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October 24, 2016

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Reactor Decommissioning Branch
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**SUBJECT: FINAL REPORT—INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR SURVEY UNITS NOL01-03, NOL01-08, OOL10-12, AND TRL 50 AT THE HUMBOLDT BAY POWER PLANT EUREKA, CALIFORNIA
DOCKET NO. 50-133 RFTA 15-006; DCN 5272-SR-02-0**

Dear Mr. Hickman:

The Oak Ridge Institute for Science and Education (ORISE), managed by ORAU for DOE, is pleased to provide the enclosed final report that details the confirmatory surveys that were performed during the period of September 30 through October 1, 2015 and August 9 through 11, 2016, at the Humboldt Bay Power Plant in Eureka, CA.

Please feel free to contact me at 865.574.6273, or Tim Vitkus at 865.576.5073, if you have any questions or comments.

Sincerely,

Nick A. Altic
Health Physicist/
Project Manager
ORAU

NAA:lw

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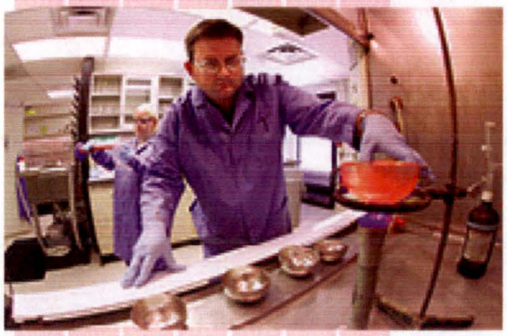

FINAL REPORT—INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR SURVEY UNITS NOL01-03, NOL01-08, OOL10-12, AND TRL 50 AT THE HUMBOLDT BAY POWER PLANT, EUREKA, CALIFORNIA

Nick Altic

Prepared for the U. S. Nuclear Regulatory Commission

October 2016

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INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR SURVEY UNITS NOL01-03, NOL01-08, OOL10-12, AND TRL 50 AT THE HUMBOLDT BAY POWER PLANT EUREKA, CALIFORNIA

OAK RIDGE INSTITUTE FOR SCIENCE AND EDUCATION
Managed by ORAU for DOE

Prepared by
N. A. Altic
ORAU

OCTOBER 2016

FINAL REPORT

Prepared for the
U.S. Nuclear Regulatory Commission

This document was prepared for the U.S. Nuclear Regulatory Commission (NRC) by the Oak Ridge Institute for Science and Education (ORISE) through an interagency agreement (NRC FIN No. F-1244) between the NRC and the U.S. Department of Energy (DOE). ORISE is managed by Oak Ridge Associated Universities under DOE contract number DE-SC0014664.

OAK RIDGE INSTITUTE FOR SCIENCE AND EDUCATION
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INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR SURVEY UNITS NOL01-03, NOL01-08, OOL10-12, AND TRL 50 AT THE HUMBOLDT BAY POWER PLANT EUREKA, CALIFORNIA

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FINAL REPORT

OCTOBER 2016

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ACRONYMS

AA	alternate action
AEC	Atomic Energy Commission
cpm	counts per minute
DCGL	derived concentration guideline level
$DCGL_W$	gross activity derived concentration guideline level
DOE	U.S. Department of Energy
dpm	disintegrations per minute
DQO	data quality objective
EPA	U.S. Environmental Protection Agency
FSS	final status survey
FSSP	final status survey planning
FSSR	final status survey report
HBPP	Humboldt Bay Power Plant
HBRP	Humboldt Bay Repowering Project
IEAV	Independent Environmental Assessment and Verification
LTP	License Termination Plan
MDC	minimum detectable concentration
MDCR	minimum detectable count rate
MeV	million electron volts
NaI	sodium iodide
NIST	National Institute of Standards and Technology
NRC	U.S. Nuclear Regulatory Commission
NRIP	NIST Radiochemistry Intercomparison Program
ORISE	Oak Ridge Institute for Science and Education
pCi/g	picocuries per gram
PG&E	Pacific Gas & Electric Company
PSQ	principal study question
ROC	radionuclide of concern
RSS	ranked set sampling
SAFSTOR	shutdown and safety storage
SOF	sum of fractions
TAP	total absorption peak
TRL 50	Trailer #50

FINAL REPORT—INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR SURVEY UNITS NOL01-03, NOL01-08, OOL10-12, AND TRL 50 AT THE HUMBOLDT BAY POWER PLANT EUREKA, CALIFORNIA

EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) requested that Oak Ridge Institute for Science and Education (ORISE), managed by Oak Ridge Associated Universities (ORAU) for the U.S. Department of Energy (DOE), perform an independent confirmatory survey at the Humboldt Bay Power Plant in Eureka, CA. Pacific Gas and Electric (PG&E) and its decommissioning contractor are currently engaged in the decontamination and decommissioning of the approximate 143-acre site.

ORISE performed independent assessment activities of one survey unit in the discharge canal (OOL10-12) and Trailer 50 (TRL 50), which included gamma, alpha, and beta radiation surveys and soil sampling during the period of September 30 through October 1, 2015. Also, during the period of August 9 through 11, 2016, ORISE performed independent assessment activities—including gamma and beta radiation surveys and soil sampling—of the accessible exterior portion of the intake canal and survey units NOL01-03 and NOL01-08. The results of ORISE gamma, alpha, and beta radiation surveys, combined with laboratory analytical results of soil samples, satisfies the NRC-approved soil and surface activity derived concentration guideline levels (DCGLs) described in PG&E's license termination plan (PGE 2014a).

FINAL REPORT—INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR SURVEY UNITS NOL01-03, NOL01-08, OOL10-12, AND TRL 50 AT THE HUMBOLDT BAY POWER PLANT EUREKA, CALIFORNIA

1. INTRODUCTION

The Pacific Gas & Electric Company (PG&E) operated the Humboldt Bay Power Plant (HBPP) Unit 3 nuclear reactor near Eureka, California under Atomic Energy Commission (AEC) provisional license number DPR-7. HBPP Unit 3 achieved initial criticality in February 1963 and began commercial operations in August 1963. Unit 3 was a natural circulation boiling water reactor with a direct-cycle design. Stainless steel fuel claddings were used from startup until cladding failures resulted in plant system contamination. Zircaloy-clad fuel was used exclusively starting in 1965, eliminating cladding-related contamination. A number of spills and gaseous releases were reported during operations, resulting in a range of mitigation activities (ESI 2008).

