters to Mr. James Sniezek dated October 26, 1979 and January 7, 1980, your reference numbers 070422/NLK Ser 284 and 070422/NLK Ser 1. Those letters relate to surveys and decontamination of the facility formerly known as the Naval Radiological Defense Laboratory, buildings 815, 364, and 816, near San Francisco.

We have reviewed the NRC files of surveys and decontamination of those facilities, and we have reviewed the results of more recent surveys and decontamination submitted as an enclosure to your letter dated October 26, 1979. The facility meets NRC guidelines for release to unrestricted use. We plan no additional on-site inspection or other action regarding this matter.

We concur in the decision to release the facility to unrestricted status.

Sincerely,

A handwritten signature in cursive script that reads "H. E. Book".

H. E. Book, Chief  
Fuel Facility and Materials  
Safety Branch

bcc: SNIEZEK  
WANDY MILLER

**NORMANDEAU ASSOCIATES**

**APPENDIX C**

**FORMULA USED BY TMA/NORCAL  
TO CALCULATE MDA FOR TRITIUM**

## **NORMANDEAU ASSOCIATES**

Line 54 - MDA calculated as 4.66 times the standard deviation of the blank cpm (equal to the blank cpm listed in line 22 of Figure 1 times the blank percent error listed in line 22 divided by 100) divided by the following correction factors: detector efficiency (line 25), yield (line 34), sample decay (line 35), aliquot (line 41), and the required conversion factor to convert from dpm to the units listed in line 44 (in this example, the conversion from dpm/ml to  $\mu\text{Ci/ml}$  is 2.22 E 06).

The following formula was used by TMA/Norcal to determine MDAs for each tritium sample. The counting times are in minutes, the aliquot size is in grams for solids, and in milliliters for water.

$$MDA = \frac{4.66 \times \text{Blank CPM} \times \text{Count Time}}{\text{Efficiency} \times \text{Yield} \times \text{Sample Decay} \times \text{Unit Conversion Factor} \times \text{Aliquot}}$$

Tritium and <sup>14</sup>C by Liquid Scintillation Counting

FIGURE 1. Tritium Data Sheet

1	28-JAN-92	TMA Corporation	4271- 6	H DST
2	10:12:22	Tritium - LSC Calculations	D 2/3 RIVER DISCHARGE CO	
3		PLSC V 1.00		
4			Reviewed _____	Date _____
5				
6	Reference Time : 334.334-91 ( 0:00 PST 30-NOV-91 )			
7				
8	Det	Count GMT	Counts	Time
9				CPM
10				% Err
11				TSIE
12				Eff
13				Sample.
14	4	22.916-92	155761.	100.00
15				1557.61
16				0.25
17				229.52
18				0.256(1)
19				Weighted Average
20				1557.61
21				0.25
22				0.256
23				Spike.
24	4	27.918-92	17607.	3.00
25				5869.00
26				0.75
27				232.20
28				0.256(1)
29				Weighted Average
30				5869.00
31				0.75
32				0.256
33				Blank.
34	4	23.749-92	442.	100.00
35				4.42
36				4.76
37				228.75
38				0.256(1)
39				Weighted Average
40				4.42
41				4.76
42				0.256
43				Eff.
44	Eff = ( 4311.39+/-44.41)/( 19486.46+/- 77.95) =			
45				0.221 +/- 0.002
46				Act Err
47				(1 sigma)
48				Sample CPM
49				1557.61 +/-
50				3.95
51				Background CPM
52				4.42 +/-
53				0.21
54				Net CPM
55				1553.19 +/-
56				3.95
57				Yield ( 1.00 Mg)
58				1.0000
59				Decay Corr (Sample)
60				0.99175
61				Decay Corr (Spike )
62				0.75882
63				Lambda(-d)
64				1.5454E-04
65				DPM of Aliquot
66				7.0784E+03
67				Aliquot
68				9.0900E+00 ml
69				DPM/ml
70				7.7870E+02
71				Uci /ml
72				3.5077E-04
73				1 Sigma Error
74				0.25 %
75				Uci Err
76				8.9256E-07
77				2 Sigma Error
78				0.50 %
79				Uci Err
80				1.7494E-06
81				Limiting Value
82				3.5224E-04
83				MDA
84				2.2078E-07

Tritium in Solid Samples by Azeotropic Distillation

EP-211

Rev. 0, October, 1992

Pg. 1

## 1.0 Introduction

This procedure extracts aqueous tritium from vegetation by azeotropic distillation with toluene. The tritium activity is then determined by LSC.

