Pacific Gas and Electric Company®

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August 13, 2014

PG&E Letter HBL-14-015

U.S Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

10 CFR 50.90
10 CFR 50.82 (a)(9)

Docket No. 50-133, License No. DPR-7
Humboldt Bay Power Plant, Unit 3

Humboldt Bay Power Plant, Unit 3, License Termination Plan, Revision 1

Dear Commissioners and Staff:

On May 3, 2013, Pacific Gas and Electric Company (PG&E) submitted a proposed License Termination Plan (LTP) for Humboldt Bay Power Plant (HBPP), Unit 3, via PG&E Letter HBL-13-007, "License Amendment Request 13-01, Addition of License Condition 2.C.5, 'License Termination Plan.'" On December 24, 2013, the NRC sent PG&E a Request for Additional Information (RAI) based on NRC review of the HBPP, Unit 3, LTP. On February 14, 2014, PG&E submitted PG&E Letter HBL-14-008 that contained a response to the RAIs except for RAIs associated with LTP Chapter 6 and Supplemental Information. Additionally, personnel from PG&E and the NRC met in a public meeting on March 12, 2014, to discuss the LTP and NRC RAIs. The PG&E response to RAIs for LTP Chapter 6 and Supplemental Information was submitted in PG&E Letter HBL-14-007 on May 13, 2014.

Enclosure 1 to this letter contains the LTP, Revision 1, which includes RAI response information contained in the above submittals, and updated information since Revision 0 and the RAI responses were submitted. Revision 1 to the LTP contains revision bars in the margins indicating revised text and RAI numbers, where applicable. In addition, Enclosure 2 to this letter contains a LTP, Revision 1 Matrix summarizing the changes and correlating LTP section numbers with RAI numbers.

The submittal of the LTP, Revision 1, does not alter the conclusions of the Determination of No Significant Hazards Consideration or the Environmental Impact Consideration as presented in Letter HBL-13-007.

PG&E makes no new or revised regulatory commitments (as defined by NEI 99-04) in this letter.

If you wish to discuss the information in the enclosure, please contact Mr. William Barley at (707) 444-0856.

FSME20
FSME

I state under penalty of perjury that the foregoing is true and correct.

Executed on August 13, 2014.

Sincerely,

Edward D. Halpin
Senior Vice President and Chief Nuclear Officer

Enclosures
cc: HBPP Humboldt Distribution
cc/enc: Marc Dapas, NRC Region IV
John B. Hickman, NRC Project Manager
Gonzalo L. Perez, California Department of Public Health

License Termination Plan, Revision 1

Terms and Acronyms

Action Level - The numerical value that will cause the decision maker to choose one of the alternative actions. It may be a regulatory threshold standard (*e.g.*, Maximum Contaminant Level for drinking water), a dose- or risk-based concentration level (*e.g.*, *DCGL*), or a reference-based standard.

AEC – Acronym for Atomic Energy Commission

AF – Area Factor

AL – ALARA action level

ALARA – "as low as reasonably achievable," which means making every reasonable effort to maintain exposures to radiation as far below the dose limits as is practical.

alpha (α) – The specified maximum probability of a Type I error. This means the maximum probability of rejecting the null hypothesis when it is true. Alpha is also referred to as the size of the test. Alpha reflects the amount of evidence the decision maker would like to see before abandoning the null hypothesis.

ANL – Argonne National Laboratory

Area of elevated activity – An area over which residual radioactivity exceeds a specified value $DCGL_{EMC}$.

beta (β) – The probability of a Type II error, i.e., the probability of accepting the null hypothesis when it is false. The complement of beta ($1-\beta$) is referred to as the power of the test.

bgs – below grade surface

BMP – Best Management Practice

BWR – Boiling Water Reactor

CAB – Citizens Advisory Board.

Caisson – An underground concrete structure at HBPP that houses the underground nuclear reactor.

CCC – California Coastal Commission

CDP – Coastal Development Permit

CEC – California Energy Commission

CFR – Code of Federal Regulations

CIRP – Caisson In Leakage Repair Project

COC – Chain of Custody refers to an unbroken trail of accountability to ensure the physical security of samples, data, and records.

Conceptual site model – A description of a site and its environs and presentation of hypotheses regarding the contaminants present, their routes of migration, and their potential impact on sensitive receptors.

Control charts – A plot of the results of a quality control action that demonstrates control is being maintained within expected statistical variation or to indicate when control is or may be lost unless intervention occurs.

CPUC – California Public Utilities Commission

Critical Group – The average group of individuals reasonably expected to receive the greatest exposure to residual radioactivity for any applicable set of circumstances.

CWT – Concentrated Waste Tank

D&D – Decontamination & Decommissioning

Data Quality Assessment (DQA) – The scientific and statistical evaluation of data used to determine if the data are of the right type, quality and quantity to support their intended use.

Data Quality Objective (DQO) – Qualitative and quantitative statements derived from the DQO process that clarify technical and quality objectives, define the appropriate type of data, and specify tolerable levels of potential decision errors that will be used as the basis for establishing the quality and quantity of data needed to support decisions.

DAW – Dry Activated Waste

DCF – Dose Conversion Factor

DCGL – Derived Concentration Guideline Level

$DCGL_{EMC}$ – A DCGL scaled, through the use of area factors, to obtain a DCGL that represents the same dose to an individual for residual radioactivity in a smaller area within a survey unit.

$DCGL_{W}$ – A DCGL for the average residual radioactivity in a survey unit. If there is no subscript associated with DCGL then it is understood to mean DCGLw. RAI 13

delta (δ) – The amount that the distribution of measurements for a survey unit is shifted to the right of the distribution of measurements of the reference area. This term is used in the evaluation of elevated areas.

delta (Δ) – The width of the gray region. Δ divided by σ, the arithmetic standard deviation of the measurements, is the relative shift expressed in multiples of standard deviations.

Derived Concentration Guideline Levels (DCGLs) – Derived radionuclide-specific activity concentration that corresponds to the release criterion (25 mrem/y) within a survey unit.

DOE – U.S. department of Energy

DP – Decommissioning plan

DPM – disintegrations per minute

DPR – Decommissioning Project Report

DQO – Data Quality Objective

DSAR – Defueled Safety Analysis Report

DTSC – Department of Toxic Substances Control

Elevated Measurement Comparison (EMC) – This comparison is used to determine if there are any measurements that exceed a specified value $DCGL_{EMC}$.

EPA – U.S. Environment Protection Agency

ETD – Easy to detect (for this purpose, nuclides that are detectable by gamma analysis)

Exposure Scenario – A description of the future land uses, human activities, and behavior of the natural system as related to a future human receptor's interaction with (and therefore exposure to) residual radioactivity. In particular, the exposure scenario describes where humans may be exposed to residual radioactivity in the environment, what exposure group habits determine exposure, and how residual radioactivity moves through the environment.

Ft^3 – cubic foot

FGEIS – Final Generic Environment Impact Statement

FGR – Federal Guidance Report

FSS – Final Status Survey

GEIS – Generic Environmental Impact Report

Gross Activity DCGLs – DCGLs established, based on the representative radionuclide mix, for gross (non-radionuclide-specific) alpha/beta surface radioactivity measurements. Field assessments will typically consist of these gross radioactivity measurements.

GTCC – Greater Than Class C

HABS – Historic American Building Survey

HAER – Historic American Engineering Record

HBGS – Humboldt Bay Generating Station

HBPP – Humboldt Bay Power Plant

HEPA – High Efficiency Particulate Air filter

Historical Site Assessment (HSA) – The identification of potential, likely, or known sources of radioactive material and radioactive contamination based on existing or derived information for the purpose of classifying a facility or site, or parts thereof, as impacted or non-impacted.

HPGe – High Purity Germanium

HSE – Health, Safety, and Environment

HTD – Hard to detect (for this purpose, nuclides that are not detectable by gamma analysis).

Investigation level – A derived media-specific, radionuclide-specific concentration or activity level of radioactivity that: 1) is based on the release criterion, and 2) triggers a response, such as further investigation or cleanup, if exceeded.

ISFSI – Independent Spent Fuel Storage Installation

ISOCS – In Situ Object Counting System

Judgmental measurement/biased measurement – A measurement performed at locations selected using professional judgment based on unusual appearance, location relative to known contaminated areas, high potential for residual radioactivity, general supplemental information, etc. Judgmental measurements are not included in the statistical evaluation of the survey unit data because they violate the assumption of randomly selected, independent measurements. Instead, judgmental measurements are individually compared to the DCGL.

LA – License Amendment

LAR – License Amendment Request

LHS – Latin Hypercube Sampling

LLRW – Low-level Radioactive Waste

LLW – Low-level Waste

Lower Bound of the Gray Region (LBGR) – Refers to the minimum value of the gray region. The width of the gray region (*DCGL-LBGR*) is also referred to as the shift, Δ.

LPG – Liquid Propane Gas

LRW – Liquid Radwaste

LTP – License Termination Plan

m^2 – square meter

m^3 – cubic meter

MARSSIM – The *Multi-Agency Radiation Site Survey and Investigation Manual (NUREG–1575)* is a multi-agency consensus manual that provides information on planning, conducting, evaluating, and documenting building surface and surface soil final status radiological surveys for demonstrating compliance with dose- or risk-based regulations or standards.

MDC – Minimum Detectable Concentration

MDCR – Minimum Detectable Count Rate

Measurement – For the purpose of MARSSIM, the term is used interchangeably to mean: (1) the act of using a detector to determine the level or quantity of radioactivity on a surface or in a sample of material removed from a media being evaluated or, (2) the quantity obtained by the act of measuring.
MEPPS – Mobile Emergency Power Plant Station

MeV – Mega electron Volts

Minimum Detectable Concentration (MDC) – This term means the *a priori* radioactivity concentration level that specific instrument or technique can be expected to detect 95% of the time; the value that should be used when stating the detection capability of an instrument for a given measurement technique. The MDC is the detection limit, L_D, multiplied by an appropriate conversion factor to give units of radioactivity concentration.

Minimum detectable count rate (MDCR) – The minimum detectable count rate is the *a priori* count rate that a specific instrument and technique can be expected to detect.

MLLW – mean lower low water, which is the average height of the lowest tide recorded at a tide station each day during the recording period.

mrem/y (millirem per year) – One one-thousandth (0.001) of a rem per year.

MSL – mean sea level

NAVD88 – North American Vertical Datum 1988

NCUAQMD – North American Unified Air Quality Management District

NDCTP – Nuclear Decommissioning Cost Triennial Proceeding

NEI – Nuclear Energy Institute

NIST – National Institute of Standards and Technology

Non-impacted Area – An area where there is no reasonable possibility (extremely low probability) for residual radioactivity to exist.

Nonparametric test – A test based on relatively few assumptions about the exact form of the underlying probability distributions of the measurements. As a consequence, nonparametric tests are generally valid for a fairly broad class of distributions. The Wilcoxon Rank Sum test and the Sign test are examples of nonparametric tests.

NRC – Nuclear Regulatory Commission

Null Hypothesis (H_0) - A statistical scenario set up to be nullified, refuted or rejected ('disproved statistically') in order to demonstrate compliance with the release criteria.

ODCM – Offsite Dose Calculation Manual

OWS – Oil/Water Separator

PCB – Polychlorinated Biphenyl

pCi/g – Picocurie per gram, a concentration scale typically used in the measurement of radioactivity in soil.

PG&E – Pacific Gas and Electric

PM_{10} – particular matter of 10 microns

Power (1-β) – This term refers to the probability of rejecting the *null hypothesis* when it is false. The power is equal to one minus the *Type II* error rate, *i.e.* (1-β).

PRCC – Partial Rank Correlation Coefficient

Precision – A measure of mutual agreement among individual measurements of the same property, usually under prescribed similar conditions, expressed generally in terms of the standard deviation.

Probabilistic – Refers to computer codes or analyses that use a random sampling method to select parameter values from a distribution. Results of the calculations are also in the form of a distribution of values. The results of the calculation do not typically include the probability of the scenario occurring.

PSDAR – Post-Shutdown Decommissioning Activities Report

QAPP – Quality Assurance Project Plan

QC – Quality Control

RA – Restricted Area

RCA – Radiological Control Area

RCRA – *Resource Conservation and Recovery Act of 1976*

Reference area – Geographical area from which representative reference measurements are performed for comparison with measurements performed in specific survey units at remediation site. A site radiological reference area (background area) is defined as an area that has similar physical, chemical, radiological, and biological characteristics as the site area being remediated, but which has not been contaminated by site activities. The distribution and concentration of background radiation in the reference area should be the same as that which would be expected on the site if that site had never been contaminated. More than one reference area may be necessary for valid comparisons if a site exhibits considerable physical, chemical, radiological, or biological variability.

Reference coordinate system – A grid of intersecting lines referenced to a fixed site location or benchmark. Typically, the lines are arranged in a perpendicular pattern dividing the survey location into squares or blocks of equal areas. Other patterns include three-dimensional and polar coordinate systems.

Relative shift (Δ/σ) – Δ divided by σ, the standard deviation of the measurements.

Release criterion – A regulatory limit expressed in terms of dose or risk.

REMP – Radiological Environmental Monitoring Program

Replicate – A repeated analysis of the same sample or repeated measurement at the same location.

RESRAD Code – A computer code developed by the U.S. Department of Energy and designed to estimate radiation doses and risks from RESidual RADioactive materials in soils.

RESRAD-BUILD Code – A computer code developed by the U.S. Department of Energy and designed to estimate radiation doses and risks from RESidual RADioactive materials in BUILDings.

Restricted Area – Any area to which access is limited by a licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials.

RGWMP – REMP Ground Work Monitoring Program

RWP – Radiation Work Permit

SAFSTOR – The alternative in which the nuclear facility is placed and maintained in a condition that allows the nuclear facility to be safely stored and subsequently decontaminated (deferred decontamination) to levels that permit release for unrestricted use.

Scanning – An evaluation technique performed by moving a detection device over a surface at a specified speed and distance above the surface to detect radiation.

SCM – Site Conceptual Model (same as Conceptual)

Scoping Survey – An initial survey performed to evaluate: 1) radionuclide contaminants, 2) relative radionuclide ratios, and 3) general levels and extent of contamination.
SFP – Spent Fuel Pool

Sign Test – A nonparametric statistical test used to demonstrate compliance with the release criterion when the radionuclide-of-interest is not present in background or present in a small fraction of the DCGL, and the distribution of data is not symmetric.

Single nuclide DCGL – A radionuclide-specific activity concentration that would result in an annual total effective dose equivalent (TEDE) of 25 mrem with no other radionuclides present.

S_o – Sensitivity Threshold

Source Term – Refers to a conceptual representation of the residual radioactivity at a site or facility.

Split Sample – A sample that has been homogenized and divided into two or more aliquots for subsequent analysis.

Standard normal distribution – A normal (Gaussian) distribution with mean zero and variance one.

Survey Area - An area established and classified based on a common radiological history, logical physical boundaries, and site landmarks for the purpose of documenting and conveying radiological information.

Survey Area Report – A report including all the survey units within a survey area providing a complete and unambiguous record of the radiological status of each survey unit relative to the established DCGLs.

Survey Package – A document developed by the DQO process providing the methodology by which to perform the final status survey.

Survey Unit – A geographical area consisting of structures or land areas of specified size and shape at a site for which a separate decision will be made as to whether or not the unit attains the site-specific reference-based cleanup standard for the designated pollution parameter. Survey units are generally formed by grouping contiguous site areas with similar use histories and having the same contamination potential (classification). Survey units are established to facilitate the survey process and the statistical analysis of survey data. One, or more, survey units makeup a survey area.

Systematic error – An error of observation based on system faults which are biased in one or more ways, *e.g.*, tending to be on one side of the true value more than the other.

TBD – HBPP Technical Basis Documents

TCP – Traffic Control Plan

Total Effective Dose Equivalent (TEDE) – The sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (CEDE) (for internal exposures).

Triangular sampling grid – A grid of sampling locations that is arranged in a triangular pattern.

Turnover Survey – A final operational radiological survey performed by the Radiation Protection (RP) Department after the completion of decommissioning activities in an area to verify that the area is ready for Final Status Survey.

Type I error – A decision error that occurs when the null hypothesis is rejected when it is true. The probability of making a Type I decision error is called alpha (α).

Type II error – A decision error that occurs when the null hypothesis is accepted when it is false. The probability of making a Type II decision error is called beta (β).

Unity rule – A rule applied when more than one radionuclide is present at a concentration that is distinguishable from background and where a single concentration comparison does not apply. In this case, the mixture of radionuclides is compared against default concentrations by applying the unity rule. This is accomplished by determining: (1) the ratio between the concentration of each radionuclide in the mixture, and (2) the concentration for that radionuclide in an appropriate listing of default values. The sum of the ratios for all radionuclides in the mixture should not exceed 1.

VSP – Visual Sample Plan software used for plotting sample/measurement locations.

Wilcoxon Rank Sum (WRS) test – A nonparametric statistical test used to demonstrate compliance with the release criterion when the radionuclide-of-interest is present in background.

W_r – This represents the sum of the ranks of the adjusted measurements from the reference area, used as the test statistic for the Wilcoxon Rank Sum test.

W_S – The sum of the ranks of the measurements from the survey unit, used with the Wilcoxon Rank Sum test.

WWI – Wastewater Impoundments

CONTENTS

FIGURES

1 INTRODUCTION

Humboldt Bay Power Plant (HBPP) Unit 3, located at 1000 King Salmon Avenue, Eureka, California, was a 63 MWe Boiling Water Reactor (BWR). Unit 3 last operated in 1976 and was permanently defueled in 1984. Transfer of spent fuel to the onsite Independent Spent Fuel Storage Installation (ISFSI) was completed in December 2008.

Pacific Gas and Electric (PG&E) began actively decommissioning Unit 3 in June 2009. The HBPP Unit 3 License Termination Plan (LTP) describes the remaining activities that PG&E will perform to complete nuclear decommissioning. The LTP will address PG&E's plans for demonstrating to the Nuclear Regulatory Commission (NRC) that the HBPP Unit 3 license for possession of radioactive material is ready to be terminated.

NRC has established specific radiological criteria for release of a nuclear power plant site for unrestricted use that must be met prior to terminating a reactor license. NRC requires that the remaining radioactivity distinguishable from background radiation, not result in a Total Effective Dose Equivalent (TEDE) that would exceed 25 mrem per year to an average member of the critical group, including that from groundwater sources of drinking water, and also that the residual radioactivity be reduced to levels that are as low as reasonably achievable (ALARA). Determination of radioactivity levels that are ALARA is made with consideration of potential detriments that may result from further decontamination and waste disposal.

A fundamental input into the development of the HBPP Unit 3 LTP is the site conceptual model. The HBPP site is currently an industrial site supplying electricity to the surrounding areas and will continue to do so for at least the life of the Humboldt Bay Generating Station (HBGS), which is 30 years. It is unlikely that the HBPP site will be used for any purpose other than an industrial site; however, PG&E has chosen the conservative approach of remediating and surveying to the resident farmer scenario at license termination.

PG&E is submitting this LTP for HBPP Unit 3. Following are the licensee name, address, license number, and docket number for HBPP Unit 3.

Pacific Gas and Electric Company
77 Beale Street
San Francisco, CA 94105
License No. DPR-7
Docket No. 50-133

1.1 Historical Background and Site Description

1.1.1 Historical Background

The HBPP site also includes the HBGS (163 MWe). The 163 MWe fossil-fueled HBGS began commercial operation in 2010, replacing HBPP Units 1 and 2 and the backup power mobile emergency power plant station (MEPPS). The HBGS will be operated for at least 30 years.

HBPP Unit 3 is a physical extension of the partially removed fossil Units 1 and 2. Unit 3 commenced commercial operations in 1963 and last operated in July 1976. Unit 3 consisted of a General Electric natural circulation, boiling water reactor, an associated turbine-generator, and the necessary support and auxiliary systems.

During its operational period, Unit 3 experienced a variety of operating events (e.g., fuel failures, maintenance, leaks, spills, and repairs) that have affected decontamination and decommissioning processes. Radiological contamination of the site is found within systems, on component and structure interiors, and in soil located inside and adjacent to the Unit 3 Restricted Area (RA). Subsequent chapters of this LTP will elaborate on these events.

Unit 3 was granted a construction permit by the Atomic Energy Commission (AEC) on October 17, 1960. Operating License DPR-7 was issued in August 1962 and the unit began commercial operation in August 1963. On May 17, 1976, NRC issued an order that required satisfactory completion of a seismic design upgrade program and resolution of specified geologic and seismic concerns prior to return to power following the upcoming 1976 shutdown. On July 2, 1976, Unit 3 was shut down for refueling. In December 1980, it became apparent that the cost of completing the required backfits would make it uneconomical to restart the unit. PG&E ultimately concluded that the seismic and other modifications required (i.e., in response to the Three Mile Island accident in 1979) were in fact not economical and in June 1983 announced its intention to decommission the unit.

The Unit 3 reactor was permanently defueled in 1984 and on July 30, 1984, PG&E submitted a license amendment request (LAR) to possess fuel for up to 30 years, but no longer operate, and to decommission using the SAFSTOR method.

On July 16, 1985, NRC issued License Amendment (LA) 19 to place Unit 3 in a possess-but-not-operate status and on July 19, 1988, NRC issued LA 23 approving the Decommissioning Plan and authorizing the decommissioning of HBPP, Unit 3.

PG&E submitted the HBPP Unit 3 Post-Shutdown Decommissioning Activities Report (PSDAR) to NRC on February 27, 1998, in accordance with 10 CFR 50.82 (a)(4)(i). The PSDAR and the Defueled Safety Analysis Report (DSAR) superseded the original Decommissioning Plan and provided the information required by 10 CFR 50.82(a)(4). By December 2008, all spent fuel had been removed from the spent fuel pool and transferred to the 10 CFR 72-licensed ISFSI.

1.1.2 Site Description

Figure 1-1 shows the geographical locations of HBPP and Unit 3 relative to the "true north" orientation. HBPP is located near the coastal community of King Salmon on the shore of Humboldt Bay in Humboldt County, in northwestern California. Unit 3 is located within the PG&E owner-controlled area at HBPP. Figure 1-2 provides an aerial view of the site with the HBPP 10 CFR 50 licensed area indicated by a line drawn on the figure. The Unit 3 structures, as well as the temporary trailers will be removed. Some office buildings will remain to support the HBGS and ISFSI.

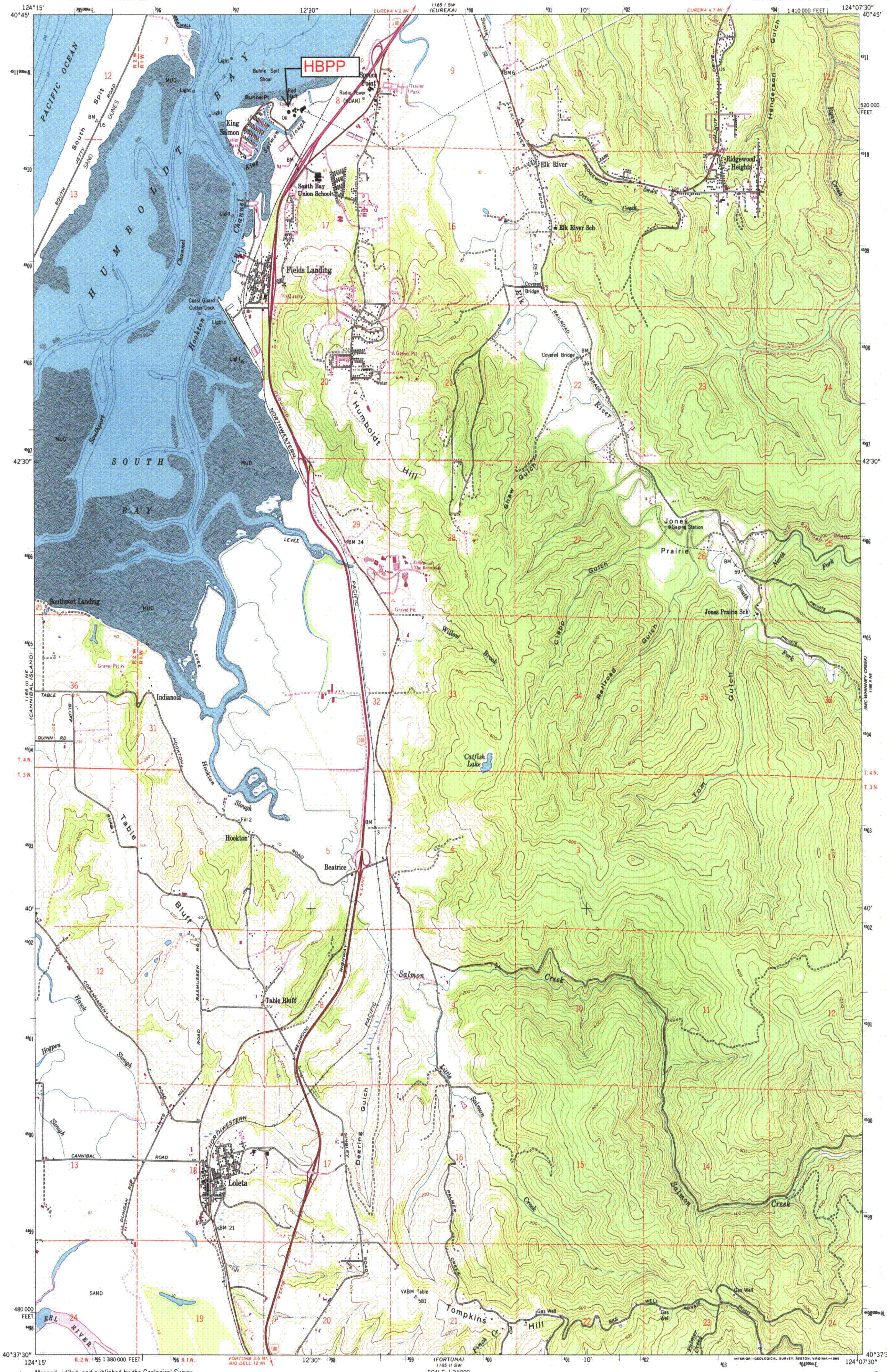
PG&E owns approximately 143 acres on the shore of Humboldt Bay opposite the bay entrance. PG&E also owns the water areas extending approximately 500 feet into Humboldt Bay from the land area. Eureka, the largest city in Humboldt County, is located approximately three miles north-northeast of the site. There are also several small residential communities within five miles of the HBPP site, including King Salmon, Humboldt Hill, Fields Landing, and the suburban communities surrounding Eureka. There are several marine landings in the community of King Salmon, which is located just west of the entrance gate to the owner-controlled area. The community of King Salmon serves frequent commercial and recreational boat traffic, including commercial and sport fishing.

HBPP site terrain varies from submerged and low tidal land, protected by dikes and tide gates, to a high precipitous bluff along the southwestern boundary. Elevations range from approximately minus 3 feet to positive 65 feet, based on a datum of the mean lower low water (MLLW) level. Figure 1-3 provides the contours of HBPP.

UNITED STATES
DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY

STATE OF CALIFORNIA
DEPARTMENT OF WATER RESOURCES

FIELDS LANDING QUADRANGLE
CALIFORNIA–HUMBOLDT CO.
7.5 MINUTE SERIES (TOPOGRAPHIC)
NW/4 FORTUNA 15' QUADRANGLE



Mapped, edited, and published by the Geological Survey

Control by USGS, USC&GS, USCE, and State of California

Topography from aerial photographs by photogrammetric methods and by planetable surveys 1959. Aerial photographs taken 1956

Hydrography compiled from USC&GS chart 5832 (1956)

Polyconic projection. 1927 North American datum
10,000-foot grid based on California coordinate system, zone 1
1000-meter Universal Transverse Mercator grid ticks, zone 10, shown in blue

To place on the predicted North American Datum 1983 move the projection lines 20 meters north and 97 meters east as shown by dashed corner ticks

Land lines are unsurveyed in part of T.3 N.-R. 1 W.

UTM GRID AND 1972 MAGNETIC NORTH DECLINATION AT CENTER OF SHEET

SCALE 1:24 000

CONTOUR INTERVAL 40 FEET
DOTTED LINES REPRESENT 10-FOOT CONTOURS
NATIONAL GEODETIC VERTICAL DATUM OF 1929
DEPTH CURVES IN FEET—DATUM IS MEAN LOWER LOW WATER
SHORELINE SHOWN REPRESENTS THE APPROXIMATE LINE OF MEAN HIGH WATER
THE MEAN RANGE OF TIDE IS APPROXIMATELY 4 FEET

THIS MAP COMPLIES WITH NATIONAL MAP ACCURACY STANDARDS
FOR SALE BY U.S. GEOLOGICAL SURVEY, P.O. BOX 25286, DENVER, COLORADO 80225
A FOLDER DESCRIBING TOPOGRAPHIC MAPS AND SYMBOLS IS AVAILABLE ON REQUEST

QUADRANGLE LOCATION

Revisions shown in purple compiled from aerial photographs taken 1972. This information not field checked

ROAD CLASSIFICATION
Heavy-duty — Light-duty
Medium-duty — Unimproved dirt
U.S. Route — State Route

FIELDS LANDING, CALIF.
NW/4 FORTUNA 15' QUADRANGLE
40124-F2-TF-024
1959
PHOTOREVISED 1972
DMA 1165 II NW–SERIES V895

Figure1- 2 Aerial View of HBPP With Site Boundary

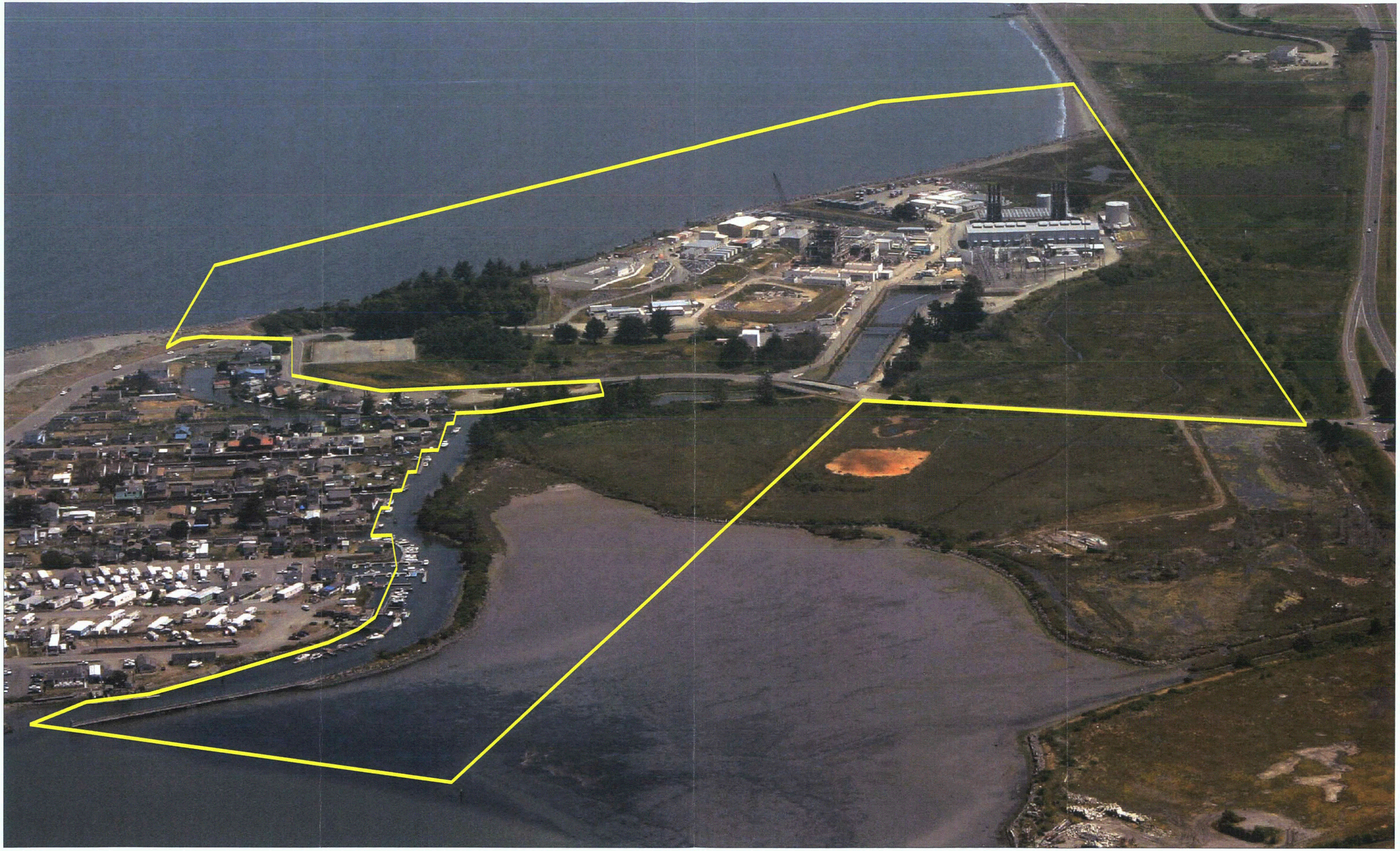
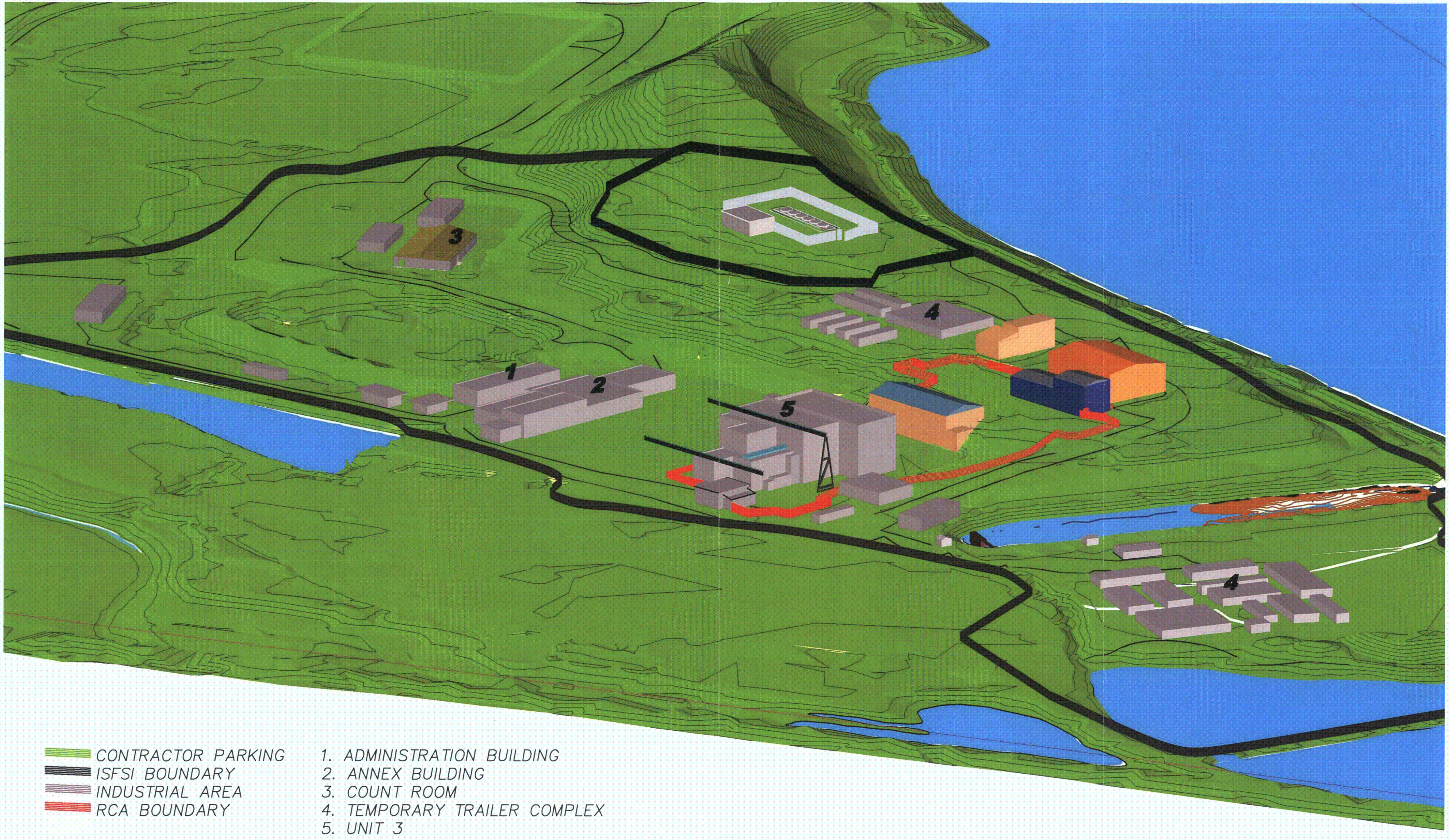


Figure 1-3 Contour Map of HBPP



Humboldt County is mostly mountainous except for the level plain that surrounds Humboldt Bay. The coastal mountains extend to the central valley. The terrain near the site rises rapidly from the bay on the north side to an elevation of approximately 65 ft MLLW at Buhne Point peninsula. Terrain to the north and east of the site is generally flat. To the south and east, the terrain rises rapidly forming Humboldt Hill, which reaches an elevation of over 500 ft MLLW within two miles of the site and is the location of several small neighborhoods.

The HBPP site is located within the hydrologic unit defined as the Redwood Creek-Mad River – Humboldt Bay Unit. The four major creeks that drain into Humboldt Bay are Freshwater Creek, Elk River, Salmon Creek, and Jacoby Creek. Several smaller tributaries also drain into the bay. Salmon Creek and Elk River are the nearest streams to the site, located 1 mile south and 1 mile north of HBPP, respectively.