In July 1973, Unit 3 was shut down for annual refueling and seismic modifications. However, by December 1980, it was concluded that completing the required upgrades and restarting Unit 3 would be cost prohibitive. PG&E decided in June of 1983 to decommission Unit 3, received a possession-only license amendment, and placed the unit into cold shutdown and safety storage (SAFSTOR). The impacted areas associated with Unit 3 are currently undergoing decommissioning. As part of the Humboldt Bay Repowering Project (HBRP), PG&E has built ten new fossil fuel units (16.3 MWe [megawatt electric] each) on the site in the vicinity of Unit 3. Decommissioning activities have also been completed on the adjacent fossil fuel Units 1 and 2, with all materials being removed to ground level (ESI 2008).

The U.S. Nuclear Regulatory Commission (NRC) requested that Oak Ridge Institute for Science and Education (ORISE), managed by Oak Ridge Associated Universities (ORAU) for the U.S. Department of Energy (DOE), perform confirmatory survey activities at the HBPP. The areas investigated included survey units NOL01-03, NOL01-08, OOL10-12, and Trailer #50 (TRL 50). This report summarizes the confirmatory survey activities associated with these areas.

2. SITE DESCRIPTION

The HBPP site, owned by PG&E, consists of 143 acres on the southern edge of Humboldt Bay, four miles southwest of the town of Eureka in Humboldt County, California. Figure 2.1 provides an aerial view of the HBPP. PG&E maintains ten new operating electric generating units at the HBPP site (in the New Generation Footprint Area) that run on fossil fuels, two non-operating fossil fuel units (Units 1 and 2), and one non-operational nuclear unit (Unit 3). Units 1 and 2, which were decommissioned to ground level, were interconnected with and west of Unit 3 (ESI 2008). The remaining property includes mostly open areas and protected wetlands.

Figure 2.1. HBPP Aerial View

3. DATA QUALITY OBJECTIVES

The data quality objectives (DQOs) described herein are consistent with the *Guidance on Systematic Planning Using the Data Quality Objectives Process* (EPA 2006) and provide a formalized method for planning radiation surveys, improving survey efficiency and effectiveness, and ensuring that the type, quality, and quantity of data collected are adequate for the intended decision applications. The seven steps in the DQO process are outlined below:

1. State the problem
2. Identify the decision/objective
3. Identify inputs to the decision/objective
4. Define the study boundaries
5. Develop a decision rule
6. Specify limits on decision errors
7. Optimize the design for obtaining data

3.1 STATE THE PROBLEM

The first step in the DQO process defines the problem that necessitates the study. NRC requested that ORISE perform confirmatory surveys at the HBPP. The objectives of the confirmatory surveys were to provide independent contractor document and field reviews and generate independent radiological data to assist NRC in evaluating the adequacy and accuracy of the licensee's final status survey (FSS) results. The problem statement is as follows:

> Perform confirmatory surveys to assess and determine the adequacy of the licensee's FSS design, implementation, and documentation for demonstrating compliance with the release criteria.

3.2 IDENTIFY THE DECISION

The second step in the DQO process identifies the principal study question (PSQ) and alternate actions (AAs), develops a decision statement, and organizes multiple decisions, as appropriate. This

is done by specifying AAs that could result from a "yes" response to the PSQ and combining the PSQ and AAs into a decision statement. Table 3.1 presents the PSQ, AAs, and combined decision statement.

Table 3.1. HBPP Confirmatory Survey Decision Process	
Principal Study Question	**Alternative Actions**
In order to select "Yes" from the AAs, the subsequent questions must all be answered "Yes." • Are the results of PG&E's FSS representative of the current radiological status? • Were field and laboratory instrumentation adequate and appropriate relative to the site's ROCs? • Were the survey units/areas classified correctly?	**Yes:** PG&E has demonstrated compliance with the release criteria through adequate measurements, sampling, and analysis—agree with PG&E's decision to release the survey unit/area. **No:** PG&E has not adequately demonstrated compliance with the release criteria. ORAU will provide technical comments and NRC may alter remedial action planning to address contamination and/or documentation issues.
Decision Statement	
Determine whether PG&E has made the appropriate decision regarding the final radiological status of the survey area/unit investigated relative to the release criteria.	

3.3 IDENTIFY INPUTS TO THE DECISION

The third step in the DQO process identifies both the information needed and the sources of this information, determines the basis for action levels, and identifies sampling and analytical methods that will meet data requirements. For this effort, information inputs included the following:

- HBPP characterization data
- HBPP Geographic Information System and Visual Sample Plan files, for survey unit boundaries
- HBPP's FSS data and supporting documentation, if available
- ORISE soil sample analytical results
- ORISE surface activity measurements

3.3.1 Radionuclides of Concern

The primary radionuclides of concern (ROCs) identified for the HBPP are beta-gamma emitters—fission and activation products—resulting from reactor operation. The HBPP DCGLs for soils and building surfaces are presented in Table 3.2 and Table 3.3, respectively (PGE 2014a).

Table 3.2. HBPP Soil DCGLs

ROC[a]	DCGL (pCi/g)[a]	ROC	DCGL (pCi/g)	ROC	DCGL (pCi/g)
Am-241	25	Cs-137	7.9	Np-237	1.1
C-14	6.3	Eu-152	10	Pu-238	29
Cm-243	29	Eu-154	9.4	Pu-239	26
Cm-244	48	H-3	680	Pu-240	26
Cm-245	17	Nb-94	7.1	Pu-241	860
Cm-246	25	Ni-59	1900	Sr-90	1.5
Co-60	3.8	Ni-63	720	Tc-99	12

[a]pCi/g = picocuries per gram

Table 3.3. HBPP Surface Activity DCGLs

ROC[a]	DCGL (dpm/100 cm^2)[a]	ROC	DCGL (dpm/100 cm^2)	ROC	DCGL (dpm/100 cm^2)
Am-241	3.00E+03	Cs-137	4.60E+04	Np-237	2.40E+03
C-14	7.00E+06	Eu-152	2.70E+04	Pu-238	3.40E+03
Cm-243	4.30E+03	Eu-154	2.50E+04	Pu-239	3.10E+03
Cm-244	5.50E+03	H-3	1.80E+08	Pu-240	3.10E+03
Cm-245	2.20E+03	Nb-94	1.90E+04	Pu-241	1.40E+05
Cm-246	2.70E+03	Ni-59	6.30E+07	Sr-90	9.70E+04
Co-60	1.30E+04	Ni-63	2.40E+07	Tc-99	9.60E+06

[a]dpm/100cm^2 = disintegrations per minute per 100 square centimeters

Each scaled radionuclide-specific $DCGL_W$ represents the concentration above background of a residual radionuclide that would result in a radiological dose of 25 millirem per year (mrem/yr) to the average member of the critical group. Because each of the individual $DCGL_W$ represents 25 mrem/yr, the sum-of-fractions (SOF) approach is used to demonstrate compliance with the dose limit. SOF calculations are performed as follows:

$$\mathrm{SOF_{TOTAL}} = \sum_{j=0}^{n} \mathrm{SOF_j} = \sum_{j=0}^{n} \frac{\mathrm{C_j}}{\mathrm{DCGL_{W,j}}}$$

Where C_j is the concentration of ROC "j," and $DCGL_{W,j}$ is the $DCGL_W$ for ROC "j." Note that gross concentrations are considered here for conservatism.

