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SURVEY UNIT PROJECT REPORTS ABSTRACT PARCEL D-1 PHASE II SANITARY SEWER AND STORM DRAIN REMOVAL

Hunters Point Naval Shipyard
San Francisco, California

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Attachment 3~~2~~ Import Fill Material (Jericho) Sample Results
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Attachment 5~~4~~ Quality Control Summary Report
Attachment 6~~5~~ Response to Comments

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LIST OF ACRONYMS AND ABBREVIATIONS

ALARA	as low as reasonably achievable
AM	Action Memorandum
Bi-214	bismuth-214
Cs-137	cesium-137
DoD	U. S. Department of Defense
DON	U. S. Department of the Navy
DQA	data quality assessment
ELAP	Environmental Laboratory Accreditation Program
EPA	U. S. Environmental Protection Agency
FSS	final status survey
ft	foot, feet
GPS	global positioning system
GWS	gamma walkover survey
HPNS	Hunters Point Naval Shipyard
HRA	Historical Radiological Assessment
IR	Installation Restoration
keV	kilo-electron volt
LBGR	lower bound of gray region
MARLAP	Multi-Agency Radiological Laboratory Analytical Protocols Manual
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
m	meter
m/sec	meters per second
m^2	square meter
MDC	minimum detectable concentration
mrem/y	millirem per year
NaI(Th)	sodium iodide (thallium activated)
NAVSEA	Naval Sea Systems Command
NCRP	National Council on Radiation Protection and Measurements
NRC	U. S. Nuclear Regulatory Commission
pCi/g	picocuries per gram
QC	quality control
Ra-226	radium-226
RASO	Naval Sea Systems Command Detachment, Radiological Affairs Support Office
ROC	radionuclide of concern
RSI	Radiation Solutions, Inc.
RSY	radiological screening yard
SAP	sampling and analysis plan
SOP	standard operating procedure
Sr-90	strontium-90
SUPR	survey unit project report
TtEC	TetraTech EC, Inc.
VSP	Visual Sample Plan
WRS	Wilcoxon Rank Sum

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1.0 INTRODUCTION

1.1 PURPOSE

This Survey Unit Project Reports (SUPRs) Abstract summarizes the scope, approach, and radiological surveys conducted during the Phase II removal of sanitary sewers and storm drains located within Parcel D-1 at the Hunters Point Naval Shipyard (HPNS) located in San Francisco, California. The location of Parcel D-1 is shown in Figure 1-1.

This SUPRs Abstract, prepared under Contract No. N62473-10-D-0808, Contract Task Order 0004, is applicable to all associated SUPRs and data sets prepared for regulatory review unless otherwise noted.

1.2 BACKGROUND

Based on the site's operational history described in the *Historical Radiological Assessment, Hunters Point Annex, Volume II, History of the Uses of General Radioactive Material 1939–2003* (HRA; Naval Sea Systems Command [NAVSEA], 2004) and site-specific investigative data, the U. S. Department of the Navy (DON) determined that the HPNS sanitary sewer and storm drain systems contain radioactive contamination in soils and debris requiring a response action. This decision is documented in the *Final Base-wide Radiological Removal Action, Action Memorandum – Revision 2006, Hunters Point Shipyard, San Francisco, California* (AM; DON, 2006), provided in Attachment 1, which was created to implement the recommendations of the HRA (NAVSEA, 2004).

HPNS is a site where radioactive materials have been used or stored, where known spills, discharges, or other instances involving radioactive materials have occurred, and where radioactive materials might have been disposed of or buried. The sanitary sewer and storm drain systems have a potential for general radioactive material contamination based on the site operating history or known contamination detected during previous radiation surveys.

The DON initiated this removal action for the extraction of piping systems and soil containing radioactive contaminants that may be present in or around the sanitary sewer and storm drain systems. This removal action serves to eliminate the potential threat posed by future migration and/or off-site release of radioactive material present at the site to the surrounding environment.

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Currently, such a release could occur as a result of wastewater or stormwater transport, erosion, weathering, seismic events, or biological activity.

The HRA (NAVSEA, 2004) states that the present-day configuration of the sanitary sewer and storm drain systems is the result of an evolutionary process. These systems were originally designed and built in the 1940s as a combined system, using the same conveyance piping and 40 separate discharge outfalls into San Francisco Bay.

This combined system grew in sections from the 1940s to its maximum size in 1958, when it underwent the first in a series of separation projects. The Building 819 pump station was constructed for conveyance of much of the sanitary sewage from HPNS to the City of San Francisco treatment works, and separation of the systems took place in the industrial areas and the southwest area of HPNS. Separation of the systems involved installation of dedicated sanitary sewer collection piping or diversion structures within the combined system piping that normally directed all combined flows through the Building 819 pump station. However, during storm events, stormwater flows would overwhelm Building 819, and much of the sewage and stormwater was diverted to various existing outfalls into the bay. Twenty-eight bay outfalls were converted to exclusive use for stormwater outlets, while 12 continued to serve as combined sanitary and stormwater sewer outlets.

A second segregation project, undertaken in 1973, provided some additional separation of the combined systems. The last of the separation projects performed in 1976 involved the installation of additional dedicated sanitary sewer piping. Complete separation of the combined systems was never achieved. Due to the evolutionary nature of the separation process, radiological contamination from the same source could have impacted the piping and other components of both systems.

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2.0 RELEASE CRITERIA

2.1 RELEASE LIMITS

The DON is taking steps to ensure that residual radioactivity in an area will not result in individuals being exposed to unacceptable levels of radiation or radioactive materials. Regulatory agencies establish radiation dose standards based on risk considerations and scientific data relating dose to risk. Residual levels of radioactive material that correspond to allowable radiation dose standards are derived by analysis of various pathways and scenarios such as direct exposure, inhalation, and ingestion.

These levels, known as release limits, are presented in terms of soil activity concentration. Release limits usually refer to average levels of radiation or radioactivity above appropriate background levels. Release limits applicable to soil are expressed in units of activity concentration, picocuries per gram (pCi/g). The development of release limits is often an iterative process, where the release limits selected or developed are modified as additional site-specific information is obtained. The development of final cleanup levels was based on the U.S. Nuclear Regulatory Commission's (NRC's) screening levels, the U.S. Environmental Protection Agency's (EPA's) preliminary remediation goals, or agreements between the DON and regulatory agencies as documented in the AM (DON, 2006).

2.2 AS LOW AS REASONABLY ACHIEVABLE

As low as reasonably achievable (ALARA) is a philosophy of striving for excellence in the practice of health physics and is an important aspect of radiation-safety regulations. .The National Council on Radiation Protection and Measurements (NCRP) in its *Report No. 116 - Limitation of Exposure to Ionizing Radiation* (NCRP, 1993), has stated, "ALARA is simply the continuation of good radiation-protection programs and practices which traditionally have been effective in keeping the average and individual exposures for monitored workers well below the limits". The application of ALARA clearly includes the consideration of economic and social factors, and thus will inherently be different for different sources or facilities.

The ALARA concept is founded in the professional judgment of radiation-safety managers and personnel and is not, therefore, able to be used as a measure as to whether or not a particular radiation-safety program is adequate in comparison with other programs. Additionally, the

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ALARA concept does not provide a numerical limit below which the ALARA concept is achieved.