The owner-controlled area is not traversed by railroad. It is bisected by King Salmon Avenue, but there have not been any changes to the original site boundary. The only access to the site is from the south via King Salmon Avenue, which also serves the community of King Salmon situated on the western part of the peninsula. Public trails run along the shoreline and along the fence to the northwest of the owner-controlled area. The major public access in the site vicinity and to other Humboldt County communities is via US Highway 101, which generally traverses north-south through Humboldt County. This highway passes about 0.2 mile east of Unit 3 and is accessible approximately 0.35 mile southeast of the site.

1.1.3 Population

The HBPP site is located on the northern California coast of Humboldt County. In 2010, the U.S. Census Bureau estimated the population of Humboldt County at 134,623.

The nearest and largest population center in Humboldt County is Eureka located approximately 3 miles north-northeast of the site with a population of approximately 26,000. King Salmon is located adjacent to and west of the site; and Fields Landing, population 222, is located approximately 0.4 mile south.

1.1.4 Land and Water Use

1.1.4.1 Land Use

The power plant site is on land zoned as coastal dependent industrial with combining district designations for coastal resource dependent, flood hazard, and coastal wetland. The project site is currently used for industrial purposes (i.e., electricity production). The majority of the project is in an unincorporated area within Humboldt County's jurisdiction. Eureka's sphere of influence extends west and south of the project site, and the city considers land within this designated area as land that may be annexed to the city in the future.

An existing public trail, included as part of the California Coastal Trail system, is located on the north and western side of the HBPP site along Humboldt Bay. Recreational opportunities within Humboldt Bay are numerous and include boating, fishing, camping, and bird watching. The following designated recreational areas are located in Humboldt Bay, within a 3-mile radius of the project site: Samoa Dunes Recreation Area, South Spit, Fields Landing County Park, Humboldt Bay National Wildlife Refuge, and Elk River Wildlife Area.

None of the areas on which decommissioning activities will occur is used for agricultural production. Prime agricultural land in Elk River Valley is located within one mile of the HBPP.

Although Humboldt County has a certified Local Coastal Program, the HBPP site is within the retained jurisdiction of the California Coastal Commission (CCC).

1.1.4.2 Water Supply

The district operates two separate water systems, a domestic water system and a raw water system. Drinking water to HBPP is supplied through the domestic water system. Raw water is taken directly from the surface of the Mad River and delivered untreated to industrial customers. HBPP does not use raw water from the Humboldt Bay Municipal Water District. The Humboldt Bay Municipal Water District produces 20 million gallons per day of water from five Ranney wells in the Mad River near Essex, located approximately 17 miles from the HBPP site, and from three wells located at the base of Humboldt Hill, approximately 4 miles from the HBPP site.

1.2 Decommissioning Approach

PG&E is submitting this LTP to address residual radioactivity on the HBPP site and discuss how the site will be remediated and verified to meet the release criteria. The LTP describes how the release criteria were determined and how they will be measured.

All structures associated with Unit 3 will be removed, along with temporary decommissioning support trailers. At license termination, only the following structures will remain:

- HBGS and associated structures
- Administration Building
- Administration Annex Building
- Security Building
- Count Room Building
- Training Building
- Waste Management Building
- ISFSI and supporting structures

PG&E wishes to perform a partial site release of the site south of King Salmon Avenue to the Humboldt Bay Harbor District. For this reason, it will be necessary to request HBPP site release for unrestricted use in two phases. The first phase would be complete when the portion of the site south of King Salmon Avenue has been verified to meet acceptance levels through final status surveys (FSSs); PG&E would then submit an LAR to release that portion of the site for unrestricted use. The second phase would be complete when the remainder of the site is verified to meet acceptance levels; PG&E would then submit a second LAR to release the remainder of the site for unrestricted use and to terminate the license.

1.3 Decommissioning Objective

The objective of decommissioning HBPP Unit 3 is to reduce the level of residual radioactivity remaining from reactor operation to levels that permit the release of the HBPP site for unrestricted use and allow for the termination of the 10 CFR 50 license. The HBPP Unit 3 LTP satisfies the 10 CFR 50.82(a)(9) requirement to submit a LTP for NRC approval. The LTP submittal is accompanied by an LAR that establishes the criteria for making changes to the LTP without prior NRC approval. Once approved, the LTP will become a supplement to the HBPP Unit 3 DSAR.

1.4 License Termination Plan Scope

PG&E prepared the LTP using the following guidance:

- Regulatory Guide 1.179, "Standard Format and Contents for License Termination Plans for Nuclear Power Reactors," [Reference 1.8.1]
- NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," [Reference 1.8.2]
- NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," [Reference 1.8.3]
- NUREG-1757, "Consolidated NMSS Decommissioning Guidance," [Reference 1.8.4]

The LTP includes a discussion of the following actions:

- Site characterization to ensure that final status surveys (FSSs) cover all areas where contamination existed, remains, or has the potential to exist
- Remaining decommissioning activities
- Plans for site remediation
- The FSS plan that will be used to confirm that the site release criteria of 10 CFR 20, Subpart E are met
- Dose modeling scenarios that ensure compliance with the site release criteria of 10 CFR 20, Subpart E
- Estimated remaining decommissioning costs
- Environmental impacts from the decommissioning of HBPP Unit 3

1.5 License Termination Plan Summary

The following subsections provide a brief summary of the seven chapters that address the requirements of 10 CFR 50.82(a)(9).

1.5.1 Chapter 2: Site Characterization

Chapter 2 summarizes the Historical Site Assessment (HSA). The HSA provided the preliminary information required to divide the site into survey areas. The survey areas were evaluated against Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) criteria to determine their classification. The HSA also provided the foundation for further site characterization based upon actual surveys to determine the extent and levels of radioactive contamination prior to remediation (Chapter 4). Data collected during site characterization may be used to change the original HSA classification of an area, within the requirements specified in this LTP, up to the time of the FSS.

1.5.2 Chapter 3: Identification of Remaining Decommissioning Activities

Chapter 3 identifies the remaining dismantlement and decontamination activities as of this LTP submittal. The information provided in this chapter includes the following:

- A summary of those activities that have already been completed
- A description of the areas requiring remediation
- Radiological conditions that may be encountered
- Estimates of occupational radiation dose
- An estimate of the remaining quantity of radioactive material to be shipped for disposal
- A description of proposed control mechanisms to ensure remediated areas are not recontaminated

1.5.3 Chapter 4: Site Remediation Plans

Chapter 4 discusses the remediation techniques that may be used to reduce residual contamination to levels that comply with the unrestricted release criteria of 10 CFR 20, Subpart E. The principal materials that will be remediated are structural surfaces and soil. Chapter 4 also discusses the ALARA evaluations, which will be used to determine if remediation is warranted beyond that required to meet the radiological dose criteria, and describes the radiation protection program that will be implemented during remediation activities.

1.5.4 Chapter 5: Final Status Survey Plan

Chapter 5 describes the process that will be used to verify that the HBPP site complies with the 10 CFR 20 criteria for unrestricted use. The plan will be implemented in accordance with approved procedures and work instructions, which comply with the FSS Quality Assurance Project Plan (QAPP). To ensure that survey results are of sufficient quality and quantity to support decision making, FSS design (e.g., scan area coverage, number of survey measurements, survey locations) will be developed using the Data Quality Objective Process described in the MARSSIM.

1.5.5 Chapter 6: Compliance with Radiological Criteria for License Termination

Chapter 6 discusses the development of the Derived Concentration Guideline Levels (DCGLs). The DCGLs are radionuclide-specific values derived from activity-dose relationships and the analyses of potential exposure pathways. DCGLs for assessing residual radioactivity levels on building surfaces and site soil have been developed for each radionuclide of concern. Also discussed in this chapter are the identification of the site inventory of radionuclides, future land use scenarios, and dose computation models, including the sensitivity analysis, exposure pathways, and derivation of area factors.

1.5.6 Chapter 7: Update of Decommissioning Costs

Chapter 7 provides a current estimate of the remaining costs to release the HBPP site for unrestricted use. This chapter also compares the remaining costs with the remaining funds, as of this LTP submittal, and gives a verification of the adequacy of financial assurance.

1.5.7 Chapter 8: Supplement to the Environmental Report

Chapter 8 identifies where HBPP decommissioning activities continue to be bounded by previously issued environmental impact statements, specifically NUREG-0586, Supplement 1, the "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities, Regarding the Decommissioning of Nuclear Power Reactors, Final Report (Reference 1.8.5)." Where the potential impact to a NUREG-0586 Supplement 1 environmental issue is considered site specific, this chapter also contains a justification that the impact to the environment from remaining decommissioning activities will be small (i.e., no detectable impact). This section of the LTP is prepared pursuant to 10 CFR 51.53(d) and 10 CFR 50.82(a)(9)(ii)(G).

1.6 License Termination Plan Change Process

PG&E is submitting this LTP as a supplement to the DSAR; thus, the LTP will be a living internal document and periodic updates will continue to be submitted in accordance with 10 CFR 50.71(e). Changes to decommissioning activities as described in the LTP must comply with the criteria in 10 CFR 50.59 and 50.82. Additionally, NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," specifies additional restrictions on changes to the LTP. A change may not be made without prior NRC approval if a change would result in any of the following:

- An increase in the DCGLs and related minimum detectable concentrations (for both scan and fixed measurement methods)
- An increase in the radioactivity level, relative to the applicable DCGL, at which investigation occurs
- A change in the statistical test applied to other than the Sign Test or Wilcoxon Rank Sum Test
- An increase in the Type 1 decision error as stated in the LTP
- A significant environmental impact not previously reviewed

Additionally, NRC must be notified at least 14 days prior to reclassification of a survey unit to a less restrictive classification (e.g., Class 2 to Class 3). Reclassification of a survey unit to a more restrictive classification (e.g., Class 2 to Class 1) may be done without prior notification.

1.7 License Termination Plan Information Contacts

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1.8 References

1.8.1 U.S. Nuclear Regulatory Commission Regulatory Guide 1.179, "Standard Format and Contents for License Termination Plans for Nuclear Power Reactors," January 1999

1.8.2 U.S. Nuclear Regulatory Commission NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," August 2000

1.8.3 U.S. Nuclear Regulatory Commission NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," April 2003

1.8.4 U.S. Nuclear Regulatory Commission NUREG-1757, "Consolidated NMSS Decommissioning Guidance," September 2006

1.8.5 U.S. Nuclear Regulatory Commission NUREG-0586, Supplement 1, "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities, Regarding the Decommissioning of Nuclear Power Reactors, Final Report," November 2002

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2 SITE CHARACTERIZATION

2.1 Historical Site Assessment Summary

2.1.1 Introduction

The Historical Site Assessment (HSA) [Reference 2-2] describes the site's physical configuration, identifies the radioactive constituents of the site contamination, assesses the migration of contaminants, identifies contaminated media, identifies non-impacted and impacted areas, and classifies impacted areas.

Pacific Gas and Electric (PG&E) has conducted the HSA of its Humboldt Bay Power Plant (HBPP), Unit 3, site in accordance with the guidance of NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," [Reference 2-1] in support of the ultimate decommissioning and license termination of the facility. The HSA formally began in 2006, following several preliminary assessments of the impact of facility operations on the remediation required prior to the performance of the Final Status Survey (FSS). These preliminary surveys included interviewing current and former HBPP site personnel during the site inspection and via telephone communications. An initial characterization survey was performed in 1997. The HSA was formally compiled in 2006 and updated in 2009 and 2011. The purpose of the HSA is to document a comprehensive investigation identifying, collecting, organizing, and evaluating historical information relevant to the HBPP site. The HSA focuses on open land areas and those structures that will remain at the time of FSS.

The HSA consisted of a review of the following items:

- Radiological Characterization Reports
- Environmental Reports
- Environmental Monitoring Reports
- Licensee Event Reports
- Construction Photographs
- Historical Photographs
- Topographical Maps
- Construction Drawings
- As Built Drawings
- Plant Operating Reports

- Plant Safety Analyses
- Radiological Surveys
- Plant Operating Logs

Concurrent with the performance of the HSA was the initial segregation of the facility into individual specific, uniquely identified, survey areas. This provides the basis for development of area-specific site drawings and survey maps required to document the characterization, remediation, and final release survey process. A major output from the HSA process was the information used as the basis for the preliminary MARSSIM classifications of the initial survey areas.

The initial classification of the site areas was based on the historical information and site characterization data. Data from subsequent characterization may be used to change the original classification of an area up to the time of the FSS as long as the classification reflects the level of residual activity existing prior to any remediation in the area.

2.1.2 Objectives of the Historical Site Assessment

PG&E conducted the HSA of the HBPP site to meet the following objectives:

- Identify known and potential sources of radioactive material and radioactively contaminated areas, including systems, structures, and environmental media based on the investigation and evaluation of existing information.
- Identify radionuclides of concern.
- Identify areas of the site with no conceivable or likely potential for radioactive or hazardous materials contamination and assign a preliminary classification of Non-Impacted while assigning a preliminary classification of Impacted to all remaining portions of the site.
- Develop the records to be used during the design of subsequent scoping, characterization, remediation, and the FSS.
- Provide preliminary information necessary to identify and segregate the site into survey areas evaluated against criteria specified in the MARSSIM guidelines for classification. This classification will designate the need for, and level of, remedial action required within a particular survey unit as well as the level of survey intensity required during the FSS.

2.1.3 Property Identification

Chapters 1 and 8 describe the HBPP site and environs.

2.1.4 HSA Methodology

2.1.4.1 Approach and Rationale

The primary objective of the HSA records search process was the identification of those events posing a significant probability of affecting the radiological characterization of the site. These included system, structure, or area contamination from system failures resulting in airborne releases, liquid spills or releases, or the loss of control over solid material management. Each event identified that posed a realistic potential to impact the characterization of the site was further investigated. This investigation focused on the scope of the contaminant sampling and analysis, remedial actions taken to mitigate the situation, and any post-remedial action sampling, survey, and analysis in an attempt to identify the "as left" condition of the event location. The following items were included in the research associated with the development of the HSA:

- Relevant excerpts from written correspondences and reports
- Personnel interviews of current and former HBPP personnel employed during the time that Unit 3 was in operation
- Site inspection, using historical site drawings, photographs, prints, and diagrams to identify, locate, confirm, and document areas of concern

Information from this research was used in the HSA development, including the compilation of data, evaluation of results, documentation of findings, and the characterization and identification of survey areas.

2.1.4.2 Documents Reviewed

Records maintained to satisfy the requirements of 10 CFR Part 50.75(g)(1) [2-7] provided a major source of documentation for the HSA records review process.

Additional documents reviewed were (HSA Section 5.3):

- Environmental Reports
- Radiological Environmental Monitoring Reports
- Radioactive Effluent Release Reports
- Licensee Event Reports
- Plant Operating Reports
- Plant Safety Analyses
- Radiological Surveys
- Plant Operating Logs

RAI 01

2.1.4.3 Site Reconnaissance

As allowed by MARSSIM Section 3.5, a formal site reconnaissance was not performed, based on the continuous occupancy of the site by the licensee, the detailed information available through the records and other documents, and the personnel interviews performed. Investigations were performed to verify locations and current conditions of questionable items or issues (radioactive liquid spills or spread of contamination) discovered during the review of historical records.

2.1.4.4 Personnel Interviews

Personal interviews of current and former HBPP site personnel were held during the site inspection and via telephone during the HSA process. Personnel were selected based on their employment history at the HBPP site. Interview efforts were focused on personnel who were employed during the time that Unit 3 was in operation. Personnel were interviewed that held positions in maintenance, qualified reactor operators, and radiation protection. Undocumented events were not discovered during this process, but the interviews did prove helpful in assessing the historical operations.

2.1.4.5 Historical Construction Photograph Review

Collections of historical photographs were reviewed to assess their contribution to the HSA. A selection of historical photographs is included as Appendix 2-A.

2.1.5 Operational History

2.1.5.1 Introduction

PG&E is the holder of the Humboldt Bay Power Plant, Unit 3 Operating License, DPR-7. Unit 3 was granted a construction permit by the Atomic Energy Commission (AEC) on October 17, 1960, and construction began in November 1960. The AEC issued Provisional Operating License No. DPR-7 for Unit 3 in August 1962. Unit 3 achieved initial criticality on February 16, 1963, and began commercial operation in August 1963.

On July 2, 1976, Unit 3 was shut down for annual refueling and seismic modifications. In December 1980, it became apparent that the cost of completing the required upgrades made the possibility of restarting Unit 3 uneconomical. Work was suspended at that time awaiting further guidance regarding modification requirements. In 1983, updated economic analyses indicated that restarting Unit 3 would probably not be cost effective and in June 1983, PG&E announced its intention to decommission the unit. A possession only license amendment was issued in 1985 and the plant was placed in a SAFSTOR status.

PG&E received approval by the NRC for its decommissioning plan (DP) in July 1988; however, since this was prior to the 1996 NRC decommissioning rule, the licensee converted the DP into its Defueled Safety Analysis Report (DSAR) [Reference 2-8], which is updated every two years.

In February 1998, PG&E issued a Post Shutdown Decommissioning Activity Report (PSDAR). The plant is currently in DECON with active decommissioning activities ongoing.

Table 2-1 summarizes the operational/post-operational history.

Table 2-1 Operational/Post-operational Chronological Summary

Unit 3 construction permit granted by AEC	October 17, 1960
Unit 3 construction begins	November 1960
AEC issues operating license DPR-7	August 1962
Unit 3 achieves criticality	February 16, 1963
Unit 3 begins commercial operation	August 1963
Unit 3 shutdown for refueling and seismic modifications	July 2, 1976
Work suspended awaiting modifications guidance	December 1980
PG&E announces decision to decommission Unit 3	June 1983
Possession only license amendment issued	July 16, 1985
NRC approves HBPP Decommissioning Plan	July 19, 1988
PG&E issues a PSDAR	February 27, 1998
NRC issues license for Humboldt Bay Independent	November 17, 2005

Spent Fuel Storage Installation (ISFSI)	
All fuel removed from spent fuel pool	December 2008

2.1.5.2 *Regulatory Overview*

NRC inspectors from Region IV offices perform routine onsite inspection of HBPP site activities. The NRC is notified of any incidents onsite per the existing protocol established with NRC Region IV and NRC reporting regulations. NRC headquarters reviews license amendment requests, exemption requests, and other submittals.

2.1.5.3 *Waste Handling Procedures*

The DSAR, Section 1.5, describes the systems and equipment for handling radioactive waste generated as a byproduct of prior plant operation and maintenance of the SFP. DSAR section 1.5 describes radioactive waste processing and disposal methods. HBPP waste handling procedures are intended to contain, adequately treat, and dispose of these radioactive byproducts. The waste disposal system uses several basic methods to treat, and dispose of radioactive material:

- Package and shipment to an permitted disposal facility
- Filtration and ion exchange to remove radioactive constituents from liquids
- Dilution of low-activity liquid and gaseous discharges

Spent fuel was removed from the site and shipped to a reprocessing facility in the early years of plant operation. The last spent fuel shipments from HBPP occurred in 1971. All the stainless steel clad fuel assemblies that were prone to failure were removed from site during this period. After that date, spent fuel remained onsite in the SFP until December 2008, when the last of the fuel in the SFP was moved to the ISFSI.

Construction of buildings and roads during and after nuclear operations at the HBPP site involved excavation of contaminated soils. By site procedure, contaminated soils were sent to an NRC-licensed disposal facility. Soils that were deemed non-contaminated by site procedures were placed onsite either west or east of the discharge canal.

Soil samples are counted down to see a lower limit of detection (LLD) of 0.18 pCi/g Cs-137. If any plant related radionuclide, other than Cs-137 is identified, the soils are considered contaminated. If Cs-137 is identified in soils greater than 6 inches from the surface, the soils are considered contaminated. If the soils are less than 6 inches from the surface any Cs-137 concentration greater than 0.4 pCi/g is considered contaminated. RAI 02

2.1.5.4 Current Site Usage

2.1.5.4.1 Description of Unit 3 Operations

Currently, site operations focus primarily on tasks and activities required to complete the dismantlement and decontamination of the facility.

2.1.5.4.2 Site Characterization

Characterizations of HBPP structures, soils, and sediments were performed on two separate occasions—one in 1997 and one in 2008. Section 2.3 explains the methodology employed for the characterization effort at HBPP.

2.1.5.5 Site Dismantlement

2.1.5.5.1 Major Dismantlement Activities within the Restricted Area (RA) **as of November 29, 2012**

- Removed Unit 3 turbine and generator
- Removed reactor vessel head
- Removed reactor feed pumps
- Removed dry well shield plug
- Removed SFP storage racks
- Removed reactor vessel internals

2.1.5.5.2 Major Dismantlement Activities outside the RA as of November 29, 2012

- Fossil units 1 and 2 have been removed
- Fuel Oil Tanks 1 and 2 have been removed
- Unit 1 and 2 Transformers have been removed
- Site is “cold and dark” with temporary power supplying Unit 3

2.1.5.6 *Radiological Sources*

2.1.5.6.1 *RA Contamination*

All areas within the RA have been identified as having been radiologically affected by the operation of the facility, unplanned events, or subsequent decommissioning activities.

2.1.5.6.2 *Areas Outside the RA Contamination*

Areas outside the RA have been affected by radiological events, by the deposition of stack releases, or through routine radioactive effluent liquid releases. The exception to this is the bay area, where no contamination has been detected. The bay area will be classified pending further characterization.

2.1.6 Event Descriptions

Table 2-2 provides a summary of events/issues that affected various HBPP areas.

Table 2-2 List of Events/Issues Affecting HBPP Areas

Event/Issue	Location	Synopsis
Liquid Radwaste (LRW) Concentrator Steam Condensate Leakage to Yard Drain North Loop	Shown as location 1 on Figure 2-1	On 9/28/67, contamination was found near a yard drain (described as either "by the Condensate Storage Tank" or "by Radwaste"). The contamination appeared to come from the radwaste concentrator after a valving error contaminated the (normally clean) condensate from the concentrator supply steam.
Possible Radwaste Spill to Radwaste Tankage Area Drain	Shown as location 2 on Figure 2-1	On 3/19/68, there was the potential that a leaking hose connection could have released one to 10 gallons of concentrator waste to the radwaste tankage area sump, which at that point was valved to the outfall canal.
Overflow of Condensate Tank	Shown as location 3 on Figure 2-1	Notes mention contamination near a "storm drain located by the Condensate Storage Tank," but the remainder of the text discusses leakage from the Concentrator. It is hypothetically possible to overflow the tank, but as yet, no specific events have been identified.

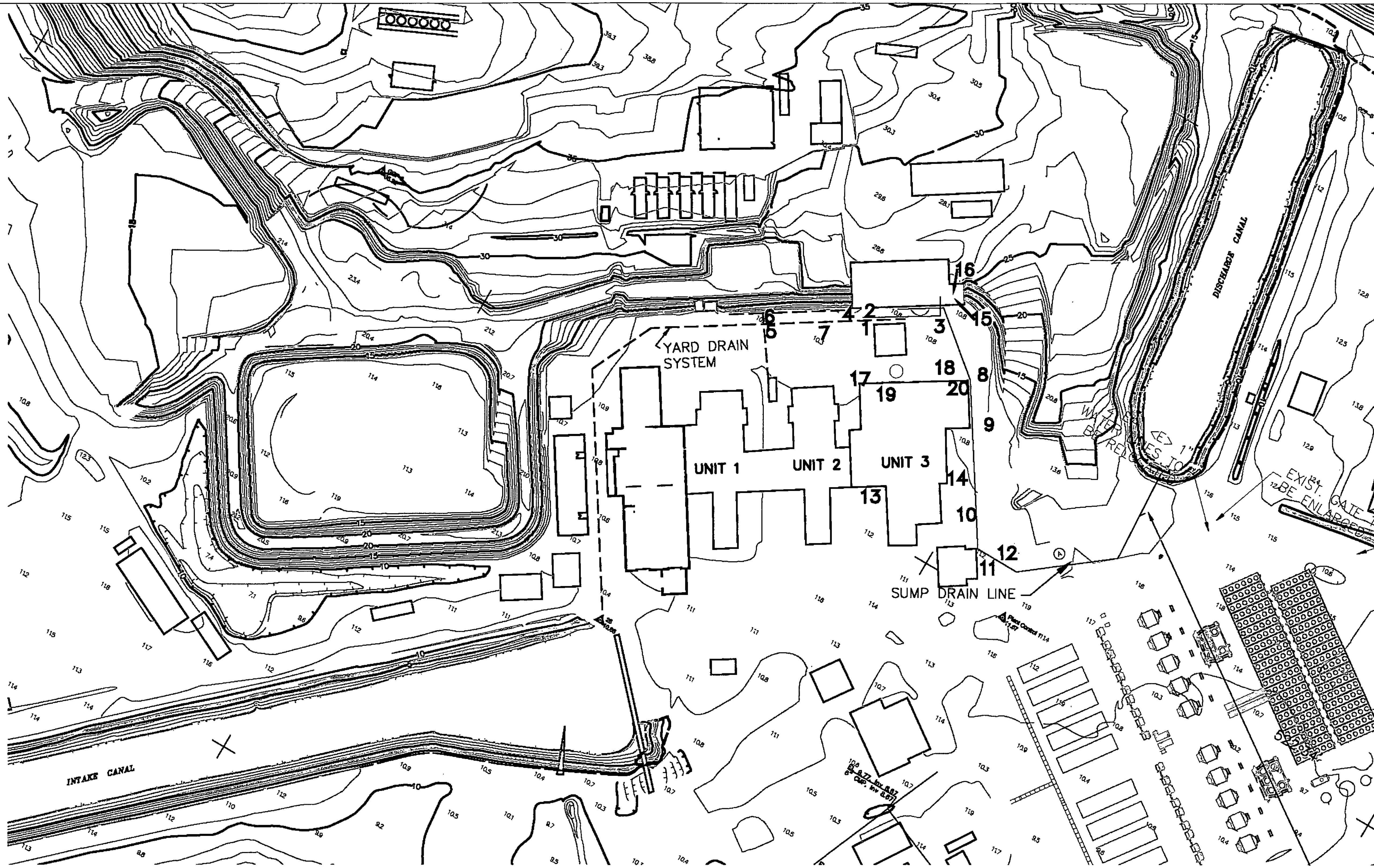
Event/Issue	Location	Synopsis
Overflow of LRW Concentrator to Concentrated Waste Tank (CWT) Vault, radwaste tankage sump and its drain to the outfall canal	Sump Drain Line referenced in Figure 2-1	On 1/26/73, concentrated waste was found to be leaking through a piping penetration from the CWT Vault to the radwaste tankage area, into the tankage area sump, and through the sump drain line to the outfall canal.
LRW Concentrator Steam Condensate Leakage to Radwaste Tankage Sump	Shown as location 4 on Figure 2-1	On 11/25/75, the condensate "drips" from the steam trap for the steam supply to the concentrator, appeared to have been temporarily contaminated. These drips drained to the radwaste tankage area sump, which at that time could be valved either to the radwaste building sump or to the outfall canal.
Subsequent contamination of electric conduit/pullbox, Yard North of Unit 2, and North Loop of Yard Drains, to intake canal, and of piping pits under #3/4 Condensate Storage Tanks, from earlier overflow of LRW Concentrator to CWT Vault	Yard Drain System shown as dotted line on Figure	On 9/7/73, after a sudden rain shower, contaminated liquid came up through openings in a manhole cover to an electric pull box located at the SW corner of the liquid radwaste tankage area. The contaminated liquid flowed across the pavement into the yard drains (in the Unit 3 yard, North of Unit 2 fans, and between the #2/3 condensate storage tanks). The liquid also followed a ditch along the bank north of Unit 2, going through drain rock and a perforated pipe into the yard drain system. The contamination originated from the overflow of the CWT vault on 1/26/73
Yard Drain System	Yard Drain System shown as dotted line on Figure 2-1	The sediment in the drains is known to have been contaminated. As much material as possible was removed from the interior of the sumps and piping, about 1999.
Pavement Contamination North of Unit 2, and under Unit 2 Fans – March, 1975	Shown as location 5 on Figure 2-1	Smearable contamination was found, associated with water puddles in the yard, in low spots south of the newly paved roadway, and under the Unit 2 fans.

Event/Issue	Location	Synopsis
Pavement Contamination North of Unit 2 – 1989	Shown as location 5 on Figure 2-1	Fixed pavement contamination and contaminated soil were found 9/7/88 and 8/30/89. Subsequent surveys found pavement/soil contamination along south side of roadway from Unit 3 RA fence to #3 condensate tank.
Unit 2 Yard Remediation – 1991	Shown as location 6 on Figure 2-1	Contaminated perforated drainpipe and contaminated drain rock (from original drainage ditch between roadway and dirt bank) were removed. Contaminated soil (above about 10 pCi/gram Cs-137 or Co-60) was removed from the area.
Contamination Under Unit 2 Fans – Approx. 1991	Shown as location 7 on Figure 2-1	Contaminated soil under fan/ducts was not accessible for removal, so was covered with concrete "Gunnite," probably about the same time as the 1991 Unit 2 Yard Remediation.
Overflow of Condensate Demineralizers	Shown as location 8 on Figure 2-1	Discussions with plant personnel suggest that condensate was released from the condensate demineralizer system, in amounts sufficient to overload the drains so that water flowed out the door to the yard. This may have happened more than once, in the period between 1967 and 1974.
Ultrasonic Water spill	Shown as location 9 on Figure 2-1	Prior to 4/3/86, a water supply hose was run from a clean water tap (below the stairway to the yard) to fill an Ultrasonic decontamination tank in the Hot Shop. On 4/3/86, the hose was disconnected from the tap and laid on the ground. Since the other end was submerged in the tank, and the hose was filled with water, water began to be siphoned from the tank to the yard, running into the adjacent yard drain.
Yard Contamination by General Contractor (G.C.) Paint Bldgs.	Shown as location 10 on Figure 2-1	On 12/2/77, concrete core drills (uses to install seismic anchors in concrete) were being cleaned in the yard, when some contaminated water was spilled. On 12/14/77, contaminated water was subsequently found outside the adjacent fence.

Event/Issue	Location	Synopsis
Railroad French Drain	Shown as location 11 on Figure 2-1	In mid-1993, while excavating to connect a new oil-water separator discharge line to the outfall tube, a previously unknown French drain (rock and perforated drainpipe) were found to be contaminated. The pipe appears to follow (alongside) the path of the (newer) rail spur into Unit 3.
Contamination at Railroad Gate	Shown as location 12 on Figure 2-1	On 10/9/80, contamination was found on the ground or pavement on both sides of the RA fence near the gate. In mid-1993, contaminated soil was found while excavating to connect a new oil-water separator discharge line to the outfall tube.
Unit 2 lube oil sump, Oily water Separator contamination	Shown as location 13 on Figure 2-1	On 11/21/83, about 1,200 gallons of Unit 3 Closed Cooling Water was spilled, going to the Unit 2 oily water sump, then to the Unit 2 oil-water separator, and then to the Low Volume Waste sump.
Condensate Pump spill to Yard Drain	Shown as location 14 on Figure 2-1	Surveys on 11/1/74 indicate a spill to the yard from a condensate pump. This may have occurred when #5 condensate pump was used to pump down the condenser.
Radwaste Treatment Bldg. roof – drum spill	Shown as location 15 on Figure 2-1	On 10/9/79, contaminated liquid was spilled on the roof of the radwaste building. Some of the contaminated liquid seems to have reached the drain at the east end of the lower section of the roof.
Radwaste Treatment Bldg. roof/truck	Shown as location 16 on Figure 2-1	On 11/14/80, concentrated waste was being solidified (in a liner on a truck) with urea formaldehyde. When the acid catalyst was added to the mix, it began to foam, and several gallons flowed onto the trailer, and then to the ground.

Event/Issue	Location	Synopsis
Reactor Water into Firewater system	Shown as location 17 on Figure 2-1	On 7/17/70, a reactor trip resulted in the low pressure core flooding system valves opening before the firewater pumps came up to full pressure. As a result, there was the potential for a short time for contaminated shutdown system water (potentially followed by reactor water) to flow into the firewater system, instead of the preferred opposite condition.
Spent Fuel Pool Leakage	Shown as location 18 on Figure 2-1	In March 1966, it was discovered that a leak in the SFP liner had developed, changing the water loss from about 0.23 inch per day to about 0.42 inch per day, or nominally between 75 to 130 gallons per day.
Caisson Leakage	Shown as location 19 on Figure 2-1	Beginning in 1992, the Caisson sump leak rate (groundwater in leakage) began to increase, from less than 100 gallons/day to about 7,000 gallons/day by 1997. The Caisson In leakage Repair Project (CIRP) plugged the leak and the leak rate decreased to about 10 gallons/day in September 1997.
Off-gas Tunnel Demineralizer Resin Spill	Shown as location 20 on Figure 2-1	In July 2005, the resin transfer line from the SFP demineralizer (through the offgas tunnel) to the resin disposal tank was found to be blocked. In January 2006, elevated radioactivity levels were identified in liquid radwaste. This led to an inspection of the offgas tunnel, which revealed resin and resin-like material in the tunnel sump near the offgas filter. The resin cleanup was completed on March 31, 2006.

Figure 2-1 Event/Activities Locations



LEGEND:

1 – 20	SPILLS
-----------	YARD DRAIN LINE
———	SUMP DRAIN LINE

PLAN VIEW
SCALE:1"=100'

DATE	REVISION DESCRIPTION	HUMBOLDT BAY POWER PLANT UNIT 3 DECOMMISSIC SITE SPILL LOCATIONS
01, NOVEMBER 2010		
PR BSE		
CK M.E.		DRAWING – SHE
AP B.B.		DSK–SITE–LTP 1 of

2.1.7 Survey Unit Identification and Classification

2.1.7.1 Survey Areas

The entire HBPP site, with the exception of the ISFSI, which is under a 10 CFR 72 license, is divided into areas. Areas are typically larger physical sections of the site that may contain one or more survey units, depending on their classification. Some characterization has been performed within the ISFSI area supporting a Class 3 area. This area will be released from the 10 CFR 50 license and will remain under the Part 72 license until such time as the spent fuel is moved to a federal repository.

2.1.7.2 Survey Units

A Survey Unit is a physical area consisting of buildings, structures, or land areas of specifically defined shapes and sizes, for which a unique decision will be made regarding whether the presence of any residual radioactive material meets or exceeds the predetermined release criteria. A Survey Unit is a single contiguous area, where size is dependent upon its physical characteristics (open land vs. structural building) and radiological conditions, and where operational conditions are reasonably consistent with the exposure modeling used to determine the classification.

2.1.7.3 Initial Designation of Areas

Using reasonable and available physical and documented references, 17 areas were identified and assigned a unique Survey Area identification. Current area designations (areas of the site are depicted in Figure 2-2, Area Designations) are summarized in Table 2-3.

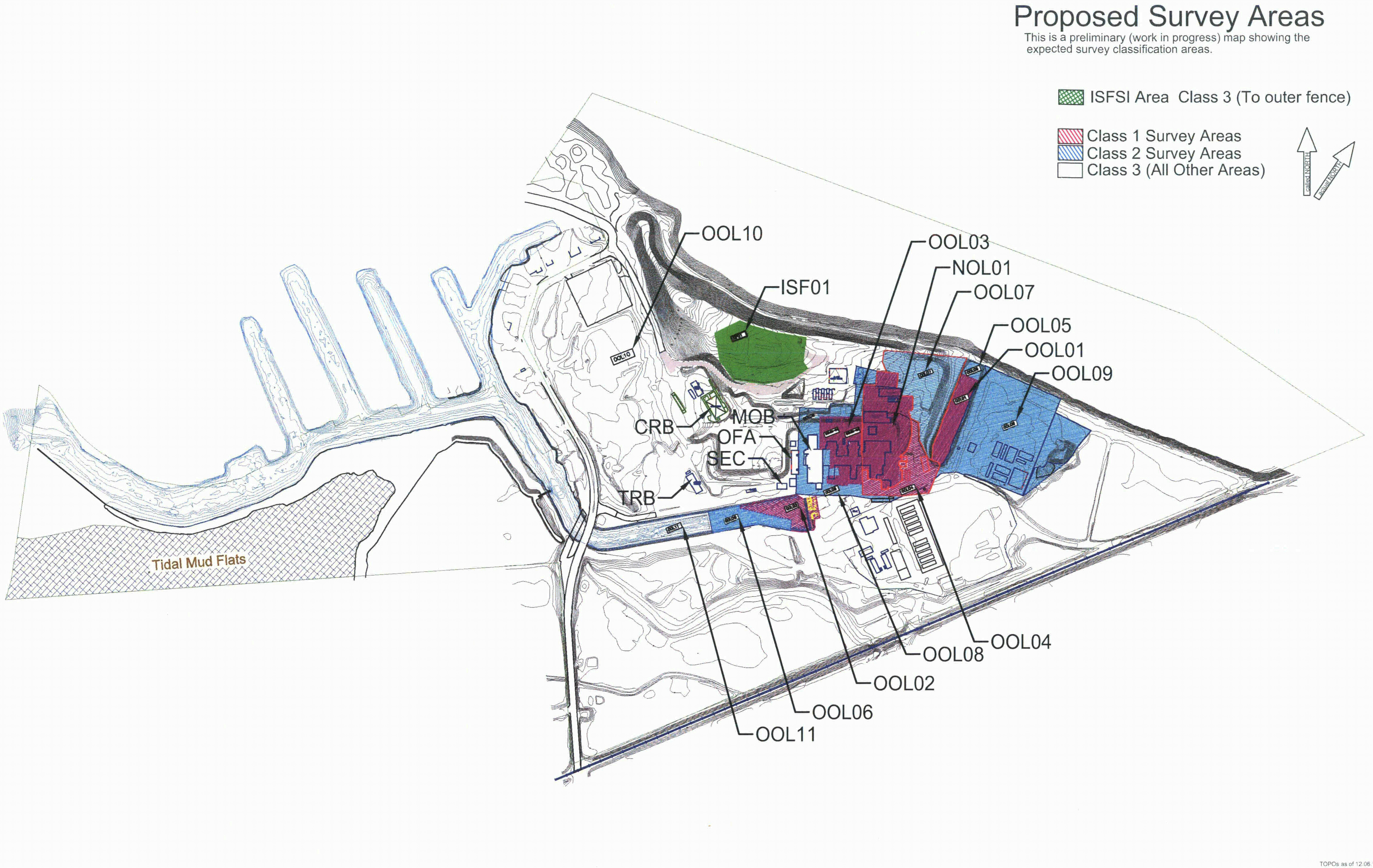
Table 2-3 Survey Area Summary Information

Survey Area Designator	Name/Building	Total Area Footprint (Square Feet)	Total Area Footprint (Square Meters)	Classification
NOL01	Open land area (inside RA)	81,989	7617	Class 1
OOL01	Discharge Canal South	26,596	2471	Class 1
OOL02	Intake East	6,755	628	Class 1
OOL03	Open Land Area Outside the RA	39,735	3692	Class 1

Survey Area Designator	Name/Building	Total Area Footprint (Square Feet)	Total Area Footprint (Square Meters)	Classification
OOL04	Sump Drain Line Land Area	4,929	458	Class 1
OOL05	Discharge Canal North	5,987	556	Class 2
OOL06	Intake Center	22,039	2047	Class 2
OOL07	NOL01 Boundary East	89,621	8326	Class 2
OOL08	NOL Boundary West	53,409	4962	Class 2
OOL09	Haz. Waste Area	11,109	1032	Class 2
OOL10	Remainder of Land Area	2,531,578	235,191	Class 3
OOL11	Intake West	26,582	2470	Class 3
OFA	Office Annex	2906	270	Class 3
ISF01	ISFSI area	59,600	5540	Class 3
TRB	Training Building	431	40	Class 3
SEC	Security Building	527	49	Class 3
MOB	Main Office Building	4402	409	Class 3
CRB	Count room Building	4004	372	Class 1
WMB	Waste Management Building	*	*	Class 1

* To be constructed

Figure 2-2 Survey Area Locations



2.1.8 Area Radiological Impact Summaries

The following were isotopes analyzed for in the 1997 characterization sampling program. The radionuclides were chosen based on historical sampling information.