3.4 Define the Study Boundaries

The fourth step in the DQO process defines target populations and spatial boundaries, determines the timeframe for collecting data and making decisions, addresses practical constraints, and determines the smallest subpopulations, area, volume, and time for which separate decisions must be made. The NRC informed ORISE of the specific survey units/areas that were subject to the confirmatory survey scope detailed in the approved survey plan (ORAU 2016a). Survey units NOL01-03 and NOL01-08, in the upper yard area, were combined into a single confirmatory unit based on like conditions and contamination potential.

ORISE will review the licensee's final status survey plans, associated procedures prior to on-site confirmatory survey activities. Reviews were performed to assess the adequacy of the licensee's documentation, while taking into account NUREG-1757, *Consolidated NMSS Decommissioning Guidance* (NRC 2006), and NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* guidance (NRC 2000).

3.5 Develop a Decision Rule

The fifth step in the DQO process specifies appropriate population parameters (e.g., mean, median), confirms action levels are above detection limits, and develops an if...then... decision rule statement. For this survey effort, the parameter of interest was the mean sum of fractions (SOF) for the survey unit/area. The decision rule can be stated as:

> If the average SOF as determined by ORISE is within the respective error of HBPP's average SOF for the investigated survey unit/area, all results are below the respective guidelines, and project documentation is complete then recommend

acceptance; if insufficient, then perform further evaluations and provide technical comments/recommendations.

For survey unit OOL10-12, in the discharge canal, judgmental only measurements may be collected to validate the licensee's classification due to accessibility issues, see Section 5.4 for details.

3.6 Specify Limits on Decision Errors

The sixth step in the DQO process specifies the decision maker's limits on decision errors, which are then used to establish performance goals for the survey. Two orders of control were implemented to confirm that HBPP's FSS data are acceptable and accurate.

The first order of control is the degree to which the SOFs reported by ORISE and HBPP should agree. The 95% confidence interval of each mean will be determined. If the two confidence intervals overlap, then HBPP's and confirmatory survey results are considered to be in agreement. A disagreement may not indicate that HBPP's results are inadequate. Class 3 survey units/areas, by definition, will have little to no residual ROC concentration. In these situations, comparing data populations where results are likely to be near or below the method minimum detectable concentrations (MDCs) with large relative uncertainties can be inconclusive. Furthermore, when the action level is substantially greater than the MDCs, comparison of the population parameters may be unnecessary.

The second order of control will be to optimize the MDCs of analyses performed by ORISE, both for field and laboratory measurements. Measurement MDCs were, at a minimum, equal to 50% of the guidelines presented in Section 3.3.1.

3.7 Optimize the Design for Obtaining Data

The seventh step in the DQO process is used to review DQO outputs, develop data collection design alternatives, formulate mathematical expressions for each design, select the sample size to satisfy DQOs, decide on the most resource-effective design of agreed alternatives, and document requisite details. Specific survey procedures are presented in Section 6.

4. DOCUMENT REVIEW

Prior to on-site activities, ORISE reviewed PG&E's FSS survey unit packages/work instructions for the upper yard area (survey units NOL01-03 and NOL01-08), portion of the discharge canal (survey unit OOL10-12), and the intake structure (PGE 2016a, PGE 2014b, PGE 2016b). The FSS survey unit packages were reviewed for adequacy and appropriateness while taking into account the MARSSIM guidance (NRC 2000).

5. PROCEDURES

During the period of August 9, 2016 through August 11, 2016, ORISE performed a confirmatory survey of upper yard land area (survey units NOL01-03 and NOL01-08) and the intake structure at the HBPP. The confirmatory survey of the intake discharge canal (survey unit OOL10-12) and TRL 50 was performed during the period of September 30, 2015 through October 1, 2015. These confirmatory surveys were performed in accordance with plans dated August 2016 and July 2015, and submitted to and approved by the NRC (ORAU 2016a, ORAU 2015a). This report summarizes the procedures and results of the survey. The ORISE survey team performed visual inspections, measurements, and sampling activities within the accessible survey areas specifically requested by the NRC. Survey activities were conducted in accordance with the *ORAU Survey Procedures Manual* and the *ORAU Environmental Services and Radiation Training Quality Program Manual* (ORAU 2015b and 2016b).

5.1 REFERENCE SYSTEM

ORISE referenced confirmatory measurement/sampling locations to the licensee's reference system for structural survey areas and global positioning system (GPS) coordinates (NAD 1983 California State Plane Zone 1 measured in feet) for land areas.

5.2 SURFACE SCANS

Surface scans of land areas were performed with Ludlum Model 44-10 NaI scintillation detectors coupled to ratemeter-scalers with audible indicators. Scan coverage was dependent on the survey unit classification and was commensurate with the potential for residual radioactivity. Instruments

were also coupled to GPS systems that enable real-time gamma count rate and spatial data capture. Locations of elevated direct radiation, suggesting the presence of residual contamination, were marked and identified for further investigation.

Surface scans of structures were performed with either Ludlum Model 43-68 gas-flow proportional detectors or Ludlum Model 44-142 plastic beta scintillators coupled to Ludlum Model 2221 ratemeter-scalers for direct alpha-plus-beta radiation or beta radiation, respectively. When available, the detectors were also coupled to dataloggers to electronically record all scanning data points. Scan coverage was dependent on the survey unit classification and was commensurate with the potential for residual radioactivity. Scans were focused on areas with the highest likelihood of contamination potential. Locations of elevated direct radiation, suggesting the presence of residual contamination nearing the $DCGL_W$, were marked and identified for further investigation.