## 2.0 References

TMA/Norcal Safety Manual

TMA/Norcal Sample Control Procedures

TMA/Norcal Health Physics Manual

TMA/Norcal Q.A. Manual

For specific material information refer to the appropriate MSDS.

## 3.0 Responsibility

3.1 The Supervisor has the responsibility to ensure that the analyst (Chemist or Technician) has been properly trained in the following procedure and that the analyst is aware of any hazards pertaining to the following procedure. Also the Supervisor must ensure that the equipment and facilities needed by the analyst to successfully complete the following procedure are available.

3.2 The analyst has the responsibility to follow sections 6.1, 6.2, and 6.3 fully, and to use professional judgement to evaluate the success of the procedure at each step. The analyst must also process any Q.C. samples associated with the group being analyzed, and coordinate with the lab Supervisor if any work must be completed by another shift.

## 4.0 Safety

4.1 The analyst shall be versed in the TMA/Norcal Safety Manual before engaging in radiochemical procedures, and know how to handle an emergency should one arise. For questions regarding proper handling procedures of chemicals refer to the TMA/Norcal Safety Manual or the appropriate MSDS, or contact the Lab Supervisor.

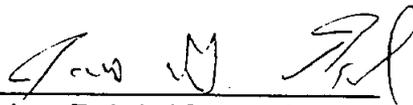
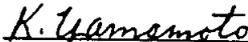
4.2 All reagents, unless otherwise specified, should be considered as contact and inhalation hazards. Wear protective gloves and use fume hoods.

4.3 Toluene is a contact and inhalation hazard and is highly flammable. Use in a fume hood and wear gloves and eye protection when in use. Do not combine with strong oxidizing agents or acids.

Approved:

11.5-92

Date

  
Cognizant Technical Personnel  
for Quality Assurance Manager

Tritium in Solid Samples by Azeotropic Distillation

EP-211

Rev. 0, October, 1992

Pg. 2

## 5.0 Materials / Reagents

Materials

Allihn condenser, S/T 24/40  
Distillation flask, S/T 24/40  
round bottom, 250 mL  
Distillation receiver, S/T 24/40  
#3642  
Heating mantle  
Centrifuge tube, 40 mL  
Transfer pipet  
Bittner bulb  
Glass scintillation vial, 25 mL

Reagents

Toluene  
Picofluor LLT Cocktail

Verify that all reagents to be used have an expiration date label on their containers and that the expiration date has not been exceeded. See QA procedure QAP-06 for details.

## 6.0 Procedure

## 6.1 Preparation

Before starting the analysis have all materials and reagents at hand. Be sure you have allowed enough time to complete the analysis. Verify that the sample planchets have been created in the computer database. If the planchet must be counted promptly upon completion of the analysis then coordinate with the counting room before the analysis is started.

## 6.2 Sample procedure

1. Hydrogen-3 in samples may be determined on a sample weight basis or on a volume of distillate basis. If analysis is on a wet weight basis, transfer aliquot to a distillation flask and record the received weight. (Note a) If analysis is on a water volume basis, simply transfer a suitable quantity of sample to the distillation flask. (Note b) Add ~100 mL of toluene.
  - a. Some samples, such as grains, may require grinding before analysis.
  - b. The percentage of H<sub>2</sub>O can vary from 10-90%. The client may also require percent moisture to be reported. In this case record the sample wet weight aliquot.
2. Connect the flask containing the sample to the distillation receiver. Connect receiver to a proper Allihn condenser. Place the unit on a heating mantle. Slowly heat the sample and avoid bumping. Adjust the temperature so that condensation is observed in the condenser above the receiver.
3. If the analysis is to be recorded on a weight basis, reflux until no more H<sub>2</sub>O is observed in the receiver, and record the total volume of H<sub>2</sub>O. If the analysis is on a volume basis, reflux until 10 mL of H<sub>2</sub>O have been collected.