RAI 10

Fe-55	Mn-54
Ni-63	Co-60
Sr-90	Zn-65
Tc-99	Sb-125
Am-241	Cs-134
Pu-238	Cs-137
Pu-239/240	Eu-154
Cm-242	Eu-155
Cm-244	Pu-241

All of the samples taken were analyzed for gamma emitters and randomly selected samples were analyzed for hard-to-detect isotope analysis. Analyses for Am-241 and Ni-63 revealed very low concentrations present in the soils relative to the concentration of Cs-137. Several of the HBPP Survey Areas fell within a single 1997 Survey Area (i.e. 03,04,05,06,07,08 and 09). For these HBPP areas the same number of samples analyzed for HTDs were reported.

2.1.8.1 NOL01-Open Land Area inside the RA

Survey Area NOL01 consists of the open land area within the boundary of the RA. Survey Area NOL01 contains about 7,617 square meters (m^2) of surface area made up of soils, engineered materials, gravel, and sand.

NOL01 is bounded by OOL07 and OOL08 on the north, OOL03 and OOL08 on the west, OOL04 and OOL08 on the south, and OOL07 on the east. NOL01 lies within the Unit 3 restricted area fencing. All the structures, systems, and components supporting Unit 3 are located within NOL01.

The following events and activities may have affected Survey Area NOL01:

- LRW concentrator steam condensate leakage to yard drain north loop
- Overflow of condensate tank
- Overflow of LRW concentrator to CWT vault, radwaste tankage sump and its drain to the outfall canal

- LRW concentrator steam condensate leakage to radwaste tankage sump
- Overflow of condensate demineralizers
- Ultrasonic water spill
- Yard contamination by G.C. Paint Bldgs
- Railroad French drain
- Contamination at railroad gate
- Condensate pump spill to yard drain
- Radwaste Treatment building roof (drum spill)
- Radwaste Treatment building roof/truck
- SFP leakage
- Caisson leakage
- Offgas tunnel demineralizer resin spill
- Wet and dry deposition from stack releases

Translocation pathways within NOL01 include:

- Radioactive liquids from events to the surface soils and downward to the subsurface
- Leakage from the SFP to the subsurface soils
- Leakage from subsurface components (e.g., French drains) to the subsurface soils
- Wet and dry deposition of radioactive materials to the surface via Unit 3 stack releases

An extensive characterization was performed in 1997 [2-5] consisting of soil and sediment sampling. Table 2-4 provides a summary of the characterization within NOL01.

Table 2-4 NOL01 Characterization Data

Location	Cs-137 (pCi/g)	Co-60 (pCi/g)	Depth (feet)
2SS034	0.20	0.07	0.5
2SS035	0.77	ND	0.5
2SS028	2.65	0.28	0.5
2SS033	1.85	0.24	0.5
1S0026	0.38	0.09	0.8
1S0027	0.54	ND	0.5
1S0027	0.09	ND	3.5
1S0028	0.07	ND	3.5
1S0028	0.09	ND	4.0

Location	Cs-137 (pCi/g)	Co-60 (pCi/g)	Depth (feet)
1S0024	0.24	0.58	1.0
1S0024	0.16	0.22	2.0
1S0029	0.11	ND	1.0
1S0023	0.24	ND	1.0
1S0023	0.42	ND	2.0
1S0022	0.11	ND	1.0
1S0022	0.28	ND	2.0
1S0021	2.18	ND	1.0
1S0021	0.16	ND	2.0
1S0020	1.23	ND	1.0
1S0020	5.90	ND	2.0
1S0020	0.40	ND	3.0
1S0020	0.13	ND	4.0
1S0019	1.35	ND	1.0
1S0019	0.30	ND	2.0
1S0018	1.55	ND	1.0
1S0017	0.26	ND	1.5
1S0017	0.06	ND	4.0
1S0059	0.31	0.08	1.0
1S0059	2.23	0.28	2.0
1S0059	0.07	ND	3.0
1S0060	0.23	ND	0.5
1S0060	0.78	ND	1.5
1S0012	0.11	ND	0.5
1S0013	0.05	ND	1.0
1S0011	1.13	0.12	0.5
1S0058	2.13	2.34	1.5
1S0062	3.98	0.3	0.5
1S0062	0.52	ND	1.5
1S0062	0.24	0.13	2.5
1S0062	0.47	0.11	3.5
1S0062	0.27	ND	4.0
1S0051	0.09	0.06	1.0
1S0051	0.64	0.12	2.0
1S0051	0.14	ND	3.0
1S0048	0.74	ND	1.0
1S0048	0.09	ND	2.0
1S0048	0.07	ND	4.0
1S0049	9.29	0.32	1.0
1S0049	21.50	0.57	2.0
1S0049	8.20	0.60	5.0
1S0049	0.46	ND	6.0
1S0049	0.31	ND	7.0
1S0049	0.24	ND	8.0
1S0049	0.17	ND	9.0
1S0049	0.11	ND	10.0
1S0050	3.80	ND	1.0
1S0050	4.55	0.06	2.0
1S0050	3.40	ND	3.0
1S0050	1.39	ND	4.0
1S0047	0.20	ND	1.0

Location	Cs-137 (pCi/g)	Co-60 (pCi/g)	Depth (feet)
1S0047	0.22	ND	2.0
1S0008	0.12	ND	1.0
1S0008	0.37	ND	2.0
1S0008	0.13	ND	3.0
1S0056	0.16	0.07	1.5
1S0056	ND	0.10	3.5
1S0046	0.08	ND	1.0
1S0046	0.09	ND	4.0
1S0076	0.09	ND	0.5
1S0076	2.3	ND	1.5
1S0076	26.15	ND	2.5
1S0076	6.98	ND	3.0
1S0053	0.67	0.08	1.0
1S0053	19.67	ND	2.0
1S0053	18.30	0.12	3.5
1S0053	17.67	0.16	4.5
1S0053	39.19	ND	5.0
1S0053	27.67	ND	6.0
1S0053	31.43	0.16	7.0
1S0053	11.00	0.15	8.0
1S0053	13.89	ND	9.0
1S0053	29.02	0.12	9.5
1S0077	17.77	3.65	0.5
1S0077	14.39	0.48	1.5
1S0077	17.96	0.34	2.5
1S0077	13.09	0.28	3.5
1S0077	11.04	0.12	4.5
1S0077	24.87	0.22	5.5
1S0077	25.97	0.15	6.5
1S0077	16.49	0.50	7.5
1S0077	12.89	0.63	8.5
1S0077	30.91	0.18	9.5
1S0077	18.82	ND	10.5
1S0077	6.29	0.14	11.5
1S0077	3.52	ND	12
1S0034	0.33	ND	1.0
1S0054	0.08	0.06	2.0
1S0045	3.09	0.39	1.0
1S0045	0.17	ND	2.0
1S0044	10.13	3.57	4.0
1S0043	10.03	0.09	1.5
1S0004	0.32	ND	1.0
1S0042	2.27	0.34	1.0
1S0005	0.21	ND	2.0
1S0041	0.16	ND	1.0
1S0041	1.99	ND	3.0
1S0041	0.13	ND	4.0
1S0040	0.14	0.08	1.5
1S0006	0.12	ND	2.0
1S0037	0.35	0.12	1.0
1S0037	0.10	ND	2.5

Location	Cs-137 (pCi/g)	Co-60 (pCi/g)	Depth (feet)
1S0038	0.26	0.08	1.0
1S0039	0.08	ND	1.5
1S0039	ND	0.10	3.5
1S0039	ND	0.14	5.5
1S0039	0.15	ND	6.0
1S0007	0.17	ND	2.0

ND = Not detected

Hard-to-detect radionuclides were analyzed for in this survey area:

- Fifty-nine samples were analyzed for Am-241. Four detections were indicated with the maximum result of 0.26 pCi/g. The MDA range for the analysis was 0.06 to 0.33 pCi/g.

- Eighteen samples were analyzed for Cm-243/244/245/246. One detection of Cm-243/244 at 0.08 pCi/g which was above the MDA of 0.06 pCi/g. Five detections of Cm-245/246 with a maximum value of 0.05 pCi/g which was above the MDA range of 0.01 to 0.07 pCi/g. RAI 03

- Eighteen samples were analyzed for Ni-63. Two detections were identified with the maximum value of 4 pCi/g with a MDA range of 1.22 to 1.69 pCi/g.

- Eighteen samples were analyzed for Pu-238/239/240/241. One detection of Pu-238 at 0.14 pCi/g at an MDA of 0.08 pCi/g. Two detections of Pu-239/240 with a maximum of 0.14 pCi/g and an MDA range of 0.01 to 0.10 pCi/g. No detections of Pu-241 were observed.

As seen in Table 2-4, activity in the soils of NOL01 vary considerably. Levels of contamination in the first 0.5 foot from the surface average approximately 1 pCi/g Cs-137 and 0.12 pCi/g Co-60. However, areas where events have occurred exhibit considerably greater contamination, not only at the surface but also at depths to 12 feet, or greater. In these areas, contamination oscillates around 17 pCi/g for some depth, until the concentrations start to decrease. It is apparent that extensive remediation will occur in NOL01; therefore, NOL01 is classified as a Class 1 area.

2.1.8.2 OOL01- Discharge Canal South

Survey Area OOL01 consists of the open land area within the southern section of the Discharge Canal. Survey Area OOL01 contains about 2,471 m^2 of surface area made up primarily of silt and sand.

OOL01 is bounded by OOL05 on the north, OOL09 on the east, OOL04 on the south, and OOL07 on the west. OOL01 is the site where circulating water, from the units, discharged to prior to entering the bay. The outer boundary of the survey area is the high water mark. Silting in of sediment has occurred since circ. water flow has ceased to a depth of approximately 10-15 feet.

The following events and activities may have affected Survey Area OOL01:

- Routine discharges of radioactive liquids from Unit 3
- Possible radwaste spill to radwaste tankage area drain
- Overflow of LRW concentrator to CWT vault, radwaste tankage sump and its drain to the outfall canal
- LRW concentrator steam condensate leakage to radwaste tankage sump
- Wet and dry deposition from stack releases

Translocation pathways within OOL01 include the following:

- Routine discharges of radioactive liquid from Unit 3 to the canal with the activity concentrating within the top 2 feet of the sediment
- Non-routine discharges of radioactive liquids to the canal with the activity depositing as described previously
- Activity from the deposition of stack releases settle on the water surface and progress downward into the sediments

An extensive characterization was performed in 1997 consisting of soil and sediment sampling. Table 2-5 provides a summary of the characterization within OOL01 to the original depth.

Table 2-5 OOL01 Characterization Data

Location	Cs-137 (pCi/g)	Co-60 (pCi/g)	Depth (feet)
6SD027	42.24	ND	0.5
6SD028	9.62	ND	0.5
6SD025	5.86	2.20	1.7
6SD023	0.37	0.17	0.5
6SD024	10.08	0.34	1.7
6SD026	10.67	2.94	1.7
6SD030	1.51	ND	1.7
6SD031	9.48	1.37	1.7
6SD032	8.54	0.45	1.7
6SD033	8.57	1.31	1.7
6SD034	8.78	0.99	1.7
6SD052	0.50	ND	1.5
6SD052	0.26	ND	2.0
6SD035	7.93	1.67	1.7
6SD036	11.77	0.45	1.7
6SD037	3.63	0.65	1.7
6SD038	13.92	0.87	1.7
6SD039	11.21	0.54	1.7
6SD053	0.56	ND	2.0
6SD040	8.96	0.69	1.7
6SD041	8.58	0.64	1.7
6SD040	8.96	0.69	1.7

ND = Not detected

Six samples were analyzed for Am-241 and five samples were analyzed for Cm-242/243/244/245/246, Ni-63 and Pu-238/239/240/241. One detection for Cm-245/246 was observed at a value of 0.04 pCi/g with an MDA of 0.01 pCi/g. The lone Cm detection represents a fraction of a DCGL (*f*DCGL) of 5.62E-04 which would neither affect the classification of the survey area, nor alter the DQOs associated with the planning of the surveys. RAI 04

As seen in Table 2-5, activity in the sediments, prior to silting in, of OOL01 vary considerably. Generally, levels of contamination average approximately 8.7 pCi/g for Cs-137 and 1.0 pCi/g for Co-60. Concentration levels at the point of entry of the circulating water into the canal are greater than the remainder of readings. An additional characterization was performed in 2008 to examine the concentrations at depths greater than those sampled in 1997. The results of those surveys determined that the contamination was limited to the top 2 feet in the sediment. The 2008 characterization survey was analyzed for the radionuclides-of-concern RAI 05

(Cs-137 and Co-60) as determined in the characterization plan. It is apparent that remediation will occur in OOL01; therefore, OOL01 is classified as a Class 1 area.

2.1.8.3 *OOL02-Intake East*

Survey Area OOL02 consists of the open land area within the eastern portion of the Intake Canal. Survey Area OOL02 contains about 628 m^2 of surface area made up primarily of silt and sand.

OOL02 is bounded by OOL08 on the north and east, OOL10 on the south, and OOL06 on the west.

The following events and activities may have affected Survey Area NOL01:

- LRW concentrator steam condensate leakage to yard drain north loop
- Wet and dry deposition from stack discharges

Translocation pathways within OOL02 include the following:

- Radioactive liquid from the yard drain north loop proceeding and discharging into the intake where the activity deposits into the sediments
- Activity from the stack depositions settling onto the surface of the water and migrating downward into the sediments

Table 2-6 provides the results of the 1997 characterization effort.

Table 2-6 OOL02 Characterization Data

Location	Cs-137 (pCi/g)	Co-60 (pCi/g)	Depth (feet)
6SD020	5.30	0.46	0.5
6SD022	22.39	0.30	1.7
6SD021	0.57	ND	1.7

ND = Not detected

Ten samples were analyzed for Am-241. Five samples were analyzed for Cm-242/243/244/245/246, Ni-63 and Pu-238/239/240/241. One detection was observed for Pu-238 at 0.13 pCi/g with an MDA of 0.12 pCi/g. One detection of Pu-239/240 was observed at 0.22 pCi/g with an MDA of RAI 06

0.08 pCi/g. There were no other detections observed with the other radionuclides greater than their respective MDAs.

As seen in Table 2-6, activity in the sediments of OOL02 will necessitate some degree of remediation. Additional characterization is scheduled for this survey area to fill in the data gaps. OOL02 is classified as a Class 1 area.

2.1.8.4 OOL03 Open Land Area outside the RA

Survey Area OOL03 consists of the open land area north of Units 1 and 2. OOL03 encompasses the north yard and embankment. Survey Area OOL03 contains about 1,989 $m^2$of surface area made up primarily of soils and engineered materials.

OOL03 is bounded by OOL08 on the north, west, and south sides; NOL01 bounds the Survey Area on the east.

The following events and activities may have affected Survey Area OOL03:

- Subsequent contamination of electric conduit/pullbox, yard north of Unit 2, and north loop of yard drains
- Pavement contamination north of Unit 2, and under Unit 2 fans
- Pavement contamination north of Unit 2 – 1989
- Unit 2 yard remediation – 1991
- Contamination under Unit 2 fans – Approximately 1991
- Unit 2 lube oil sump, oily water separator contamination
- Wet and dry deposition from Unit 3 stack discharges

Translocation pathways within OOL03 include the following:

- Radioactive liquid traveling across the yard area from Unit 3 and into the yard drain north of Unit 2
- Liquids from the above event followed a ditch along the bank north of Unit 2, going through drain rock and a perforated pipe into the soils
- Activity from the deposition of stack releases settling onto the surfaces of the Survey Area

Table 2-7 provides the results of the 1997 characterization effort.

Table 2-7 OOL03 Characterization Data

Location	Cs-137 (pCi/g)	Co-60 (pCi/g)	Depth (feet)
1S0071	0.71	ND	0.5
1S0071	0.09	ND	1.5
1S0072	1.26	ND	0.5
1S0072	0.56	0.18	1.5
1S0072	0.23	0.10	2.5
1S0073	0.47	ND	0.5
1S0073	0.17	ND	1.5
4S0040	3.19	0.49	0.5
4S0040	0.72	ND	1.5
1S0074	0.93	0.12	0.5
1S0074	0.17	ND	1.5
4S0039	3.57	0.25	0.5
4S0039	0.30	0.10	1.5
4S0039	0.33	ND	2.5
4S0039	0.20	ND	3.0
1S0075	0.87	ND	0.5
1S0075	0.06	ND	1.5
1S0075	0.17	ND	2.5
4S0038	11.30	0.23	0.5
4S0038	1.79	ND	1.5
4S0038	1.67	0.06	2.5
4S0038	1.48	0.09	3.0
4S0038	0.21	ND	3.5
4S0037	0.21	ND	0.5
1S0065	18.22	0.19	0.5
1S0066	23.70	0.14	0.5
1S0066	1.11	ND	1.0
1S0067	14.06	0.13	0.5
1S0067	3.12	ND	1.0
1S0068	11.84	0.11	0.5
1S0068	0.88	ND	1.0
1S0069	9.98	ND	0.5
1S0069	0.42	ND	1.0
1S0070	2.76	ND	0.5
1S0070	2.32	ND	1.0
1S0063	11.81	0.18	0.5
1S0033	0.41	ND	4.0
1S0064	19.55	0.48	0.5
1S0064	0.8.	ND	1.0
1S0032	0.23	ND	1.0
1S0034	0.33	ND	1.0

ND = Not detected

Twenty-one samples were analyzed for Am-241 and one sample was analyzed for Cm-242/243/244/245/246, Ni-63

RAI 07

and Pu-238/239/240/241. No detections above the MDA were observed for these isotopes.

As seen in Table 2-7, activity in the soils of OOL03 will necessitate some degree of remediation. OOL03 is classified as a Class 1 area.

2.1.8.5 *OOL04 - Sump Drain Line Land Area*

Survey Area OOL04 consists of a narrow strip of open land area traveling from Unit 3 to the discharge canal. Buried beneath OOL04 is the sump drain line. Survey Area OOL04 contains about 458 m^2 of surface area made up primarily of soil.

During the initial phase of the HBGS construction in 2008, utilities and obstructions were removed or relocated from the HBGS footprint area to prepare for the HBGS builder to begin construction. While a utility line was being relocated to an area outside of the HBGS footprint area, the soil was removed from the top of the discharge tubes. An access portal or manhole was discovered in this area. The radwaste tankage drain line connected into this concrete monolith. The line then exited the monolith toward the discharge canal. The drain line, as well as the concrete monolith, was significantly contaminated (20 mrem/hr on the concrete surface).

The concrete monolith was removed as well as most of the drain line toward the Unit 3 RA. Soil samples in the area were greater than 50pCi/g. The soil was removed to "near background levels." The area above the discharge tubes and around the radwaste tankage drain line is designated a Class 1area.

OOL04 is bounded by OOL08 and OOL10 on the south, OOL10 and OOL09 on the east NOL01 and OOL08 on the west and NOL01, OOL01 and OOL07 bound the Survey Area on the north.

The following events and activities may have affected Survey Area OOL04:

- Routine discharges of radioactive liquids from Unit 3
- Possible radwaste spill to radwaste tankage area drain

- Overflow of LRW concentrator to CWT Vault, radwaste tankage sump and its drain to the outfall canal
- Wet and dry deposition from Unit 3 stack discharges

Translocation pathways within OOL04 include the following:

- Routine, as well as non-routine, discharges through the drain line with the potential for the migration of activity from the piping to the subsurface soils below
- Activity from the stack deposition settling onto the surfaces of the soils

Twenty-one samples were analyzed for Am-241 and one sample was analyzed for Cm-242/243/244/245/246, Ni-63 and Pu-238/239/240/241. No detections above the MDA were observed for these isotopes. RAI 07

Since there have been significantly contaminated soils in this area, the area above the discharge tubes and around the radwaste tankage drain line is designated a Class 1 area.

2.1.8.6 *OOL05 - Discharge Canal North*

Survey Area OOL05 consists of the north end of the discharge canal. Characterization sampling has identified a reduction in the concentration of activity, prompting a different classification from the remainder of the canal. Survey Area OOL05 contains about 556 m^2 of surface area made up primarily of silt and sediment.

In 2008, 19 sediment sampling borings were advanced to characterize radiological and environmental chemical soil conditions in the HBPP discharge canal and to determine the environmental impacts at the plant. The reasoning and rationale for the locations of the 19 sample borings was to provide a sufficient spatial characterization of the discharge canal soils for decision-making purposes. The samples collected in 1997 by IT/Duratek ("Structural Characterization Report for Humboldt Bay Power Plant," March, 1998) were all collected down the centerline of the discharge canal. Several samples had elevated concentrations of Cs-137 at the lowest depth of sample (approximately 1.7 feet) into the native sediment; however, the northernmost end of the canal indicated considerably less activity than the remainder of the canal.

OOL05 is bounded by OOL01 on the south, OOL09 on the east, OOL07 on the west side; OOL10 bound the Survey Area on the north.

The following events and activities may have affected Survey Area OOL04:

- Routine discharges of radioactive liquids from Unit 3
- Possible radwaste spill to radwaste tankage area drain
- Overflow of LRW concentrator to CWT vault, radwaste tankage sump and its drain to the outfall canal
- LRW concentrator steam condensate leakage to radwaste tankage sump
- Wet and dry deposition from Unit 3 stack releases

Translocation pathways within OOL05 include the following:

- Some of the activity from the routine and non-routine liquid releases into the discharge canal reaching the northern section and progressing downward into the sediments below
- Activity settling onto the surfaces of the water from the deposition of stack releases settling into the sediments below

Table 2-8 provides the results of the 1997 characterization effort.

Table 2-8 OOL05 Characterization Data

Location	Cs-137 (pCi/g)	Co-60 (pCi/g)	Depth (feet)
6SD043	1.75	0.19	1.7
6SD044	1.73	0.28	1.7
6SD045	0.18	ND	1.7

ND = Not detected

Twenty-one samples were analyzed for Am-241 and one sample was analyzed for Cm-242/243/244/245/246, Ni-63 and Pu-238/239/240/241. No detections above the MDA were observed for these isotopes. RAI 07

Since activity has been found in this area, albeit at low levels, OOL05 is classified as a Class 2 Survey Area. Further characterization is scheduled for this area.

2.1.8.7 OOL06 – Intake Center

Survey Area OOL06 consists of the open land area within the center portion of the Intake Canal. Survey Area OOL06 contains about 2,047 m^2 of surface area made up primarily of silt and sand.

OOL06 is bounded by OOL08 on the north, OOL02 on the east, OOL10 on the south, and OOL11 on the west.

The following events and activities may have affected Survey Area NOL01:

- LRW concentrator steam condensate leakage to yard drain north loop
- Wet and dry deposition of activity from Unit 3 stack releases

Translocation pathways within OOL06 include the following:

- A portion of the activity from the yard drain discharge migration into the Survey Area, due to tidal influences, traveling downward into the sediments below
- Activity from the deposition from stack releases settling onto the water surface and traveling downward into the sediments below

Table 2-9 provides the results of the 1997 characterization effort.

Table 2-9 OOL06 Characterization Data

Location	Cs-137 (pCi/g)	Co-60 (pCi/g)	Depth (feet)
6SD018	0.23	ND	1.7
6SD017	0.18	ND	1.7

ND = Not detected

Twenty-one samples were analyzed for Am-241 and one sample was analyzed for Cm-242/243/244/245/246, Ni-63 and Pu-238/239/240/241. No detections above the MDA RAI 07 were observed for these isotopes.

Since limited characterization data exist for this Survey Area, and the potential exists for the area to be impacted, OOL06 is classified as a Class 2 Survey Area. Further characterization is scheduled for this area.

2.1.8.8 *OOL07 – NOL Boundary East*

Survey Area OOL07 consists of the open land area bordering the eastern side of the Class 1 Survey Area NOL01. Survey Area OOL07 contains about 8,326 m^2 of surface area made up primarily of soils and engineered materials.

OOL07 is bounded by OOL10 on the north, OOL01 and OOL05 on the east, OOL04 on the south, and NOL01 on the west.

The following events and activities may have affected Survey Area OOL07:

- Wet and dry deposition from the stack releases
- Liquid spills crossing over into OOL07 from NOL01

Translocation pathways within OOL07 include the following:

- Radioactive liquids migrating from NOL01 crossing into OOL07 settling onto and beneath the soils
- Activity from the deposition from stack releases settling onto the surface of the Survey Area

Table 2-10 provides the results of the 1997 characterization effort.

Table 2-10 OOL07 Characterization Data

Location	Cs-137 (pCi/g)	Co-60 (pCi/g)	Depth (feet)
2SS026	0.08	ND	0.5
2SS029	0.62	ND	0.5
4SO027	0.07	0.11	1-3
4SO028	0.12	0.16	0.5

ND = Not detected

Twenty-one samples were analyzed for Am-241 and one sample was analyzed for Cm-242/243/244/245/246, Ni-63 and Pu-238/239/240/241. No detections above the LLD were observed for these isotopes. RAI 07

Since limited characterization data exist for this Survey Area, further characterization is scheduled for this area. OOL06 is classified as a Class 2 Survey Area.

2.1.8.9 OOL08 – NOL Boundary West

Survey Area OOL08 consists of the open land area bordering the western side of the Class 1 Survey Area NOL01. Survey Area OOL08 contains about 6,837 m^2 of surface area made up primarily of soils and engineered materials.

OOL08 is bounded by OOL10 on the north and west, OOL10 and OOL02 on the south, and NOL01, OOL04, and OOL07 on the east.

The following events and activities may have affected Survey Area OOL08:

- Wet and dry deposition from the stack releases
- Liquid spills crossing over into OOL08 and OOL03 and NOL01

Translocation pathways within OOL08 include the following:

- Radioactivity in this Survey Area would translocate in much the same way as activity in OOL07

Table 2-11 provides the results of the 1997 characterization effort.

Table 2-11 OOL08 Characterization Data

Location	Cs-137 (pCi/g)	Co-60 (pCi/g)	Depth (feet)
2SS003	0.24	ND	0.5
2SS024	0.71	ND	0.5
2SS016	0.16	ND	0.5
4SO030	0.20	ND	0.5
2SS019	0.23	ND	0.5
4SO002	0.26	ND	0.5
4SO002	0.09	ND	3.5

ND = Not detected

Twenty-one samples were analyzed for Am-241 and one sample was analyzed for Cm-242/243/244/245/246, Ni-63 and Pu-238/239/240/241. No detections above the MDA were observed for these isotopes. RAI 07

OOL08 is classified as a Class 2 Survey Area. Further characterization is scheduled for this area.

2.1.8.10 OOL09 – Hazardous Waste Area

Survey Area OOL09 consists of the open land area east of the discharge canal and the site occupied by temporary trailers. Survey Area OOL09 contains about 1,032 m^2 of surface area made up primarily of soils and engineered materials.

OOL09 is bounded by OOL10 on the north, south and east and OOL01, OOL04 and OOL05 on the west side.

The following events and activities may have affected Survey Area OOL09:

- The placement of slightly contaminated hazardous waste spoils in the area

Translocation pathways within OOL09 include the following:

- Small quantities of radioactive material could leach from the spoils pile onto the soils where it was placed
- Activity from the deposition of stack releases could settle onto the soils

Table 2-12 provides the results of the 1997 characterization effort.

Table 2-12 OOL09 Characterization Data

Location	Cs-137 (pCi/g)	Co-60 (pCi/g)	Depth (feet)
4SO036	0.1	ND	3.5
4SO036	ND	0.1	4.0
4SO035	3.87	1.28	0.5
4SO042	0.08	ND	0.5
4SO042	0.09	ND	1.5
4SO042	0.11	ND	2.5
4SO044	0.12	ND	0.5

Location	Cs-137 (pCi/g)	Co-60 (pCi/g)	Depth (feet)
4SO044	0.08	ND	1.5
4SO044	0.07	0.05	2.5
4SO045	0.14	ND	0.5
4SO045	0.07	ND	1.5
4SO045	0.12	ND	2.5

ND = Not detected

Twenty-one samples were analyzed for Am-241 and one sample was analyzed for Cm-242/243/244/245/246, Ni-63 and Pu-238/239/240/241. No detections above the MDA were observed for these isotopes. RAI 07

OOL09 is classified as a Class 2 Survey Area. Further characterization is scheduled for this area.

2.1.8.11 OOL10 – Remainder of Land Area

Survey Area OOL10 consists of the remainder of the open land areas, with the exception of the western portion of the Intake. Survey Area OOL10 contains about 234,584 m^2 of surface area made up primarily of soils and engineered materials.

OOL10 is bounded by the bay on the north and non-PG&E property on the remaining sides.

The following events and activities may have affected Survey Area OOL10:

- Wet and dry deposition of activity from Unit 3 stack releases

Translocation pathways within OOL10 include the following:

- Activity from the deposition from Unit 3 stack releases settling onto the soil surfaces

Table 2-13 provides a summary of the results of the 1997 characterization effort.

Table 2-13 OOL10 Characterization Data

Nuclide	Samples Analyzed	Number Detections	Mean pCi/g
Cs-137	35	21	0.30
Cs-137	36	30	0.46

All samples were analyzed for Cs-137, Co-60 and Mn-54. Four samples were analyzed for Am-241 with no activity above the MDA for Co-60 and Am-241. RAI 08

OOL10 is classified as a Class 3 Survey Area.

2.1.8.12 *OOL11 – Intake West*

Survey Area OOL11 consists of the open land area within the western portion of the Intake Canal. Survey Area OOL11 contains about 2,470 m^2 of surface area made up primarily of silt and sand.

OOL11 is bounded by OOL11 on the north and south sides, non-PG&E property on the west, and OOL06 on the east.

The following events and activities may have affected Survey Area OOL11:

- LRW concentrator steam condensate leakage to yard drain north loop
- Wet and dry deposition of activity from Unit 3 stack releases

Translocation pathways within OOL10 include the following:

- The possible migration of activity from the yard drain discharge and deposition from stack releases into the Survey Area due to tidal influences

Table 2-14 provides the results of the 1997 characterization effort.

Table 2-14 OOL11 Characterization Data

Location	Cs-137 (pCi/g)	Co-60 (pCi/g)
6SD010	0.08	ND
6SD013	ND	0.08

ND = Not detected

Ten samples were analyzed for Am-241. One sample wasanalyzed for Cm-242/243/244/245/246, Ni-63 and Pu-238/239/240/241. One detection was observed for Cm- 245/246 with an activity of 0.05 pCi/g with an MDA of 0.03pCi/g. One each detection was observed for Pu-238 RAI 09 and Pu-239/240 with activities of 0.13 and 0.22 pCi/g respectively. The MDAs for the Pu-238 and Pu-239/240 were 0.12 and 0.08 pCi/g respectively.

Since limited characterization data exist for this Survey Area, and the potential exists for the area to be impacted, OOL11 is classified as a Class3 Survey Area. Further characterization is scheduled for this area.

2.1.8.13 OFA01 – Office Annex

Survey Area OFA01 consists of the Office Annex Building. Survey Area OFA01 footprint contains about 270 m^2 of surface area. The Office Annex is a concrete block structure for administrative offices constructed in 1980s.

OFA01 is bounded by OOL10 on the north, west, and south sides, and OOL08 on the east.

The following events and activities may have affected Survey Area OFA01:

- Wet and dry depositions from Unit 3 stack discharges

Translocation pathways within OFA01 include the following:

- Activity from the deposition from stack releases settling onto the building surfaces (more so on the roofs than the remainder of the surfaces)

A characterization was performed September 2008 consisting of 41 fixed-point measurements. No measurements on the building's exterior walls exceeded the Maximum Detectable Activity (MDA) for the instrument. The 19 measurements that exceeded the instrument's MDA are listed in Table 2-15.

Table 2-15 provides the results of the 2008 characterization effort.

Table 2-15 OFA01 Characterization Data

Location	Dpm/100cm2	Location	Dpm/100cm2
Floor	223	Roof	458
Floor	627	Roof	465
Roof	559	Roof	397
Roof	455	Roof	386

Location	Dpm/100cm2	Location	Dpm/100cm2
Roof	429	Roof	437
Roof	415	Roof	483
Roof	382	Roof	368
Roof	447	Roof	408
Roof	451	Roof	307
Roof	433		

OFA01 is classified as a Class 3 Survey Area.

2.1.8.14 *ISF01 – ISFSI area*

Survey Area ISF01 consists of the ISFSI. Survey Area ISF01 footprint contains about 5540 m^2 of surface area. ISF01 is located at the top of the hill on the west side of the site.

ISF01 is bounded by OOL10 on all sides.

The following events and activities may have affected Survey Area ISF01:

- Wet and dry depositions from Unit 3 stack discharges

Translocation pathways within ISF01 include the following:

- Activity from the deposition from stack releases settling onto the soil surfaces.

Table 2-16 provides the results of the 2008 characterization effort.

Table 2-16 ISF01 Characterization Data

Location	dpm/100cm2
2SS017	0.13
2SS039	0.12
2SS038	0.21
2SS036	0.17
2SS037	ND

ND = Not detected

ISF01 is classified as a Class 3 Survey Area. The ISFSI security building was constructed after Unit 3 shutdown and is therefore classified as non-impacted. The ISF01 will remain under the HBPP 10 CFR 72 license after it is released from the Part 50 license.

2.1.8.15 TRB01 – Training Building

Survey Area TRB01 consists of the Training Building. Survey Area TRB01 footprint contains about 40 m^2 of surface area. The Training Building is located adjacent to the PG&E employee parking lot.

The Training Office Building was used by the Training Coordinator. It is a concrete block structure, constructed about 1974, originally intended to be a security search area. It has been used for training and is now used by security personnel.

TRB01 is bounded by OOL10 on all sides.

The following events and activities may have affected Survey Area TRB01:

- Wet and dry depositions from Unit 3 stack discharges

Translocation pathways within TRB01 include the following:

- Activity from the deposition from stack releases settling onto the building surfaces (more so on the roofs than the remainder of the surfaces)

A characterization was performed September 2008 consisting of 30 fixed-point measurements. Of the 30 measurements taken, 17 exceeded the instrument's MDA and are listed in Table 2-17.

Table 2-17 provides the results of the 2008 characterization effort.

Table 2-17 TRB01 Characterization Data

Location	dpm/100cm2	Location	dpm/100cm2
Roof	307	Roof	281
Roof	256	Roof	289
Roof	213	Roof	404
Roof	296	Interior	733
Roof	361	Interior	265
Roof	242	Interior	249
Roof	307	Exterior Walls	340
Exterior Walls	257	Exterior Walls	310
Exterior Walls	234		

TRB01 is classified as a Class 3 Survey Area.

2.1.8.16 SEC01 – Security Building

Survey Area SEC01 consists of the Security Building. Survey Area SEC01 footprint contains about 49 m^2 of surface area. The Security Building is located adjacent to the access road at the entrance to the Industrial Area.

The Security Building is a small concrete block building housing the security officers and the site entry port. It was constructed about 1974 and originally intended as a security search area. It has been used for training, and is now office space for the plant security force.

SEC01 is bounded by OOL10 on all sides.

The following events and activities may have affected Survey Area SEC01:

- Wet and dry depositions from Unit 3 stack discharges

Translocation pathways within SEC01 include the following:

- Activity from the deposition from stack releases settling onto the building surfaces (more so on the roofs than the remainder of the surfaces)

A characterization was performed September 2008 consisting of 31 fixed-point measurements. Of the 31 measurements taken, 14 exceeded the instrument's MDA and are listed in Table 2-18.

Table 2-18 provides the results of the 2008 characterization effort.

Table 2-18 SEC01 Characterization Data

Location	dpm/100cm2	Location	dpm/100cm2
Roof	328	Interior	483
Roof	224	Interior	292
Roof	213	Exterior Walls	495
Roof	516	Exterior Walls	306
Interior	296	Exterior Walls	246
Interior	253	Exterior Walls	355
Interior	260	Exterior Walls	299

SEC01 is classified as a Class 3 Survey Area.

2.1.8.17 MOB01 Main Office Building

Survey Area MOB01 consists of the Main Office Building. Survey Area MOB01 footprint contains about 409 m^2 of surface area. The Main Office Building is located across from the Office Annex.