5.3 SURFACE ACTIVITY MEASUREMENTS

For the intake structure, total beta surface activity measurements were performed with Ludlum Model 43-68 gas-flow proportional detectors coupled to Ludlum Model 2221 ratemeter-scalers. A mylar thickness of 3.8 mg/cm^2 was used to shield any alpha component for the beta measurements. Surface activity measurements were collected side-by-side the licensee from random-start systematic locations as determined by the licensee for their FSS.

Alpha and beta total surface activity measurements were performed at judgmental locations in TRL 50 using a Ludlum Model 43-92 and 44-142 detector coupled to a Ludlum Model 2221 ratemeter-scaler. Smear samples, to determine removable alpha and beta activity levels, were collected from each of the direct measurement locations.

5.4 SOIL SAMPLING

A ranked set sampling (RSS) process, following U.S. Environmental Protection Agency (EPA) guidance, was used to generate the random locations from which the soil samples were collected (EPA 2006). The process combines random sampling with the use of professional judgment to select sampling locations. One-minute static gamma measurements collected from a population of random locations provided the measurable field screening method. Gamma measurements were performed at the surface of the RSS locations. The count data was then ranked (high, medium, or

low) to determine which location is sampled and submitted for laboratory analysis. Samples were collected at a depth of 0–15 cm from the remediated surface. Additional samples were collected from locations of elevated direct gamma radiation detected during surface scans or if field investigations indicated the potential for subsurface contamination. Since the licensee's FSS data was unavailable prior to the confirmatory survey, nine confirmatory samples were collected from 27 field ranking locations. Due to co-location, and similar likelihood for contamination, accessible portions of survey units NOL01-03 and NOL01-08, in the upper yard, were combined into a single confirmatory survey unit. The RSS process was applied to the single confirmatory survey unit, resulting in the collection of nine soil samples.

When ORISE staff arrived on-site during the September 2015 confirmatory survey, survey unit OOL10-12 in the discharge canal contained significant water. Therefore, only two judgmental soil samples were collected from the small portion of OOL10-12 that was accessible.

6. SAMPLE ANALYSIS AND DATA INTERPRETATION

Samples were returned to the ORISE Radiological and Environmental Analytical Laboratory in Oak Ridge, Tennessee for analysis and interpretation. Sample analyses were performed in accordance with the *ORAU Radiological and Environmental Analytical Laboratory Procedures Manual* (ORAU 2016c). Soil samples were analyzed by solid-state gamma spectroscopy for gamma-emitting ROCs. Analytical results were reported in units of picocuries per gram (pCi/g). Smear samples were analyzed for alpha/beta activity using a low-background proportional counter and results reported in units of disintegrations per minute per 100 square centimeters (dpm/100 cm^2). Direct measurement data were converted to units dpm/100 cm^2.

Sample custody was transferred to the ORISE Radiological and Environmental Analytical Laboratory in Oak Ridge, Tennessee. Sample analyses will be performed in accordance with the *ORAU Radiological and Environmental Analytical Laboratory Procedures Manual* (ORAU 2016c). Results of confirmatory survey activities are presented in this draft report and provided to NRC for review and comment. Data collected as part of this survey will be archived by ORISE. Surplus soil and miscellaneous material samples will be returned to the site for disposal.

7. FINDINGS AND RESULTS

The results for each of the verification activities are discussed below.

7.1 DOCUMENT REVIEW

The ORISE reviews of project documentation indicated that all procedures and methods implemented for the FSSs were appropriate and that the resultant data were acceptable. After review of the FSS data packages provided by the licensee, it was determined that they had accurately and adequately explained the sequence of FSS activities at the HBPP.

7.2 SURFACE SCANS

Surface scan results for the interior and exterior investigated areas are discussed below.

7.2.1 Structural Surfaces

Surface scans of the available portion of the intake structure did not identify radiation levels distinguishable from background. Instrument response during the intake structure scan ranged from 200 to 600 cpm. Due to the configuration of the scaffolding used to access the intake structure the surface scans were not electronically recorded for personnel safety reasons.

Alpha and beta surface scans of TRL 50 did not identify any locations with elevated direct radiation.

7.2.2 Land Areas

Exterior open-land scans performed with a NaI scintillation detector coupled with a ratemeter-scaler were approximately background levels for the upper yard area, with the exception noted below. Scan ranges were approximately 4,000 to 8,200 cpm. As scans progressed toward the northern most corner of NOL01-03, the gamma scan count rates peaked at approximately 8,000 cpm. This location was flagged for further investigation and sampling. A judgmental soil sample, 5272S0011, was collected from this location (see Section 8.4 for more details). Figure A-1 shows the gamma scan coverage and the gamma radiation count rates for the upper yard area.

Surface scans of survey unit OOL10-12 in the discharge canal. Instrument response ranged from 3,000 to 5,000 cpm. No elevated direct gamma radiation was identified.

7.3 SURFACE ACTIVITY MEASUREMENTS

Total surface activity levels for the intake structure and TRL 50 are provided in Table B-1 and Table B-2, respectively. Total surface activity levels were not corrected for material-specific background, but rather an ambient background was applied for the surface activity calculation. The total surface activity levels reported by PG&E represent gross activity levels that were not corrected for background. Therefore, total surface activity levels reported by PG&E are significantly greater that those reported by ORISE and a direct comparison is not appropriate.

Table 7.1. Total Surface Activity Measurement Summary (dpm/100 cm^2)

Statistical Parameter	Intake Structure[a,b]		TRL 50[c]
	ORISE	PG&E	ORISE
Mean	360	1,133	17 / -17
Min	300	987	-9 / -210
Max	480	1,256	100 / 280

[a]Results represent total beta surface activity levels.
[b]Ambient background used in ORISE calculations; PG&E did not correct for background.
[c]Format is alpha/beta total surface activity level.

7.4 RADIONUCLIDE CONCENTRATIONS IN SOIL

A total of 10 soil samples (9 random and 1 judgmental) were collected from the upper yard area—which consists of survey units NOL01-03 and NOL01-08. Individual soil sample results for the gamma-emitting fission/activation products that PG&E has identified as site-related contaminants for the upper yard area are presented in Table B-3. The analysis results are summarized in Table 7.2 below. Of the nine random soil samples collected, only one (sample ID 5272S0018) had a Cs-137 concentration above the analytical MDC. Cobalt-60 was not identified above the analytical MDC in any of the random samples. One judgmental soil sample (sample ID 5272S0011) was collected from a location exhibiting elevated direct gamma radiation. Neither Co-60 nor Cs-137 was identified in the judgmental sample above the analytical MDC. The elevated direct gamma radiation was due to detector geometry as the judgmental location was inside of a small hole.