Tritium in Solid Samples by Azeotropic Distillation

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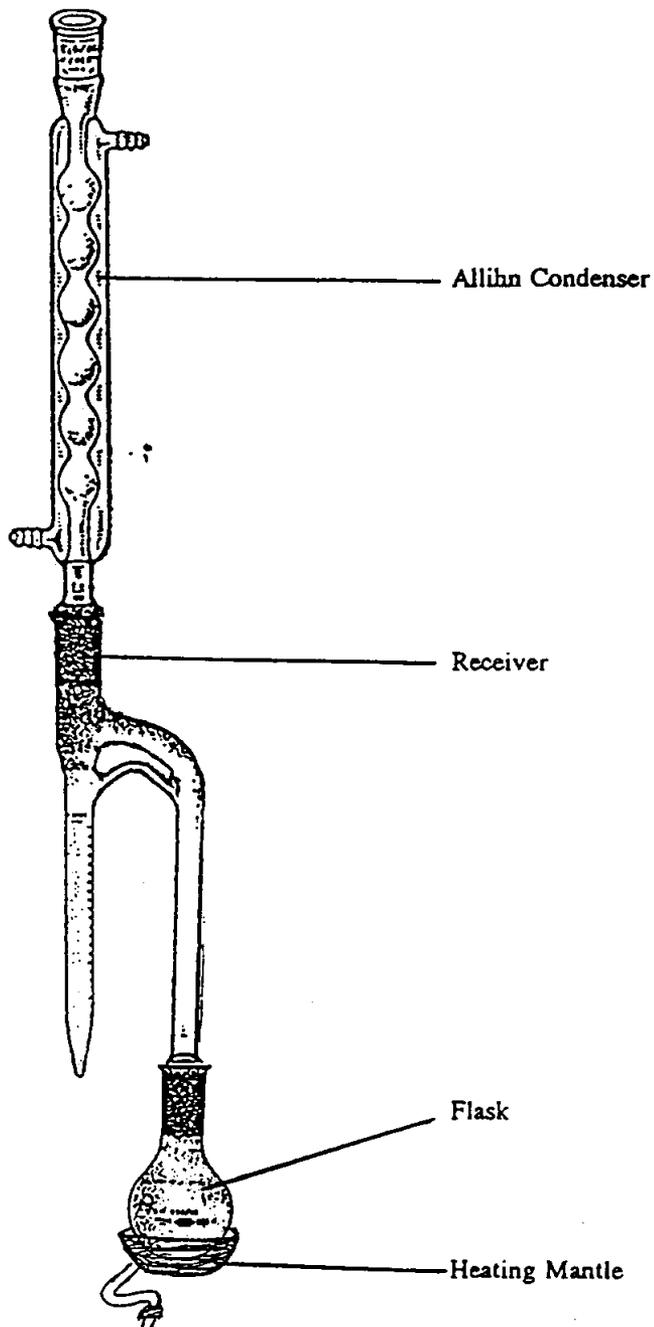
4. Remove the receiver and transfer 10.0 g of distillate to a plastic scintillation vial. Add 10 mL of Picofluor cocktail.
5. Shake the sample until it is homogeneous, label the cap of the vial, and turn in with a Log-in form to Radiometrics for counting.
6. Process a blank with each batch of samples. Add 20 mL H<sub>2</sub>O and ~100 mL Toluene to a flask and continue with Step 2.

**6.3 Waste Disposal**

For disposal of other wastes generated in this procedure see EP-010.