The original Main Office Building was built during Unit 1 plant construction. An addition was added to the structure in the 1970s and remodeling has been performed on the structure. Currently, the structure is used for offices, tool room, and an electrical shop.

MOB01 is bounded by OOL08 on all sides with the exception of a small area on the east side, which is bounded by OOL03.

The following events and activities may have affected Survey Area MOB01:

- Wet and dry depositions from Unit 3 stack discharges

Translocation pathways within MOB01 include the following:

- Activity from the deposition from stack releases settling onto the building surfaces (more so on the roofs than the remainder of the surfaces)

A characterization was performed September 2008 consisting of 120 fixed-point measurements. Of the 120 measurements taken, 47 exceeded the instrument's MDA and are listed in Table 2-19.

Table 2-19 provides the results of the 2008 characterization effort.

Table 2-19 MOB01 Characterization Data

Location	dpm/100cm2	Location	dpm/100cm2
Walls/Ceilings	331	Roof	335
Walls/Ceilings	255	Roof	281
Floor	221	Roof	249
Floor	347	Roof	415
Roof	263	Roof	361
Roof	285	Roof	285
Roof	451	Roof	249

Location	dpm/100cm2	Location	dpm/100cm2
Roof	390	Roof	281
Roof	303	Roof	328
Roof	361	Roof	566
Roof	299	Roof	411
Roof	563	Roof	397
Roof	415	Roof	397
Roof	516	Roof	292
Roof	415	Roof	624
Roof	519	Roof	530
Roof	595	Roof	548
Roof	1025	Roof	631
Roof	1021	Roof	967
Roof	1126	Roof	877
Roof	224	Roof	628
Roof	696	Roof	548
Roof	520	Roof	404
Roof	628		

MOB01 is classified as a Class 3 Survey Area.

2.1.8.18 CRB01 Count Room Building

Survey Area CRB01 consists of the count room building. Survey Area CRB01 footprint contains about 372 m^2 of surface area. The count room building is located in the southwest corner of the site.

The original count room building was built during 2010. Currently, the structure is used for preparing and counting samples and houses the whole-body counter.

CRB01 is bounded by OOL10 on all sides.

The following events and activities may have affected Survey Area CRB01:

- Sample preparation activities potentially impact the survey area

Characterization data is unavailable for this survey area and will be scheduled prior to the survey package development.

Since no events have been identified that would have affected this survey area, it has been classified as a Class 1 area.

2.1.9 HSA Findings

2.1.9.1 Potential Contaminates

The primary contaminants of concern for the HBPP site are Fe-55, Co-60, Cs-134, Cs-137, Ni-63, Pu-238/241, and Am-241. Since the plant has been in cold shutdown and SAFSTOR since 1976, the more abundant activation and fission products, Fe-55 and Co-60, have decayed to 0.1 percent and 1.6 percent, respectively, of their total activity because of their short half-life. This has led to Cs-137 and Ni-63 as the most abundant radionuclides in the HBPP inventory. Personnel at HBPP have seen an increase in Am-241 since the shutdown of Unit 3. The increase is most likely from the beta decay of Pu-241 to Am-241. The radionuclide inventory performed in 1981 did not include analysis for Pu-241, possibly due to detection limits. Plutonium-241 decays by a very weak beta at 20.8 KeV. It also decays by alpha emission to Np-237; however, this mode of decay has a relative abundance of less than 1 percent. No equilibrium point will be reached between Pu-241 and Am-241 because of their short to long half-lives, 14.4 years and 432.7 years, respectively. The increase of Am-241 should reach 90 percent of its maximum in approximately 48 years from the date of the last fuel cladding failure, which occurred in1965. This will occur around the year 2013 and the maximum should occur about 73 years after the last fuel failure (or 2038).

2.1.10 HSA Conclusions

Data from the HSA investigation suggest that the land and structures that may require remediation lie very near to the Unit 3 nuclear reactor. The Unit 3 reactor and buildings will require remediation before they are demolished to ensure the offsite dose limits delineated in the plant's Offsite Dose Calculation Manual (ODCM) are not exceeded and remediated such that FSS can be achieved. All materials above the DCGLs would be disposed at an NRC-licensed waste disposal facility.

The migration of surface and subsurface contamination appears to be limited to areas within proximity of Unit 3. The areas of concern for the HBGS facility and the ISFSI show little to no affect from operations at HBPP and the available data suggest that these areas do not require remediation.

All classifications are subject to change if new data become available.

2.2 Hydrogeological Investigations

2.2.1 Previous Reports and Studies

A substantial amount of subsurface investigative work has been done on the HBPP site beginning in the 1950s. Historical subsurface studies at the HBPP site have ranged in purpose and specific area and/or depth of interest. Types of exploration include borings for geotechnical, hydrogeologic, seismic, and stratigraphic investigations; shallow trenches for fault investigations; and installation of groundwater monitoring wells for contamination detection and monitoring. Numerous data-review documents and hydrogeologic studies have been produced for the HBPP site. The following subsurface studies and documents were considered most relevant available:

Bechtel Civil & Minerals, Inc. "Interoffice Memorandum, Humboldt Bay Power Plant Unit #3 Report of 1984 Geologic Activities." August 1984.

Bechtel's investigation consisted of the installation of 12 boreholes near Unit 3, 11 of which were subsequently constructed as monitor wells (MW-1 through MW-11). The purpose of Bechtel's study was "to provide input to geology, groundwater and seismology sections of an environmental report to be filed with PG&E's decommissioning permit request" and to collect data for use in "evaluating the direction and rate of possible contaminant migration." A flowmeter survey was conducted in five of the monitor wells (MW-2, MW-3, MW-4, MW-10, and MW-11) to assess groundwater flow direction and velocity. Five of the monitor wells installed by Bechtel (MW-1, MW-2, MW-4, MW-6, and MW-11) were recently redeveloped and play an active role in the current groundwater monitoring program.

PG&E Department of Engineering Research. Effects of Tides on Groundwater Flow at Humboldt Bay Power Plant. January 1987

This report presents the results of a groundwater flow analysis within the first and second water bearing zones near the Wastewater Impoundments (WWI) and the Oil/Water Separator (OWS). Although the WWI area is north of Unit 3 and outside the study area, the OWS is adjacent to the southern boundary of Unit 3. The study used pressure transducers and continuous data acquisition systems to track the influence of the tides on groundwater flow.

PG&E Technical and Ecological Services, Water Resources Unit. Humboldt Bay Power Plant Wastewater Treatment Impoundments Hydrogeologic Characterization Study. November 1988

This study characterizes groundwater flow within the area of the wastewater treatment impoundments east of the discharge canal. The study is a follow-up to the 1987 study mentioned previously and included an analysis of data acquired from 30 wells (piezometer and monitoring wells) installed at four different levels within the first and second water-bearing zones.

PG&E Geosciences. Technical Report TR-HBIP-2002-01, Seismic Hazard Assessment for the Humboldt Bay ISFSI Project, Revision 0." December 27, 2002

This technical report presents the results of a comprehensive review of both regional and local seismic hazards for the HBPP site. It includes a review of historical subsurface work and presents updated cross-sections and geologic mapping.

ENERCON Services, Inc. Humboldt Bay Power Plant Tritium Evaluation. December 2006

In this report, ENERCON presents the results of a review of the existing groundwater-monitoring program at HBPP, in which they identify sources of potential radiological release into the groundwater, identify potential migration pathways, and evaluate the existing monitor well network in terms of its effectiveness for detecting radiological releases. The report lays the groundwork for meeting the requirements set forth in the Nuclear Energy Institute (NEI) groundwater protection initiative. Recommendations included the installation of seven new monitoring wells. (SHN Consulting installed these seven monitoring wells in August 2008.)

PG&E. "DECON-POS-H011: Groundwater Investigation History, Control, and Management, Revision B." May 2009

This position paper, one of a series developed to aid in the decommissioning of Unit 3, outlines the issues, strategies, and costs for groundwater monitoring and control during decommissioning activities, particularly with respect to the planned removal of subgrade structures.

As previously noted, this strategy is good for the current condition and to assess offsite migration of contaminants, but not the final as left remediated site with a well network that will support the Resident Farmer Scenario and resident farmer well.

2.2.2 Current Groundwater Monitoring

Seven monitor wells, which are referred to as the "Intermediate Screened Wells," have screened intervals within an elevation range of approximately -24 to -40 feet North American Vertical Datum 1988 (NAVD88). The remaining five monitor wells are referred to as the "Deep Screened Wells," and have a screen interval at an elevation range of approximately -62 to -82 feet NAVD88. Unit 3 area monitor wells were originally sampled on a quarterly basis and, as part of modifications to the Radiological Environmental Monitoring Program (REMP), HBPP increased groundwater gauging events to monthly intervals in 2010. At this time, the shallow zone monitor wells installed by Arcadis in 2009 are not part of the REMP groundwater monitoring program (RGWMP), though several of these wells have been sampled and groundwater measurements are taken from selected wells during quarterly monitoring events. Table 2-20 provides the HBPP monitor well elevations and depths. Additional wells are slated to be installed to provide additional future monitoring to determine the impact of decommissioning activities on groundwater (e.g. within the slurry wall area).

Table 2-20 Monitoring Well Elevation and Depth

Well Location	Date Installed	Top of Casing Elevation (feet[1])	Screen Interval (BGS)[2]	Screen Interval (feet)[1,3]
MW-1	Jun-84	10.84	39.9 to 44.3	-28.91 to -33.31
MW-2	Jun-84	10.94	39.8 to 49.2	-28.9 to -38.3
MW-4	Jun-84	11.13	41.0 to 50.2	-29.71 to -38.91
MW-6	Jun-84	10.79	44.2 to 48.5	-33.54 to -37.84
MW-11	Jun-84	11.39	35.8 to 45.0	-23.98 to -33.18
RCW-SFP-	Aug-08	26.22	56.5 to 66.0	-30..01 to -
RCW-SFP-	Aug-08	32.40	57.1 to 66.6	-28.29 to -37.79
RCW-CS-1	Aug-08	10.50	73.0 to 82.5	-62.27 to -71.77
RCW-CS-2	Aug-08	10.62	73.5 to 83.0	-62.70 to -72.20
RCW-CS-3	Aug-08	10.91	73.5 to 83.0	-62.44 to -71.94
RCW-CS-4	Aug-08	10.90	83.5 to 93.0	-72.38 to -81.88
RCW-CS-5	Aug-08	10.92	84.0 to 93.5	-72.84 to -82.34
5G-MW-03	Jan-09	23.90	20.0 to 30.0	3.90 to -6.10
1C-MW-06	Jan-09	10.28	15.0 to 25.0	-4.72 to -14.72
1C-MW-07	Jan-09	10.36	15.0 to 25.0	-4.64 to -14.64
1C-MW-08	Jan-09	10.69	15.0 to 25.0	-4.31 to -14.31

Well Location	Date Installed	Top of Casing Elevation (feet[1])	Screen Interval (BGS)[2]	Screen Interval (feet)[1,3]
1E-MW-12	Jan-09	10.42	15.0 to 25.0	-4.58 to -14.58
1E-MW-13	Jan-09	11.39	15.0 to 25.0	-3.61 to -13.61

Referenced to NAVD88 (North American Vertical Datum 1988)

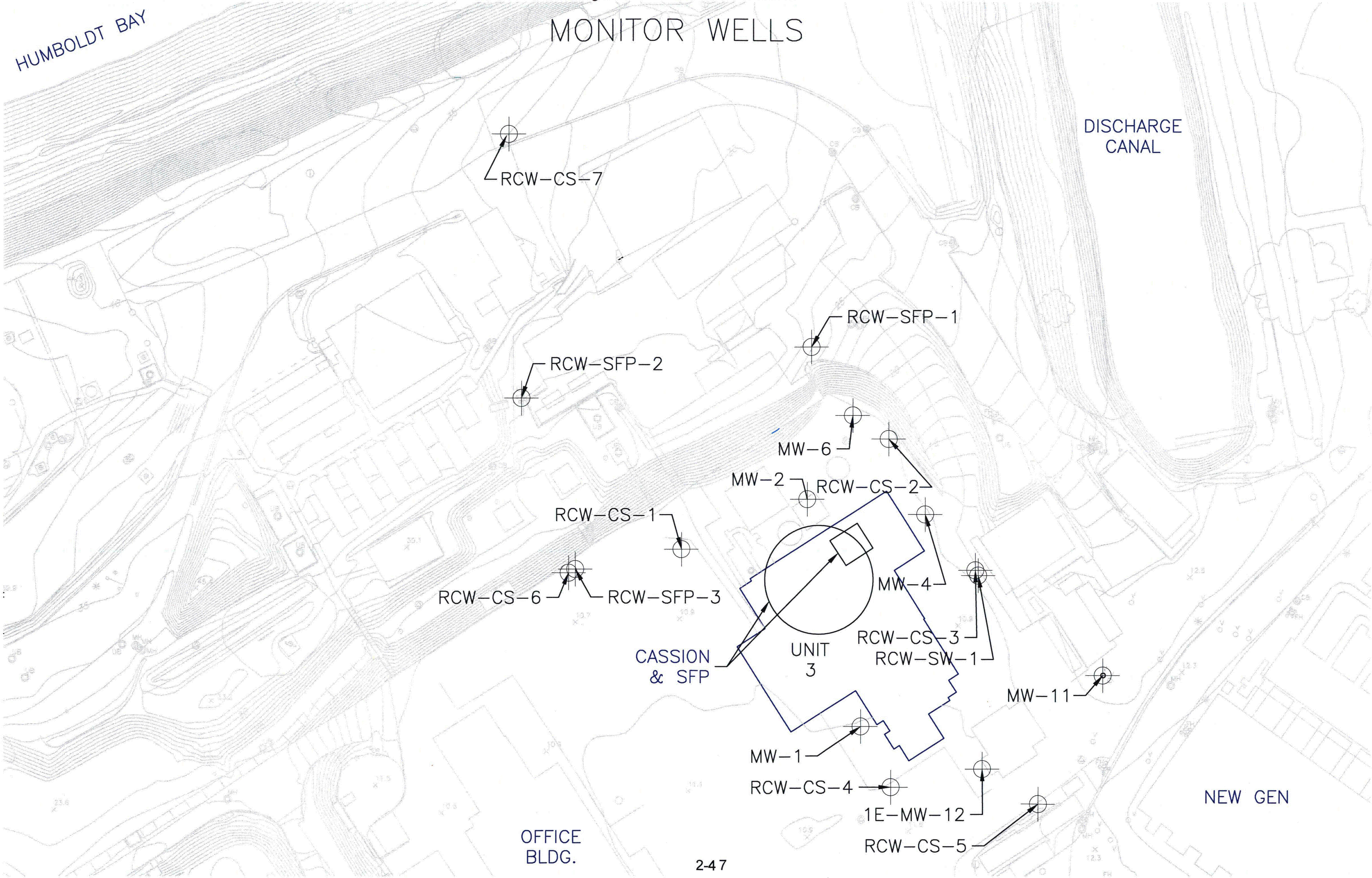
BGS: Below Ground Surface

Well screen depth adjusted to top of casing elevation

2.2.3 Groundwater Monitoring Results

Appendix 2-C provides the results of groundwater monitoring for 2009 and 2010.

Figure 2-3 HBPP Well Locations



2.3 Site Characterization Surveys

2.3.1 1997 Characterization Survey

The objective of this radiological survey was to assess the nature, degree, and extent of radiological contamination in sediments and shallow soils at HBPP. The primary purpose of the survey was to provide a decision-making basis for developing remediation requirements and cost estimates leading to the future decommissioning of Unit 3. Additional objectives of the site characterization survey included the following:

- Identifying areas not affected by HBPP operations and in which radioactivity is indistinguishable from background
- Confirming and updating survey unit classifications
- Providing a basis for development of data quality objectives for the final survey
- Obtaining data that may be used in the final site survey

The scope of the investigation included sampling of sediments and surface and shallow subsurface soils to a nominal depth of 4 feet below ground surface.

2.3.1.1 Methodology

The radiological survey of shallow soils and sediment at HBPP was conducted using a graded approach that assumed all areas of the plant were either Class 1, 2, or 3. Prior to the sampling, a preliminary classification of sediments and soils at HBPP was performed, based on the facility layout, operational history, interviews with PG&E staff, and information presented in *Residual Radionuclide Distribution and Inventory at the Humboldt Bay Nuclear Power Plant* (Battelle, 1983). Environmental media were initially divided into six survey units.

Survey unit 1, the Unit 3 RA comprised surface and shallow subsurface soils hypothesized to be affected by release of liquids to the land surface and by aerial deposition from the Unit 3 stack. Survey unit 2, the relatively undisturbed upland soils in outlying areas, was hypothesized to be affected only by aerial deposition from the stack. Survey unit 3, the relatively undisturbed low-lying or wetland-type soils in outlying areas, was hypothesized to be similarly affected by emissions from the stack, but was also hypothesized to be affected by contaminated sediments transported from the central portions of the plant during rainfall events. Survey unit 4, the disturbed soils surrounding Units 1 through 3, was hypothesized to be affected by stack emissions, as well as earthmoving activities that may have resulted in the relocation or shallow burial of contaminated soil. Survey unit 5, the sediments of Humboldt Bay, was hypothesized to be affected by stack emissions and sediment transported from the plant. Survey unit 6, the sloughs, canals, and ditches of HBPP, was hypothesized to be affected by stack emissions, liquid releases, or transport of sediments from the plant.

The minimum number of samples expected to be required to meet the statistical requirements of the final survey were estimated using the approach documented in MARSSIM. This approach includes the use of nonparametric statistical methods including the Wilcoxon Rank Sum test and Sign test. For survey units subjected to random or systematic sampling, a decision rule was developed as follows: "If the mean concentration of the survey unit adjusted to account for background radiation exceeds the investigation levels, then the survey unit is assumed to require remediation." It was assumed that this decision rule would be tested using the Wilcoxon Rank Sum test or Sign test. Next, a relative shift of 1.6 was estimated by assuming that the Lower Bound of the Gray Region was 50 percent of the investigation level, and the relative standard deviation of survey results was 30 percent for all radionuclides. Using a relative shift of 1.6 and a decision error rate of 5 percent for alpha and beta type errors, the estimated numbers of samples required to perform the Wilcoxon Rank Sum and Sign tests were 16 and 17, respectively (Tables 5.3 and 5.5 in MARSSIM). As described in MARSSIM, these numbers included a 20 percent contingency to account for unusable sample results. The number of samples was further increased to 30 in most survey units to account for the following uncertainties:

- Site-specific standard deviations were unavailable.
- Site-specific cleanup standards had not been agreed upon at that time.

- Acceptable decision error rates for the final survey had not been agreed upon at that time

2.3.1.2 *Survey Instrumentation*

Survey instrumentation was selected to ensure that sensitivities were sufficient to detect the expected radionuclides at the minimum detection requirements. A list of the survey instrumentation, radiations detected, and calibration sources is provided in Table 2-21.

Table 2-21 Survey Instrumentation

Instrument/ Detector	Detector Type	Radiation Detected	Calibration Source	Use
Eberline ESP-1/SPA-3	NaI Detector (gamma scintillator)	Gamma	Cesium-137	Qualitative Soil Contamination Measurement
TSA Large Area Detector	Plastic Scintillator	Beta/Gamma	Cesium-137	Qualitative Soil Contamination Measurement
EG&G Ortec NOMAD Gamma Spectroscopy System	High Purity Germanium (HPGe)	Gamma	Mixed Gamma Standard	Radionuclide identification and quantification

Soil and sediment samples were analyzed onsite using gamma spectroscopy. The samples were collected, prepared, and analyzed in accordance with the Sample Analysis and Data Management Plan (GTS Duratek, 1997) and approved procedures. Once analyzed, the samples were archived and turned over to HBPP personnel for storage pursuant to sample chain-of-custody procedure, unless the samples were shipped offsite for further analyses. A total of 706 samples were analyzed onsite, not including sample splits and duplicates.

2.3.2 2008 Characterization Survey

The purpose of the 2008 HBPP Characterization activities was to assess the radiological status of the HBPP site in accordance with MARSSIM guidance. The characterization activities were guided by HBPP-PP-003, "Site Characterization Plan," which used the MARSSIM Data Quality Objective (DQO) process to establish the necessary requirements and methods for obtaining high quality characterization data. The scope of this survey was as follows:

- Identify and quantify the nature and extent of radiological materials
- Determine the distribution of radioactive material contamination in each area that contained radioactive materials contamination
- Obtain data to provide guidance for decontamination/remediation activities planning
- Obtain data to provide guidance for waste management planning
- Provide information to support the development of the site-specific DCGLs
- Provided the information needed to develop the FSS for each survey area

2.3.2.1 *Methodology*

The survey package development involved performing walk-downs of each area. During the walk-down, details regarding the physical survey area were compiled in the survey package, such as type of area (structure, system, or environ), surfaces in the area (wall, floor, ceiling, surface soil, or other feature), and dimensions. Data from previous HBPP characterization and scoping surveys were reviewed and used as appropriate. Each survey package contained the following eight sections of information:

1. Detailed description of the survey area and/or survey units
2. Photographs, drawing, or drawings of the survey area and/or survey units
3. Survey area operational history including summary data from previous surveys
4. Characterization survey instructions—types and number of survey measurements and/or samples prescribed for the survey
5. Survey support requirements such as shovels, scoops, ladders, GPS, and coring tools
6. Health and safety requirements
7. Radiation Work Permit (RWP) requirements
8. Characterization Data—survey instrument data downloads, survey reports, and sample analysis reports

For each survey area in a designated survey unit, ENERCON used 30 samples/measurements as a standard number of characterization survey locations for all areas that were designated as MARSSIM Class 2 or Class 3 at HBPP.

The basis for using this number is that the quantity is an important standard number in large population statistics and was used historically in NRC guidance, NUREG/draft 5849, "Manual for Conducting Surveys in Support of License Termination." NUREG 5849 states that for survey areas with a low potential for residual contamination, 30 random measurements should be collected to identify the condition of an area within a 95 percent confidence level. Regarding the statistical number of locations using MARSSIM, the number 30 corresponds to designing a survey using the more conservative relative shift of 1, which typically provides a sample number requirement of 29. These statistical methods are only able to work if all the surfaces in the survey area have the same potential for having residual contamination (e.g., walls, floors, horizontal pipes/beams, ceilings). For Class 1 areas such as the soils around Unit 3, a more direct bounding survey was used since the data would not be used for final status surveys, given that the area would most likely require remediation.

2.3.2.2 *Instrumentation*

Radiological survey instrumentation was selected to ensure that sensitivities were sufficient to detect the expected radionuclides at the minimum detection requirements. A list of the survey instrumentation, radiations detected, and calibration sources is provided in Table 2-22.

Table 2-22 2008 Characterization Instrumentation

Instrument/ Detector	Detector Type	Radiation Detected	Calibration Source	Use
Ludlum 2221 (2"X 2")	NaI Detector (gamma scintillator)	Gamma	Am--241	Qualitative Soil Contamination Measurement
Ludlum Model 2360 (126 cm2 area)	Gas flow proportional	Alpha/Beta/ Gamma	Th-230/Tc-99	Surface static/scan measurements
Ludlum Model 19	NaI	Gamma	Cs-137	Area exposure measurements
Ludlum Model 2929	ZnS	Alpha/beta/ Gamma	Pu-239/Tc-99/I-131/C-14	Swipe/smear counting

2.4 Continuing Characterization

Characterization data will be collected as necessary throughout the project. Results of future characterization sample analysis will be evaluated to determine the impact, if any, on the radionuclide identities, nuclide fractions, and the classification of structures, soils and other site media

2.5 Summary

The characterization data collected and analyzed to date are of sufficient quantity and quality to provide the basis for the initial classification of survey areas, planning remedial activities, estimating radiological waste types and volumes, and for the development of the DCGLs. However, characterization is an ongoing process that will continue as necessary during decommissioning.

2.6 References

2-1 NUREG-1575, “Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM),” August 2000

2-2 Humboldt Bay, “Historical Site Assessment 2011 Update,” July 2011

2-3 HBPP Plant Operation Reports

2-4 HBPP Plant Operating Logs

2-5 IT Corporation, "Calculation of Preliminary Soil Cleanup Guidelines for Residual Radionuclides at the PG&E Humboldt Bay Power Plant, Eureka, California" Rev. 0, March 1998.

2-6 Enercon Services, Inc., “Radiological Characterization Report, Humboldt Bay Power Plant, Eureka, California,” HBPP-RPT-001, Rev. 1, November 21, 2008.

2-7 10 CFR 50.75, “Reporting and recordkeeping for decommissioning planning”

2-8 HBPP DSAR, “Defueled Safety Analysis report for the Humboldt bay power plant unit 3,” Revision 9-19, 8/17/2011

2-9 HBPP PSDAR “Post-Shutdown Decommissioning Activities Report,” Revision 3

2-10 *“Residual Radionuclide Distribution and Inventory at the Humboldt Bay Nuclear Power Plant" Battelle, 1983*

2-11 “Sample Analysis and Data Management Plan” GTS Duratek, 1997

2-12 NUREG/draft 5849, "Manual for Conducting Surveys in Support of License Termination."