PG&E provided ORISE with their FSS soil sample results for the upper yard (PGE 2016c). However, non-detects were reported as "ND" making a comparison impractical. Of the 35 FSS soil

samples taken by PG&E to-date, only five had a detectable concentration of Cs-137; no Co-60 was detected in any sample. The maximum Cs-137 concentration reported by PG&E was 0.668 pCi/g.

A total of two judgmental soils samples were collected from survey unit OOL10-12 in the discharge canal. Individual soil sample results for OOL10-12 are provided in Table B-4. Neither of the two samples had radionuclide concentrations above the analytical MDC for the site-related contaminants.

Table 7.2. Radionuclide Concentration in Confirmatory Soil Samples Summary (pCi/g)				
Statistical Parameter	**Upper Yard**		**OOL10-12**	
	Cs-137	**Co-60**	**Cs-137**	**Co-60**
Number of Samples	10 (9 random, 1 judgmental)		2 (both judgmental)	
Mean	0.014	-0.004	N/A	N/A
Min	-0.034	-0.017	0.006	-0.016
Max	0.140	0.005	0.027	-0.006

8. COMPARISON OF RESULTS WITH GUIDELINES

The total beta surface activity values for the intake structure and TRL 50 were directly compared with the $DCGL_W$ for Co-60, as this is the most conservative value for beta radiation. Total alpha surface activity values for TRL 50 were directly compared with the $DCGL_W$ for Np-237, as this is the most conservative value for alpha radiation. All values were less than their corresponding $DCGL_W$.

Laboratory analysis results of all soil samples were significantly less than the $DCGL_W$ values. The unity rule was applied to the upper yard confirmatory unit, and the average SOF was 0.00.

9. SUMMARY

At the request of the NRC, ORISE conducted confirmatory surveys of at the HBPP during the period of September 30 through October 1, 2015 and August 9 through 11, 2016. The survey

activities included visual inspections, measurement and sampling activities. Confirmatory activities also included the review and assessment of the licensee's project documentation and methodologies. The licensee was forthcoming with information requested by ORISE staff.

The FSS data packages that were reviewed accurately and adequately described the sequence of FSS activities and the radiological conditions at the site. The only recommendation, which is minor, is for PG&E to report the actual value of all analytical results rather than "ND," as recommended in the MARSSIM. All confirmatory surface activity measurement results were below the respective $DCGL_W$ values for the ROCs as specified in the final status survey planning (FSSP). Soil sample activities were below significantly less than the $DCGL_W$ values. ORISE's survey data verifies that the radiological conditions of the confirmatory survey units are below the $DCGL_W$ value requirements stated in the licensee's FSS plan (PGE 2014a). Confirmatory survey activities validated the licensee's classifications, radiological status, and satisfaction of the guidelines.