2.3 ENVIRONMENTAL ALARA PROCESS

This section describes the steps taken to implement the environmental ALARA policy.

2.3.1 Identification of Potential Radiological Impacts

Each new radiological operation was subjected to ALARA reviews before work began to ensure that radiation exposures to workers, the public, and the environment met ALARA principles. ALARA reviews were conducted for all operations, practices, and procedures that have potential for individual or collective doses to workers. Where appropriate, reviews culminated in changes or additions to work planning documents, standard operating procedures (SOPs), and radiological work permits.

These documents were used to identify activities that have potential for radiological environmental impacts and could require environmental ALARA analysis. If a radiological impact was identified that could impact the environment significantly, the impact was communicated to the work staff and the DON's Radiological Affairs Support Office (RASO).

2.3.2 Review of Radiological Impacts

Radiological impacts to the environment, workers, and the public from field operations were assessed for compliance with ALARA principles. Results from radiological survey activities, and air, soil, sediment, and water samples were used to assess the radiological impacts of HPNS activities.

Environmental radiological impacts from radiological activities were determined using nine methods: 1) air monitoring stations located around the excavation site perimeter to track radiological impacts (results were provided to RASO on a regular basis); 2) field monitoring and sampling to identify areas requiring additional remediation; 3) remediation of contaminated areas at or above the release criteria; 4) control of radiologically impacted areas and work sites; 5) frisking of personnel and examining equipment leaving a radiologically controlled area; 6) use of release criteria that equate to dose and risk; 7) review of historical radiological operations to allow complete investigation of all areas of radiological concern; 8) characterization of radiologically impacted sites to ensure complete removal of radioactive material above the

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release criteria; and 9) dosimetry worn by personnel to measure time-averaged doses from gamma radiation.

2.3.3 Performance of Qualitative ALARA Analyses

Laboratory analyses were performed for radionuclides of concern as well as for a number of naturally occurring radionuclides to ensure that any possible radioactive contamination was identified. After each excavation was completed, the qualitative radiological impacts from operations were evaluated by performing a dose and risk assessment (see Section 6.0). The results of analysis and assessments are provided to the RASO and regulatory agencies for review.

2.3.4 Performance of Quantative ALARA Analyses

Based on qualitative ALARA analyses, excavation projects that could cause a potential dose to the public to exceed 100 millirem (individual) or 10 person-rem (collective) were subjected to quantitative ALARA analyses using the steps described below. To date, no operations at HPNS have resulted in an individual dose to the public greater than 100 millirem or a collective dose greater than 10 person-rem.

Quantitative ALARA analyses include societal, technological, economic, and public policy considerations. In addition, these ALARA analyses consider NRC guidance for performing the following environmental ALARA assessments:

- Identification of possible radiation protection systems, such as alternative operating methods or controls, that are reasonably achievable. The options should range from the most rudimentary (base case) to the most technologically sophisticated systems.
- Quantification of exposures and doses to workers and the public in the vicinity of the work through air monitoring and dosimetry.
- Quantification of the economic factors, including the costs of purchasing, installing, operating, and maintaining the radiological equipment, and the potential health effects associated with the exposure of people and any other direct or indirect cost resulting from exposures to radiation during investigations and/or remediation.
- Identification and estimation of other health and non-health detriments and benefits, such as equipment loss and accidents.
- Evaluation of process alternatives using a quantitative cost-benefit analysis, when possible. When evaluations included assumptions, judgments, and limitations that could be quantified, and potential doses were well below the dose limit, qualitative analyses were used with full documentation.

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- Implementation of the ALARA principles and monitoring of the results.

The following specific factors are used in performing a quantitative ALARA analysis:

- Dose to workers, the public, and the environment before and during work processes
- Residual dose to the local population
- Applicable alternative processes (treatments, operating methods, or controls) for site investigations or remediation
- Costs for each alternative evaluated
- Societal and environmental (positive and negative) impacts associated with alternatives

Based on recent estimates of dose to the public from HPNS operations, only qualitative ALARA analyses were required. Much of the data and analysis used for environmental ALARA evaluations were developed as part of the routine work processes.

To ensure that ALARA levels are met, removal actions conducted in accordance with the AM (DON, 2006) were designed to 1) substantially reduce ionizing radiation below cleanup goals and 2) eliminate identified pathways of exposure to ionizing radiation.

2.4 RADIONUCLIDES OF CONCERN

As listed in Table 3-1 of the *Basewide Storm Drain and Sanitary Sewer Removal Plan* [Attachment 1 to the *Basewide Radiological Management Plan, Hunters Point Naval Shipyard, San Francisco, California* (ITSI Gilbane, 2013a)], the primary radionuclides of concern (ROCs) for most survey units were cesium-137 (Cs-137), radium-226 (Ra-226), and strontium-90 (Sr-90). Table 2-1 lists those ROCs. The ROCs, their half-lives, and principal types of radiation (alpha, beta, and gamma) are provided in Table 2-1.

Table 2-1
Radionuclides of Concern

Radionuclide	Half-Life	Radiations
Cs-137	30.17 years	beta (β^-)/gamma (γ)
Ra-226	1,600 years	alpha (α)/gamma (γ)
Sr-90	28.6 years	beta (β^-)

Exceptions to the ROCs are noted individually in the specific SUPR, as necessary. The samples collected were analyzed by gamma spectroscopy at a U. S. Department of Defense (DoD)

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Environmental Laboratory Accreditation Program (ELAP) and California Department of Public Health accredited laboratory, Curtis and Tompkins, for definitive data. Ten percent of the samples were randomly selected and also analyzed for Sr-90 at the DoD ELAP accredited laboratory, GEL Laboratories, LLC.

The presence of Sr-90 would be attributed primarily to fission products associated with OPERATIONS CROSSROADS and decontamination of ships that participated in atomic weapons testing. Since Cs-137 would have also been present, any sample containing activity greater than the release criteria for Cs-137 was also analyzed for Sr-90.

2.5 RELEASE CRITERIA FOR RADIONUCLIDES OF CONCERN

Radionuclide-specific release criteria were obtained for each ROC using the criteria established by the NRC and EPA, as discussed in Section 2.1. The release criteria are shown in Table 2-2 along with their calculated residual dose in units of millirem per year (mrem/y). Survey units for the sanitary sewer and storm drain removal project are considered acceptable when the radionuclide concentrations meet the soil release criteria defined in Table 2-2.

Table 2-2
Release Criteria

Radionuclide	Soil			
	Outdoor Worker (pCi/g)	Residual Dose (mrem/y)[a]	Residential (pCi/g)	Residual Dose (mrem/y)[a]
Cs-137	0.113	0.2142	0.113	0.2561
Ra-226	1.0[b]	6.342	1.0[b]	17.24
Sr-90	10.8	0.1931	0.331	1.648

Notes:
[a] The resulting dose is based on modeling lead-210 in equilibrium using RESRAD Version 6.5 with radon pathways turned off.
[b] Limit is 1 pCi/g above background, per agreement with EPA.