Appendix 2-A

Historical Site Photographs

HBPP Site Pre-Construction

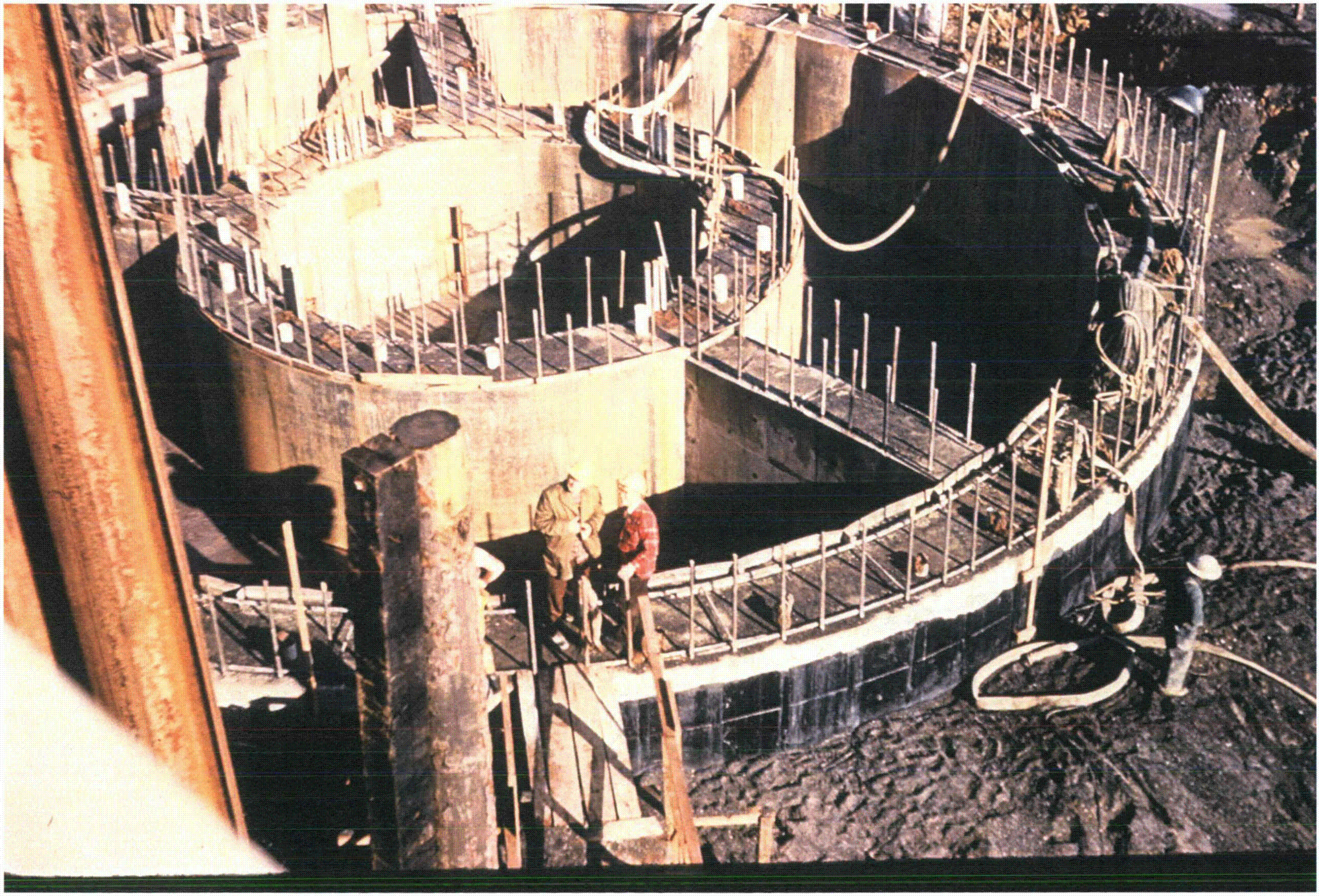
HBPP Unit 3 Excavation

Driving Pilings

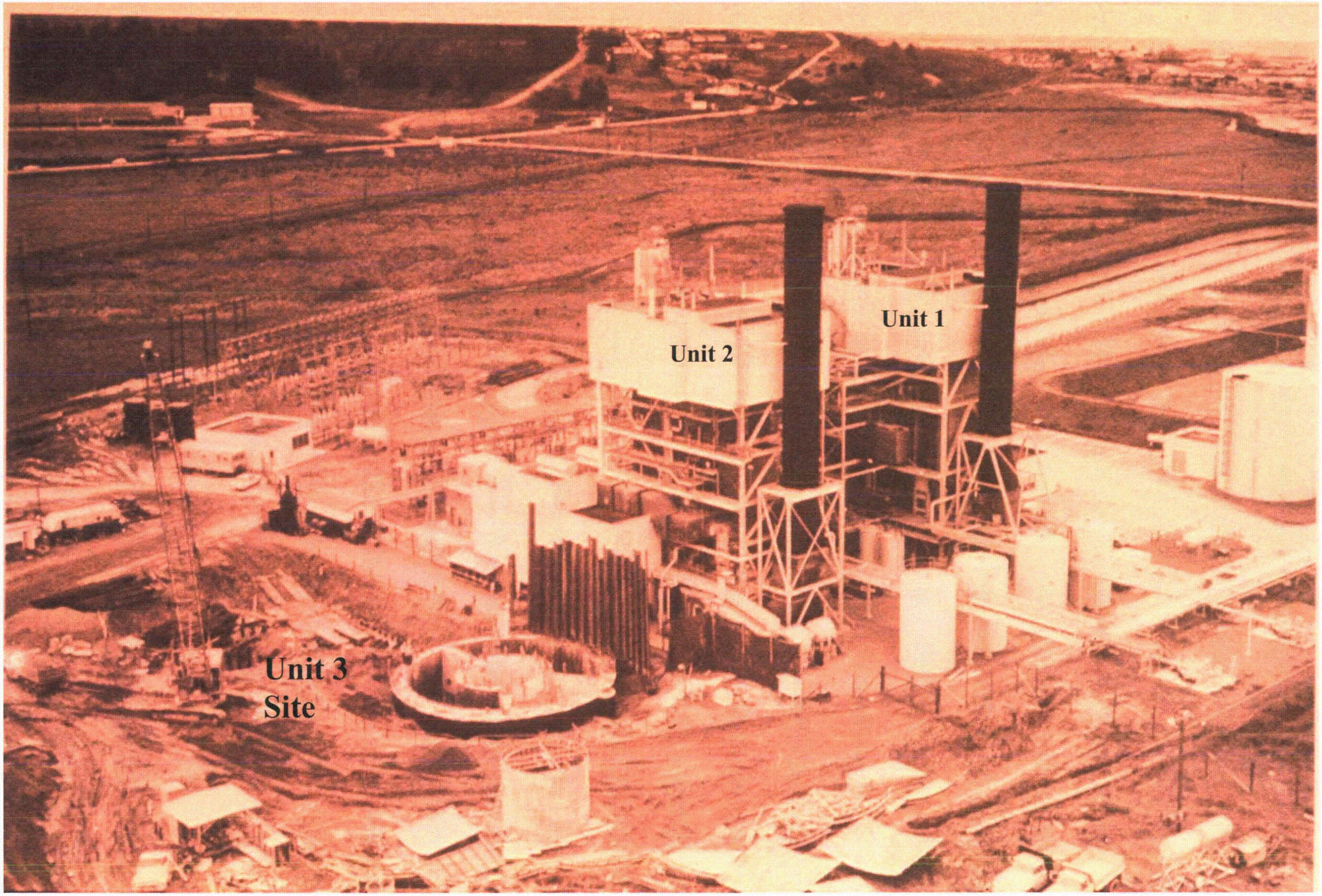
Reactor Drywell in Place

SFP Excavation

Inner and Outer Caisson

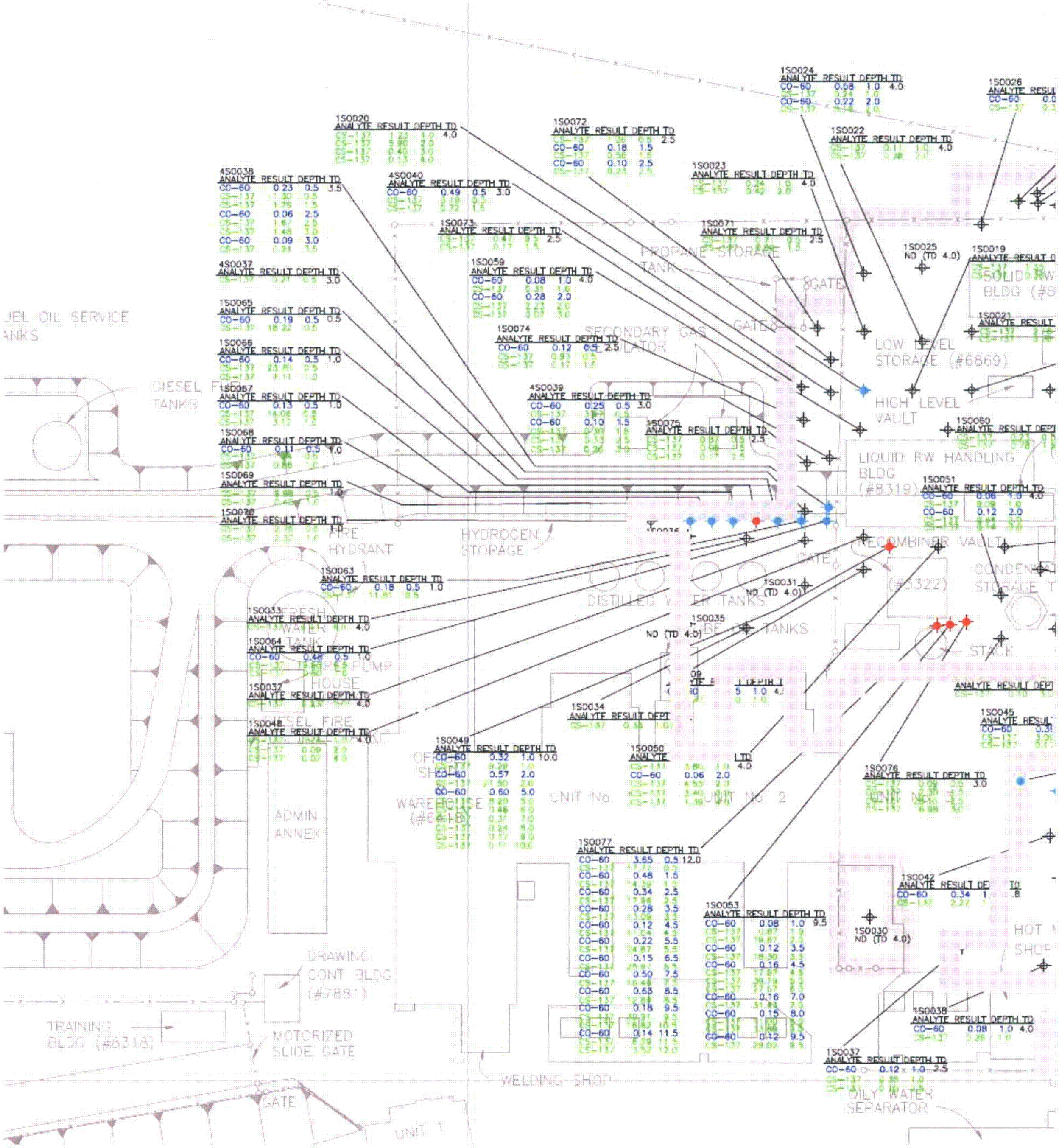


Unit 3 during Construction



Appendix 2-B
Characterization Sample Location

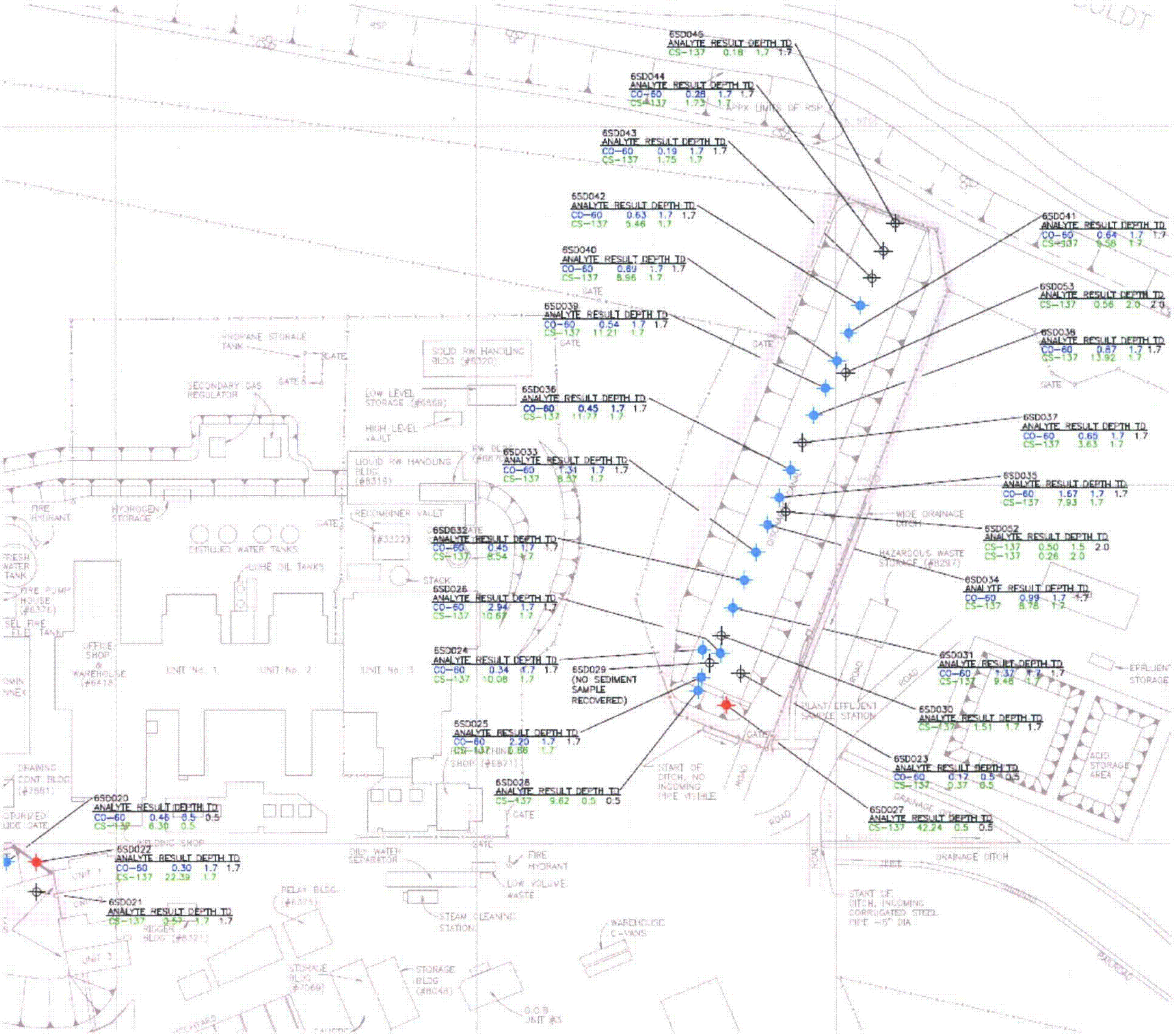
RA West and North Yard



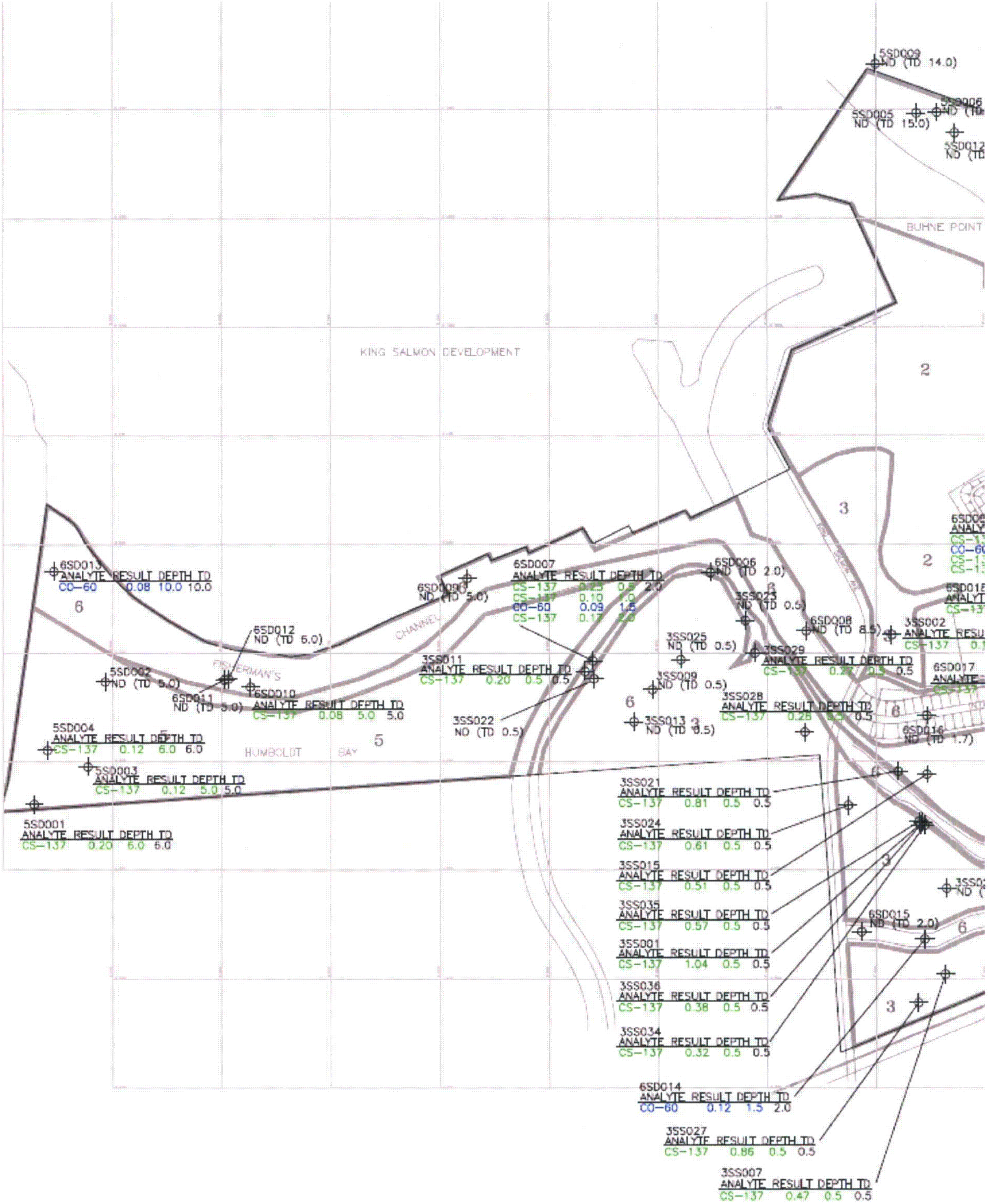
RA East


WIDE DRAINAGE DITCH
HAZARDOUS WASTE STORAGE (#8297)
PLANT EFFLUENT SAMPLE STATION
START OF DITCH, NO INCOMING PIPE VISIBLE
DRAINAGE DITCH
ROAD
GATE
HOT MACHINE SHOP (#6871)
CONDENSATE STORAGE TANK
STACK
HYDRANT

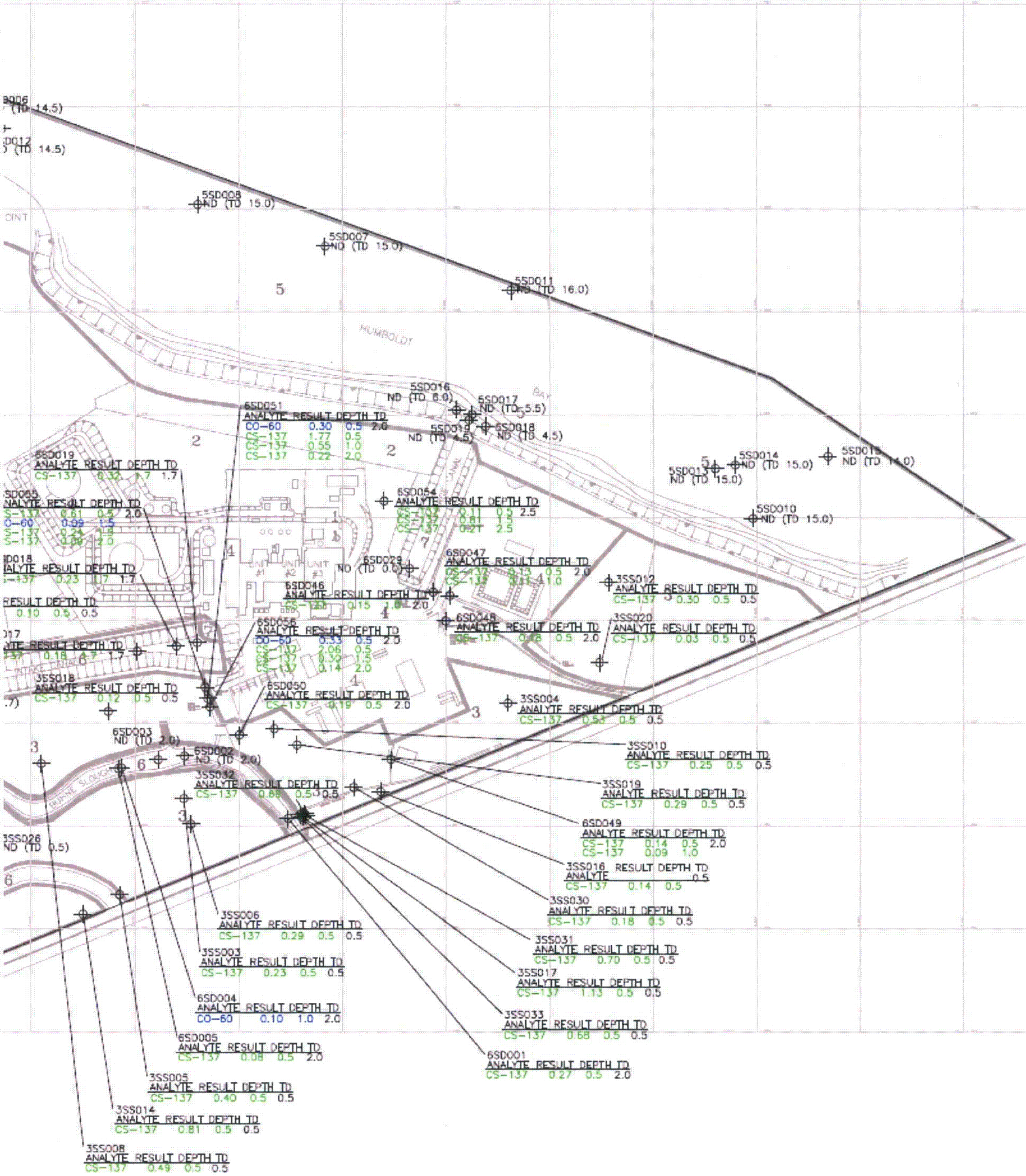
Intake and Discharge Canals



West Side of Site



East Side of Site



Appendix 2-C
HBPP Groundwater Monitoring Summary

HBPP Groundwater Monitoring Summary

Well/Quarter	Gross Beta	H-3	Sr-90	Gross Alpha	Am-241	Co-60	Cs-137
1C-MW-07/ 2nd 2009	13.4 pCi/L	<MDC	<MDC	10.5 pCi/L	<MDC	<MDC	<MDC
RCW-CS-1/ 2nd 2009	19.4 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1EW-MW-12/ 2nd 2009	5.59 pCi/L	<MDC	<MDC	4.82 pCi/L	<MDC	<MDC	<MDC
5G-MW-03/ 2nd 2009	3.59 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-5/ 2nd 2009	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-1/ 2nd 2009	5.19 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1C-MW-08/ 2nd 2009	5.32 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-4/ 2nd 2009	5.92 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-2/ 2nd 2009	3.60 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-6/ 2nd 2009	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-2/ 2nd 2009	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-4/ 2nd 2009	6.93 pCi/L	<MDC	0.59 pCi/L	<MDC	<MDC	<MDC	<MDC
MW-11/ 2nd 2009	6.00 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-3/ 2nd 2009	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1E-MW-13/ 2nd 2009	2.67 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
SFP-1/ 2nd 2009	4.30 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-SFP-2/ 2nd 2009	4.83 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1H-MW-02/ 2nd 2009	N/A	<MDC	N/A	N/A	N/A	N/A	N/A
RCW-CS-2/ 3rd 2009	11.0 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-4/ 3rd 2009	9.19 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-1/ 3rd 2009	12.6 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
SFP-1/ 3rd 2009	5.78 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-11/ 3rd 2009	8.88 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1E-MW-13/ 3rd 2009	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-SFP-2/ 3rd 2009	4.86 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC

Well/Quarter	Gross Beta	H-3	Sr-90	Gross Alpha	Am-241	Co-60	Cs-137
1C-MCW-8/ 3rd 2009	5.57 pCI/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-1/ 3rd 2009	28.1 pCi/L	<MDC	<MDC	80.6 pCi/L	<MDC	<MDC	<MDC
1C-MW-07/ 3rd 2009	13.7 pCi/L	<MDC	<MDC	7.09 pCi/L	<MDC	<MDC	<MDC
5G-MW-03/ 3rd 2009	4.19 pCi/L	<MDC	<MDC	2.87 pCi/L	<MDC	<MDC	<MDC
RCW-CS-5/ 3rd 2009	3.55 pCi/L	<MDC	<MDC	2.62 pCi/L	<MDC	<MDC	<MDC
1E-MW-12/ 3rd 2009	15.5 pCi/L	<MDC	<MDC	11.1 pCi/L	<MDC	<MDC	<MDC
RCW-CS-3/ 3rd 2009	3.63 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-4/ 3rd 2009	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-2/ 3rd 2009	6.83 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-6/ 3rd 2009	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-2/ 4th 2009	7.19 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-4/ 4th 2009	7.25 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-1/ 4th 2009	8.30 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-SFP-1/ 4th 2009	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-11/ 4th 2009	6.99 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1E-MW-13/ 4th 2009	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-SFP-2/ 4th 2009	41.4 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1C-MCW-08/ 4th 2009	3.50 pCi/L	<MDC	<MDC	2.22 pCi/L	<MDC	<MDC	<MDC
RCW-CS-1/ 4th 2009	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-2/ 4th 2009	4.08 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-6/ 4th 2009	3.19 pCi/L	952 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-5/ 4th 2009	6.73 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1E-MW-12/ 4th 2009	14.3 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-3/ 4th 2009	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-4/ 4th 2009	5.24 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-1/ 1st 2010	4.54 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC

Well/Quarter	Gross Beta	H-3	Sr-90	Gross Alpha	Am-241	Co-60	Cs-137
MW-2/ 1st 2010	7.64 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-4/ 1st 2010	13.2 pCi/L	<MDC	<MDC	8.35 pCi/L	<MDC	<MDC	<MDC
MW-6/ 1st 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-11/ 1st 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-1/ 1st 2010	9.02 pCi/L	<MDC	<MDC	29.7 pCi/L	<MDC	<MDC	<MDC
RCW-CS-2/ 1st 2010	10.2 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-3/ 1st 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-4/ 1st 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-5/ 1st 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-SFP-1/ 1st 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1E-MW-12/ 1st 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1E-MW-13/ 1st 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-SFP-2/ 1st 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1C-MW-08/ 1st 2010	2.82 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1C-MW-07/ 1st 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-1/ 2nd 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-2/ 2nd 2010	3.85 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-4/ 2nd 2010	12.6 pCi/L	N/A	<MDC	9.19 pCi/L	<MDC	<MDC	<MDC
MW-6/ 2nd 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-11/ 2nd 2010	7.09 pCi/L	N/A	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-1/ 2nd 2010	17.8 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-2/ 2nd 2010	31.4 pCi/L	N/A	<MDC	27.4 pCi/L	<MDC	<MDC	<MDC
RCW-CS-3/ 2nd 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-4/ 2nd 2010	5.19 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-5/ 2nd 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-SFP-1/ 2nd 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC

Well/Quarter	Gross Beta	H-3	Sr-90	Gross Alpha	Am-241	Co-60	Cs-137
1E-MW-12/ 2nd 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1E-MW-13/ 2nd 2010	2.98 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-SFP-2/ 2nd 2010	<MDC	N/A	<MDC	<MDC	<MDC	<MDC	<MDC
1C-MW-08/ 2nd 2010	2.87 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1C-MW-07/ 2nd 2010	2.97 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-1/ 3rd 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-2/ 3rd 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-4/ 3rd 2010	6.41 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-6/ 3rd 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-11/ 3rd 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-1/ 3rd 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-2/ 3rd 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-3/ 3rd 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-4/ 3rd 2010	10.8 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-5/ 3rd 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-SFP-1/ 3rd 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-SFP-2/ 3rd 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1C-MW-07/ 3rd 2010	7.32 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1C-MW-08/ 3rd 2010	8.65 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1E-MW-12/ 3rd 2010	86.0 pCi/L	<MDC	<MDC	103 pCi/L	<MDC	<MDC	<MDC
1E-MW-13/ 3rd 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-1/ 4th 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-2/ 4th 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-4/ 4th 2010	3.92 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-6/ 4th 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
MW-11/ 4th 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC

Well/Quarter	Gross Beta	H-3	Sr-90	Gross Alpha	Am-241	Co-60	Cs-137
RCW-CS-1/ 4th 2010	18.6 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-2/ 4th 2010	17.2 pCi/L	<MDC	<MDC	4.85 pCi/L	<MDC	<MDC	<MDC
RCW-CS-3/ 4th 2010	5.13 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-4/ 4th 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-CS-5/ 4th 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-SFP-1/ 4th 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
RCW-SFP-2/ 4th 2010	6.33 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1C-MW-07/ 4th 2010	3.28 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1C-MW-08/ 4th 2010	11.5 pCi/L	<MDC	<MDC	8.71 pCi/L	<MDC	<MDC	<MDC
1E-MW-12/ 4th 2010	9.32 pCi/L	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC
1E-MW-13/ 4th 2010	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC	<MDC

CONTENTS

TABLES

3 IDENTIFICATION OF REMAINING DECOMMISSIONING ACTIVITIES

3.1 Introduction

In accordance with 10 CFR 50.82 (a)(9)(ii)(B), the License Termination Plan (LTP) must identify the major remaining dismantlement and decontamination activities. This chapter was written following the guidance of NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," (Reference 3-1) and Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," (Reference 3-2) and will discuss those remaining dismantlement activities as of April 12, 2013. Information is presented to demonstrate that these activities will be performed in accordance with 10 CFR 50 and will not be detrimental to the common defense and security or to the health and safety of the public pursuant to 10 CFR 50.82(a) (10). Information that demonstrates that these activities will not have a significant effect on the quality of the environment is provided in LTP Chapter 8, Supplement to the Environmental Report.

The information includes those areas and equipment in need of further remediation, and an estimate of radiological conditions that may be encountered. Included are estimates of associated occupational radiation dose and projected volumes of radioactive waste. Humboldt Bay Power Plant (HBPP's) primary goals are to decommission HBPP safely and successfully terminate the HBPP license. HBPP will decontaminate and dismantle HBPP in accordance with the DECON alternative, as described in NUREG-0586, "Final Generic Environmental Impact Statement" (GEIS) (Reference 3-3). Completion of the DECON option is contingent upon access to one or more low-level waste (LLW) disposal sites. Currently, HBPP has access to the disposal facilities in Utah, Texas, and Idaho.

HBPP is currently conducting decontamination and dismantlement (D&D) activities at the HBPP site in accordance with HBPP procedures and approved work packages. Decommissioning activities are being coordinated with the appropriate federal and state regulatory agencies.

Decommissioning activities at HBPP are conducted in accordance with the HBPP PSDAR, Radiation Protection Program, written work plans, existing 10 CFR Part 50 license, and the requirements of 10 CFR 50.82(a)(6) and (a)(7). If an activity requires prior NRC approval under 10 CFR 50.59(c)(2) or a change to the HBPP Technical Specifications or license, a submittal will be made to the NRC for review and approval prior to implementation of the activity in question.

The activities listed in Section 3.3, "Future Decommissioning Activities," include activities up to future release of the site. This section provides an overview of the major remaining decommissioning activities.

Information related to the remaining D&D tasks is provided in section 3.4. This information includes an estimate of the quantity of radioactive material to be disposed in accordance with 10 CFR 20.2001, a description of proposed control mechanisms to ensure areas are not recontaminated, estimates of occupational exposures, and characterization of radiological conditions to be encountered and the types and quantities of radioactive waste. This information supports the assessment of impacts considered in other sections of the LTP and provides sufficient detail to identify inspections or technical resources needed during the remaining dismantlement activities. Many of these dismantlement tasks require coordination with other federal, state, or local regulatory agencies or groups.

The dismantlement activities described in Section 3.3 provide the NRC the information to support site release and future license termination pursuant to 10 CFR 50.82(a)(11)(i). Therefore, this section was written in order to indicate clearly each major dismantlement activity that remains to be completed prior to qualifying for license termination. The final state of the HBPP site will be an electrical production facility for approximately 30 years (as defined in Chapter 1). The impact of decommissioning activities performed will be to reduce residual radioactivity to a level of 25 mrem/year and as low as reasonably achievable (ALARA) from all potential pathways to the average member of the critical group (Residential Farmer).

3.2 Completed Decommissioning Activities and Tasks

3.2.1 *Spent Fuel Storage*

The Humboldt Bay Independent Spent Fuel Storage Installation (ISFSI) was loaded with five Hi-Star HB casks between August 2008 and December 2008 containing all the spent nuclear fuel stored onsite at the HBPP, licensed under a 10 CFR 72 Site Specific License. This removed all spent fuel assemblies from the spent fuel pool. A sixth Hi-Star cask constructed for use in storing the Greater Than Class C (GTCC) waste that includes items from dismantlement activities of the reactor vessel has been placed in the ISFSI. The casks will eventually be transferred to a national permanent repository.

3.2.2 *Spent Fuel Pool Activities*

The spent fuel racks have been removed.

3.2.3 *Reactor Building*

The drywell shield plugs as well as the drywell head have been removed.

Reactor and drywell heads have been shipped

The reactor internals have been removed.

3.2.4 *Liquid Radwaste (LRW) Building*

The evaporator and miscellaneous tanks have been removed.

3.2.5 *Turbine Building*

The following activities have been completed in the Turbine Building:

- Main Turbine removal
- Condenser removal
- Steam, feedwater, and seal oil piping removal
- The building has been demolished and the concrete removed. Subgrade structures still remain

3.2.6 *Miscellaneous Structures*

The following miscellaneous structures activities have been completed:

- Aboveground portions of fossil units 1 and 2 have been demolished and removed
- Mobile Emergency Power Plant Stations (MEPPs) 1 and 2 have been removed from site
- All fuel oil tanks associated with Units 1 and 2 have been demolished and removed
- A section of the circulation water piping has been removed
- A majority of the ventilation stack has been demolished and removed

3.3 Future Decommissioning Activities

3.3.1 *Remaining Component Removal*

The following table lists the remaining major activities associated with the decommissioning of HBPP and their projected completion date:

Table 3-1 Major Remaining Activities and Completion Dates

Activity	Projected Completion Date
Reactor vessel removal	Late 2014
Reactor Building above grade removal	Late 2015
Spent Fuel Pool removal	Mid 2016
Caisson removal	Early2017
LRW Building removal	Late 2018
Waste Buildings and vaults removal	Late2014
Slurry Wall installation	Early 2015
Intake Canal dredging/remediation	Mid 2015
Discharge Canal dredging/ remediation	Mid 2015
Site restoration	Late 2018
FSS activities	Late 2019

3.3.2 *Control Mechanisms to Ensure No Recontamination*

Due to the large scope of remaining structures and systems to be decontaminated and the need for some FSS activities to be performed in parallel with dismantlement activities, a systematic approach to controlling areas is established. Upon commencement of the FSS for survey areas where there is a potential for recontamination, implementation of one or more of the following control measures will be implemented:

- Personnel training
- Installation of barriers to control access to surveyed areas
- Installation of barriers to prevent the migration of contamination from adjacent areas
- Installation of postings requiring personnel to perform contamination monitoring prior to surveyed area access
- Locking entrances to surveyed areas of the facility
- Installation of tamper-evident labels or seals
- Upon completion of FSS, the area will be placed under periodic routine surveillance survey by the FSS department to ensure no recontamination occurs. If recontamination is identified, an investigation will be initiated that could result in corrective actions up to and including reperformance of the FSS for that area.

3.4 Occupational Exposure

Table 3-2 provides HBPP cumulative site dose and estimates for the decommissioning project. These estimates were developed to provide site management ALARA goals. The goals are verified by summation of actual site dose, as determined by appropriate dosimetry. Exposure estimates are a compilation of radiation work permit estimates for the period. The total nuclear worker exposure during decommissioning is currently estimated to be less than 86 person-rem. This estimate is significantly below the 1,874 person-rem estimate of the GEIS for immediate dismantlement and below the ten-year SAFSTOR estimate 834 person-rem.

Table 3-2 HBPP Cumulative Site Dose

Year	Exposure (person-rem)
2009	0.6
2010	7.7
2011	6.7
2012	15.9
2013	24.1
2014-2018	30.0*

*estimated exposure

3.4.1 *Public Exposure*

Continued application of HBPP's current and future Radiation Protection and Radiological Effluent Programs ensures public protection in accordance with 10 CFR Part 20 and 10 CFR Part 50, Appendix I. Sections 3.4.4 and 3.4.5 conclude that the public exposure as a result of decommissioning activities is bounded by the evaluation in the GEIS, which concludes the impact is small.

3.4.2 *Estimate of Quantity of Radioactive Material to be Shipped for Disposal.*

HBPP has shipped for radioactive disposal approximately 6,016 cubic meters (m^3) (212,450 cubic feet [ft^3]) of waste through December 31, 2013. The estimate of remaining waste is 60,000 m^3 (2,118,900 ft^3), most of which is very low activity soils, sediments, and concrete debris. This volume of waste exceeds the NUREG-0586, volume for the reference boiling water reactor of 18,343 m^3 (647,777 ft^3). The additional waste generated is mainly due to the removal of the caisson and the removal of low-level sediments in the discharge canal. An environmental impact assessment due to the additional volume of waste generated is provided in Chapter 8 of the LTP.

3.4.3 *Solid Waste Activity and Volume*

HBPP's Annual Radioactive Effluent Release Report (Reference 3-4, 3-5 and 3-6), required by Section 3.7.3 of the Humboldt Bay Power Plant Unit 3 SAFSTOR Quality Assurance Plan, includes a report on solid waste activity and volumes. This report is submitted annually. A summary of solid waste disposal for 2009 through 2013, with an estimate for the remainder of the project is provided in Table 3-3. Future updates may be obtained from HBPP for inspection.

Table 3-3 Solid Waste Effluent Release Report Summary

Year	Volume (m^3)	Total Curies
2009	252.8	0.101
2010	1312	1.12
2011	654.73	3.661
2012	343	390
2013	3454	1727
2014-2019	60,000*	1000*

* Estimated values

3.4.4 *Liquid Waste Activity and Volume*

HBPP also reports, in the Annual Radioactive Effluent Release Report, data on liquid waste discharged in effluents from the facility. The set of data provided in Table 3-4 provides a compilation of this information. The following text summarizes the liquid waste effluent release reports for 2009 through 2013. Liquid radioactive discharges ceased in 2014.

Table 3-4 Liquid Waste Effluent Releases

Year	Tritium Release (Ci)	Dissolved and Entrained Gas Release (Ci)	Alpha Release (Ci)	Other Fission and Activation Release (Ci)	Volume (Liters)	Volume of Dilution Water (Liters)
2009	2.74E-03	0	2.29E-06	5.08E-04	1.01E+05	7.23E+10
2010	1.88E-03	0	1.06E-05	5.39E-03	2.72E+05	7.57E+10
2011	7.66E-04	0	4.60E-06	4.48E-03	2.47E+05	1.04E+09
2012	3.15E-03	0	2.01E-06	1.92E-03	4.00E+05	1.93E+09
2013	8.98E-03	0	1.3E-06	2.04E-03	3.89E+05	2.47E+09

Radiation doses for the maximally exposed individuals both actual and projected due to liquid waste effluent releases are 0.07 mrem for the decommissioning period which is bounded by the evaluation in the GEIS (less than 0.1 person-rem).

3.4.5 *Gaseous Waste Activity and Volume*

HBPP also reports in the Annual Radioactive Effluent Release Report, data on gaseous waste. The set of data provided in Table 3-5 provides a compilation of this information. A summary of the gaseous waste effluent release reports for 2009 through 2013 with an estimate for the remainder of the project follows.

Table 3-5 Gaseous Waste Effluent Releases

Year	Fission and Activation Gas Release (Ci)	Iodines (Ci)	Particulates (Ci)
2009	<MDA	<MDA	<8.89E-06
2010	<MDA	<MDA	<1.47E-05
2011	<MDA	<MDA	<2.05E-05
2012	<MDA	<MDA	< 2E-05
2013	<MDA	<MDA	<2.4E-05
2014*	<MDA	<MDA	<2.4E-05
2015*	<MDA	<MDA	<2.4E-05
2016*	<MDA	<MDA	<2.4E-05
2017*	<MDA	<MDA	<2.4E-05
2018*	<MDA	<MDA	<2.4E-05
2019*	<MDA	<MDA	<2.4E-05

* Estimated values

Radiation doses for the maximally exposed individuals both actual and projected due to gaseous waste effluent releases are 0.00 mrem (total body-teen age group) and 0.00 mrem (bone-teen age group) which is bounded by the evaluation in the GEIS (less than 0.1 person-rem).

3.5 Site Description after License Release

Currently, the Count Room Building, Waste Management Building, Security buildings, Admin buildings, Training building, ISFSI, and Humboldt Bay Generating Station (HBGS) are the only structures scheduled to remain onsite at the time of license termination. All other above-grade structures will have been removed and the site will have been graded.

3.6 Coordination with Outside Entities

The decommissioning and termination of HBPP's 10 CFR Part 50 license involves, among others, the US NRC, several State of California regulatory agencies, US Army Corp of Engineers and the US Department of Transportation.

Chapter 8, "Supplement to the Environmental Report," discusses some of the related requirements.

3.7 References

3-1 U.S. Nuclear Regulatory Commission NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans" April 2000

3-2 U.S. Nuclear Regulatory Commission Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" January 1999

3-3 U.S. Nuclear Regulatory Commission NUREG-0586, "Final Generic Environmental Impact Statement (GEIS) on Decommissioning of Nuclear Facilities" October 2002, Supplement 1

3-4 HBPP Annual Radioactive Effluent Release Report March 30, 2010

3-5 HBPP Annual Radioactive Effluent Release Report March 31, 2011

3-6 HBPP Annual Radioactive Effluent Release Report March 30, 2012

3-7 HBPP Annual Radioactive Effluent Release Report for 2013

CONTENTS

TABLES

4 SITE REMEDIATION PLAN

4.1 Remediation Actions and ALARA Evaluations

This chapter of the License Termination Plan (LTP) describes various remediation and decontamination actions that may be used during the decommissioning of Humboldt Bay Power Plant (HBPP), Unit 3. Additionally describes are the methods used to reduce residual contamination to levels that comply with the NRC's annual dose limit of 25 mrem, and as low as reasonably achievable (ALARA). Finally, the Radiation Protection Program requirements for the remediation are described.

4.2 Remediation Actions

Remediation actions are performed throughout the decommissioning process. The remediation action taken is dependent on the material contaminated. The principal materials that may be subjected to remediation are hardened structural surfaces and soils. Activities performed solely to accommodate final status survey (FSS) measurements (e.g., wiping down of surfaces, shaving concrete to allow for proper instrument probe geometries) will not be evaluated for ALARA.

4.2.1 *Structures*

Following the removal of designated equipment and components, structures will be surveyed as necessary, contaminated materials will be remediated or removed and disposed as radioactive waste. Contaminated structural surfaces that will remain onsite after license termination will be remediated to levels that will meet the established radiological criteria provided in Chapter 6. Remediation techniques that may be used for the structural surfaces include washing, wiping, pressure washing, vacuuming, scabbling, chipping, and sponge or abrasive blasting. Washing, wiping, abrasive blasting, vacuuming, and pressure washing techniques may be used for both metal and concrete surfaces. Scabbling and chipping are mechanical surface removal methods intended for concrete surfaces. Concrete removal, if required, may include using machines with hydraulic-assisted, remote-operated, articulating tools. These machines have the ability to exchange scabbling, shear, chisel, and other tool heads.

4.2.1.1 *Scabbling and Shaving*

As stated above, the principal remediation methods expected to be used for removing contaminants from concrete surfaces are scabbling and shaving. Scabbling is a surface removal process that uses pneumatically operated air pistons with tungsten-carbide tips that fracture the concrete surface to a nominal depth of 0.25 inch at a rate of about 20 ft^2 per hour. The scabbling pistons (feet) are contained in a closed-capture attachment that is connected by hoses to a sealed vacuum and collector system. Shaving uses a series of diamond cutting wheels on a spindle, and performs at similar rates to scabbling. The wheels are also contained in a closed-capture attachment similar to scabbling equipment. The fractured media and dusts from both methods are deposited into a sealed removable container. The exhaust air passes through both roughing and absolute high efficiency particulate air (HEPA) filtration devices. Dust and debris generated through these remediation processes is collected and controlled during the operation.

4.2.1.2 *Needle Guns*

A second method of scabbling is accomplished using needle guns. The needle gun is a pneumatic air-operated tool containing a series of tungsten-carbide or hardened steel rods enclosed in a housing. The rods are connected to an air-driven piston to abrade and fracture the media surface. The media removal depth is a function of the residence time of the rods over the surface. Typically, one to two millimeters are removed per pass. Generated debris collection, transport, and dust control are accomplished in the same manner as other scabbling methods. Use of needle guns for removing and chipping media is usually reserved for areas not accessible to normal scabbling operations. These include, but are not limited to, inside corners, cracks, joints, and crevices. Needle gunning techniques can also be applied to painted and oxidized surfaces.

4.2.1.3 *Chipping*

Chipping includes the use of pneumatically operated chisels and similar tools coupled to vacuum-assisted collection devices. Chipping activities are usually reserved for cracks and crevices. This action is also a form of scabbling.

4.2.1.4 *Sponge and Abrasive Blasting*

Sponge and abrasive blasting are similar techniques that use media or materials coated with abrasive compounds such as silica sands, garnet, aluminum oxide, and walnut hulls. Sponge blasting is less aggressive, incorporating a foam media that, upon impact and compression, absorbs contaminants. The medium is collected by vacuum and the contaminants are washed from the medium so the medium may be reused. Abrasive blasting is more aggressive than sponge blasting but less aggressive than scabbling. Both operations use intermediate air pressures. Sponge and abrasive blasting are intended for the removal of surface films and paints.

4.2.1.5 *Pressure Washing*

Pressure washing uses a nozzle of intermediate water pressure to direct a jet of pressurized water that removes superficial materials from the suspect surface. A header may be used to minimize overspray. A wet vacuum system is used to suction the potentially contaminated water into containers for filtration or processing.

4.2.1.6 *Washing and Wiping*

Washing and wiping techniques are actions that are normally performed during the course of remediation activities and will not always be evaluated as a separate ALARA action. When washing and wiping techniques are used as the sole means to reduce residual contamination below Derived Concentration Guideline Levels (DCGL) levels, ALARA evaluations are performed. Washing and wiping techniques used as housekeeping or good practice measures will not be evaluated. Examples of washing and wiping activities for which ALARA evaluations would be performed include the following:

- Decontamination of structural materials, metals, or media for which decontamination reagents may be required

- Structure areas that do not provide sufficient access for use of other decontamination equipment such as pressure washing.

4.2.1.7 *Grit Blasting*

Most contaminated piping will be removed and disposed as radioactive waste. Any remaining contaminated piping buried or embedded in concrete may be remediated using methods such as grit blasting. Grit blasting uses grit media such as garnet or sand under intermediate air pressure directed through a nozzle that is pulled through the closed piping at a fixed rate. The grit blasting action removes the interior surface layer of the piping. A HEPA vacuum system maintains the sections being cleaned under negative pressure and collects the media for reuse or disposal. The final system pass is performed with clean grit to remove any residual contamination.

4.2.1.8 *Removal of Activated Concrete*

Activated concrete will be evaluated and remediated or removed, as necessary.

4.2.1.9 *Additional Remedial Actions*

Mechanical abrasive equipment, such as hones, may be used to remove contamination from the surfaces of embedded or buried piping. Chemical removal means may be used, as appropriate, for the removal of certain contaminants.

4.2.2 *Soil*

Soil contamination above the site specific DCGL that is removed will be disposed as radioactive waste. Operational constraints and dust control will be addressed in site excavation and soil control procedures. In addition, work package instructions for remediation of soil may include additional constraints and mitigation or control methods. The site characterization process established the location and extent of soil contamination. As needed, additional investigations will be performed to ensure that any changing soil contamination profile during the remediation actions is adequately identified and addressed. It should also be noted that soil remediation volume estimates in the LTP may vary from section to section, as appropriate, depending on their use (e.g., decommissioning cost estimates, ALARA evaluations, or dose assessment). Chapter 5 discusses soil sampling and survey methods. Soil remediation equipment will include, but not be limited to, shovels, backhoe and trackhoe excavators. As practical, when the remediation depth approaches the soil interface region between unacceptable and acceptable contamination, a squared edge excavator bucket design or similar technique may be used. This simple methodology minimizes the mixing of contaminated soils with acceptable lower soil layers as would occur with a toothed excavator bucket. Remediation of soils will include the use of established Excavation Safety and Environmental Control procedures. Additionally, work package instructions will augment the previous guidance and procedural requirements to ensure adequate erosion, sediment, and air emission controls during soil remediation.

Characterization data available to date indicates that no remediation of surface or ground waters will be required at the HBPP to meet the site release criteria.