10. REFERENCES

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APPENDIX A
FIGURES

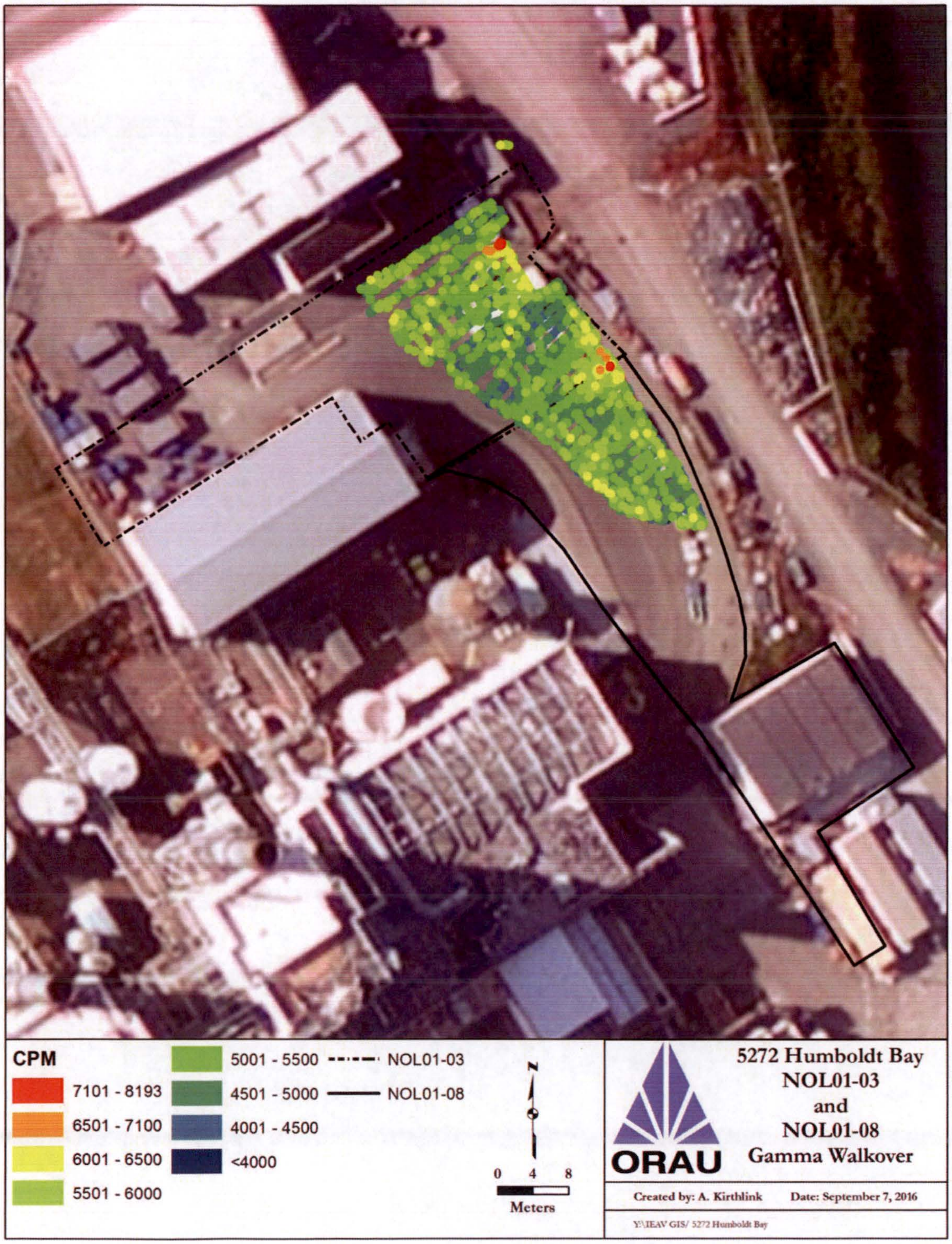

Figure A-1. Upper Yard Area Gamma Walkover Results

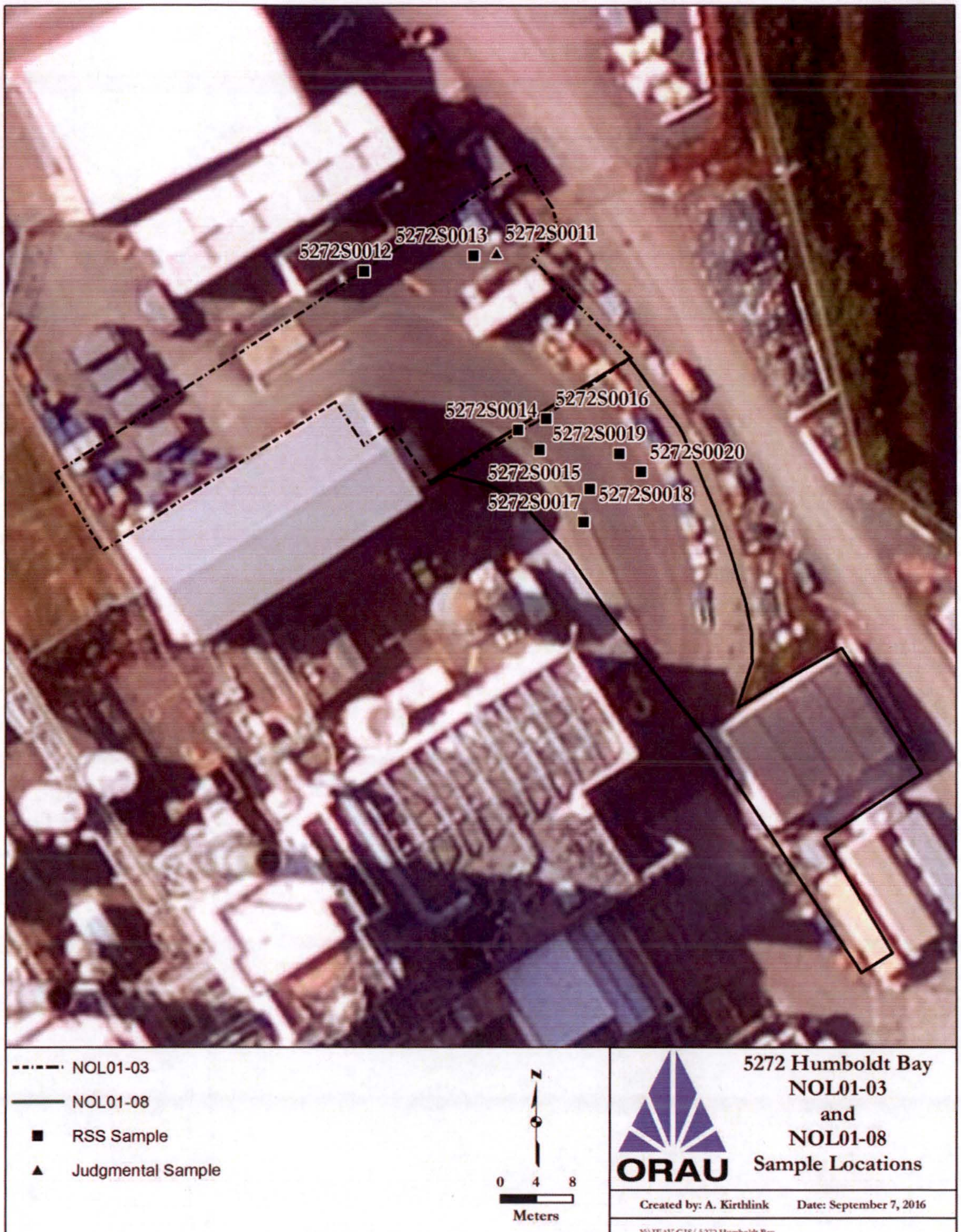

Figure A-2. Upper Yard Area Confirmatory Sample Locations

Figure A-3. OOL10-12 Sample Locations

APPENDIX B
TABLES

Table B-1. Total Beta Surface Activity for the Intake Structure				
Location ID	Gross Count Rate (cpm)		Total Beta Surface Activity (dpm/100 cm²)	
	ORAU	PG&E	ORAU	PG&E
1	211	233	300	1,064
2	213	250	320	1,142
3	222	275	390	1,256
4	233	246	480	1,124
5	218	216	360	987
6	215	250	330	1,142
7	220	252	370	1,151
8	215	262	330	1,197

Table B-2. Surface Activity Measurements from TRL 50								
Material	Location ID	Surface location	Raw Count Rate (cpm)		Total Surface Activity (dpm/100 cm2)		Removable Surface Activity (dpm/100 cm2)	
			Alpha	Beta	Alpha	Beta	Alpha	Beta
Carpet	R0012	Floor	4	251	26	-38	-1	1
Carpet	R0013	Floor	1	254	0	-15	1	4
Carpet	R0014	Floor	2	266	9	76	1	0
Linoleum	R0015	Floor	2	243	9	-99	1	7
Linoleum	R0016	Floor	0	262	-9	46	-1	0
Carpet	R0017	Floor	13	293	100	280	-1	-4
Carpet	R0018	Floor	2	228	9	-210	1	0
Carpet	R0019	Floor	1	254	0	-15	-1	1
Carpet	R0020	Floor	2	232	9	-180	1	-1

Table B-3. Radionuclide Concentration in Soil for the Upper Yard Confirmatory Unit								
Sample ID	Sample Type	Concentration (pCi/g)[a]						SOF
		Cs-137			Co-60			
NOL01-03								
5272S0011	Judgmental	-0.025	±	0.020	-0.012	±	0.025	0.00
5272S0012	Random	0.008	±	0.015	0.003	±	0.016	0.00
5272S0013	Random	0.0075	±	0.0099	0.000[b]	±	0.013	0.00
NOL01-08								
5272S0014	Random	0.014	±	0.011	-0.009	±	0.011	0.00
5272S0015	Random	0.003	±	0.018	-0.007	±	0.019	0.00
5272S0016	Random	-0.034	±	0.018	-0.004	±	0.021	0.00
5272S0017	Random	-0.014	±	0.020	-0.017	±	0.016	0.00
5272S0018	Random	0.140[b]	±	0.023	-0.009	±	0.015	0.02
5272S0019	Random	0.016	±	0.020	0.005	±	0.011	0.00
5272S0020	Random	-0.014	±	0.017	-0.002	±	0.020	0.00
Mean		0.014			-0.004			0.00
Std. Deviation		0.050			0.007			0.01
Min		-0.034			-0.017			0.00
Max		0.140			0.005			0.02

[a]Uncertainty represents the total propagated uncertainty reported at the 95% confidence level.
[b]Zero values are due to rounding.
[c]This value represents the only result above the analytical MDC.