2.6 INVESTIGATION LEVELS

Investigation levels are specific levels of radioactivity used to indicate when additional investigation may be necessary. Investigation levels also serve as a quality control (QC) check. For example, in addition to indicating potential contamination, a measurement that exceeds the investigation level may have indicated a failing instrument.

When determining an investigation level using a statistical based parameter (for example, standard deviation), the following may be considered: survey objectives, underlying radionuclide

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distributions (normal, log-normal, nonparametric), data population descriptors (standard deviation, mean, median), and prior survey and historical information.

When an investigation level was exceeded, the measurement was confirmed to ensure that the initial measurement actually exceeded the investigation level. This involved taking further measurements to confirm the initial result and, as appropriate, quantifying the area of elevated residual radioactivity.

An investigation level for gamma walkover scan (GWS) surveys was established as the reference area mean + 3 σ where σ is the standard deviation of the gamma readings in the reference area. See also discussion in Section 3.2.1.

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3.0 FINAL STATUS SURVEY

The objective of the final status survey (FSS) was to demonstrate that identified residual radioactivity levels inside the excavated trench and subsequent overburden soils to be used as backfill met the release criteria. The *Multi-Agency Radiation Survey and Site Investigation Manual* (MARSSIM; DoD et al., 2000); NUREG-1505, *Nonparametric Statistical Methodology for the Design and Analysis of the Final Status Decommissioning Survey Guide* (NRC, 1998); and NUREG-1507, *Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions Guide* (NRC 1997) were used as guidance in designing and conducting the FSS.

3.1 SURVEY UNITS

Sanitary sewer and storm drain systems were divided into geographic areas referred to as zones, which are equivalent to numerically identified survey units (e.g., Survey Unit 283) used in Phase I reporting. The zones are shown in Figure 3-1. Within each zone are two types of survey units: trench and excavated soil. The zones and the survey units within each zone are summarized in Table 3-1. The installation restoration (IR) site associated with each zone, if any, is also listed. For convenience, the average trench depth is given in units of feet (ft) and the approximate trench surface area – which includes floor and walls – is given in square meters (m^2).

Trench survey units consist of the surface area remaining in the trench as sanitary sewer and storm drain lines, as well as surrounding soil, are removed. Excavated soil survey units are created from the soil excavated during sanitary sewer and storm drain removal, so that the soil can be surveyed to determine its acceptability for use as fill material. Although the size of each survey unit may vary, each received the same scan survey and systematic sampling frequency.

3.1.1 Trench Survey Units

GWS surveys and systematic sampling were performed for each identified survey unit. Trench survey units varied in size but did not exceed 1,000 m^2 in surface area. Systematically located post-excavation samples were collected after establishing a grid using Visual Sample Plan (VSP). Additional surveys may have been performed if the investigation levels were exceeded.

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Each trench unit was mapped using computer-aided design technologies. First, the trench was plotted, and then the extent of the trench excavation was drawn, with walls splayed out flat, into a two-dimensional plane. The design was then extracted into VSP to determine the locations of systematic sample points.

Table 3-1
Summary of Zones and Survey Units

Zone	Related IR Site	Survey Units		Average Trench Depth (ft)	Trench Surface Area (m^2)
		Trench	Soil		
A	---	1	5	3-6	980
B	---	1	5	6	1,000
C	---	1	6	8	1,000
D	---	1	3	7	860
E	---	1	3	3-6	950
F	IR-48	1	3	5	1,000
G	---	1	2	5-6	965
H	IR-17	1	1	3	320
I	---	1	1	2-4	450
J	---	1	2	5	940
K	IR-53	1	1	3	85
L	---	1	2	2-5	500
M	IR-70	1	2	5	575
N	---	1	1	3	120
O	---	1	2	4	1,000
P	IR-70	1	1	4	320
Q	---	1	2	4	600

A minimum of 20 systematic sample locations, based on a random start point and the square grid pattern, were generated using VSP and are provided in each SUPR. Sample collection points were placed in the field using approved land surveying technologies and identified with the global positioning system (GPS).

3.1.2 Excavated Soil Survey Units

During the course of excavation, soil from the excavations was removed to dewatering/screening pads located in the radiological screening yard (RSY). The soils were then spread out in lifts not to exceed 6 inches in thickness and 1,000 m^2 in total surface area. Each of these areas was then identified as an excavated soil survey unit and assigned an incremental number.

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The entire area of each excavated soil survey unit was plotted in VSP, and 20 systematic sampling locations were determined using a random start point on a square grid pattern. Typical systematic sampling locations used for each excavated soil survey unit are shown on Figure 3-2. The final sampling locations of excavated soil survey units are provided in each SUPR.

3.2 SURVEY AND SAMPLING

Three types of data were collected: GWS data, gamma static measurements, and soil samples. GWS and gamma static measurements were performed to identify areas of potentially elevated residual radioactivity. Instruments were calibrated prior to initial use. On a daily basis, instruments were inspected to ensure their proper working condition and response-checked before use. GWS and gamma static measurements were used to guide soil sampling efforts by identifying locations with potentially elevated radioactivity. Soil samples were collected and analyzed to quantify radionuclide concentrations.

The sanitary sewer and storm drain piping removed from the trenches were surveyed and sampled, but separately from the soil excavated from the trenches and the trenches themselves and not as part of the FSS. Contaminated piping was turned over to the LLRW Disposal Contractor for disposal as low-level radioactive waste. Non-contaminated pipe was turned over to the Non-hazardous Waste Disposal Contractor to be disposed as non-hazardous waste. These activities are discussed in the *Parcel D-1 Phase II Remedial Action Closure Report, Hunters Point Naval Shipyard, San Francisco, California* (Gilbane, 2015).

3.2.1 GWS Data

A GWS of the excavated soil survey units and accessible surface areas of each trench survey unit was performed using the Radiation Solutions, Inc. (RSI) RS-700 self-contained mobile gamma ray detection system. System specifications are found in Attachment 2~~1~~. The RS-700 system consists of an integrated controller and data acquisition system, a digital gamma ray spectrometer/multi-channel analyzer (or console), and a data controller coupled to a sodium iodide (thallium activated) [NaI(Tl)] gamma scintillation detector. Custom-built copper-lined lead shielding with a steel surround encases the top and sides of the detector. The shield focuses the detection capability on radiation coming from the ground, i.e., forces the detector to "scan downward," optimizing the detection of gamma-rays at surface or near-surface locations.

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The RS-700 system was used in two configurations as shown in Figure 3-3. The GWS survey of the RSY pads was performed using the RS-700 system with two detectors mounted end-to-end on a cart pulled behind a small tractor equipped with an adjustable throttle to allow for speed control. The trench floor and walls were surveyed using a single RSI detector mounted on a frame suspended from a boom lift allowing the detector to be moved by the lift along the floor and walls of the trench. The detector was maintained at a height of approximately 0.1 meters (m) above the ground and moved over the surface at a speed of 0.5 meters per second (m/sec), with each pass spaced 0.5 m (more or less, based on the configured detector field of view) from the previous pass to achieve 100-percent coverage of the area being surveyed. GWS data are provided as an attachment to each individual SUPR.