4.3 Remediation Activities Impact on the Radiation Protection Program

The Radiation Protection Program used for decommissioning is similar to the program in place during power operation. During power and SAFSTOR operations, contaminated structures, systems, and components were decontaminated in order to perform maintenance or repair actions. The techniques used during operations are the same or similar to the techniques used during decommissioning to reduce personnel exposure to radiation and contamination and to prevent the spread of contamination from established contaminated areas. Remediation activities have had an impact on the HBPP Radiation Protection Program, given that alpha contamination is present on the interior surfaces of Unit 3 systems and components. Subsequent challenges to the program due to the alpha are:

- Alpha monitoring is required in the ventilation system
- Alpha contaminated surfaces require application of a fixative prior to dismantling the system or component
- Mechanical cutting is the only current method allowed on alpha contaminated material removal, causing challenges in the exposure control to personnel

Decommissioning planning allows radiation protection personnel to focus on each area of the site and plan each activity well before execution of the remediation technique. Low levels of surface contamination are to be remediated by washing and wiping. These techniques have been used successfully in the decommissioning process. Wiping with a detergent has been the method of choice for large area decontamination. Wiping with detergent soaked or oil-impregnated media has been used on small items, overhead spaces, and small hand tools to remove surface contaminants. These same techniques will be applied to remediate lightly contaminated structural surfaces during remediation actions. Scabbling or other surface removal techniques will reduce high levels of contamination on contaminated concrete. Mechanical or diamond wire cutting will be used to section the reactor vessel. The current Radiation Protection Program provides adequate controls for these actions.

The Decommissioning Organization is experienced in and capable of applying these remediation techniques on contaminated systems, structures, or components during decommissioning. The existing Radiation Protection Program is adequate to control the radiological aspects of remediation work safely.

4.4 ALARA Evaluation

In order to terminate the NRC 10 CFR 50 license, HBPP must demonstrate that the dose criteria in 10 CFR 20, Subpart E, have been met, and should demonstrate whether it is feasible to further reduce the levels of residual activity to below those necessary to meet the dose criteria (i.e., to levels that are ALARA). For the HBPP decommissioning, the ALARA cleanup levels are established at one of two levels: a pre-defined generic ALARA screening, or a survey unit-specific ALARA evaluation. In either case, an ALARA action level (AL) is applied.

The AL corresponds to a residual activity concentration at which the averted radiation dose converted into dollars is equal to the costs of remediation. An ALARA analysis ensures that the efforts to remove residual contamination are commensurate with the risk that exists from leaving the contamination in place. “Reasonably achievable” is judged by considering the state of technology and the economics of improvements in relation to all the benefits from these improvements. However, a comprehensive consideration of risks and benefits will include risks from non-radiological hazards. An action taken to reduce radiation risks should not result in a significantly larger risk from the other hazards.

NUREG-1757, “Consolidated NMSS Decommissioning Guidance” (Reference 4-3) recognizes that remediation of soils beyond the DCGLs is not likely to be cost-beneficial due to the high costs of waste disposal. For

HBPP, if remediation of soils beyond the DCGL is determined not to be cost-beneficial, then residual activity in soils that meet the DCGL will be considered ALARA. Similarly, if residual radioactivity on remaining structures is below a pre-determined generic ALARA screening level or a unit specific level, then the levels associated with the structure will be considered ALARA. The methodology and equations used are consistent with those provided in Volume 2 of NUREG-1757. Copies of ALARA evaluations will be included in the FSS Report for each survey area.

4.5 Unit Cost Estimates

In order to effectively perform ALARA evaluations and remediation actions, unit cost values are required. These values are used to perform the NUREG-1757, Volume 2, Cost-Benefit Analysis.

4.5.1 *Calculation of Total Cost*

When performing a fairly simple evaluation, the costs generally include the monetary costs of: (1) the remediation action being evaluated, (2) transportation and disposal of the waste generated by the action, (3) workplace accidents that occur because of the remediation action, (4) traffic fatalities resulting from transporting the waste generated by the action, (5) doses received by workers performing the remediation action, and (6) doses to the public from excavation, transport, and disposal of the waste. Other costs that are appropriate for the specific case may also be included. Values of some standard parameters are contained in Table 4-1.

The total cost, ($Cost_T$), which is balanced against the benefits, has several components and may be evaluated according to Equation N-3 of NUREG-1757, Vol. 2 Appendix N:

$$Cost_T = Cost_R + Cost_{WD} + Cost_{ACC} + Cost_{TF} + Cost_{WDose} + Cost_{PDose} + Cost_{other}$$

Where:

$Cost_R$	=	monetary cost of the remediation action (including mobilization costs)
$Cost_{WD}$	=	monetary cost for transport and disposal of the waste generated by the action
$Cost_{ACC}$	=	monetary cost of worker accidents during the remediation action

$Cost_{TF}$ = monetary cost of traffic fatalities during transportation of the waste

$Cost_{WDose}$ = monetary cost of dose received by workers performing the remediation action and transporting waste to the disposal facility

$Cost_{PDose}$ = monetary cost of dose to the public from excavation, transport and disposal of the waste

$Cost_{other}$ = other costs as appropriate for the particular situation

4.5.1.1 *Remedial Action Costs*

Calculations of the incremental remedial action costs include the standard manpower and mechanical costs. Lower concentrations may change sampling/survey requirements. Increased survey costs can be considered in the remedial action (e.g., confined spaces, difficult to access areas, ceilings and walls above 6 feet) and will raise standard remediation costs due to the increase in man-hours, but note that these are the incremental costs of surveying below the dose limit.

4.5.1.2 *Transport and Disposal of the Waste*

The cost of waste transport and disposal ($Cost_{WD}$) may be evaluated according to Equation N-4 of NUREG-1757, Vol. 2 Appendix N:

$$Cost_{WD} = V_A \times Cost_V$$

Where:

V_A = volume of waste produced, remediated in units of m^3

$Cost_V$ = cost of waste disposal per unit volume, including transportation cost, in units of $\$/m^3$

4.5.1.3 *Non-radiological Risks*

The cost of non-radiological workplace accidents ($Cost_{ACC}$) may be evaluated using Equation N-5 of NUREG-1757, Vol. 2 Appendix N:

$$Cost_{ACC} = \$3{,}000{,}000 \times F_W \times T_A$$

Where:

\$3,000,000 = monetary value of a fatality equivalent to \$2,000 Person-Rem (see pages 11-12 of "Reassessment of NCR's Dollar per Person-Rem Conversion Factor

Policy," NUREG-1530, December 1995)(Reference 4-6)

F_W = workplace fatality rate in fatalities/hour worked; and

T_A = worker time required for remediation in units of worker-hours

4.5.1.4 *Transportation Risks*

The cost of traffic fatalities incurred during the transportation of waste ($Cost_{TF}$) may be evaluated using Equation N-6 of NUREG-1757, Vol. 2, Appendix N:

$$Cost_{TF} = \$3{,}000{,}000 \times \left[\frac{V_A}{V_{SHIP}}\right] \times F_T \times D_T$$

Where:

V_A = volume of waste produced in units of m^3

F_T = fatality rate per truck-kilometer (km) traveled in units of fatalities/truck-km

D_T = distance traveled in km

V_{SHIP} = volume of a truck shipment in m^3

4.5.1.5 *Worker Dose Estimates*

The cost of the remediation worker dose (*$Cost_{WDose}$*) may be evaluated using Equation N-7 of NUREG-1757, Vol. 2 Appendix N:

$$Cost_{WDose} = \$2{,}000 \times D_R \times T$$

Where:

D_R = total effective dose equivalent (TEDE) rate to remediation workers in units of rem/hr

T = time worked (site labor) to remediate the area in units of person-hour

4.5.1.6 *Loss of Economic Use of Property*

A cost in the "other" category could include the fair market rental value or economic use for the site during the time the additional remediation work is being performed.

4.5.1.7 *Parameters*

For performing these calculations, acceptable values for some of the parameters are shown in Table 4-1 below:

Table 4-1 Parameter Values for use in ALARA Analysis

Parameter	Value	Reference and Comments
Workplace accident fatality rate, F_W	4.2×10^{-8}/hr	NUREG–1496, "Final Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities," and NUREG–1496, July 1997, Volume 2, Appendix B, Table A.1 (Reference 4-2)
Transportation fatality rate, F_T	Trucks: 3.8×10^{-8}/km	NUREG–1496, Volume 2, Appendix B, Table A.1
Dollars per person-rem	$2,000	NUREG/BR–0058 (Reference 4-8)
Monetary discount rate, r	0.07/y for the first 100 years and 0.03/y thereafter, or 0.07 for buildings and 0.03 for soil	NUREG/BR–0058
Number of years of exposure, N	Buildings: 70 years Soil: 1000 year	NUREG–1496, Volume 2, Appendix B, Table A.1
Population density, P_D	Building: 0.007 Land: 0.0001	HBPP Site-Specific Sensitivity Analysis
Excavation, monitoring, packaging, and handling soil	1.62 person-hours/m^3 of soil	NUREG–1496, Volume 2, Appendix B, Table A.1
Waste shipments volume	Truck: 13.6 m^3/shipment	NUREG–1496, Volume 2, Appendix B, Table A.1

4.5.2 *Calculation of Benefits*

In the simplest form of the analysis, the only benefit estimated from a reduction in the level of residual radioactivity is the monetary value of the collective averted dose to future occupants of the site. For buildings, the collective averted dose from residual radioactivity is based on the occupational scenario. For land, the averted dose is based on the resident farmer scenario. In general, the ALARA analysis should use the same critical group scenario that is used for the compliance calculation.

The benefit from collective averted dose (B_{AD}) is calculated by determining the present worth of the future collective averted dose and multiplying it by a factor to convert the dose to a monetary value using Equation N-1 of NUREG-1757, Vol. 2 Appendix N:

$$B_{AD} = \$2{,}000 \times PW(AD_{collective})$$

Where:

B_{AD} = benefit from an averted dose for a remediation action, in current U.S. dollars

\$2,000 = value in dollars of a person-rem averted (see NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," Revision 4, 2004)

$PW(AD_{collective})$ = present worth of a future collective averted dose

An acceptable value for a collective dose is \$2000 per person-rem averted, discounted for a dose averted in the future (see Section 4.3.3 of "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," NUREG/BR-0058, Revision 4, 2004). For doses averted within the first 100 years, a discount rate of 7 percent should be used. For doses averted beyond 100 years, a 3 percent discount rate should be used.

The present worth of the future collective averted dose can be estimated from Equation N-2 of NUREG-1757, Volume 2, Appendix N, for relatively simple situations:

$$PW(AD_{collective}) = P_D \times A \times 0.025 \times F \times \frac{Conc}{DCGL_W} \times \frac{1 - e^{-(r+\lambda)^N}}{r + \lambda}$$

Where:

P_D = population density for the critical group scenario in people/m^2

A = area being evaluated in square meters (m^2)

0.025 = annual dose to an average member of the critical group from residual radioactivity at the DCGL concentration in rem/y

F = effectiveness, or fraction of the residual radioactivity removed by the remediation action

$Conc$ = average concentration of residual radioactivity in the area being evaluated in units of activity per unit area for buildings or activity per unit volume for soils

$DCGL_W$ = derived concentration guideline equivalent to the average concentration of residual radioactivity that would give a dose of 0.25 mSv/y (25 mrem/y) to the average member of the critical group, in the same units as "Conc"

r = monetary discount rate in units per year

λ = radiological decay constant for the radionuclide in units per year

N = number of years over which the collective dose will be calculated

The present worth of the benefit calculated by Equation N-2 assumes that the peak dose occurs in the first year. This is usually true for the building occupancy scenario, but not always true for the residential scenario where the peak dose can occur in later years. When the peak dose occurs in later years, Equation N-2 would overestimate the benefit. A more exact calculation may be used that avoids this overestimation of the benefit of remediation by calculating the dose during each year of the evaluation period and then calculating the present worth of each year's dose.

The $DCGL_W$ used should be the same as the $DCGL_W$ used to show compliance with the 25 mrem/y dose limit. The population density, P_D, should be based on the dose scenario used to demonstrate compliance with the dose limit. Thus, for buildings, the estimate P_D for the occupational scenario should be used. For soil, P_D should be based on the resident farmer scenario. The factor at the far right of the equation, which includes the exponential terms, accounts for both the present worth of the monetary value and radiological decay.

If more than one radionuclide is present, the total benefit from a collective averted dose, B_{AD} is the sum of the collective averted dose for each radionuclide. When multiple radionuclides have a fixed concentration, residual radioactivity below the dose criteria is normally demonstrated by measuring one radionuclide and comparing its concentration to a $DCGL_W$ that has been calculated to account for the dose from the other radionuclides. In this case, the adjusted $DCGL_W$ may be used with the concentration of the radionuclide being measured. The other case is where the ratio of the radionuclide concentrations is not fixed, but varies from location to location within a survey unit; this benefit is the sum of the collective averted dose from each.

4.5.3 *Residual Radioactivity Levels that are ALARA*

The residual radioactivity level that is ALARA is the concentration *(Conc)* at which the benefit from removal equals the cost of removal. If the total cost ($Cost_T$) is set equal to the present worth of the collective dose averted in Equation N-2, the ratio of the concentration *(Conc)* to the $DCGL_W$ can be determined by using Equation N-8 of NUREG-1757, Vol. 2, Appendix N below:

$$\frac{Conc}{DCGL_W} = \frac{Cost_T}{\$2{,}000 \times P_D \times 0.025 \times F \times A} \times \frac{r+\lambda}{1-e^{-(r+\lambda)N}}$$

All the items in Equation N-8 are as previously defined. Since P_D, N, λ and r are constants that have generic values for all locations on the site for each scenario, HBPP only needs to determine the total cost, $Cost_T$, and the effectiveness, F, for a specific remediation action for a specific area. If the concentration at a location exceeds *Conc*, it may be cost effective to remediate the location by a method whose total cost is $Cost_T$. Note that the concentration, *Conc,* which is ALARA, can be higher or lower (more or less stringent) than the $DCGL_W$, although the $DCGL_W$ must be met in order to meet the criteria for license termination.

4.6 Radionuclides Considered for ALARA Calculations

As discussed in Chapter 6, the site-specific suite of radionuclides identified for use at HBPP contains 22 radionuclides. Only two of these radionuclides have been identified above minimum detectable concentration (MDC) levels in soil samples and structural surface samples. For purposes of the ALARA calculations, only Cs-137 and Co-60 are used along with their associated DCGL values.

4.7 References

4-1 U.S. Nuclear Regulatory Commission NUREG/CR-5512, Volume 3, “Residual Radioactive Contamination from Decommissioning—Parameter Analysis, Draft Report for Comment,” October 1999

4-2 U.S. Nuclear Regulatory Commission, NUREG-1496, Volume 2, “Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities,” July 1997

4-3 U.S. Nuclear Regulatory Commission, NUREG-1757, Volume 2, Final Report, “Consolidated NMSS Decommissioning Guidance—

Characterization, Survey, and Determination of Radiological Criteria," September 2003

4-4 U.S. Nuclear Regulatory Commission, NUREG/CR-5884, Volume 2, "Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station," November 1995

4-5 U.S. Nuclear Regulatory Commission, NUREG/CR-5884, Volume 1, "Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station," November, 1995

4-6 U.S. Nuclear Regulatory Commission, NUREG-1530, "Reassessment of NRC's Dollar per Person-Rem Conversion Factor Policy," December 1995

4-7 U.S. Nuclear Regulatory Commission, NUREG–1496, Volume 1, Final Report, "Final Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities," July 1997

4-8 U.S. Nuclear Regulatory Commission, NUREG/BR–0058, Revision 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," September 2004

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5 FINAL STATUS SURVEY PLAN

5.1 Introduction to the Final Status Survey Plan

The Humboldt Bay Power Plant (HBPP), Unit 3, Final Status Survey (FSS) Plan has been prepared using the applicable regulatory and industry guidance. Survey results are documented by survey unit in corresponding survey packages.

5.1.1 *Purpose*

The FSS Plan describes the final survey process used to demonstrate that the HBPP facility and site comply with radiological criteria for unrestricted use specified in 10 CFR 20.1402 (i.e., annual dose limit of 25 millirem as well as ensure dose will be As Low As Reasonably Achievable (ALARA) for all dose pathways). Nuclear Regulatory Commission (NRC) regulations applicable to radiation surveys are found in 10 CFR 50.82(a)(9)(ii)(D), 10 CFR 50.82(11)(ii), and 10 CFR 20.1501(a) and (b).

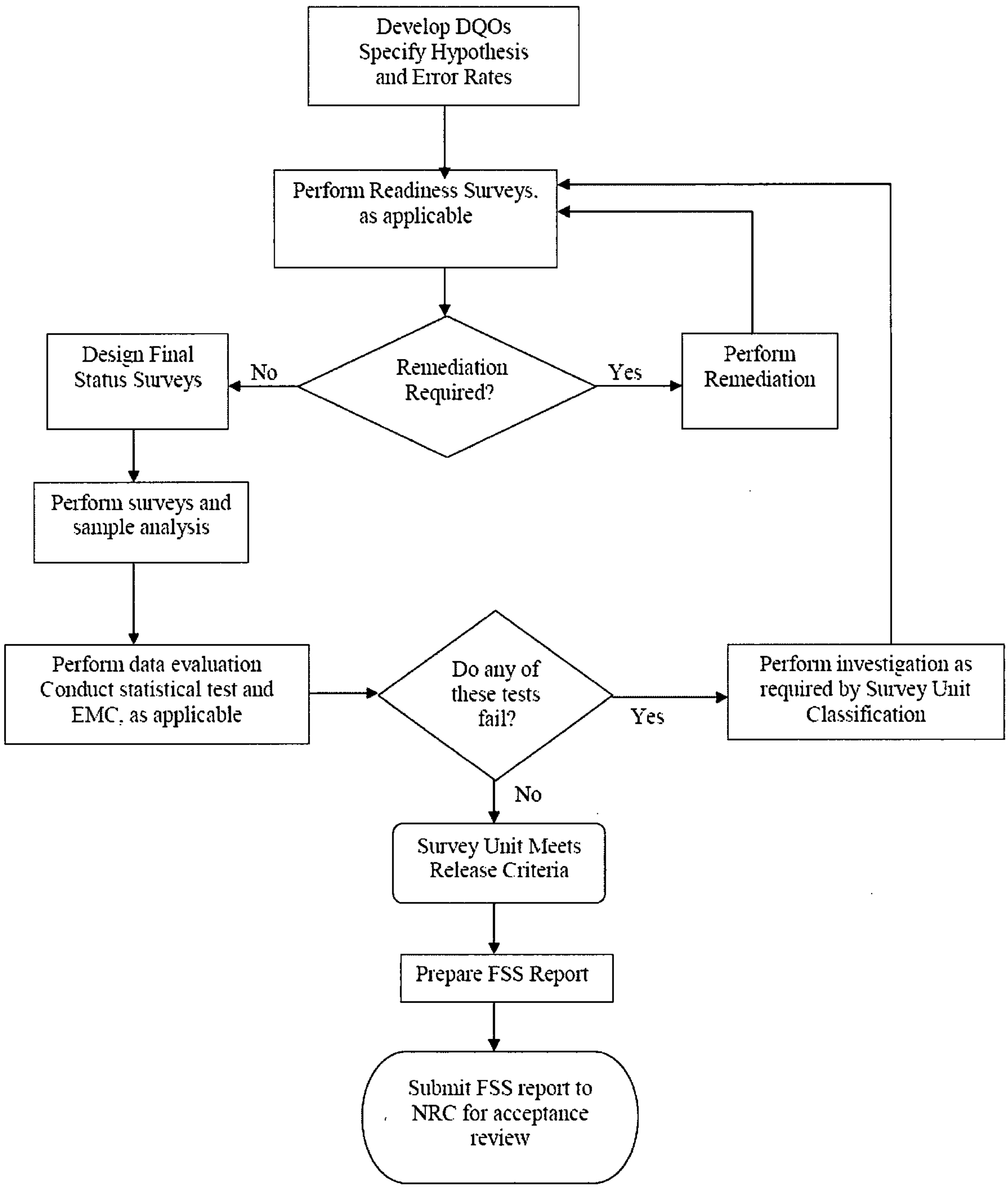
5.1.2 *Scope*

PG&E intends to release site land from the 10 CFR Part 50 license. An Independent Spent Fuel Storage Installation (ISFSI) located on the site is licensed under 10 CFR Part 72 and will be released from the Part 50 license per this release plan. This Plan addresses only facilities and land areas that are identified as contaminated or potentially contaminated (affected) resulting from activities associated with commercial nuclear plant operation.

5.1.3 *Final Status Survey Preparation and Implementation Overview*

The FSS Plan contained in this chapter will be used as the basis for developing FSS procedures and applying existing procedures to the FSS process (Figure 5-1). Section 5.1.4 contains a list of regulatory documents used in preparing the FSS Plan. Quality assurance requirements, which are outlined in Section 5.8, apply to activities associated with FSS. An FSS Package will be produced for each survey unit; the survey package is a collection of documentation detailing survey design, survey implementation, and data evaluation for the FSS. The following sections describe specific elements of the organization, preparation, and implementation of the HBPP FSS. All processes associated with final status surveys will be conducted in accordance with approved site procedures.

Figure 5-1 FSS Process Overview



5.1.3.1 *FSS Organization*

The general FSS organization will consist of the HBPP Site Closure Manager, the FSS Supervisor, FSS Engineers, and technicians. Since the License Termination organization has not been fully implemented at the time of License Termination Plan (LTP) development, it is expected that specific job titles may vary over the period of project execution. These titles are used within this document to describe various functional areas of responsibility. Section 5.8.1.1 outlines the basic responsibilities and functions of the FSS organization.

5.1.3.2 *Survey Preparation*

Survey preparation is the first step in the FSS process and occurs after any necessary remediation has been completed. In areas where remediation is required, a remediation survey or equivalent evaluation will be performed to confirm that remediation was successful prior to initiating FSS activities. Remediation surveys, turnover surveys, or equivalent evaluation for areas not requiring remediation may be performed using the same process and controls as FSS so that data from these surveys may be used as part of the FSS data. In order for survey data to be used for FSS, it will be designed and collected in compliance with approved procedures and in accordance with Sections 5.3 through 5.5 or as specified by the License. Additionally, the area will be controlled in accordance and implemented via approved procedures. Any surveys performed prior to the NRC approval of the LTP are understood to have been performed "at risk." Survey design and the data collected will be carefully evaluated to ensure the intent of the LTP and associated procedures were met before using the data. Following turnover/remediation surveys (if required) or post-remediation evaluation, the FSS is performed. Areas to be surveyed are isolated and/or controlled to ensure that radioactive material is not reintroduced into the area from ongoing activities nearby and to maintain the "as left" condition of the area. Section 5.2 addresses specific survey preparation requirements and considerations. All tools and equipment that would impede the survey must be removed, the area must be free of obstructions to the survey, and the area must be in a condition that will allow FSS activities.

Routine access, material storage, and worker transit through the area are not allowed, unless authorized by the FSS Supervisor, or designee. An inspection of the area is conducted by FSS personnel to ensure that work is complete and the area is ready for final status survey. Approved procedures provide isolation and control measures until the area is released for unrestricted use.

5.1.3.3 *Survey Design*

The survey design process establishes the methods and performance criteria used to conduct the survey. Survey design assumptions are documented in Survey Packages for each survey unit in accordance with approved procedures. The site land, structures, and systems[1] are organized into survey areas and classified by contamination potential as Class 1, Class 2, Class 3, or nonimpacted in accordance with Section 5.2.2. See Chapter 2 for illustrative representations of the HBPP survey areas. Survey unit size is based on the assumptions in the dose assessment models in accordance with the guidance provided in NUREG-1757, Volume 2, "Consolidated NMSS Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria, Final Report." The percent coverage for scan surveys is determined in accordance with Section 5.3.2. The number and location of structure surface measurements (and structure volumetric samples, if required) and soil samples are established in accordance with Sections 5.3.3 and 5.3.5. Investigation levels are established in accordance with Section 5.3.6. A survey map is prepared for each survey unit and a reference grid is superimposed on the map to allow use of an (*x*, *y*) coordinate system. Random numbers between 0 and 1 are generated, which are then multiplied by the maximum *x* and *y* axis values of the sample grid. This provides coordinates for each random sample location, or a random start location for a systematic grid, as appropriate. Grid

[1] embedded and buried piping/conduit are the principal potentially contaminated systems that will remain after decommissioning

points may be automatically designated on the map, with grid locations, if generated, using Visual Sample Plan (VSP) software. The measurement/sample locations are plotted on the map. Each measurement/sample location is assigned a unique identification code, which identifies the measurement/sample by survey area, survey unit, and sequential number. The appropriate instruments and detectors, instrument operating modes, and survey methods used to collect and analyze data are also specified. Replicate measurements are performed as part of the quality process established to identify, assess, and control errors and uncertainty associated with sampling, survey, or analytical activities. This quality control process, described in Section 5.8.1, provides assurance that the survey data meet the accuracy and reliability requirements necessary to support the decision to release or not release a survey unit. Written survey instructions that incorporate the requirements set forth in the survey design and direction are provided, as applicable to survey design, for selection of instruments, count times, instrument modes, survey methods, required documentation, investigation set points, investigative actions, background requirements, and other appropriate instructions. In conjunction with the survey instructions, survey data forms may be prepared to assist in survey documentation as well as using the data-logging capabilities of the instruments. The survey design is reviewed and quality verification steps applied to ensure that appropriate instruments, survey methods, and sample locations have been properly identified. A two-tiered review process will be used with a review by a peer Engineer and a review and approval by the FSS Supervisor, or designee.

5.1.3.4 *Survey Data Collection*

After preparation of a survey package, the FSS data are collected. Trained and qualified personnel will perform the necessary measurements using calibrated instruments in accordance with approved procedures and instructions contained in the survey package. Section 5.5 addresses FSS data collection requirements. Survey areas and/or locations are identified by gridding, markings, or flags as appropriate. A FSS Engineer, or qualified designee, performs a pre-survey briefing with the survey technicians during which the survey instructions are reviewed and additional survey unit considerations are discussed (e.g., safety). The technicians gather instruments and equipment as indicated and perform surveys in accordance with the appropriate procedures and survey package specifications. Technicians are responsible for documenting survey results and maintaining custody of samples and instrumentation. At the completion of surveys, technicians return instruments and prepare samples for analysis. Survey instruments provided to the technicians are prepared in accordance with approved site procedures and the survey instructions. Instrument calibration, except for onsite lab instrumentation, is performed either onsite or by an offsite vendor and performance checks are conducted in accordance with applicable site procedures. Data are reviewed to flag any measurements that exceed investigation criteria so that appropriate investigation surveys can be performed and any required remediation can be planned and performed as necessary. Corrective action documents will be initiated as necessary to document problems and to implement appropriate corrective actions.

If a survey unit has been selected to receive a Quality Control (QC) survey (replicate surveys, etc.), a QC survey package is developed and implemented. QC measurement results are compared to the original measurement results. If QC results do not reach the same conclusion as the original survey, an investigation is then performed. Section 5.8 provides additional detail regarding QC survey requirements.

5.1.3.5 *Data-Assessment*

Survey data assessment is performed to verify that the data are sufficient to demonstrate that the survey unit meets the unrestricted use criterion. Statistical analyses are performed on the data and compared to predetermined investigation levels (see Section 5.3.6). Depending on the results of the data assessment and any required investigation, the survey unit may either be released or require further remediation, reclassification, and/or resurvey. Assumptions and requirements in the survey package are reviewed for applicability and completeness; additional data needs are identified during this review. Specific data assessment requirements are contained in Section 5.6. A review is performed of survey data and sample counting reports to verify completeness, legibility, and compliance with survey design and associated instructions. As directed by the FSS supervisor, or designee, the following types of activities may be performed:

- Convert data to reporting units
- Calculate mean, median and range of the data set
- Review the data for values that vary more than three standard deviations from the data mean
- Calculate the standard deviation of the data set
- Calculate Minimum Detectable Concentration (MDC) for each survey type performed
- Create posting, frequency and quartile plots for visual interpretation of data

Computer programs may be used for these activities if they have been approved by the Site Closure Manager, or designee. FSS personnel include data quality verifications in their evaluations of statistical calculations. Integrity and usefulness of the data set and the need for further data or investigation are also included in the evaluations. The Site Closure Manager, or designee, will review the data for statistical evaluation. The results of the data evaluation are documented and filed in the survey package.

5.1.3.6 *Final Status Survey Package Completion*

Survey results are documented by survey unit in corresponding survey packages. The data are reviewed, analyzed, and processed and the results documented in the FSS Package. This documentation file provides a record of the information necessary to support the decision to release the survey units for unrestricted use. The FSS Reports will be prepared to provide the necessary data and analyses from FSS packages for submittal to the NRC. Section 5.7 addresses reporting of survey results and conclusions.

5.1.4 *Regulatory Requirements and Industry Guidance*

This FSS Plan has been developed using the guidance contained in the following documents:

- NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM), Revision 1, August 2000"
- NUREG-1505, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys, Revision 1, June 1998"
- NUREG-1507, "Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions, June 1998"
- NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," Revision 1, April 2003
- NUREG-1757, Vol. 2, "Consolidated NMSS Decommissioning Guidance Characterization, Survey, and Determination of Radiological Criteria, Final Report", Revision 1, September 2006
- Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," Revision 1, June 2011

Other documents used in the preparation of this plan are listed in Section 5.9. PG&E anticipates the NRC may choose to conduct confirmatory measurements during HBPP FSS activities. The NRC may take confirmatory measurements to make a determination the FSS and associated documentation demonstrate that the site is suitable for release in accordance with the criteria established in 10 CFR Part 20 subpart E.

5.2 Development of Survey Plan

5.2.1 *Radiological Status*

The following sections provide a summary of site characterization and dose modeling results applicable to development of the HBPP FSS Plan.

5.2.1.1 *Identification of Radiological Contaminants*

A site-specific suite of radionuclides potentially present at HBPP has been developed. This suite contains 22 radionuclides that are potentially present in HBPP environs, structures, and systems/components.

Development of this site-specific suite of radionuclides is described in Chapter 6, "Compliance with the Radiological Criteria for License Termination," Section 6.2. PG&E has conducted radiological characterization of the site property to identify and document residual contamination resulting from nuclear plant operation, SAFSTOR operations and decommissioning activities. The effort included reviews of historical information as well as physical measurements of onsite soils, structures, and systems during scoping and characterization surveys. Chapter 2, Site Characterization, contains a detailed discussion of this effort.

5.2.1.2 *Dose Modeling Summary*

Dose models allow the translation of residual radioactivity levels into potential radiation doses to the public. For the HBPP site, dose models have been developed based on the guidance found in NUREG/CR-5512, Volumes 1, 2, and 3. The conceptual model reflects the anticipated site conditions at the time of unrestricted release. The dose modeling approach for the HBPP site is consistent with the information for site specific modeling provided in Appendix I of NUREG-1757, including source term abstraction and scenarios, pathways, and critical groups.

There are three defining factors for a dose model: (1) the scenario, (2) the critical group, and (3) the exposure pathways. The scenarios described in NUREG/CR-5512 Volume 1 address the major exposure pathways of direct exposure to penetrating radiation and inhalation and ingestion of radioactive materials. The scenarios also identify the critical group, which is defined as the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity within the assumptions of a particular scenario. The design for scenarios and the site-specific modeling provide reasonable and conservative estimates of the potential doses associated with residual radioactivity.

The dose models supporting the building surface and soil Derived Concentration Guideline Levels (DCGLs) were developed using the approach outlined previously. The scenarios described in NUREG/CR-5512, Volume 1, were selected for the HBPP site to estimate potential radiation doses from radioactive material in buildings based on (building occupancy) and soil (resident farmer scenario).

Table 5-1 provides a list of significant radionuclides that may be present in onsite soils and on structural surfaces along with their corresponding single nuclide DCGL values derived in Chapter 6. The DCGL values have been rounded down to two significant figures.

Table 5-1 DCGLs by Radionuclide and Medium Type

Nuclide	Building Surface ($dpm/100\ cm^2$)	Soils (pCi/g)
	25 mrem/y DCGL	25mrem/y DCGL
Am-241	3.0E+03	2.5E+01
C-14	7.0E+06	6.3E+00
Cm-243	4.3E+03	2.9E+01
Cm-244	5.5E+03	4.8E+01
Cm-245	2.2E+03	1.7E+01
Cm-246	2.7E+03	2.5E+01
Co-60	1.3E+04	3.8E+00
Cs-137	4.6E+04	7.9E+00
Eu-152	2.7E+04	1.0E+01
Eu-154	2.5E+04	9.4E+00
H-3	1.8E+08	6.8E+02
Nb-94	1.9E+04	7.1E+00
Ni-59	6.3E+07	1.9E+03
Ni-63	2.4E+07	7.2E+02
Np-237	2.4E+03	1.1E+00
Pu-238	3.4E+03	2.9E+01
Pu-239	3.1E+03	2.6E+01
Pu-240	3.1E+03	2.6E+01
Pu-241	1.4E+05	8.6E+02

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Nuclide	Building Surface (dpm/100 cm^2)	Soils (pCi/g)
	25 mrem/y DCGL	25mrem/y DCGL
Sr-90	9.7E+04	1.5E+00
Tc-99	9.6E+06	1.2E+01

5.2.1.3 *Surrogate Ratio DCGLs*

Generally, surrogate ratio DCGLs are developed and applied to land areas and materials with volumetric residual radioactivity where constant radionuclide concentration ratios can be demonstrated to exist. They are derived using pre-remediation site characterization data collected prior to the FSS. The established ratio among the radionuclide concentrations allows the concentration of every radionuclide to be expressed in terms of any one of them. Likewise, a surrogate ratio DCGL allows the DCGLs specific to Hard-to-Detect (HTD) radionuclides in a mixture to be expressed in terms of a single radionuclide that is more readily measurable. The measured radionuclide is called the surrogate radionuclide. Cs-137 is expected to be the surrogate radionuclide for HBPP. A sufficient number of measurements, representative of the area of interest, are taken to establish a consistent ratio of radionuclide concentrations. The number of measurements needed to determine the ratio is based on the chemical, physical, and radiological characteristics of the radionuclides and the site. Measurements from different media types will not be mixed to derive the ratio. The surrogate ratio is acceptable if the mean values for individual samples for a given media are within two standard deviations of the overall mean value for the media. Once an appropriate surrogate ratio is determined, the DCGL of the measured radionuclide is modified to account for the represented radionuclide according to the following Equation 5-1 (MARSSIM Equation 4-1):

$$DCGL_{SR} = DCGL_{SUR} \times \frac{DCGL_{REP}}{[(C_{REP} \div C_{SUR})(DCGL_{SUR})] + DCGL_{REP}}$$

Equation 5-1

where:

$DCGL_{SR}$ = modified DCGL for surrogate ratio

$DCGL_{SUR}$ = DCGL for surrogate radionuclide

$DCGL_{REP}$ = DCGL for represented radionuclide

C_{REP} = Concentration of represented radionuclide

C_{SUR} = Concentration of surrogate radionuclide

The following process will be applied to assess the need to use surrogate ratios for final status surveys:

- Determine whether HTD radionuclides (e.g., TRU, Sr-90, H-3) are likely to be present in the survey unit based on process knowledge and historical data or characterization.

- When HTD radionuclides are likely to be present, establish a relationship using a representative number of samples (typically 6 or more). The samples may come from another survey unit if the source of the contamination and expected concentrations are reasonably the same. These samples will be analyzed for ETD and HTD radionuclides using gross alpha, alpha spectroscopy, gross beta analysis, or gamma spectroscopy techniques. RAI 11

Surrogate relationships will be determined using one of the methods described below:

- Develop a surrogate relationship for each HTD radionuclide.
- Determine the average surrogate DCGL and the standard deviation from the surrogate relationships.

If the mean values for individual samples for a given media are within two standard deviations of the overall mean value for the media, the surrogate ratio is acceptable. If this criterion is not met, the following steps will be applied:

- The lowest surrogate DCGL from the observed radionuclide mix may be applied to the entire survey unit.

- Additional samples may be collected and analyzed to allow for a detailed analysis and documented evaluation of the radionuclide distribution in order to establish a DCGL specific to that survey unit.
- A corrective action document will be initiated and entered into the corrective action system.

A general expression for the surrogate equation based on recursive relationships is provided by the following equation for i HTD radionuclides. RAI 11

$$DCGL_{surrogate} = \frac{DCGL_{ETD}\prod_{i=1}^{n} DCGL_i}{\prod_{i=1}^{n} DCGL_i + DCGL_{ETD}\sum_{i=1}^{n} f_1 \prod_{\substack{m=1 \\ m\neq i}}^{n} DCGL_m}$$

Equation 5-2

where:

$DCGL_{ETD}$ = the DCGL for the easy-to-detect radionuclide
$DCGL_i$ = the DCGL for the ith hard-to-detect radionuclide
$DCGL_m$ = the DCGL for the mth hard-to-detect radionuclide for which the corresponding f_i is applied
f_i = the activity ratio of the ith hard-to-detect radionuclide to the easy-to-detect radionuclide

Physical or chemical differences between the radionuclides may produce different migration rates, causing the radionuclides to separate and changing the radionuclide ratios. Remediation activities have a reasonable potential to alter the surrogate ratio established prior to remediation. Additional post-remediation samples will be collected to ensure that the data used to establish the ratio are still appropriate and representative of the existing site condition. If these additional post-remediation samples are not consistent with the pre-remediation data, surrogate ratios will be re-established.

Surrogate relationships will be verified by either performing HTD analyses on post-remedial samples (e.g. 6 or more) or by analyzing a minimum of 10% of the FSS samples for HTD. All FSS samples are held in storage on-site until the survey unit is approved for release by the NRC. In the event that additional analyses are required to reconfirm HTD ratios, these FSS samples will be available for analysis. RAI 11

Post-remediation surveying will be accomplished utilizing instrumentation and methodologies consistent with FSS surveying:

- Field screening will be performed using 2350-1 instruments with NaI detectors. Scanning rates will be determined so that activity at the $DCGL_W$ will be detected. Scanning may be performed using the ISOCS provided the assay sensitivity allows for the detection of activity at the $DCGL_W$.
- Field sampling analysis will be performed to the MDC criteria addressed in Section 5.5.3.

The remedial action support survey relies on a simple radiological parameter, such as direct radiation near the surface (i.e. surface scans using a 44-10 detector), as an indicator of effectiveness. The investigation level (the level below which there is an acceptable level of assurance that the established DCGLs have been attained) is determined and used for immediate, in-field decisions. There will be radionuclides and media that cannot be evaluated at the $DCGL_W$ using field monitoring techniques. For these cases, field samples will be collected and analyzed and compared to the release DCGLs. RAI 28

Characterization surveys will be performed of the remediated areas to the rigors of FSS to determine if the area is ready for a FSS (i.e. the area will pass an FSS).