Table B-4. Radionuclide Concentration in Soil for Survey Unit OOL10-12							
Sample ID	Sample Type	Concentration (pCi/g)[a,b]					
		Cs-137			Co-60		
5272S0009	Judgmental	0.027	±	0.015	-0.016	±	0.020
5272S0010	Judgmental	0.006	±	0.011	-0.006	±	0.020

[a]Uncertainty represents the total propagated uncertainty reported at the 95% confidence level.
[b]No gamma-emitting ROC was identified above the analytical MDC.

APPENDIX C
MAJOR INSTRUMENTATION

The display of a specific product is not to be construed as an endorsement of the product or its manufacturer by the author or his employer.

C.1 SCANNING AND MEASUREMENT INSTRUMENT/DETECTOR COMBINATIONS

C.1.1 GAMMA

Ludlum NaI Scintillation Detector Model 44-10, Crystal: 5.1 cm × 5.1 cm
(Ludlum Measurements, Inc., Sweetwater, Texas)
coupled to:
Ludlum Ratemeter-scaler Model 2221
(Ludlum Measurements, Inc., Sweetwater, Texas)
coupled to:
Trimble Data Logger (Trimble Navigation Limited, Sunnyvale, California)

C.1.2 BETA

Ludlum Gas-Flow Proportional Detector Model 43-68, 126 cm^2 physical area
coupled to:
Ludlum Ratemeter-scaler Model 2221
(Ludlum Measurements, Inc., Sweetwater, Texas)

Ludlum Plastic Scintillation Detector Model 44-142, 100 cm^2 physical area
coupled to:
Ludlum Ratemeter-scaler Model 2221
(Ludlum Measurements, Inc., Sweetwater, Texas)

C.1.3 ALPHA

Ludlum Gas-Flow Proportional Detector Model 43-68, 126 cm^2 physical area
coupled to:
Ludlum Ratemeter-scaler Model 2221
(Ludlum Measurements, Inc., Sweetwater, Texas)

Ludlum Plastic Scintillation Detector Model 43-92, 100 cm^2 physical area
coupled to:
Ludlum Ratemeter-scaler Model 2221
(Ludlum Measurements, Inc., Sweetwater, Texas)

C.2 LABORATORY ANALYTICAL INSTRUMENTATION

High-Purity, Extended Range Intrinsic Detector
CANBERRA/Tennelec Model No: ERVDS30-25195
(Canberra, Meriden, Connecticut)
Used in conjunction with:
Lead Shield Model G-11
(Nuclear Lead, Oak Ridge, Tennessee) and
Multichannel Analyzer
Canberra's Gamma Software
Dell Workstation
(Canberra, Meriden, Connecticut)

High-Purity, Intrinsic Detector
Model No. GMX-45200-5
CANBERRA Model No: GC4020
(Canberra, Meriden, Connecticut)
Used in conjunction with:
Lead Shield Model G-11
Lead Shield Model SPG-16-K8
(Nuclear Data)
Multichannel Analyzer
Canberra's Gamma Software
Dell Workstation
(Canberra, Meriden, Connecticut)

Low Background Gas Proportional Counter
Model LB-5100-W
(Tennelec/Canberra, Meriden, CT)

Tri-Carb Liquid Scintillation Analyzer
Model 3100
(Packard Instrument Co., Meriden, CT)

APPENDIX D
SURVEY AND ANALYTICAL PROCEDURES

D.1 PROJECT HEALTH AND SAFETY

ORISE performed all survey activities in accordance with the *ORAU Radiation Protection Manual*, the *ORAU Health and Safety Manual*, and the *ORAU Radiological and Environmental Survey Procedures Manual* (ORAU 2014, ORAU 2015e, and ORAU 2015b). Prior to on-site activities, a work-specific hazard checklist was completed for the project and discussed with field personnel. The planned activities were thoroughly discussed with site personnel prior to implementation to identify hazards present. Additionally, prior to performing work, a pre-job briefing and walk-down of the survey areas were completed with field personnel to identify hazards present and discuss safety concerns. Should ORISE have identified a hazard not covered in the *ORAU Radiological and Environmental Survey Procedures Manual* (ORAU 2015b)or the project's work-specific hazard checklist for the planned survey and sampling procedures, work would not have been initiated or continued until it was addressed by an appropriate job hazard analysis and hazard controls.

D.2 CALIBRATION AND QUALITY ASSURANCE

Calibration of all field instrumentation was based on standards/sources, traceable to National Institute of Standards and Technology (NIST).

Field survey activities were conducted in accordance with procedures from the following ORAU documents:

- *ORAU Radiological and Environmental Survey Procedures Manual* (ORAU 2015b)
- *ORAU Radiological and Environmental Analytical Laboratory Procedures Manual* (ORAU 2016c)
- *ORAU Environmental Services and Radiation Training Quality Program Manual* (ORAU 2016b)

The procedures contained in these manuals were developed to meet the requirements of U.S. Department of Energy (DOE) Order 414.1D and the U.S. Nuclear Regulatory Commission (NRC) *Quality Assurance Manual for the Office of Nuclear Material Safety and Safeguards* and contain measures to assess processes during their performance.

Quality control procedures include:

- Daily instrument background and check-source measurements to confirm that equipment operation is within acceptable statistical fluctuations.

- Participation in Mixed-Analyte Performance Evaluation Program, NIST Radiochemistry Intercomparison Testing Program (ITP), and ITP Laboratory Quality Assurance Programs.
- Training and certification of all individuals performing procedures.
- Periodic internal and external audits.