Figure 3-3
RSI RS-700 System Detector Configurations

Two detectors mounted on pull behind cart

Single detector mounted on suspended frame

The detector height of 0.1 m above the ground was selected for two reasons. First, to maximize sensitivity, the detector was maintained as close to the ground as possible, with the distance of 0.1 m above the ground considered the smallest distance at which the detector can travel above the ground without being damaged. Second, the surface-to-detector geometry mimics that used in the modeling analysis documented in Section 6.8.2 of NUREG-1507 (NRC, 1998), for the calculation of the scan minimum detectable concentration (MDC) for land areas assuming the use of a NaI(Tl) scintillation detector. Similarly, the scan rate of 0.5 m/sec was selected as the slowest reasonable scan rate achievable, yet also consistent with the referenced NUREG-1507 (NRC, 1998) modeling analysis.

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The RSI RS-700 data were captured and position-correlated by means of a Trimble SPS852 GPS rover mounted on the detector. The GPS rover pulled position data both from U. S. NAVSTAR and from Russian GLONASS satellites, which increased positional accuracy.

The RS-700 data were plotted on a cumulative frequency distribution to provide a visual representation of the data distribution and assist in identifying dissimilar populations and data set outliers. The plotted data appearing in a relatively straight line suggest a single normally distributed population, while non-linear results suggest multiple populations or represent a departure of the population from normality.

The mean and standard deviation of the RS-700 data set were calculated and used to develop color-coded contour maps based on sigma values (i.e., the number of standard deviations each measurement lies from the mean). The contouring process involved creating a regularly spaced grid and assigning values to every spot on the grid. Grid node values were assigned using a weighted average based on the inverse square law, which law describes how radiation levels drop off with distance from a source. Once the grid was complete, color-coded contours were created from grid node values within the specified ranges of values. The contouring process tends to smooth over single data points with lower sigma values while accentuating clustered areas or single locations with higher sigma values. This is the desired effect which aids in the data analysis by focusing attention on those areas most likely to be contaminated.

Biased sample locations were selected using a subjective process involving the three types of contour mapping methods outlined in Table 3-2.

Discrete locations on the investigated area map with sigma values above 3.0 (i.e., outlying locations) were initially selected for biased sampling. These locations were overlaid onto the background area and combined area maps to see how outlying locations shown on those maps compared to the biased sampling locations initially selected. Where appropriate, additional biased sampling locations were selected.

The overall goal of the process was to identify a reasonably number of biased sample locations that are most representative of discrete outlying locations that appear on one or more of the maps. Using the contour maps in conjunction with each other maximized their value, in that

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each map accentuates locations that could yield optimal information regarding the radiological characteristics of the material being sampled. Likewise, the limitations of each method, which are primarily related to masking the presence of contamination or creating an inaccurate appearance of contamination, were minimized. This would not have been the case if each mapping method was used independently.

Table 3-2
Contour Mapping Methods Used to Identify Biased Sampling Locations

Description	Advantage	Disadvantage
Method No. 1 – Investigated Area Sigma		
Compares sample population against statistical properties of a normally distributed population. Statistically, a very small number of points with sigma values above 2.0 are expected for a normal distribution (2.28%), and an even smaller number of points with sigma values above 3.0 (0.13%). A higher number of points indicates dissimilar populations or outliers, either of which could represent contamination.	Non-uniform contamination within sample population will be readily apparent	Uniformly distributed contamination may be masked within sample population and not revealed
Method No. 2 – Background Area Sigma		
Compares sample population against statistical properties of reference area population. Basic assumption is the site from which the sample population was collected possesses similar radiological characteristics as the reference area, and the sample population, if uncontaminated, possesses the same statistical characteristics as reference area population.	Compares sample population to a known radiological standard (i.e, reference area)	Dissimilar radiological characteristics between sample site and reference area will distort contour map and may mask presence of contamination or create an inaccurate appearance of contamination
Method No. 3 – Combined Area Sigma		
Compares sample population against statistical properties of a normally distributed population centered around reference area mean. Basic assumption is sample population and reference area population share a common centroid or mean value, but vary in their distribution around the mean. This method is an attempt to partially compensate for dissimilar radiological characteristics that may exist between sample site and reference area population.	Reduces the potential impact of any dissimilarity in radiological characteristics that may exist between sample site and reference area	Dissimilar mean values between populations will distort contour map and may mask presence of contamination or create an inaccurate appearance of contamination

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3.2.2 Gamma Static Measurements

A gamma static measurement was taken at each systematic and biased soil sample location identified using the process described in the preceding section. Gamma static measurements were collected using a Ludlum Model 44-10 2-inch by 2-inch NaI detector with a Ludlum 2221 ratemeter/scaler. The detector was held 0.1 m from the surface and a 1 minute scaler count was collected.

A method for calculating minimum detectable concentrations (MDCs) for gamma scanning of land areas, presented in Section 6.8.2 of NUREG-1507 (NRC, 1998), may be modified to develop MDCs for gamma static measurements of land areas. The method is based on assumptions regarding the areal extent, depth, and radionuclides distributed in the soil and instrument response to the theoretical photon fluence based on those assumptions as modeled by software such as MicroShield®. Due to the broad nature of the assumptions regarding the matrix of interest, the modeled fluence performs well to predict instrument response when it used as a bounding source term, though it is usually a poor representation of actual field conditions. Also, measurements collected using the Ludlum Model 44-10 detector are influenced not solely by the matrix of interest, but also by gamma-rays from ambient and cosmic sources of radiation, as well as subsurface contributors. Therefore, gamma static measurements are used primarily as a qualitative comparative indication of radioactivity in soil and are not used in trench or excavated soil suitability decisions. The direct measurement of radioactivity in soil, as was done with soil sampling described below, provides the information upon which suitability decisions are made.

3.2.3 Soil Sample Collection

Soil samples were collected at biased sample locations identified using the process described in Section 3.2.1. In addition, systematically spaced samples were collected from the laydown pads, with the number and locations of samples to be collected calculated in accordance with Section 5.2.1. Samples were also collected of soil in the trench along the pipe footprint based on contamination found on or inside the removed sanitary sewer and storm drain piping. Where a sample was found to have a radioactivity exceeding the release criteria, a series of samples were collected around the elevated sample location to bound the area of elevated activity.

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3.3 TRENCH BACKFILL

Once the survey and sampling of the trench and excavated soil were complete, the data sets were reviewed and sent to RASO for concurrence. With Navy authorization, the trenches were then backfilled by zone and the screened excavated soil used as backfill material. Where necessary, imported fill material was used to complete the backfill. Table 3-3 summarizes the backfill material sources, the trench depths, and the exposure rates measured at 1-ft lifts as the material was placed into the trenches.

3.4 IMPORT FILL

In 2011, import fill material was purchased from Jericho. The soil was screened both chemically and radiologically before being delivered for use as backfill material at HPNS. The import fill material also met the standards specified in Worksheet #15.1 of the TetraTech EC, Inc. (TtEC) *Sampling and Analysis Plan, Basewide Radiological Support, Hunters Point Naval Shipyard, San Francisco, California* (TtEC SAP; TtEC, 2011). Approximately 40,000 cubic yards of the material was stockpiled at HPNS by TtEC.