5.2.1.4 *Gross Activity DCGLs*

As a rule, gross activity DCGLs ($DCGL_{GA}$) are developed and applied to structures and plant systems with surface residual radioactivity where multiple radionuclides are present at concentrations that exceed 10 percent of their respective DCGLs. The $DCGL_{GA}$ is determined in a manner similar to surrogate DCGLs, taking into account nuclide detectability to enable field measurement of gross activity, rather than the determination of individual radionuclide activity, for comparison to the radionuclide specific DCGL. The $DCGL_{GA}$, for surfaces with multiple radionuclides is calculated using the following Equation 5-3 (MARSSIM, Equation 4-4):

$$DCGL_{GA} = \frac{1}{\frac{f_1}{DCGL_1} + \frac{f_2}{DCGL_2} + \cdots \frac{f_n}{DCGL_n}}$$

Equation 5-3

where:

f_n= fraction of the total activity contributed by radionuclide n

DCGLn = DCGL for radionuclide n

Different radionuclides or radionuclide combinations may exist on different portions of the site and require the calculation of one or more site-specific $DCGL_{GA}$. $DCGL_{GA}$ are calculated using the relative nuclide fractions determined from samples of building surface or plant system material, as appropriate, prior to remediation. For areas where the radionuclide distribution has not been determined, the most conservative distribution resulting in the lowest DCGL of those specified areas will be used. The distributions are based on the radionuclides identified in samples collected from the specific areas prior to FSS. If new radionuclide distribution data are obtained and determined to be more appropriate for use, the $DCGL_{GA}$ may be reevaluated and altered during the course of the FSS; however, the single nuclide DCGLs will not be revised without NRC approval.

5.2.2 *Classification of Areas*

Prior to beginning the FSS, a characterization of the radiological status and historical review of the site was performed. Additional data may be collected and evaluated throughout the decommissioning. The methods and results from site characterization are described in Chapter 2. Based on the characterization results, the structures and open land areas were classified following the guidance in Appendix A, of NUREG-1757, Volume 2 and Section 4.4 of NUREG-1575. Area classification ensures that the number of measurements and the scan coverage is commensurate with the potential for residual contamination to exceed the unrestricted use criteria. Initial classification of site areas is based on historical information and site scoping and characterization data. Data from operational surveys performed in support of decommissioning, routine surveillance, or any other applicable survey data may be used to change the initial classification of an area up to the time of commencement of the FSS for that area as long as the classification reflects the levels of residual radioactivity that existed prior to remediation. Once the FSS of a given survey unit begins, the basis for any reclassification will be documented, requiring a redesign of the survey unit package, if required (e.g. a Class 3 to a Class 2) and the initiation of a new survey using the redesigned survey unit package. If during the conduct of a FSS, sufficient evidence is accumulated to warrant an investigation and reclassification of the survey unit, in accordance with Section 5.3.6, the FSS may be terminated without completing the current survey unit package.

Reclassification to a more restrictive classification will be performed in accordance with Section 5.3.6.4 of the L TP. New DQOs will be developed with a new survey plan. Reclassification to a more restrictive classification does not require prior NRC notification provided that the Type I error is not increased. The reclassification will be addressed in the new survey plan, as well as the final report on the survey area. RAI 12

5.2.2.1 *Non-Impacted Areas*

Non-impacted areas have no reasonable potential for residual contamination because there was no demonstrable impact from site operations. These areas are not required to be surveyed beyond what has already been completed as a part of the Historical Site Assessment (HSA) as described in Chapter 2, or scoping/site characterization surveys performed to confirm the area's non-impacted classification.

5.2.2.2 *Impacted Areas*

Impacted areas may contain residual radioactivity from licensed activities. Based on the levels of residual radioactivity present, impacted areas are further divided into Class 1, Class 2, or Class 3 designations. The following definitions are from NUREG-1757, Volume 2, Page A2.

- Class 1 Areas: are impacted areas that are expected to have concentrations of residual radioactivity that exceed the *DCGL*
- Class 2 Areas: are impacted areas that are not likely to have concentrations of residual radioactivity that exceed the *DCGL*

- Class 3 Areas: are impacted areas that have a low probability of containing residual radioactivity

If the available information is not sufficient to designate an area as a particular class, the area will either be classified as Class 1 or be further characterized. Areas that are considered to be on the borderline between classes will receive the more restrictive classification.

5.2.2.3 *Initial Classification of Structural Surfaces and Land Areas*

All land areas and structural surfaces to remain after decommissioning were assigned an initial classification. Characterization was performed and reported by initial survey area designation. The area designations developed for the characterization process were used, for the most part, to delineate and classify areas for FSS. This allows characterization data to be efficiently used for final survey area classification and for estimating the sigma value for sample size determination. For operational efficiency, each of the final survey areas listed in Table 5-2 may be subdivided into multiple survey units. The classification of all subdivided survey units will be the same as indicated in Table 5-2, unless reclassified in accordance with this LTP. No individual survey unit will have more than one classification. Areas within the Restricted Area (RA) will require further characterization once demolition activities are in progress. These areas are classified as Class 1 areas and will remain Class 1 areas. Chapter 2 provides the data for the information provided in Table 5-2.

Table 5-2 Survey Area Summary

Survey Area Designator	Name/Building	Total Area Footprint m^2	Classification	σ pCi/g	Mean pCi/g
NOL01	Open land area (inside RA)	7617	1	5.15	2.68
OOL01	Discharge canal south	2471	1	8.55	8.73
OOL02	Intake canal east	628	1	11.48	9.42
OOL03	Open Land Area Outside the RA	1989	1	1.01	0.77
OOL04	Sump Drain Line Land Area	458	1	7.70	11.25
OOL05	Discharge Canal North	556	2	0.90	1.22

Survey Area Designator	Name/Building	Total Area Footprint m^2	Classification	σ pCi/g	Mean pCi/g
OOL06	Intake Center	2047	2	0.04	0.20
OOL07	NOL01 Boundary East	8326	2	0.25	0.27
OOL08	NOL01 Boundary West	6837	2	0.20	0.27
OOL09	Haz. Waste Area	1032	2	1.19	0.48
OOL10	Remainder of Land Area	235,191	3	0.18	0.38
OOL11	Intake West	2470	3	0.08	0.08
OFA	Office Annex	270	3	85*	428*
ISF01	ISFSI Area	5540	3	0.08	0.13
TRB	Training Bldg.	40	3	118*	314*
SEC	Security Bldg.	49	3	101*	326*
MOB	Main Office Bldg.	409	3	89*	348*
CRB	Count Room Bldg.	372	1	TBD**	TBD**
WMB	Waste Management Building	***	1	***	***

* Units are in dpm/100 cm^2
** The building is in use and will require further characterization
*** Building to be constructed. Data will be available once constructed

5.2.2.4 *Changes in Classification*

Initial classification of site areas is based on historical information, scoping surveys, and site characterization data. Data from operational surveys performed in support of decommissioning, routine surveillance, and any other applicable survey data may be used to change the initial classification of an area up to the time of commencement of the FSS for that area as long as the classification reflects the levels of residual radioactivity that existed prior to remediation. Units within initial survey areas may be upgraded in classification due to future requirements for lay down and storage areas during demolition activities or incorrect initial classification. If during FSS, sufficient evidence is accumulated to warrant an investigation and reclassification of the survey unit in accordance with Section 5.3.6, the survey may be terminated without completing the current survey unit package.

5.2.3 *Establishing Survey Units*

The survey units contained within the survey areas are divisions that have similar characteristics and contamination levels. Survey units are assigned only one classification. The site is surveyed and evaluated on a survey unit basis. The site is released on a survey area basis (i.e., through survey area FSS reports).

5.2.3.1 *Survey Unit Size*

Survey unit sizes will be selected based on area classification, survey execution logistics, and applicable regulatory guidance documents. Typical survey unit sizes for structural surfaces and open land area soil are listed in Table 5-3. Survey unit sizes are consistent with NUREG-1575. Class 1 and 2 areas provided in Table 5-2 may be further subdivided into smaller areas to meet the guidelines present in Table 5-3. If survey unit areas larger than the sizes in Table 5-3 are used, a technical evaluation will be presented in the FSS Package for the specific survey unit justifying the survey unit size.

Table 5-3 Suggested Survey Unit Sizes

Class	Structural Surfaces*	Open Land Soil Area
1	100 m^2	2000 m^2
2	100 to 1000 m^2	2000 to 10000 m^2
3	No Limit	No Limit

* Based on floor area

5.2.3.2 *Reference Coordinate System for Open Land Areas (Reference Grid)*

A reference coordinate system is used for impacted areas to facilitate the identification of sample points within the survey unit. The reference coordinate system is basically an X-Y plot of the site area referenced to a fixed structure(s) on the site (e.g., the corner of a building) or by the utilization of a Global Positioning System (GPS) referenced to the State of California Mercator projections. The metadata used is North American Datum (NAD) 83, California zone 1. Elevations are in North American Vertical Datum (NAVD) 88. Once the reference points are established, grids may be overlaid parallel to lines of latitude and longitude.

5.2.4 *Access Control Measures*

5.2.4.1 *Turnover*

Due to the scope of decommissioning activities, it is anticipated that some surveys will be performed in parallel with dismantlement activities. This will require a systematic approach to be established to turnover of the areas. Prior to acceptance of a survey unit for FSS, the following conditions must be satisfied in accordance with applicable procedures:

- Decommissioning activities having the potential to contaminate a survey unit shall be complete or measures taken to eliminate such potential.
- Tools and equipment that would impede the FSS of the survey must be removed, and housekeeping and cleanup shall be complete.
- Decontamination activities in the area shall be complete.
- Access control or other measures to prevent recontamination must be implemented.
- Turnover or remediation surveys may be performed and documented to the same standards as FSS, so that data can be used for the FSS.

When an area is turned over for FSS, an FSS Area Turnover Sheet will be initiated. The Site Closure Manager will ensure all decommissioning activities in areas either adjacent to the area to be isolated or that could otherwise impact it are either complete or deemed not to have the potential to spread plant-related radioactive material to the area. The Site Closure Manager will determine what combination of measures will be employed to prevent recontamination of the FSS area in accordance with HBPP procedure RCP FSS-4. A combination of personnel training (General Employee Training), postings (RCP FSS-4) and periodic surveillance surveys (RCP FSS-13) are some of the measures routinely employed. RAI 14

5.2.4.2 *Walkdown*

The principal objective of the walkdown is to assess the physical scope of the survey unit. The walkdown ensures that the area has been left in the necessary configuration for FSS or that any further work has been identified. The walkdown provides detailed physical information for survey design. Details such as structural interferences or areas requiring special survey techniques can be determined. Specific requirements will be identified for accessing the survey area and obtaining support functions necessary to conduct FSS, such as interference removal or dewatering. Industrial safety and environmental concerns will also be identified during this walkdown.

5.2.4.3 *Transfer of Control*

Once a walkdown has been performed and the turnover requirements have been met, access control to the area is transferred from the HBPP Radiation Protection (RP) Department to the FSS group. Access control and isolation methods are described in the following subsection.

5.2.4.4 *Isolation and Control Measures*

Since all site decommissioning activities will not be completed prior to the start of the FSS, measures will be implemented to protect survey units from contamination during and subsequent to the FSS. Decommissioning activities creating a potential for the spread of contamination will be completed within each survey unit prior to the FSS. Additionally, decommissioning activities that create a potential for the spread of contamination to adjacent areas will be evaluated and controlled. Upon commencement of the FSS for survey units where there is a potential for recontamination, implementation of a combination of the following control measures will be required as needed for appropriate area control:

- Personnel training
- Installation of barriers to control access to surveyed areas
- Installation of barriers to prevent the migration of contamination from adjacent or overhead areas from water runoff, etc.
- Installation of postings requiring contamination monitoring prior to surveyed area access
- Locking entrances to surveyed areas of the facility
- Installation of tamper-evident devices at entrance points

- Periodic surveillance/inspection to monitor and verify adequacy of isolation and control measures
- Installation of postings restricting the introduction of radioactive materials into the area

Periodic surveillances/inspections will not be required for open land areas that are not normally occupied and are unlikely to be impacted by decommissioning activities. If the periodic surveillance/inspection indicates that the adequacy of isolation and control measures has been compromised with the potential for recontamination of the area, post-FSS radiation survey locations will be selectively determined for survey, based on technical or site-specific knowledge and current conditions present in or near the survey area. The selected locations will be surveyed using the same instruments and techniques used for the FSS, and the results will be compared with those obtained during the FSS to determine whether the area had been recontaminated. The primary function of these surveys is to detect the potential migration of contaminants from decommissioning activities taking place in adjacent areas.

5.3 Survey Design and Data Quality Objectives

This section describes the methods and data required to determine the number and location of measurements or samples in each survey unit and the coverage fraction for scan surveys. The design activities described in this section will be documented in a survey package for each survey unit. Survey design considers the following:

- Type I and II Errors
- Scan Survey Coverage
- Sample Size Determination
- Instrumentation and Required Minimum Detectable Concentrations MDCs
- Sample Location
- DCGL and $DCGL_{EMC}$ ($DCGL_{EMC}$ is defined in Section 5.3.6.3)

5.3.1 *Data Quality Objectives (DQOs)*

The appropriate design for a given survey area is developed using the DQO process as outlined in MARSSIM, Appendix D. These seven steps are:

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

The DQO process will be used for designing and conducting all final status surveys at HBPP. Each survey package will contain the appropriate information, statistical parameters, and contingencies to support the DQO process.

5.3.2 *Scan Survey Coverage*

The area covered by scan measurement is based on the survey unit classification as described in NUREG-1757, and as shown in Table 5-4. The accessible area scan required of Class 1 survey units will be 100 percent. For Class 2 survey units, the emphasis will be placed on scanning the higher risk areas such as soils, floors, and lower walls. Scanning percentage of Class 3 survey units will be performed on likely areas of contamination based on the judgment of the FSS Engineer. The FSS Engineer has the discretion to increase the scan coverage beyond 10 percent, if desired.

Table 5-4 Scan Survey Coverage Requirements

	Class 1	Class 2	Class 3
Scan Coverage	100%	10-100 %*	Judgmental (1-10%)

* For Class 2 Survey Units, the amount of scan coverage will be proportional to the potential for finding areas of elevated activity or areas close to the release criterion in accordance with MARSSIM Section 5.5.3. Accordingly, HBPP will use historical information and the results of individual measurements collected during characterization to correlate this activity potential to scan coverage levels.

5.3.3 *Sample Size Determination*

NUREG-1757, Volume 2, Appendix A, describes the process for determining the number of survey measurements necessary to ensure a data set sufficient for statistical analysis. Sample size is based on the relative shift, the Type I and II errors, standard deviation, and the specific statistical test used to evaluate the data.

5.3.3.1 *Determining Which Statistical Test Will Be Used*

Appropriate tests will be used for the statistical evaluation of survey data. Tests such as the Sign test and Wilcoxon

Rank Sum (WRS) Test will be implemented using unity rules, surrogate methodologies, or combinations of unity rules and surrogate methodologies, as applicable, as described in MARSSIM and NUREG-1505 chapters 11 and 12. If the contaminant is not in the background or constitutes a small fraction of the DCGL, the Sign Test will be used. If background is a significant fraction of the DCGL, the WRS Test will be used.

5.3.3.2 *Establishing Decision Errors*

The probability of making decision errors is controlled by hypothesis testing. The survey results will be used to select between one condition of the environment (the null hypothesis) or an alternate condition (the alternative hypothesis). These hypotheses, chosen for MARSSIM Scenario A, are defined as follows:

- Null Hypothesis (H_0): The survey unit does not meet the release criteria.
- Alternate Hypothesis (H_a): The survey unit does meet the release criteria.

HBPP will use the Null Hypothesis concept in the design of all final status surveys.

A Type I decision error would result in the release of a survey unit containing average residual radioactivity above the release criteria. The Type I decision error occurs when the Null Hypothesis is rejected when it is true. The probability of making this error is designated as "α." A Type II decision error would result in the failure to release a survey unit when the average residual radioactivity is below the release criteria. This occurs when the Null Hypothesis is accepted when it is not true. The probability of making this error is designated as "β." Appendix E of NUREG-1757, Volume 2, recommends using a Type I error probability (α) of 0.05 and states that any value for the Type II error probability (β) is acceptable. Following the NUREG-1757, Volume 2, guidance, α will be set at 0.05. A β of 0.05 will be selected initially, based on site-specific considerations. The β may be modified, as necessary, after weighing the resulting change in the number of required survey measurements against the risk of unnecessarily investigating and/or remediating survey units that are truly below the release criteria.

5.3.3.3 *Relative Shift*

The relative shift (Δ / σ) is calculated. Delta (Δ) is equal to the $DCGL_W$ minus the Lower Boundary of the Gray Region (LBGR). Calculation of sigma (σ) is discussed in Section 5.3.3.3.2 and initial values are provided in Table 5-2. The sigma values used for the relative shift calculation may be recalculated based on the most current data obtained from post remediation or post-demolition surveys or from background reference areas, as appropriate. The LBGR is initially set at 0.5 times the *DCGL*, but may be adjusted to obtain an optimal value, normally between 1 and 3 for the relative shift.

5.3.3.3.1 Lower Boundary of the Gray Region

The LBGR is the point at which the Type II (β) error applies. The default value of the LBGR is set initially at 0.5 times the DCGL. If the relative shift is greater than 3, then the number of data points, N, listed for the relative shift values of 3 from Table 5-5 or Table 5-3 in MARSSIM, will normally be used as the minimum sample size. If the minimum sample size results in a sample density less than the required minimum density, the sample size will be increased accordingly.

5.3.3.3.2 Standard Deviation (Sigma)

Sigma values (estimate of the standard deviation of the measured values in a survey unit and/or reference area) were initially calculated from characterization data. These sigma values can be used in FSS design or more current post-remediation sigma values can be used. The use of the sigma values from the characterization data will be conservative for the sample size determination since the post-remediation sigma values are expected to be smaller.

5.3.3.3.3 *Wilcoxon Rank Sum (WRS) Test Sample Size*

The number of data points, N, to be obtained from each reference area or survey unit are determined using Table 5-3 in MARSSIM. This table includes the recommended 20 percent adjustment to ensure an adequate sample size.

5.3.3.3.4 *Sign Test Sample Size*

The number of data points is determined from Table 5-5 in MARSSIM for application of the Sign Test. This table includes the recommended 20 percent adjustment to ensure an adequate sample size.

5.3.3.3.5 *Elevated Measurement Comparison Sample Size Adjustment*

If the Scan MDC is greater than the $DCGL_W$, the sample size will be calculated using Equation 5-4 (NUREG-1757, Equation A-8) provided below. If N_{EMC} exceeds the statistically determined sample size *(N)*, N_{EMC} will replace *N*.

$$N_{EMC} = \frac{A}{A_{EMC}}$$

Equation 5-4

where:

N_{EMC} = the elevated measurement comparison sample size

A = the survey unit area

A_{EMC} = the area corresponding to the area factor calculated using the MDC concentration

5.3.4 *Background Reference Area*

Background reference area measurements are required when the WRS test is used, and background subtraction may be used with the Sign Test under certain conditions such as those described in Chapter 12 of NUREG-1505. Reference area measurements, if needed, will be collected using the methods and procedures required for Class 3 final survey units. For soil, reference areas will have a soil type as similar to the soil type in the survey unit as possible.

When there is a reasonable choice of possible soil reference areas with similar soil types, consideration will be given to selecting reference areas that are most similar in terms of other physical, chemical, geological, and biological characteristics. For structure survey units that contain a variety of materials with markedly different backgrounds, a reference area will be selected containing similar materials. If one material is predominant, or if there is not a large variation in background among materials, a background from a reference area containing a single material is appropriate when it is demonstrated that the selected reference area will not result in underestimating the residual radioactivity in the survey unit.

It is understood that background reference areas should have physical characteristics (including soil type and rock formation) similar to the site and shall not contain areas contaminated by site activities. Offsite areas (outside the HBPP Owner Controlled Area) should be chosen to serve as background reference areas.

Should significant variations in background reference areas be encountered, appropriate evaluations will be performed to define the background concentration. As noted in NUREG-1757, Appendix A, Section A.3.4, the Kruskal-Wallis test can be conducted in such circumstances to determine that there are no significant differences in the mean background concentrations among potential reference areas. HBPP will consider this and other statistical guidance in the evaluation of apparent significant variations in background reference areas.

If material background subtraction is performed, the sigma value used will account for the variability of the material background.

5.3.5 *Reference Grid and Sample Location*

Sample location is a function of the number of measurements required, the survey unit classification, and the contaminant variability.

5.3.5.1 *Reference Grid*

The reference grid is primarily used for reference purposes and is illustrated on sample maps. Physical marking of the reference grid lines in the survey unit will be performed only when necessary. For the sample grid in Class 1 and Class 2 survey units, a randomly selected sample start point will be identified. Beginning at the random starting coordinate, a row of points is identified, parallel to the X-axis, at intervals of L, the grid spacing. A second row of points is then developed, parallel to the first row, at a distance of 0.866 x L from the first row. The sample and reference grids are illustrated on sample maps and may be physically marked in the field. For Class 3 survey units, all sample locations are randomly selected, based on the reference grid point(s). GPS instruments will be used in open land areas to determine reference or sample grid locations within the survey area. Locations within a survey area also may be tied to a site United States Geological Survey (USGS) survey benchmark. Digital cameras may be employed to provide a record of survey locations within the survey unit and will be used extensively at HBPP.

5.3.5.2 *Measurement Locations*

Measurement locations within the survey unit are clearly identified and documented for purpose of reproducibility. Actual measurement locations are identified by tags, labels, flags, stakes, paint marks, geopositioning units, or photographic records. An identification code matches a survey location to a particular survey unit.

Sample points for Class 1 and Class 2 survey units are positioned in a systematic pattern or grid throughout the survey unit by first randomly selecting a start point coordinate. A random number generator is used to determine the start point of the grid pattern. The grid spacing, L, is a function of the area of the survey unit as shown in Equation 5-5 (MARSSIM Equation 5-5) for a triangular grid:

$$L = \sqrt{\frac{A}{0.866\, n_{EA}}}$$

Equation 5-5

Where:

A = Area of the survey unit

n_{EA} = Calculated number of survey locations

Beginning at the random starting coordinate, a row of points is identified, parallel to the X-axis, at intervals of L. A second row of points is then developed, parallel to the first row, at a distance of 0.866 x L from the first row.

Software may be used to generate grid patterns and sample/measurement locations (i.e., Visual Sample Plan (VSP)).

Random measurement patterns are used for Class 3 survey units. Sample location coordinates (x and y) are randomly picked using a random number generator or VSP.

Measurement locations selected using either a random selection process or a randomly started systematic pattern that do not fall within the survey unit or that cannot be surveyed due to site conditions are replaced with other measurement locations as determined by the FSS Supervisor, FSS Engineer, or designee.

5.3.6 *Investigation Levels and Elevated Areas Test*

During survey unit measurements, levels of radioactivity may be identified that warrant investigation. Depending on the results of the investigation, the survey unit may require no action, remediation, and/or reclassification and resurvey. The following subsections describe investigation process and investigation levels.

5.3.6.1 *Investigation Process*

During the survey process, locations with potential residual activity exceeding investigation levels are documented and marked for further investigation. The elevated survey measurement is verified by resurvey. For Class 1 areas, size and average activity level in the elevated area is acceptable if it complies with the area factors and other criteria that may apply to evaluation of the DCGL for elevated measurements $DCGL_{EMC}$. As discussed in Section 5.3.6.3, the $DCGL_{EMC}$ is applicable only for Class 1 areas. If any location within a Class 2 area exceeds the DCGL, scanning coverage in the vicinity is increased in order to determine the extent and level of the elevated reading(s) and the area is evaluated for reclassification. If the elevated reading occurs in a Class 3 area, the scanning coverage is increased and the area is evaluated for reclassification and resurvey under the criteria of the new classification. All survey unit investigations will be conducted in accordance with the applicable FSS DQOs.

Investigations should address the following items:

- The assumptions made in the survey unit classification
- The most likely or known cause of the contamination
- The effects of summing multiple areas with elevated activity within the survey unit

Depending on the results of the investigation, a portion of the survey unit may be reclassified or combined with an adjacent area with similar characteristics if there is sufficient justification. Either action would result in resurvey of the (new) area(s). The results of the investigation process are documented in the survey package. Section 5.6 provides additional discussion regarding potential reclassification of the survey unit.

5.3.6.2 *Investigation Levels*

Technicians will respond to all instrument indications of elevated activity while surveying. Upon receiving an indication, the technician will stop and resurvey the last square meter of area surveyed to verify the increase. Technicians are cautioned, in training, about the importance of the verification survey and are given specific direction in the procedure as to survey extent and scan speed. If the indication is verified, the technician will mark the area with a flag or other appropriate means. Each area marked will be addressed in an investigation survey instruction prepared for the survey unit. The instruction will specify the required actions, such as a rescan of the area, direct measurements, and collection of a soil sample (for land surveys). Each investigation will be evaluated and reported in the FSS survey area report. Investigation levels are shown in Table 5-5.

Table 5-5 Investigation Levels

Classification	Scan Investigation Levels	Direct Investigation Levels
Class 1	> $DCGL_{EMC}$	> $DCGL_{EMC}$ or > $DCGL_W$ and > a statistical parameter-based value
Class 2	> $DCGL_W$ or > MDC_{SCAN} if MDC_{SCAN} is greater than $DCGL_W$	> $DCGL_W$
Class 3	Detectable over Background	> 0.5 $DCGL_W$

In Class 1 areas, the size and average activity level in the elevated area is determined to demonstrate compliance with the area factors. If any location in a Class 2 area exceeds the DCGL, scanning coverage in the vicinity is increased in order to determine the extent and level of the elevated reading(s). If the elevated reading occurs in a Class 3 area, the scanning coverage is increased and reclassification of the area should be considered.

5.3.6.3 *Elevated Measurement Comparison (EMC)*

5.3.6.3.1 Open Land Areas and Structural Surfaces

The elevated measurement comparison is applied to Class 1 survey units when one or more verified scan or static measurement exceeds the investigation level. As stated in MARSSIM, the EMC is intended to flag potential failures in the remediation process and should not be considered the primary means to identify whether or not a survey unit meets the release criterion. The EMC provides assurance that unusually large measurements receive the proper attention and that any area having the potential for significant dose contribution is identified. Locations identified by scan methodology or soil sample analyses measurements with levels of residual radioactivity that exceed the $DCGL_{EMC}$ are subject to additional surveys to determine compliance with the elevated measurement criteria. The size of the area containing the elevated residual radioactivity and the average level of residual activity within the area are determined. The average level of activity is compared to the $DCGL_W$ based on the actual area of elevated activity. An a priori $DCGL_{EMC}$ for the area between direct measurements (the likely size of an elevated area) is established during the survey design and is calculated as follows:

$$DCGL_{EMC} = Area\ Factor \times DCGL_W$$

Equation 5-6

The area factor is the multiple of the $DCGL_W$ that is permitted in the area of elevated residual radioactivity without remediation. The area factor is related to the size of the area over which the elevated activity is distributed. The actual area is generally bordered by levels of residual radioactivity below the $DCGL_W$ and its size is determined during the investigation process. Area factor calculations are described in Section 6.6 and summarized in Tables 5-6 and 5-7. The actual area of elevated activity is determined by investigation surveys and the area factor is adjusted for the actual area of elevated activity. The product of the adjusted area factor and the $DCGL_W$ determines the a posteriori $DCGL_{EMC}$. Additional measurements are made to determine the average activity of the elevated area, if necessary. If the $DCGL_{EMC}$ is exceeded, the area is remediated and resurveyed. The results of the elevated area investigations in a given survey unit that are below the $DCGL_{EMC}$ limit are evaluated using

Equation 5-6. If more than one elevated area is identified in a given survey unit, the unity rule with Equation 5-6 is used to determine compliance. If the formula value is less than unity, no further elevated area testing is required and the EMC test is satisfied.

Table 5-6 Soil Area Factors

Radionuclide of Concern (ROC)	Area Factor for Area Contaminated Zone (m^2):							
	2000	**1000**	**500**	**100**	**50**	**10**	**5**	**1**
Am-241	1.0E+00	1.0E+00	2.0E+00	8.7E+00	1.6E+01	4.9E+01	7.7E+01	1.9E+02
C-14	1.0E+00	1.5E+00	4.0E+00	4.2E+01	1.1E+02	1.0E+03	2.5E+03	1.8E+04
Cm-243	1.0E+00	1.0E+00	1.6E+00	3.4E+00	4.3E+00	7.3E+00	1.1E+01	3.2E+01
Cm-244	1.0E+00	1.0E+00	2.0E+00	9.7E+00	1.9E+01	7.6E+01	1.2E+02	2.8E+02
Cm-245	1.0E+00	1.0E+00	1.9E+00	6.2E+00	9.2E+00	1.9E+01	3.0E+01	8.1E+01
Cm-246	1.0E+00	1.0E+00	2.0E+00	9.7E+00	1.9E+01	7.6E+01	1.3E+02	2.8E+02
Co-60	1.0E+00	1.0E+00	1.1E+00	1.3E+00	1.4E+00	2.2E+00	3.3E+00	1.0E+01
Cs-137	1.0E+00	1.0E+00	1.3E+00	1.7E+00	1.9E+00	3.0E+00	4.5E+00	1.4E+01
Eu-152	1.0E+00	1.0E+00	1.0E+00	1.1E+00	1.3E+00	2.0E+00	3.0E+00	9.1E+00
Eu-154	1.0E+00	1.0E+00	1.0E+00	1.2E+00	1.3E+00	2.0E+00	3.0E+00	9.2E+00
H-3	1.0E+00	1.1E+00	2.1E+00	1.0E+01	2.1E+01	1.0E+02	2.0E+02	9.3E+02
I-129	1.0E+00	1.1E+00	2.2E+00	1.1E+01	2.2E+01	9.9E+01	1.9E+02	8.3E+02
Nb-94	1.0E+00	1.0E+00	1.0E+00	1.2E+00	1.3E+00	2.0E+00	3.0E+00	9.0E+00
Ni-59	1.0E+00	1.2E+00	2.3E+00	1.2E+01	2.3E+01	1.2E+02	2.3E+02	1.2E+03
Ni-63	1.0E+00	1.2E+00	2.3E+00	1.2E+01	2.3E+01	1.2E+02	2.3E+02	1.2E+03
Np-237	1.0E+00	1.0E+00	2.0E+00	9.0E+00	1.6E+01	5.6E+01	9.8E+01	3.5E+02
Pu-238	1.0E+00	1.0E+00	2.0E+00	9.7E+00	1.9E+01	7.6E+01	1.3E+02	2.8E+02
Pu-239	1.0E+00	1.0E+00	2.0E+00	9.7E+00	1.9E+01	7.6E+01	1.3E+02	2.9E+02
Pu-240	1.0E+00	1.0E+00	2.0E+00	9.7E+00	1.9E+01	7.6E+01	1.3E+02	2.9E+02
Pu-241	1.0E+00	1.0E+00	2.0E+00	8.8E+00	1.6E+01	4.9E+01	7.8E+01	1.9E+02
Sr-90	1.0E+00	1.0E+00	2.0E+00	1.0E+01	2.0E+01	9.9E+01	2.0E+02	9.6E+02
Tc-99	1.0E+00	1.0E+00	2.0E+00	1.0E+01	2.0E+01	1.0E+02	2.0E+02	1.0E+03

Table 5-7 Building Surfaces Area Factors

	Area Factor Value:										
(m^2)	Am-241	C-14	Cm-243	Cm-244	Cm-245	Cm-246	Co-60	Cs-137	Eu-152	Eu-154	H-3
1	9.7E+01	9.7E+01	8.9E+01	1.0E+02	3.9E+01	6.4E+01	1.3E+01	1.5E+01	1.3E+01	1.3E+01	1.0E+02
2	4.9E+01	4.9E+01	4.5E+01	5.0E+01	2.1E+01	3.3E+01	7.2E+00	8.2E+00	7.2E+00	7.2E+00	5.0E+01
3	3.3E+01	3.3E+01	3.0E+01	3.3E+01	1.5E+01	2.3E+01	5.3E+00	6.0E+00	5.3E+00	5.3E+00	3.3E+01
4	2.5E+01	2.4E+01	2.3E+01	2.5E+01	1.2E+01	1.8E+01	4.3E+00	4.9E+00	4.3E+00	4.3E+00	2.5E+01
5	2.0E+01	2.0E+01	1.9E+01	2.0E+01	9.9E+00	1.5E+01	3.7E+00	4.2E+00	3.7E+00	3.7E+00	2.0E+01
6	1.6E+01	1.6E+01	1.6E+01	1.7E+01	8.6E+00	1.2E+01	3.3E+00	3.8E+00	3.3E+00	3.3E+00	1.7E+01
8	1.2E+01	1.2E+01	1.2E+01	1.2E+01	6.9E+00	9.7E+00	2.8E+00	3.2E+00	2.8E+00	2.8E+00	1.2E+01
10	9.9E+00	9.9E+00	9.5E+00	1.0E+01	5.9E+00	8.0E+00	2.5E+00	2.8E+00	2.5E+00	2.5E+00	1.0E+01
50	2.0E+00	2.0E+00	2.0E+00	2.0E+00	1.8E+00	1.9E+00	1.2E+00	1.3E+00	1.2E+00	1.2E+00	2.0E+00
100	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00
	Area Factor Value:										
(m^2)	I-129	Nb-94	Ni-59	Ni-63	Np-237	Pu-238	Pu-239	Pu-240	Pu-241	Sr-90	Tc-99
1	6.5E+01	1.3E+01	1.0E+02	1.0E+02	8.9E+01	1.0E+02	1.0E+02	1.0E+02	9.8E+01	9.0E+01	8.7E+01
2	3.4E+01	7.2E+00	5.0E+01	5.0E+01	4.5E+01	5.0E+01	5.0E+01	5.0E+01	4.9E+01	4.5E+01	4.4E+01
3	2.3E+01	5.3E+00	3.3E+01	3.3E+01	3.0E+01	3.3E+01	3.3E+01	3.3E+01	3.3E+01	3.0E+01	3.0E+01
4	1.8E+01	4.3E+00	2.5E+01	2.5E+01	2.3E+01	2.5E+01	2.5E+01	2.5E+01	2.5E+01	2.3E+01	2.3E+01
5	1.5E+01	3.7E+00	2.0E+01	2.0E+01	1.8E+01	2.0E+01	2.0E+01	2.0E+01	2.0E+01	1.9E+01	1.8E+01
6	1.3E+01	3.3E+00	1.7E+01	1.7E+01	1.6E+01	1.7E+01	1.7E+01	1.7E+01	1.7E+01	1.6E+01	1.5E+01
8	9.7E+00	2.8E+00	1.3E+01	1.2E+01	1.2E+01	1.3E+01	1.2E+01	1.2E+01	1.2E+01	1.2E+01	1.2E+01
10	8.0E+00	2.5E+00	1.0E+01	1.0E+01	9.4E+00	1.0E+01	1.0E+01	1.0E+01	9.9E+00	9.5E+00	9.4E+00
50	1.9E+00	1.2E+00	2.0E+00	2.0E+00	2.0E+00	2.0E+00	2.0E+00	2.0E+00	2.0E+00	2.0E+00	2.0E+00
100	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00

Equation 5-7 applies to a single radionuclide contaminant. When multiple radionuclides are present, the calculation in Equation 5-7 is made with a unitized DCGL.

$$\frac{\delta}{DCGL_W} + \frac{(Conc_{AVE} - \delta)}{(Area\ Factor)(DCGL_W)} < 1$$

Equation 5-7

where:

δ = Estimate of average concentration of residual radioactivity, and

$Conc_{AVE}$= average concentration in elevated area.

If more than one elevated area exists in the survey unit, a separate term will be included for each in Equation 5-7 (refer to Section 5.6.2.2).

5.3.6.3.2 Embedded/Buried Piping

DCGLs for HBPP embedded and buried piping will be in accordance with HBPP Technical Basis Documents (TBDs). The HBPP embedded/buried piping DCGL TBD will be submitted to NRC for approval prior to implementation.

5.3.6.4 *Remediation and Reclassification*

As shown in Table 5-8, Class 1 areas of elevated residual activity above the $DCGL_{EMC}$ are remediated to reduce the residual radioactivity to acceptable levels. Based on survey data, it may be necessary to remediate an entire survey unit or only a portion of it. If an individual survey measurement (scan or direct measurement) in a Class 2 survey unit exceeds the $DCGL_W$, the survey unit or a portion of it may be evaluated for a change of classification to a Class 1 survey unit and the survey redesigned and re-performed accordingly. If an individual survey measurement in a Class 3 survey unit exceeds 0.5 $DCGL_W$, the survey unit, or portion of a survey unit, will be evaluated, and if necessary, reclassified to a Class 2 survey unit and the survey redesigned and re-performed accordingly.

Table 5-8 Investigative Actions for Individual Survey Units

Area Classification	Action if Investigation Results exceed:		
	$DCGL_{EMC}$	$DCGL_W$	0.5 $DCGL_W$
Class 1	Remediate and re-survey as necessary	Acceptable*	N/A
Class 2	Remediate, reclassify portions as necessary, and investigate**	Reclassify portions as necessary and investigate**	N/A
Class 3	Remediate, reclassify portions as necessary ,and investigate**	Reclassify portions as necessary, increase scan coverage, and investigate**	Reclassify portions as necessary and resurvey, increase scan coverage

* For individual measurements above DCGL, the Sign Test will be conducted on the survey unit and an EMC evaluation performed.
**Requires an investigation of the initial classification process and a survey unit evaluation of sufficient intensity to satisfy the requirements of new classification status.