Detectors used for assessing surface activity were calibrated in accordance with ISO-7503[1] recommendations. Total alpha and beta efficiencies (ε_{total}) were determined for each instrument/detector combination and consisted of the product of the 2π instrument efficiency (ε_i) and surface efficiency (ε_s): $\varepsilon_{total} = \varepsilon_i \times \varepsilon_s$. ISO-7503 recommends an ε_s of 0.25 for alpha emitters and also beta emitters with a maximum energy of less than 0.4 MeV and an ε_s of 0.5 for maximum beta energies greater than 0.4 MeV.

Beta total efficiencies were determined based on a multi-point energy calibration using C-14, Tc-99, Tl-204, and Sr-90; development of instrument efficiency to beta energy calibration curves; and the selection of the ε_i and ε_s that represented the primary radionuclide of concern. Based on the data in PG&E's FSSP worksheet, a weighted efficiency for the fractional contributions of Co-60 and Cs-137 was calculated. That total weighted efficiency used to quantify beta activity was 0.13 for the plastic scintillators and 0.10 for the gas-proportional.

Th-230 was selected as the alpha calibration source. The 2π alpha instrument efficiency (ε_i) factor was 0.46 for the ZnS scintillation detectors, resulting in a total efficiency of 0.12.

D.3 SURVEY PROCEDURES

D.3.1 SURFACE SCANS

Scans for elevated gamma radiation were performed by passing the detector slowly over the surface. The distance between the detector and surface was maintained at a minimum. Specific scan minimum detectable concentration (MDCs) for the sodium iodide scintillation detectors (NaI) were not determined as the instruments were used solely as a qualitative means to identify elevated gamma radiation levels in excess of background. Identifications of elevated radiation levels that

[1] International Standard. ISO 7503-1, Evaluation of Surface Contamination - Part 1: Beta-emitters (maximum beta energy greater than 0.15 MeV) and alpha-emitters. August 1, 1988.

could exceed the site criteria were determined based on an increase in the audible signal from the indicating instrument.

Beta scans were performed using small, hand-held detectors. Identification of elevated radiation levels was based on increases in the audible signal from the indicating instrument. Beta surface scan MDCs were estimated using the approach described in NUREG-1507. The scan MDC is a function of many variables, including the background level. Additional parameters selected for the calculation of scan MDCs included a two-second observation interval, a specified level of performance at the first scanning stage of 95% true positive and 25% false positive rate, which yields a d' value of 2.32 (NUREG-1507, Table 6.1), and a surveyor efficiency of 0.5. The parameters for the two beta scanning detectors and the resulting detectable count rate (MDCR) and scan MDC was calculated and presented in Table D-1.

Table D-1. Scan MDC Summary		
Instrument Parameter	**Ludlum 44-142**	**Ludlum Model 43-68**
Mylar Window	1.2 mg cm^2	0.8 mg cm^2
Background	305	480
Total Efficiency	0.13	0.2
B_i	=(305)(2 s)(1 min/60 s) = 10 counts	=(480)(2 s)(1 min/60 s) = 16 counts
MDCR	= (2.32)(10 counts)$^{1/2}$[(60 s/min)/2s] = 222 cpm	= (2.32)(16 counts)$^{1/2}$[(60 s/min)/2s] = 278 cpm
$MDCR_{surveyor}$	= 222/(0.5)$^{1/2}$ = 314 cpm	= 278/(0.5)$^{1/2}$ = 393 cpm
Scan MDC	= (314)/(0.13) = 2,400 dpm/100 cm^2	= (393)/(0.2) = 2,000 dpm/100 cm^2

D.3.2 SURFACE ACTIVITY MEASUREMENTS

Measurements of total beta and alpha surface activity levels were performed using hand-held scintillation detectors coupled to portable ratemeter-scalers. Count rates (cpm), which were integrated over one minute with the detector held in a static position, were converted to activity levels (dpm/100 cm^2) by dividing the count rate by the total static efficiency ($\varepsilon_i \times \varepsilon_s$) and correcting for the physical area of the detector, which for both detectors is 100 cm^2. ORISE did not determine construction material-specific background for each surface type encountered for determining net count rates. Instead, ORISE took a conservative approach and applied an ambient detector background. The ambient beta and alpha background (1 cpm) count rates for the area would be

used (305 cpm used in the example below) when determining surface activity. An example *a priori* MDC for beta activity is given by:

$$MDC = \frac{3 + \left(4.65\sqrt{B}\right)}{G\ \varepsilon_{tot}}$$

Where:

B	=	background
ε_{tot}	=	total efficiency
G	=	geometry correction factor (1.0 for the Ludlum 44-142, and 1.26 for the Ludlum 43-68)

The *a priori* beta static MDC was 650 dpm/100 cm^2 for the Ludlum 44-142 and 670 dpm/100 cm^2 for the Ludlum 43-68. The alpha static MDC was 64 dpm/100 cm^2 for the Ludlum 43-93.

D.3.3 SOIL SAMPLING

Soil samples (approximately 0.5 kilogram each) were collected, using a clean garden trowel, then transferred into a new sample container by ORISE personnel. In total, ORISE collected 10 soil samples from the upper yard area and two soil samples from the discharge canal. ORISE personnel labeled each sample in accordance with ORISE survey procedures and completed the required custody documentation.

D.4 RADIOLOGICAL ANALYSIS

D.4.1 GAMMA SPECTROSCOPY

Samples were analyzed as received, mixed, crushed, and/or homogenized as necessary, and a portion sealed in a 0.5-liter Marinelli beaker. The quantity placed in the beaker was chosen to reproduce the calibrated counting geometry. Net material weights were determined and the samples counted using intrinsic, high purity, germanium detectors coupled to a pulse height analyzer system. Background and Compton stripping, peak search, peak identification, and concentration calculations were performed using the computer capabilities inherent in the analyzer system. All total absorption peaks (TAPs) associated with the ROCs were reviewed for consistency of activity. Spectra were also reviewed for other identifiable TAPs. TAPs used for determining the activities of ROCs and the typical associated MDCs for a one-hour count time were:

Radionuclide[a]	TAP (MeV)	MDC (pCi/g)
Am-241	0.0595	0.15
Co-60	1.173	0.06
Cs-137	0.662	0.05
Eu-152	0.344	0.10
Eu-154	0.723	0.15
Nb-94	0.871	0.05
Np-237	0.312	0.08

[a]Spectra were also reviewed for other identifiable TAPs.

D.4.2 DETECTION LIMITS

Detection limits, referred to as MDCs, were based on 95% confidence level. Because of variations in background levels, measurement efficiencies, and contributions from other radionuclides in samples, the detection limits differ from sample to sample and instrument to instrument.