Twenty random soil samples were collected from the Jericho import fill stockpile. The samples were analyzed by Curtis and Tompkins using gamma spectroscopy. The results of the random soil samples were used to perform dose modeling when import fill material was used to backfill a trench. The samples were analyzed by gamma spectroscopy. Ten percent of the samples (two samples total) were also analyzed for Sr-90 at the DoD ELAP accredited laboratory, GEL Laboratories.

Table 3-3
Trench Backfill Summary

Zone	Trench (Survey Unit ID 04-PDT-XXX)	Backfill Material Source(s) (Excavated Soil Survey Unit ID 04-PD-XXXXX)	Exposure Rate @ 1 m (µR/hr)
A	001	D0001, D0002, D0011, D0013, D0042	5-7
B	002	D0003, D0004, D0005, D0012, D0014	5-7
C	003	D0006, D0007, D0008, D0009, D0010, D0041	6-8
D	004	D0015, D0016, D0017	6-8
E	005	D0029, D0039, D0040, Imported Fill	5-9
F	006	D0018, D0024, D0030	5-7
G	007	D0019, D0027	5-7
H	008	D0020	5-7

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I	009	D0022	5-7
J	010	D0021, D0023, Imported Fill	5-7
K	011	D0025	5-7
L	012	D0026, D0028, Imported Fill	5-9
M	013	D0031, D0032	5-7
N	014	D0033	5-6
O	015	D0034, D0035, Imported Fill	5-7
P	016	Imported Fill	6-8
Q	017	D0037, D0038	6-7

The import fill material gamma spectroscopy and Sr-90 results are provided in Attachment 3~~2~~. Table 3-4 provides a summary of the Ra-226 sample results of the import fill material from Jericho.

Table 3-4
Import Fill Material (Jericho) Sampling Results

Sample Number	Ra-226 (pCi/g)	Sample Number	Ra-226 (pCi/g)	Sample Number	Ra-226 (pCi/g)	Sample Number	Ra-226 (pCi/g)
1	0.289	6	0.335	11	0.387	16	0.330
2	0.276	7	0.344	12	0.376	17	0.410
3	0.313	8	0.363	13	0.408	18	0.346
4	0.278	9	0.413	14	0.253	19	0.346
5	0.383	10	0.372	15	0.366	20	0.342

Ra-226 was detected above the MDC in all 20 samples. Cs-137 was not detected above the MDC in any of the samples. The MDC for Cs-137 ranged from 0.016 to 0.023 pCi/g. Two samples were also analyzed for Sr-90. One sample reported Sr-90 at 0.156 pCi/g. The second sample did not detect Sr-90 above the sample MDC of 0.148 pCi/g.

3.5 REFERENCE AREA

Reference area samples for the sanitary sewer and storm drain removal project were obtained in April 2006 in an area of the Building 813 parking lot in Parcel D-2. Building 813 was identified in the HRA as being impacted, but the parking lot area was not identified as being impacted and is considered of like material to that encountered in the soils in survey units. Eighteen samples were collected systematically from this area for reference area purposes. The samples were analyzed by Curtis and Tompkins using gamma spectroscopy. Ten percent of the samples (two samples total) were also analyzed for Sr-90 at the Curtis and Tompkins laboratory. The

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reference area samples provide a basis for net activity concentration. The reference area gamma spectroscopy and Sr-90 results are provided in Attachment 4~~3~~.

A summary of the Ra-226 sample results is provided in Table 3-5. Ra-226 was detected above the MDC in all 18 samples. The average reference area activity for Ra-226, measured by a 21-day in-growth of the 609.31 kilo-electron volt (keV) gamma energy peak for bismuth-214 (Bi-214), was determined to be 0.488 pCi/g. This places the release criterion at 1.488 pCi/g of Ra-226. The average value (0.488 pCi/g) was used in each SUPR for background subtraction of Ra-226 for dose and risk modeling. Based on the observed variability in the reference area data population (standard deviation = 0.073 pCi/g), Ra-226 concentrations less than the mean plus 3 standard deviations, or 0.706 pCi/g, are considered representative of background levels for Ra-226 in soil.

Cs-137 was not detected above the MDC any of the samples. The MDC for Cs-137 ranged from 0.017 to 0.026 pCi/g. Neither of the two samples analyzed for Sr-90 detected Sr-90 above the MDC of 0.194 pCi/g.

Table 3-5
Reference Area Ra-226 Sampling Results

Sample Number	Ra-226 (pCi/g)	Sample Number	Ra-226 (pCi/g)
1	0.498	10	0.554
2	0.454	11	0.477
3	0.598	12	0.480
4	0.331	13	0.421
5	0.590	14	0.410
6	0.600	15	0.559
7	0.507	16	0.495
8	0.426	17	0.431
9	0.454	18	0.496

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4.0 LABORATORY ANALYSIS

Sample analytical results were used to determine whether the soil met the release criteria or if additional remediation was required. The definitive data were sent to RASO for concurrence and Navy authorization prior to using the soil as backfill material, backfilling the trenches, or disposing of the soil as low-level radioactive waste.

4.1 SAMPLE PREPARATION

Samples for radiological analysis were collected by trained and qualified field technicians. Soil samples were approximately 1,000 grams in size. Visually identifiable foreign objects and debris were removed manually in the field. Samples were bagged in one-gallon resealable plastic bags, numbered, logged, and sent for laboratory analysis. Each sample was labeled and assigned a unique sample identification number.

The samples were turned over to the laboratory using the chain-of-custody (COC) procedure described in the ITSI Gilbane *Sampling and Analysis Plan, Phase II Radiological Remediation and Support, Parcel D-1, Hunters Point Shipyard, San Francisco, California* (ITSI Gilbane SAP; ITSI Gilbane, 2013b). Each sample was verified to contain identical information on the sample container and the associated COC form when received at the laboratory.

Once received by the laboratory, samples were prepared by drying, grinding, mixing, sifting, and weighing as appropriate prior to analysis in accordance with the laboratory's approved procedures.

For counting by gamma spectroscopy, 300 to 400 grams of sample material was placed and sealed in a 250-milliliter "tuna can" metal container. The Curtis and Tompkins laboratory analyzed the samples directly for Ra-226 using the EPA 901.1 MOD/DOE HASL 4.3.5.2 method and analysis of the 46.09 percent abundant Bi-214 609.31 keV gamma energy peak after an in-growth period of 21 days to allow the Bi-214 to approach secular equilibrium with Ra-226.

4.2 GAMMA SPECTROSCOPY ANALYSIS

Gamma spectroscopy analyses were performed by Curtis and Tompkins. Curtis and Tompkins is a California Department of Public Health certified laboratory, as well as a DoD ELAP accredited laboratory.

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Gamma spectroscopy analysis was performed using EG&G ORTEC® detector systems equipped with beryllium end caps (windows), which allowed for enhanced quantification of low-energy gammas (such as Ra-226). Hardware features included a high-purity germanium gamma photon detector supported by a multi-channel analyzer and analysis software. Instrument hardware was calibrated using a multi-energy National Institute of Standards and Technology traceable source ranging from 50 keV to 2.6 mega-electron volts. All results were reported in pCi/g.