**7.0 Supersession / Recision**

This procedure supersedes TMA/Norcal EA-48, rev. 4 "Aqueous <sup>3</sup>H in Vegetation (Azeotropic Distillation)", of July 1991.



Azeotropic Distillation Assembly

***NORMANDEAU ASSOCIATES***

**APPENDIX D**

**EVALUATION OF ACCEPTABILITY OF PROPOSED DECOMMISSIONING ACTIVITIES**



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

ENCLOSURE C

MAY 06 1987

MEMORANDUM FOR: William E. Cline, Chief  
Nuclear Materials Safety and Safeguards Branch  
Region II

FROM: John W. N. Hickey, Chief  
Operations Branch  
Division of Fuel Cycle, Medical, Academic and  
Commercial Use Safety

SUBJECT: EVALUATION OF ACCEPTABILITY OF PROPOSED  
DECOMMISSIONING ACTIVITIES

11 11 07

In your memorandum of April 16, 1987, you requested guidance on the acceptable concentration of Co-60 and Cs-137 in soil to allow the release of the Department of the Army, Fort McClellan, Alabama, facility. The primary pathway for exposure of individuals for these nuclides is by direct radiation. Therefore, the determination of acceptability for surface contamination of ground areas should be based on the following criteria:

External Radiation

The gamma exposure at 1 meter above the ground surface shall not exceed 10 uR/h above background for an area of greater than 30 ft x 30 ft and shall not exceed 20 uR/h above background for any discrete area (i.e., less than 30 ft x 30 ft).

Concentration criteria have also been developed for Co-60 and Cs-137 for situations in which subsurface contamination may be present, such as when burials of material have been made. These criteria are as follows:

<u>Radionuclide</u>	<u>Concentration Limit Above Background (pCi/g)</u>
Co-60	8
Cs-137	15

Where more than one radionuclide is present, the sum of the ratios of the individual radionuclide concentrations to their respective concentration limits shall not exceed 1.

I hope that this information is satisfactory as you evaluate the adequacy of the Department of the Army's decommissioning. If you have further questions, please feel free to contact me.

John W. N. Hickey, Chief  
Operations Branch  
Division of Fuel Cycle, Medical,  
Academic, and Commercial Use Safety

B70626009A B70616  
REG2 LIC30 PDR  
01-02B61-04

Table 2. Additional Guidelines on Acceptable Levels of Contamination in Soil To Be Released for Unrestricted Use

Radionuclide <sup>a</sup>	Maximum Soil Concentration (in pCi/g)
Cobalt-60 <sup>b</sup>	8
Strontium-90 <sup>b</sup>	5
Cesium-137 <sup>b</sup>	15
Radium-226 <sup>b</sup>	5
Radium-228 <sup>b</sup>	5

<sup>a</sup>If only one radionuclide is present, then the maximum concentration is the value listed in the table. However, if more than one radionuclide is present, determine for each radionuclide the ratio between the measured concentration in soil and the concentration listed in the table. The sum of all such ratios may not exceed one (i.e., unity).

<sup>b</sup>Memorandum to W. E. Cline, Chief, Nuclear Materials Safety and Safeguards Branch, NRC, Region II, from J. W. N. Hickey, Chief, Operations Branch, Division of Fuel Cycle, Medical, Academic, and Commercial Use Safety: Evaluation of Acceptability of Proposed Decommissioning Activities, May 6, 1987.

NUDOCs - DOCTYPE cmmemo cat: LIC ASS: 8706260094  
<sup>c</sup>United Nuclear Corporation Resources Company, Approved Soil Decontamination Criteria for the Decommissioning of the UNC Facility, Docket No. 70-820, May 12, 1981.

<sup>d</sup>U.S. Environmental Protection Agency, Health and Environmental Protection Standards for Uranium and Thorium Mill Tailings, 40 CFR Part 192, Subparts B and E, July 1, 1991.

***NORMANDEAU ASSOCIATES***

**APPENDIX E**

**LABORATORY ANALYTICAL DATA  
(WILL BE PROVIDED ON REQUEST)**