Data reviews consisted of ensuring that 100 percent of the data comply with the method- and project-specific requirements as outlined in the ITSI Gilbane SAP (ITSI Gilbane, 2013b). Quality assurance data reviews included checking for compliance with the required QC practices, SOPs, and method and project requirements.

4.3 GAMMA SPECTROSCOPY ANALYSIS FLAGS

The gamma spectroscopy summary reports are presented for each sample in the individual SUPR. The summary reports also identify flags (or symbols) that may be associated with each radionuclide, if appropriate. These flags and their associated meanings are defined in the individual summary report.

4.4 TOTAL STRONTIUM/STRONTIUM-90 ANALYSIS

The DoD ELAP accredited laboratory, GEL Laboratories, determines total strontium and Sr-90 through separation and isotopic measurement. For soils, the strontium is transferred from the soil into a liquid matrix prior to precipitation as insoluble carbonate. Interferences from calcium and other radionuclides are removed by one or more precipitations of the strontium carrier as strontium nitrate. The yttrium-90 daughter of Sr-90 is removed by a hydroxide precipitation step. The yttrium-90 daughter grows in again and is then separated with stable yttrium carrier as hydroxide and finally precipitated as oxalate and beta counted. Total strontium and Sr-90 are counted for beta particle activity by gas flow proportional counters.

Total strontium analysis includes both Sr-89 and Sr-90. However, due to the short half-life of Sr-89 (50.5 days) versus that of Sr-90 (28.6 years), this result was conservatively reported as the Sr-90 result.

The results of total strontium/Sr-90 analysis were reported in pCi/g and are included as an attachment to each SUPR. Any sample exhibiting Cs-137 activity greater than the release

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criterion was also analyzed for total strontium/Sr-90. In addition, a minimum of 10 percent of samples were randomly chosen for Sr-90 analysis.

4.5 COUNTING UNCERTAINTY

Measurement of radioactivity has some degree of inherent uncertainty. The uncertainty quoted is the standard deviation of the average, obtained from the statistical uncertainties of the individual measurements. This uncertainty results from the combination of all possible inaccuracies in the measurement process, including such factors as the reading of the result, peak integration interpretation, peak background determination, the calibration of the measurement device, numerical rounding errors, and the random nature of radioactivity. Individual radioactivity measurements are accompanied by a plus or minus (±) value, which is an uncertainty term known as either the two-sigma counting error or the total propagated uncertainty.

Total propagated uncertainty includes both counting uncertainty and analytical uncertainty. Because measuring a radionuclide requires a process of counting random radioactive emissions from a sample, the counting uncertainty gives information on what the measurement might be if the same sample were counted again under identical conditions. The counting uncertainty implies that at the specified level of uncertainty a recount of the same sample would give a value somewhere between the reported value minus the counting uncertainty and the reported value plus the counting uncertainty.

The uncertainty in the background is not as simple because the background is a calculated number. The background area uncertainty is the uncertainty in the channels used to calculate the end points of the background multiplied by the ratio of the number of channels in the peak to the number of channels used to calculate the background. For wide peaks and low counts per channel, there is high uncertainty in the calculated background.

4.6 DATA ASSESSMENT

Laboratory data were assessed to determine whether the objectives of the survey process were met. The assessment process consisted of four data phases: verification, validation, evaluation, and quality assessment. The assessment of HPNS laboratory data ensured the objective of the survey, final unrestricted release, was met.

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4.6.1 Data Verification

Data verification ensured that the requirements were implemented as prescribed. Data verification activities included technical reviews of data and audits, as appropriate.

4.6.2 Data Validation

As stated in the ITSI Gilbane SAP (ITSI Gilbane, 2013b), Worksheet #36, radiological analytical results will be validated using Chapter 8 of the Multi-Agency Radiological Laboratory Analytical Protocols Manual (MARLAP; EPA et. al., 2004) and the QC criteria specific in the SAP. Therefore, the guidance document and SAP QC guidelines were used in validation of radiological data. Data not meeting method and/or SAP specifications were flagged as estimated or rejected.

Relative percent difference requirements for laboratory duplicate samples for the DoD ELAP accredited laboratories are provided in Worksheet 28.1 of the ITSI Gilbane SAP (ITSI Gilbane, 2013b).

4.6.3 Data Evaluation

Radiochemical and laboratory data were evaluated prior to submittal to the RASO. The evaluation of data was based on method requirements and results of QC checks, the level of contamination of method blanks and method spikes (as appropriate), and the overall indication of interference due to contamination. The data qualifiers, if used, were listed at the bottom of the data report.

4.6.4 Data Quality Assessment

Data quality assessment (DQA) is a scientific and statistical evaluation that indicates whether the data are of the right type, quality, and quantity to support their intended use. DQA provides the assessment needed to decide if the planning objectives were met. All data presented in the SUPRs were subject to the DQA process, and the data were determined suitable for use. Details of the DQA are provided in the Quality Control Summary Report, presented as Attachment 54.

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5.0 STATISTICAL TEST

As stated in Section 8.2.2.1 of MARSSIM (DoD et al., 2000), there is no need to conduct statistical tests when all sample results are less than the release criteria. Because all FSS sample results (post-remediation) for the SUPRs were less than the release criteria, no statistical tests were included in the SUPRs. However, this section describes the process by which the sampling density was determined such that statistical testing, if necessary, would be sufficient.

5.1 DECISION ERRORS

There are two types of decision errors that were used when performing the statistical tests outlined in MARSSIM (DoD et al., 2000). The first type of decision error, called a Type I error, occurs when the null hypothesis is rejected when it is actually true. This type of error is sometimes called a "false positive." The probability of a Type I error is denoted by α. The Type I error is often referred to as the significance level or size of the test.

The second type of decision error, called a Type II error, occurs when the null hypothesis is not rejected when it is actually false. This type of error is sometimes called a "false negative." The probability of a Type II error is denoted by β. The power of a statistical test is defined as the probability of rejecting the null hypothesis when it is actually false. It is numerically equal to $1-\beta$, where β is the Type II error rate.

Each survey was designed to limit Type I and Type II errors to a maximum probability of 5 percent. It was important to minimize the chances of concluding that a survey unit met the release criteria (reject the null hypothesis) when it actually exceeded the limits (Type I error), and concluding that a survey unit exceeded the release criteria (accept the null hypothesis) when it actually met the limits (Type II error).

5.2 WILCOXON RANK-SUM TEST

The Wilcoxon Rank-Sum (WRS) test is designed to test a hypothesis about the location of a population distribution. It is most often used to test the hypothesis about a population median and often involves the use of matched pairs. For example, reference area and survey unit data were tested for a median difference of zero. This test is also a nonparametric test that may be used when it is only necessary, or possible, to know if observed differences between two conditions were significant. The WRS test is structured to denote a change in magnitude, as

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opposed to any attempt at the quantitative measurement. Per Section 2.5.1.2 of MARSSIM (DoD et al., 2000), the WRS test is the recommended statistical test for comparison of survey unit radionuclide concentrations with background radionuclide concentrations. Although the WRS test was not conducted for the SUPRs because the results for all survey units were less than the release criteria, the number of sampling points was calculated to ensure that sufficient sample results from the survey unit areas were available for statistical comparison had all FSS results not all been lower than each release criterion.

5.2.1 Determining the Numbers of Data Points for the WRS Test

Since the ROCs were present in background, N is calculated in the manner specified for the WRS test using Equation 5-1:

Equation 5-1

$$N = \left(\frac{\left(Z_{1-\alpha} + Z_{1-\beta}\right)^2}{3(P_r - 0.5)^2}\right)(1.2)$$

where:

$Z_{1-\alpha}$ = 1.645 Type I decision error level
$Z_{1-\beta}$ = 1.645 Type II decision error level
P_r = 0.983039 random measurement probability
(from Table 5.1 in MARSSIM [DoD et al., 2000])
(1.2) = 20% increase in number of samples over the minimum

The variable used to calculate N not already specified in the *Basewide Radiological Management Plan, Hunters Point Naval Shipyard, San Francisco, California* (ITSI Gilbane, 2013a) was the random measurement probability P_r. P_r in Equation 5-1 above is based on the relative shift. The relative shift is equal to Δ/σ, where Δ was equal to the release criterion minus the lower boundary of the gray region (LBGR), and σ was an estimate of the standard deviation of the measured values in a survey unit. Half the value of the release criterion was used as the LBGR, as suggested in MARSSIM (DoD et al., 2000). Likewise, σ was assigned the value of the standard deviation of the measurement values in the reference area. For Ra-226, the reference area standard deviation is 0.073 pCi/g (see Table 3-3), which results in a Δ/σ value of 6.8. MARSSIM (DoD et al., 2000), Appendix D.6, states that the design goal should be to achieve Δ/σ values between 1 and 3 as values greater than 3 will not result in significant changes in the

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number of measurements required to support a decision. Therefore, the survey design goal for the relative shift (Δ/σ) was adjusted to 3.

From MARSSIM (DoD et al., 2000), Table 5.1, the P_r value for $\Delta/\sigma = 3.0$ is 0.9830339. Solving Equation 5-1, N was calculated as 19 samples. Since N is the total number of data points for each survey unit/reference area combination, only 10 samples collected from the survey unit are ~~approximately two times the number of samples were collected as were~~ needed to support the WRS test.

5.2.2 ~~Determining Data Points for~~ Limiting the Size of Small Areas of Elevated Activity That Could Be Missed During Sampling

The statistical test described above evaluated whether or not the residual radioactivity in the survey unit exceeded the release criterion for contamination conditions that were approximately uniform across the survey unit. To limit the size of ~~obtain reasonable assurance that~~ any small areas of elevated residual radioactivity ~~were not~~ that could be missed during sampling~~the survey~~, the total number of samples was increased. Equation 5-2 was used to determine the number of samples. The required number of samples, n_{EA}, was developed for an area of elevated activity based on a survey unit size of 1,000 m^2 and a square grid with spacing, L, of 7 meters. Substituting these values into the following equation, n_{EA} was calculated to be 20.

Equation 5-2

$$L = \sqrt{\frac{A}{n_{EA}}}$$

where:

L = length of spacing (m)
A = surface area of the survey unit (1,000 m^2)
n_{EA} = number of samples

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6.0 DOSE MODELING

The intent of a SUPR is to achieve unrestricted release for the associated survey units. In doing so, the most conservative approach was used to model the area using RESRAD Version 6.5. The default residential farmer scenario was used with only two minor changes. The first is that the actual surface area of the trench unit was used (in m^2). The second is that modeling was performed using the mean concentrations of Cs-137, Ra-226 (with the lead-210 daughter), and Sr-90 using the actual activity and replacing it with the MDC value if the reported activity was less than the MDC. The activity used to model for Ra-226 was based on the isotopic net concentration (survey unit mean isotopic concentration minus background mean isotopic concentration). No background is subtracted from the Cs-137 or Sr-90 values, as these radionuclides are not naturally occurring or expected at drain piping depths from global fallout from nuclear weapons testing.

The default parameters included external and inhalation exposures and exposure from the following ingestion pathways: drinking water, food grown with contaminated irrigation water, food grown on contaminated soil, fish, and inadvertent ingestion of soil. Land-based foods considered were leafy vegetables, root vegetables, fruit, grain, beef, poultry, milk, and eggs. Animal feeds included fresh forage, as well as stored grain and hay. This was the most conservative scenario as it assumed that persons living on the site can use the land for any purpose without land use restrictions.

If none of the concentrations for the ROCs for the specific survey unit were greater than zero, the net dose and increased health risk for the survey unit were expected to be negligible, and no further modeling was determined to be reasonable or prudent.

The final dose and health risk modeling summary, if needed, are included in each SUPR. Demonstration of both a net residual dose of less than 25 mrem/y and a net increased health risk in the range of 10^{-4} to 10^{-6} to the average member of the critical group (or member of the public) will be considered sufficient for unrestricted release of the survey unit.

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7.0 RECOMMENDATION OF FINAL UNRESTRICTED RELEASE

A SUPR will be prepared for each zone with recommendations for free release. Each SUPR provides general background information about the trench survey unit and associated excavated soil survey units that are part of the survey unit package. Any exceptions to the ROCs are addressed in each SUPR.

A discussion of the FSS includes the activities performed for the trench survey unit and each excavated soil survey unit or import fill that was used for backfill. The discussion includes results of the soil samples. Gamma scan data are provided as attachments in each SUPR.

The version of RESRAD used and the results of the dose and risk modeling are provided as an attachment and discussed in each SUPR.

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8.0 REFERENCES

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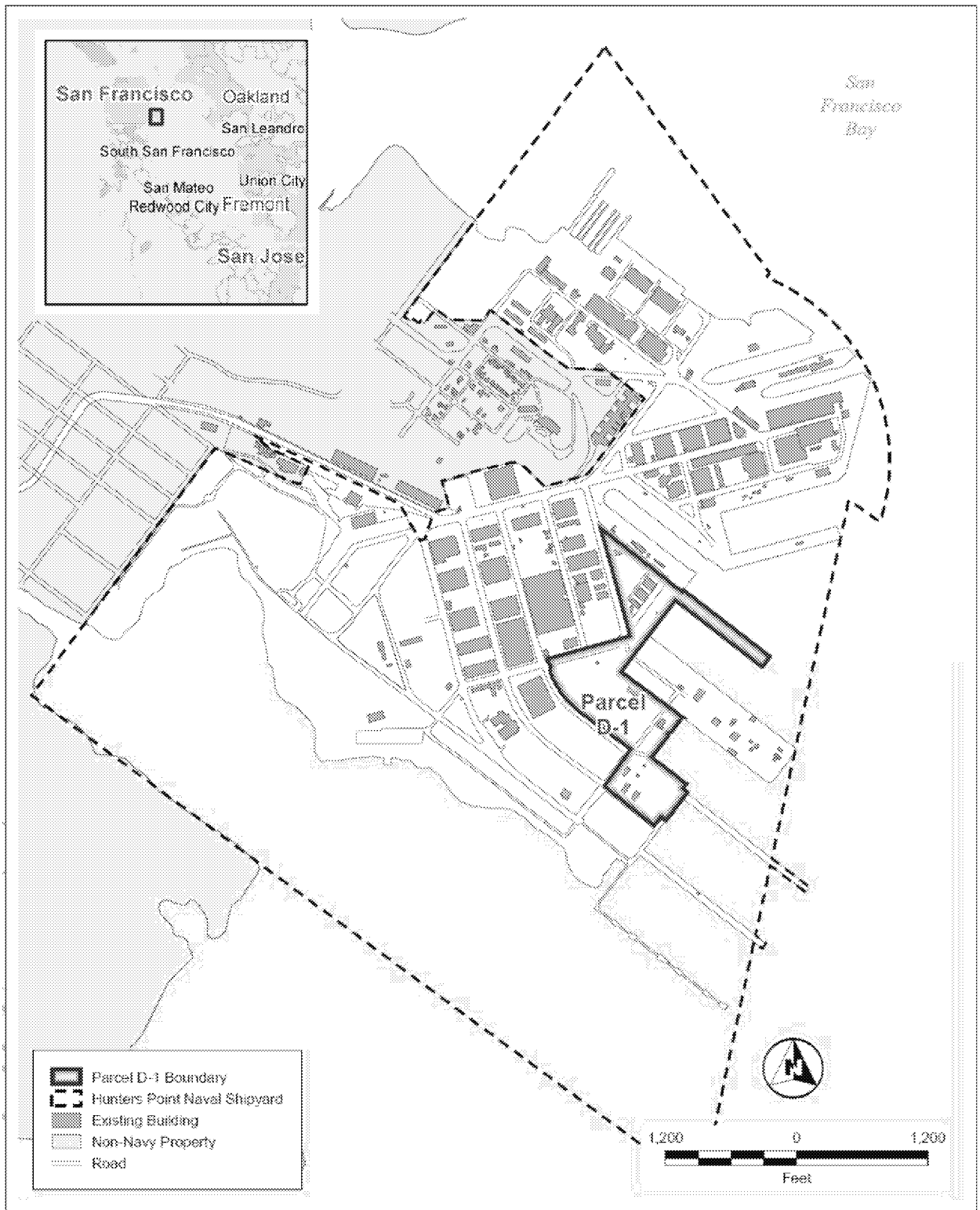
FIGURES



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Radiological Remediation and Support
Parcel D-1, Phase II
Hunters Point Naval Shipyard
San Franscisco, California

Figure 1-1
Parcel D-1 Vicinity Map

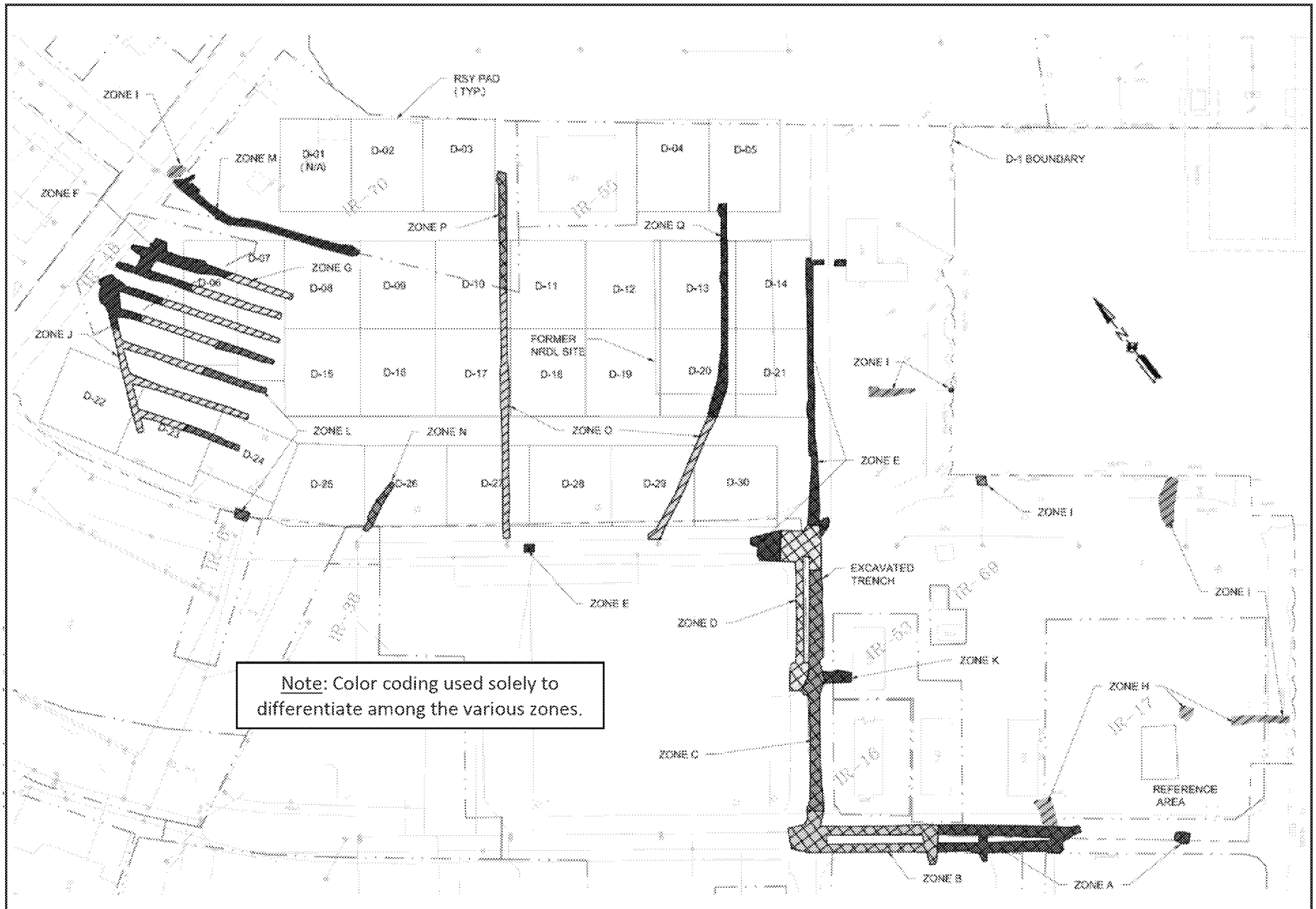

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Figure 3-1
Parcel D-1 Phase II Zones

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Radiological Remediation and Support
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Figure 3-2
Typical RSY Systematic Sample Locations

ATTACHMENTS
(On CD)

ATTACHMENT 1

HUNTERS POINT FINAL BASEWIDE RADIOLOGICAL REMOVAL ACTION ACTION MEMORANDUM – REVISION 2006

(On CD)

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ATTACHMENT ~~21~~

RSI RS-700 GAMMA-RAY DETECTION SYSTEM SPECIFICATIONS (On CD)

(replace page with .pdf document – 23 pages)

ATTACHMENT 3~~2~~

IMPORT FILL MATERIAL (JERICHO) SAMPLE RESULTS (On CD)

(replace page with .pdf document – 23 pages)

ATTACHMENT ~~43~~

REFERENCE AREA SAMPLE RESULTS (On CD)

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ATTACHMENT 54

QUALITY CONTROL SUMMARY REPORT

(replace page with Quality Control Summary Report document)

ATTACHMENT 65

RESPONSE TO COMMENTS