5.3.6.5 *Resurvey*

Following an investigation, if a survey unit is reclassified to a more restrictive classification or if remediation activities were performed, a resurvey is performed in accordance with approved procedures. If a Class 2 area had contamination greater than the $DCGL_W$, the area should be reclassified to a Class 1 area. If the average value of Class 2 direct survey measurements was less than the $DCGL_W$, the Scan MDC was sensitive enough to detect the $DCGL_{EMC}$ and there were no areas greater than the $DCGL_{EMC}$, the survey redesign may be limited to obtaining a 100 percent scan without having to re-perform the static measurements or soil sample analyses. This condition assumes that the sample density meets the requirements for a Class 1 area.

5.4 Survey Methods and Instrumentation

5.4.1 *Survey Measurement Methods*

Survey measurements and sample collection are performed by personnel trained and qualified in accordance with the applicable HBPP procedures. The techniques for performing survey measurements or collecting samples are specified in approved HBPP procedures. FSS measurements include surface scans, direct surface measurements, and gamma spectroscopy of volumetric materials. Advanced Survey Technologies, not specifically described in this LTP also may be used for final status surveys. If so, HBPP will give NRC 30 days notice to provide an opportunity to review the associated basis document that will be provided on the Advanced Survey Technology(s). Onsite, as well as offsite, laboratory facilities are used for gamma spectroscopy, liquid scintillation, and gas proportional counting in accordance with applicable procedures. "Approved" off-site facilities "as required by Section 5.8" are used as necessary. No matter which facilities are used, analytical methods will be administratively established to detect levels of radioactivity at 10 percent to 50 percent of the DCGL value.

5.4.2 *Structures*

Structures will receive scan surveys, direct measurements, and, when necessary, volumetric sampling.

5.4.2.1 *Scan Surveys*

Scanning is performed in order to locate small elevated areas of residual activity above the investigation level. Structures are scanned for beta/gamma radiation with appropriate instruments such as those listed in Table 5-9. The measurements will typically be performed at a distance of 1 cm or less from the surface and at a nominal scan speed of 5 cm/sec for hand-held instruments. Adjustments to scan speed and distance may be made in accordance with approved technical guidance.

5.4.2.2 *Direct Measurements*

Direct measurements are performed to detect surface activity levels. Direct measurements are conducted by placing the detector on or very near the surface to be counted and acquiring data over a predetermined count time. A count time of one minute is typically used for HBPP surface measurements and generally provides detection levels well below the DCGL (the count time may be varied provided the required detection level is achieved).

5.4.2.3 *Concrete with Activated Radionuclides*

Activated concrete that does not meet FFS criteria at HBPP will be removed and shipped to a suitable burial site.

5.4.2.4 *Volumetric Concrete Measurements*

Volumetric sampling of contaminated concrete, as opposed to direct measurements, may be necessary if the efficiency or uncertainty of the gross beta measurements is too high.

In this case, the surface layer is removed from the known area by using a commercial stripping agent (coated surfaces) or by physically abrading the surface. The removed coating material is analyzed for activity content and the level converted to appropriate units (i.e., dpm/100 cm^2) for comparison with surface activity DCGLs. Direct measurements can then be performed on the underlying surface after removal of the coating.

The thickness of the layer of building surface to be removed as a sample should be consistent with the development of the HBPP site model and the DCGLs (i.e., less than 10mm in depth).

Input parameters in the RESRAD-BUILD model assumes all the activity at the surface (LTP, Chapter 6, Appendix B), therefore it would be appropriate to posit that the activity is less than 10 mm in depth. For the radionuclides-of-concern, a 10 mm thickness provides a minimal degree of shielding. RAI 16

5.4.2.5 *Soils*

Soil will receive scan surveys at the coverage level described in Table 5- 4 and volumetric samples will be taken at designated locations. Surface soil samples will normally be taken at a depth of 0 to 15 cm. Samples will be collected and prepared in accordance with approved procedures.

5.4.2.5.1 **Scans**

Open land areas are scanned for gamma emitting nuclides. The gamma emitters are used as surrogates for the HTD radionuclides. Sodium iodide detectors are typically used for scanning. For detectors such as the Ludlum 44-10, the detector is held within 2.5 to 7.5 cm off the ground surface and is moved at a speed of 0.5 m/sec, traversing each square meter three times. The area covered by scan measurements is based on the survey unit classification, as described in Section 5.3.2.

5.4.2.5.2 *Volumetric Samples*

Soil materials are analyzed by gamma spectroscopy. Soil samples of approximately 1,500 grams are normally collected from the surface layer (top 15 cm). Sample preparation includes removing extraneous material, homogenizing, and drying the soil for gamma isotopic analysis. Separate containers are used for each sample and each container is moved through the analysis process following site procedures. Samples are split, when required, by the HBPP FSS Quality Assurance Project Plan (QAPP). If a survey area has already been excavated and remediated to the soil DCGL, this area will be treated as surface soil, and the FSS will be performed on the excavated area. Soil samples will be collected to depths at which there is high confidence that deeper samples will not result in higher concentrations.

Alternatively, a sodium-iodide detector or in situ object counting system (ISOCS) of sufficient sensitivity to detect DCGL concentrations may be used to identify the potential presence of subsurface contamination (i.e. greater than 15 cm in depth) triggering an investigation. RAI 18

All subsurface sampling will be performed in accordance with the guidance in Section G.2.1 of NUREG- 1757, Volume 2. The sample size for subsurface samples will be determined using the same methods described for surface soil. Per NUREG-1757, Volume 2, scanning is not applicable to subsurface areas; however, HBPP FSS will employ scanning techniques commensurate with the survey unit classification. Scanning subsurface soils, where accessible as an excavated surface, will be used for characterization data.

Soil sample depth will be determined during the DQO phase of the survey design. Surface soil samples will normally be taken at a depth of 0 to 15 cm. Areas of potential subsurface soil contamination (e.g. areas identified where spills were present, areas RAI 17 found to contain contamination during remediation, etc.) may require sampling at a depth exceeding 15 cm up to a depth of 1 meter. If contamination below 15 cm is suspected, split spoon sampling or similar methods, will be used for the final survey.

The advantage of a conservative statistical MARSSIM approach for surficial soils may be lessened by potential waste migration along channels or lineation to subsurface sites. Areas where subsurface activity exists at levels challenging the release criteria will require additional geological and historical assessments or additional sampling, as identified in the DQO process. If HBPP intends to use subsurface samples for FSS compliance purposes, potential complications will be considered in the DQO process, and additional subsurface soil sampling/assessment details will be provided to the NRC on a case-by-case basis to ensure that sampling and evaluation methods are appropriate.

5.4.2.5.3 *Alternative Survey Plan in Excavations*

Over the course of the decommissioning project at HBPP, there will be instances where deep excavations are made. These are necessary to remove radiologically contaminated soils and to remove both clean and contaminated foundations and underground utilities. Due to the instability of the soils and seismic risks present, shoring or trench boxes will be required for personnel access to ensure safety to personnel entering the excavation. The shoring and/or trench boxes will prevent the survey of 100% of the surface areas in Class 1 survey areas, primarily the walls of steep excavations. In order to assess the residual activity present in these areas safely the following methodology will be utilized: RAI 20

- The excavations will be remediated until soil characterization indicates values are less than the release criteria. The contaminated media removed will be disposed of as waste material.

- Soil that must be removed below the above excavated depth may be removed and either surveyed as a Class 1 material (I.e. 100% survey) at 6 inch lifts or surveyed by a bulk monitor system for reuse. A TBD will be developed for the bulk assay system and submitted to the NRC prior to being used. RAI 20
- FSS will be performed on the bottom of the excavation prior to any backfill.
- Sides wall soils where shoring or trench boxes limit safety of scanning will be assessed by combinations of soils removed from within the trench, soils attached to the exterior of the boxes/shoring as removed, or specific depth sampling of soils behind shoring on a case-by-case basis.

Note: Where known contaminated systems may exist below the remediated soils level or unidentified underground utilities are encountered and deemed to be potentially contaminated, then additional measurements will be taken during the excavation to provide for appropriate remediation.

5.4.3 *Specific Survey Area Considerations*

5.4.3.1 *Pavement-Covered Areas and Shallow Concrete Slabs*

Survey of paved areas will be required along the roadways providing ingress and egress to HBPP. The survey design of paved/concrete areas will be based on soil survey unit sizes since they are outdoor areas where the exposure scenario is most similar to direct radiation from surface soil. The applicable DCGL will be the soil DCGL. Scan and static gamma and beta-gamma surveys are determined by the survey unit design. Samples will be obtained of not only the asphalt/concrete, but of the soil present under the asphalt/concrete. Paved areas may be separate survey units or they may be incorporated into surveys of adjacent open land areas of like classification.

5.4.3.2 *Bulk Materials*

Controls will be instituted to prevent mixing of soils from different survey areas prior to evaluation. Soils satisfying the criteria for unrestricted release will be stockpiled for use as onsite backfill material. (Class 2 material could be used in either Class 1 or 2 areas and Class 1 material could only be used in Class 1 areas.) The radiological evaluation of soils resulting from minor trenching and digging efforts in Class 3 defined areas (no reasonable potential for subsurface contamination) will be performed by characterization survey in accordance with site procedures. Excavated soils that demonstrate residual radioactivity consistent with Class 3 status will be released for use as onsite excavation backfill. RAI 20

All soil stockpiles at HBPP are under the Stormwater Pollution Prevention Plan Best Management Practices and are therefore required to have a lower and upper cover and be waddled in the middle. RAI 21

5.4.3.3 *Embedded Piping and Buried Piping*

Embedded and buried piping may remain after decommissioning HBPP. Separate FSS survey plans will be developed for embedded/buried piping, which will include survey unit DQOs. These FSS plans will include the following items:

- radionuclides of interest and chosen surrogate
- levels and distribution of contamination
- internal surface condition of the piping
- internal residues and sediments and their radiation attenuation properties
- removable and fixed surface contamination
- instrument sensitivity and related scan and fixed minimum detectable concentrations
- piping geometry and presence of internally inaccessible areas/sections
- instrument calibration

Accessible internal surfaces are surveyed the same as other structural surfaces. Scale and sediment samples will be obtained, if appropriate, as well as smears and wipes to assist in the identification of the total radionuclide deposits within the piping. The activity of the internal surfaces will be compared to the building surface DCGLs, which is a conservative measure. If the amount of activity observed on the internal surfaces is so great as to fail a survey unit, specialized embedded piping DCGLs will be developed in a technical basis document. Some buried piping, storm drains, sewer systems, plumbing and floor drains may be free released or assessed. All remaining embedded and buried piping will be grouted after surveying unless it is to be used as an active system (e.g., drainage piping).

5.4.3.4 *Cracks, Crevices, Wall-to-Floor Interfaces, and Small Holes*

Surface contamination on irregular structure surfaces (e.g., cracks, crevices, and holes) is difficult to survey directly. Where no remediation has occurred and residual activity has not been detected above background, these surface blemishes may be assumed to have the same level of residual activity as that found on adjacent surfaces. The accessible surfaces are surveyed in the same manner as other structural surfaces and no special corrections or adjustments are required. In situations where remediation has taken place or where residual activity has been detected above background, a representative sample of the contamination within the crack or crevice may be obtained or an adjustment for instrument efficiency may be made. If an instrument efficiency adjustment cannot be justified based on the depth of contamination or other geometry factors, volumetric samples will be collected. As an alternative method, radionuclide specific analysis, coupled with application of the unity rule, may be used. Volumetric samples analyzed by gamma spectroscopy will detect the presence of radioactivity below the surface. Typically, such sampling is performed following removal of paint and other surface coatings during remediation. After analysis, the data may be converted to equivalent surface activity. The accessible surfaces on irregular structure surfaces are surveyed in the same manner as other structure surfaces except that they are included in areas receiving judgmental scans when scanning is performed over less than 100 percent of the area.

5.4.3.5 *Paint Covered Surfaces*

Painted surfaces will be evaluated prior to the start of the FSS for that survey unit. In the event of suspected activity beneath painted surfaces, the coating will be removed prior to performing the survey. No special consideration will be given to wall or ceiling areas painted before plant startup and which have not been subjected to repeated exposure to materials that would have penetrated the painted surface. If the thickness of the coating can be determined with certainty, then a source efficiency correction may be applied to the measurement as described in NUREG-1507.

5.4.3.6 *Exterior Surfaces of Building Foundations*

Exterior surfaces of below-grade foundations will be evaluated using the historical site assessment and other pertinent records to determine the potential for sub-surface contamination on the exterior surfaces of below-grade foundations. One method available to evaluate the exterior surfaces is the use of core bores through foundation or walls and the taking of soil samples at locations having a high potential for the accumulation and migration of radioactive contamination to sub-surface soils. These biased locations for soil and concrete assessment could include stress cracks, floor and wall interfaces, penetrations through walls and floors for piping, run-off from exterior walls, and leaks or spills in adjacent outside areas, etc. If the soil is found to be free of residual radioactivity at the biased locations, it will be assumed that the exterior surface of the foundation is also free of residual activity. Otherwise, additional sampling may be necessary to determine the extent of decontamination and remediation efforts. Another method available for evaluating the exterior surfaces of below-grade foundations is gamma well logging. Soil in biased locations next to the exterior of the buildings may be evaluated using this technique. This technique can provide for rapid isotopic analysis of soils without sampling.

The HBPP Unit 3 caisson will be removed. The caisson structure is a Class 1 structure. The structure will have a disposition survey performed in order to determine the appropriate burial site. The structure will not undergo an FSS since it will not be present on-site at the time of license termination. The excavation will undergo a Class 1 FSS. RAI 23

5.4.3.7 *Groundwater*

Assessments of any residual activity in groundwater at HBPP will be via groundwater monitoring wells. The monitoring wells installed at the site will monitor groundwater at both deep and shallow depths. Section 2.2.2 describes the groundwater monitoring conducted.

The data collected from the monitoring wells will be used to ensure that the concentration of well water available, based upon the well supply requirements assumed in Section 6 for the resident farmer (i.e. resident farmer's well), is below the U.S. Environmental Protection Agency (EPA) maximum contaminant levels (MCLs) (e.g., 20,000 pCi/l for H-3). This will ensure that the dose contribution from groundwater is a small fraction of the limit in 10CFR20.1402.

5.4.4 *Instrumentation*

Radiation detection and measurement instrumentation for the FSS is selected to provide both reliable operation and adequate sensitivity to detect the radionuclides identified at the site at levels sufficiently below the DCGL. Detector selection is based on detection sensitivity, operating characteristics, and expected performance in the field. The instrumentation will, to the extent practicable, use data logging. Commercially available portable and laboratory instruments and detectors typically are used to perform the three basic survey measurements: (1) surface scanning, (2) direct surface contamination measurements, (3) and spectroscopy of soil and other bulk materials, such as concrete.

HBPP procedures control the issuance and use of instrumentation. Records supporting the instrumentation program are maintained in accordance with HBPP procedures.

5.4.4.1 *Instrument Selection*

Radiation detection and measurement instrumentation is selected based on the type and quantity of the radiation to be measured. The instruments used for direct measurements are capable of detecting the radiation of concern below the applicable DCGL. MDCs of less than 50 percent of the DCGL allow detection of residual activity in Class 3 survey units at an investigation level of 0.5 times the DCGL. Instruments used for scan measurements in Class 1 areas are required to be capable of detecting radioactive material at the $DCGL_{EMC}$. Instrumentation currently proposed for use in the HBPP FSS is listed in Table 5-9. Instrument MDCs are discussed in Section 5.4.3.4 and nominal MDC values are listed in Table 5-10. Other measurement instruments or techniques may be used. The acceptability of any alternate technologies for use in the FSS Program will be justified in a technical basis evaluation document. Technical basis evaluations for Advanced Survey Technologies will be provided for NRC review 30 days prior to use. An instrument technical analysis will include the following:

- Description of the conditions under which the method would be used
- Description of the measurement method, instrumentation, and criteria
- Justification that the technique would provide the required sensitivity for the given survey unit classification in accordance with Table 5-10
- Demonstration that the instrument provides sufficient sensitivity for measurement below the release criteria with Type I error equivalent to 5 percent or less

Table 5-9 Typical FSS Instrumentation

Measurement Type	Detector Type	Effective Detector Area and Window Density	Instrument and Model	Detector Model
Alpha Scan	Gas-flow Proportional	126 cm^2 0.8 mg/cm^3 Aluminized Mylar	Ludlum 2350-1	Ludlum 43-68
Beta-gamma static and scan	Gas-flow Proportional	126 cm^2 0.8 mg/cm^3 Aluminized Mylar	Ludlum 2350-1	Ludlum 43-68
Beta-gamma scan	Gas-flow Proportional	584 cm^2 0.8 mg/cm^3 Aluminized Mylar	Ludlum 2350-1	Ludlum 43-37
Gamma scan	Scintillation	2" diameter x 2" length NaI	Ludlum 2350-1	Ludlum 44-10
Soil, structure surface and bulk material	High purity germanium	N/A	Canberra and off site Laboratory	N/A

Table 5-10 Typical FSS Detection Sensitivities

Instruments and Detectors	Radiation	Background Count Time (minutes)	Background (cpm)	Instrument Efficiency (2pi)	Count Time (minutes)	Static MDC (dpm/100 cm^2)	Scan MDC (dpm/100 cm^2)
Model 43-68	Alpha	1	2	0.1500	5	61	N/A
Model 43-68	Beta-Gamma	1	300	0.3200	1	920	13291
Model 43-37	Beta-Gamma	1	600	0.2800	1	320	7267
Model 4410	Gamma	1	4000	0.0350	0.04	N/A	See Table 5-13 for E_i
HPGe	Gamma	Up to 60	N/A	0.40 Relative	10-60	0.15	N/A
Tennelec Low Bkg. Counter	Alpha Beta	20 20	0.175 3.9	0.348 0.377	3 3	<11 <16	N/A N/A

5.4.4.2 *Calibration and Maintenance*

Instruments and detectors are calibrated by HBPP for the radiation types and energies of interest at the site. Approved suppliers will calibrate instruments, as necessary that will be utilized for analysis under their approved Quality Assurance Program as described in Section 5.8. Calibration may also be performed in accordance with approved procedures at HBPP or Diablo Canyon Power Plant (DCPP). The calibration source for beta survey instruments is Cs-137, because the average beta energies approximate the beta energy of the radionuclides found on surfaces at HBPP. The alpha calibration source is Am-241 that has an appropriate alpha energy for plant-specific alpha emitting nuclides. Gamma scintillation detectors are calibrated using Cs-137. Radioactive sources used for calibration are traceable to the National Institute of Standards and Technology (NIST). When characterized High Purity Germanium (HPGe) detectors are used, using approved procedures, suitable NIST-traceable sources are used for onsite calibration, and the software is set up appropriately for the desired geometry.

5.4.4.3 *Response Checks*

Instrumentation response checks are conducted to ensure proper field survey instrument response and operation. An acceptable response for field instrumentation is an instrument reading within plus or minus 20 percent of the established check source value as documented on a control chart. Response checks are performed daily before instrument use and again at the end of use. Check sources contain the same type of radiation of that being measured in the field and are held in fixed geometry jigs for reproducibility. If an instrument fails a response check, it is tagged "Out of Service" to prevent inadvertent use and is removed from service until the problem is corrected in accordance with applicable HBPP procedures. Measurements made between the last acceptable check and the failed check will be evaluated to determine if they should remain in the data set.

5.4.4.4 *Minimum Detectable Concentration*

The MDC is determined for the instruments and techniques used for final status surveys (Table 5-9). The MDC is the concentration of radioactivity that an instrument can be expected to detect 95 percent of the time.

5.4.4.4.1 Static MDC for Structure Surfaces

For static (direct) surface measurements, with conventional detectors, such as those listed in Table 5-9, the MDC is calculated by Equation 5-8 as follows:

$$MDC_{static} = \frac{3 + 4.65\sqrt{B}}{(K)(t)}$$

Equation 5-8

where:

3 = Poisson probability sum for α and β squared and corrected to 3 (Brodsky 1992)

MDCstatic = minimum detectable concentration for direct counting (dpm/100 cm^2)

B = number of background counts during the count interval t

t = count interval (for paired observations of sample and blank, usually 1 minute)

K = calibration constant (counts/min per dpm/100 cm^2) The value of *K* includes correction factors for efficiency (e_i and e_s). The value of e_s is dependent on the material type. Corrections for radionuclide absorption have been made.

5.4.4.4.2 Structural Surface Beta-Gamma Scan MDCs

Following the guidance of Sections 6.7 and 6.8 of NUREG- 1507, MDCs for surface scans of structural surfaces for beta and gamma emitters will be computed by Equation 5-9. For determining Scan MDCs, a rate of 95 percent of correct detections is required and a rate of 60 percent of false positives is determined to be acceptable: therefore, a sensitivity index value of 1.38 was selected from Table 6.1 of NUREG-1507 and Equation 5-9 becomes:

$$MDC_{structural\ surface\ scan}(dpm/cm^2) = \frac{1.38\sqrt{B}}{\sqrt{p}\, e_i e_s \left(\frac{A}{100}\right) t}$$

Equation 5-9

where:

B = number of background counts during the count interval t

p = surveyor efficiency

e_i = instrument efficiency (2π) for the emitted radiation (cpm per dpm)

e_s = source efficiency (intensity) in emissions per disintegration

A = sensitive area of the detector (cm^2)

t = time interval of the observation while the probe passes over the source (minutes)

The numerator in Equation 5-9 represents the minimum detectable count rate that the observer would "see" at the performance level represented by the sensitivity index. The surveyor efficiency (p) will be taken to be 0.5, as recommended by Section 6.7.1 of NUREG-1507. The factor of 100 corrects for probe areas that are not 100 cm^2. In the case of a scan measurement, the counting interval is the time the probe is actually over the source of radioactivity. This time depends on scan speed, the size of the source, and the fraction of the detector's sensitive area that passes over the source, with the latter depending on the direction of probe travel. The source efficiency term (e_s) in Equation 5-9 may be adjusted to account for effects such as self absorption, as appropriate.

5.4.4.4.3 *Total Efficiency (e_t) and Source Efficiency (e_s) for Concrete Contamination*

The source term inventory on contaminated concrete appears to be primarily located within the top few millimeters of the concrete surface.

The practical application of choosing the proper instrument efficiency may be determined by averaging the surface variation (peaks and valleys narrower than the length of the detector) and adding 0.5 centimeter, the spacing that should be maintained between the detector and the highest peaks of the surface. Selection of the source to detector distance is based on Table 5-11 that best reflects the predetermined geometry.

Table 5-11 Source to Detector Distance Effects on Instrument Efficiencies for α/β Emitters

Source to Detector Distance (cm)	Instrument Efficiency ei	
	Cs-137 Distributed	Am-241 Distributed
Contact	(1) (2π eff)	(1)(2π eff)
0.5	(0.894)(2π eff)	(0.833)(2π eff)
1.0	(0.816)(2π eff)	(0.724)(2π eff)
2	(0.659)(2π eff)	(0.362) (2π eff)

Source efficiency (e_s) reflects the physical characteristics of the surface and any surface coatings. The source efficiency is the ratio between the number of particles emerging from surface and the total of particles released within the source. The source efficiency accounts for attenuation and backscatter. Source efficiency (e_s) is nominally 0.5 (no self-absorption/attenuation, no backscatter). Backscatter increases the value, self-absorption decreases the value. Source efficiencies may either be derived empirically or simply selected from the guidance contained in International Organization for Standardization (ISO) 7503-1 (Reference 5-8). ISO 7503-1 takes a conservative approach by recommending the use of factors to correct for alpha and beta self-absorption/attenuation when determining surface activity. However, this approach may prove to be too conservative for radionuclides with max beta energies that are marginally lower than 0.400 MeV, such as Co-60 with a βmax of 0.3 18 MeV. In this situation, it may be more appropriate to determine the source efficiency by considering the energies of other beta emitting radionuclides. Using this approach, it is possible to determine weighted average source efficiency. For example, a source efficiency of 0.375 may be calculated based on a 50/50 mix of Co-60 and Cs-137. The source efficiencies for Co-60 and Cs-137 are 0.25 and 0.5 respectively, since the radionuclide fraction for Co-60 and Cs-137 is 50 percent for each, the weighted average source efficiency for the mix may be calculated in the following manner:

$$(.25)(.50) + (.50)(.50) = 0.375$$

Table 5-12 lists the ISO 7503-1 source efficiencies.

Table 5-12 Source Efficiencies as Listed in ISO 7503-1

	>.400 Mev_{max}	**<.400 Mev_{max}**
Beta Emitters	$e_s = 0.50$	$e_s = 0.25$
Alpha Emitters	$e_s = 0.25$	$e_s = 0.25$

The total efficiency for any given condition can now be calculated from the product of the instrument efficiency e_i and the source efficiency e_s.

$$e_{total} = (e_i)\,(e_s)$$

Equation 5-10

where:

e_{total} = Total efficiency
e_i = Instrument efficiency
e_s = Source efficiency

5.4.4.4.4 Structural Surfaces Alpha Scan MDCs

In cases where alpha scan surveys are required, MDCs must be quantified differently from those for beta-gamma surveys because the background count rate from a typical alpha survey instrument is nearly zero (1 to 3 counts per minute, typically). Since the time that an area of alpha activity is under the probe varies and the background count rates of alpha survey instruments is so low, it is not practical to determine a fixed MDC for scanning. Instead, it is more useful to determine the probability of detecting an area of contamination at a predetermined DCGL for given scan rates. For alpha survey instrumentation with a background around 1 to 3 counts per minute, a single count will give a surveyor sufficient cause to stop and investigate further. Thus, the probability of detecting given levels of alpha emitting radionuclides can be calculated by use of Poisson summation statistics (see MARSSIM Section 6.7.22 and Appendix J for details). Doing so, one finds that the probability of detecting an area of alpha activity of 300 dpm/100 cm^2 at a scan rate of 3 cm per second (roughly 1 inch per second) is 90 percent if the probe dimension in the direction of the scan is 10 cm. If the probe dimension in the scan direction is halved to 5 cm, the detection probability is still 70 percent. Choosing appropriate values for surveyor efficiency, instrument and surface efficiencies will yield MDCs for alpha surveys for structure surfaces. If for some reason lower MDCs are desired, then scan speeds may be adjusted, within practicable limits, via the methods of Section 6.7.2.2 and Appendix J of MARSSIM.

5.4.4.4.5 *Open Land Area Gamma Scan MDCs*

In addition to the minimum detectable count rate (MDCR) and detector characteristics, the Scan MDC (in pCi/g) for land areas is based on extent of the elevated area, depth of the elevated area, and the radionuclide (i.e., energy and yield of gamma emissions). If one assumes constant parameters for each of these variables, with the exception of the specific radionuclide in question, the Scan MDC may be reduced to a function of the radionuclide alone. The evaluation of open land areas requires a detection methodology of sufficient sensitivity for the identification of small areas of potentially elevated activity. Scan measurements are performed by passing a 2-inch x 2-inch NaI (Tl) gamma scintillation detector in gross count rate mode across the land surface under investigation. The centerline of the detector is maintained at a source-to-detector distance of approximately 6 cm and moved from side to side in a 1 meter wide pattern at a rate of 0.5 m/sec. This serpentine scan pattern is designed to cross each survey cell (one square meter [1 m^2]) a minimum of three times in approximately 10 seconds with a maximum separation of less than 150 cm between one pass. The audible signal is monitored for detectable increases in count rate. An observed count rate increase results in further investigation to verify findings and define the level and extent of residual radioactivity. This method represents the Stage 1 and Stage 2 surface scanning process for land areas defined in NUREG-1507 and is the basis for calculation of the scanning detection sensitivity (Scan MDC). The sensitivity is only slightly affected by the relative amounts of Cs-137 and Co-60 in the soil, giving typical Scan MDC values in the range of 5 to 6 pCi/g for instrument backgrounds of 8,000 to 10,000 cpm. Alternative methods of sufficient sensitivity for the identification of small areas of elevated radioactivity may be used where appropriate.

An a priori determination of scanning sensitivity is performed to ensure that the measurement system is able to detect concentrations of radioactivity at levels below the regulatory release limit. Expressed in terms of scan MDC, this sensitivity is the lowest concentration of radioactivity for a given background that the measurement system is able to detect at specified performance level and surveyor efficiency.

The Scan MDC value (in pCi/g) for open land area surface scanning with a desired performance level of 95 percent correct detections and 60 percent false positive rate, the sensitivity index has a value of 1.38, resulting in the following MDCR:

$$MDCR = 1.38\sqrt{b_i} \times \left(\frac{60\ sec}{1\ min}\right)$$

Equation 5-11

where:

b_i = background counts in the observation interval

Introducing the human factor performance element of surveyor efficiency, the surveyor minimum detectable count rate becomes:

$$MDCR_{surveyor} = \frac{MDCR}{\sqrt{p}}$$

Equation 5-12

Where:

$MDCR_{surveyor}$ = Minimum detectable surveyor count rate (cpm), and

p = Surveyor efficiency = *0.5*

A corresponding minimum detectable exposure rate can be determined for a specified detector and radionuclide by dividing the $MDCR_{surveyor}$ value by the detector manufacturer's count rate to exposure rate ratio (cpm per µR/h) to give a minimum detectable exposure rate in units of µR/h. The minimum detectable exposure rate then is used to determine the minimum detectable radionuclide concentration (i.e., the Scan MDC) by modeling a specified small area of elevated activity using MicroShield™ to yield a conversion factor (E_i) of cpm per pCi/g. The minimum detectable exposure rate is then divided by the MicroShield™ conversion factor to give a Scan MDC in units of pCi/g.
Table 5-13 provides the E_i for HBPP predominant gamma emitting radionuclides as determined by HBPP Technical Base Documents (TBD) *Instrument Efficiency Determination for use in Minimum Detectable*

Concentration Calculations in Support of the Final Status Survey at HBPP.

Table 5-13 Efficiency for Photon Emitting Isotopes

Isotope	E_i (cpm/pCi/g)
Co-60	315
Nb-94	387
Cs-137	202
Eu-152	419
Eu-154	230

5.4.4.4.6 *HPGe Spectrometer Analysis*

The onsite chemistry laboratory maintains gamma isotopic spectrometers that are calibrated to various sample geometries, including one liter Marinelli geometry for soil analysis. These systems are calibrated using a National Institute of Standards and Technology (NIST) traceable mixed gamma source using approved procedures. The detectors are manufactured by Canberra Industries. Approved off-site laboratories may also be used to perform gamma analyses.

Laboratory counting system count times are set to meet a maximum MDC of 10 percent of the DCGL for HBPP radionuclides.

5.4.4.4.7 *Pipe Survey Instrumentation*

Accessible portions of any remaining embedded piping will be surveyed to ensure residual remaining activity is less than the DCGL. Pipe survey instruments proposed for use at HBPP are scintillation detectors and/or Geiger-Mueller (GM) arrays. Pipe survey instruments proposed for use will have a level of sensitivity adequate to detect residual activity below the embedded piping DCGLs.

Class 1 piping will be surveyed at 1 foot intervals with 100 percent coverage. Inaccessible portions will be made accessible by cutting access ports in the piping. In Class 2 and Class 3 piping where 100 percent coverage is not required, an evaluation

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will be performed as to the percent of survey.

5.5 Data Collection and Processing

This section describes data collection, review, validation, and record keeping requirements for final status surveys.

5.5.1 *Sample Handling and Record Keeping*

A chain-of-custody (COC) record will accompany each sample from the collection point through obtaining the final results to ensure the validity of the sample data. COC records are controlled and maintained in accordance with applicable procedures.

Each survey unit has an associated document package that covers the design and field implementation of the survey requirements. Survey unit records are considered quality records.

5.5.2 *Data Management*

Survey data are collected from several sources during the data life cycle and are evaluated for validity throughout the survey process. QC replicate measurements are not used as FSS data. Measurements performed during turnover and investigation surveys can be used as FSS data if they were performed according to the same requirements as the FSS data, as follows:

- Survey data shall reflect the as-left survey unit condition (i.e., no further remediation required).
- The application of isolation measures to the survey unit to prevent recontamination and to maintain final configuration are in effect.
- The data collection and design were in accordance with FSS methods and procedures, (e.g., Scan MDC, investigation levels, survey data point number and location, statistical tests, and EMC tests or as specified by the LAR submitted for the HBGS).

Measurement results stored as final status survey data constitute the final survey of record and are included in the data set for each survey unit used for determining compliance with the site release criteria. Measurements are recorded in units appropriate for comparison to the applicable DCGL. Numerical values, even negative numbers, are recorded. Measurement records include, at a minimum, the surveyor's name, the location of the measurement, the instrument used, measurement results, the date and time of the measurement, any surveyor comments, and records of applicable reviews.

5.5.3 *Data Verification and Validation*

The FSS data are reviewed prior to data assessment to ensure that they are complete, fully documented, and technically acceptable. The review criteria for data acceptability will include at a minimum, the following items:

- The instrumentation MDC for fixed or volumetric measurements are less than 10% of the DCGL (preferable) while MDCs up to 50% of the DCGL are acceptable. RAI 26
- The instrument calibration was current and traceable to NIST standards.
- The field instruments were source checked with satisfactory results before and after use each day data were collected or data were evaluated accordance with Section 5.4.4.3.
- The MDCs and assumptions used to develop them were appropriate for the instruments and techniques used to perform the survey.
- The survey methods used to collect data were appropriate for the types of radiation involved and for the media being surveyed.
- "Special methods" for data collection were properly applied to the survey unit under review. These special methods are described in this LTP section or will be the subject of an NRC notice of opportunity for review.
- The sample was controlled from the point of sample collection to the point of obtaining results.
- The data set is comprised of qualified measurement results collected in accordance with the survey design, which accurately reflects the radiological status of the facility.
- The data have been properly recorded.

If the data review criteria were not met, the discrepancy will be reviewed and the decision to accept or reject the data will be documented, reviewed, and approved by the FSS Supervisor.

5.5.4 *Graphical Data Review*

Survey data will be graphed to identify patterns, relationships, or possible anomalies that might not be apparent using other methods of review. A posting plot and a frequency plot will be made. Other special graphical representations of the data set will be made as the need dictates. The FSS Supervisor will review all data for acceptance.

5.5.4.1 *Posting Plots*

Posting plots will be used to identify spatial variability in the data. The posting plot consists of the survey unit map with the numerical data shown at the location from which it was obtained. Posting plots can reveal areas of elevated radioactivity or local areas in which the DCGL is exceeded. Posting plots can be generated for background reference areas to point out spatial trends that might adversely affect the use of the data. Anomalies in the background data may be the result of residual, undetected activity, or may just reflect background variability.

5.5.4.2 *Frequency Plots*

Frequency plots will be used to examine the general shape of the data distribution. Frequency plots are basically bar charts showing data points within a given range of values. Frequency plots reveal such things as skewness and bimodality (having two peaks). Skewness may be the result of a few areas of elevated activity or may be the result of very little activity present in the survey unit such as a log-normal data distribution. Multiple peaks (bi-modal, tri-modal, etc.) in the data may indicate the presence of isolated areas of residual radioactivity or background variability due to soil types or differing materials of construction. Variability may also indicate the need to match background reference areas to survey units more carefully or to subdivide the survey unit by material or soil type.

5.5.4.3 *Contour and 3-D Surface Plots*

Contour and 3-D surface plots may be used to represent graphically a trend in collected survey data. This can be an aid in visualizing the location of activity outside the area that affects the collected data. Contour and 3-D surface plots typically require that a plotting algorithm be applied to interpolate data at a predetermined frequency.

5.6 Data Assessment and Compliance

An assessment is performed on the FSS data to ensure that they are adequate to support the determination to release the survey unit. Simple assessment methods such as comparing the survey data to the DCGL or comparing the mean value to the DCGL are first performed. The statistical tests are then applied, as necessary, to the final data set and conclusions are made as to whether the survey unit meets the site release criterion.

5.6.1 *Data Assessment Including Statistical Analysis*

The results of the survey measurements are evaluated to determine whether the survey unit meets the release criteria. In some cases, the determination can be made without performing complex, statistical analyses.

5.6.1.1 *Interpretation of Sample Measurement Results*

An assessment of the measurement results is used to determine quickly whether the survey unit passes or fails the release criteria or whether one of the statistical analyses must be performed. The evaluation matrices are presented in Tables 5-14 and 5-15.

Table 5-14 Interpretation of Sample Measurements When the WRS Test is Used

Measurement Results	Conclusion
Difference between maximum survey unit concentration and minimum reference area concentration is less than $DCGL_W$	Survey Unit meets the release criteria
Difference of survey unit average concentration and reference average concentrations greater than $DCGL_W$	Survey Unit fails
Difference between any survey unit concentration and any reference area concentration is greater than $DCGL_W$. A difference of survey unit average concentration and reference area average concentration is less than $DCGL_W$	Conduct WRS test and elevated measurements test

Table 5-15 Interpretation of Sample Measurements When the Sign Test is Used

Measurement Results	Conclusion
All concentrations less than $DCGL_W$	Survey Unit meets the release criteria
Average concentration greater than $DCGL_W$	Survey Unit fails
Any concentration greater than $DCGL_W$ and average concentration less than $DCGL_W$	Conduct Sign test and elevated measurements test

When required, one of four non-parametric statistical tests will be performed on the survey data:

1. WRS Test
2. Sign Test
3. WRS Test Unity Rule
4. Sign Test Unity Rule

In addition, survey data are evaluated against the EMC criteria as previously described in Section 5.3.6.3 and as required by NUREG-1757, Volume 2. The statistical test is based on the null hypothesis (H_0) that the residual radioactivity in the survey unit exceeds the DCGL. There must be sufficient survey data at or below the DCGL to reject the null hypothesis and conclude the survey unit meets the site release criterion for dose. Statistical analyses are performed using a specially designed software package or, if necessary, using hand calculations.

5.6.1.2 *Wilcoxon Rank Sum Test*

The WRS test, or WRS Unity Rule (NUREG-1505, Chapter 11), may be used when the radionuclide of concern is present in the background or measurements are used that are not radionuclide-specific. In addition, this test is valid only when "less than" measurement results do not exceed 40 percent of the data set.

The WRS test is applied as follows:

1. The background reference area measurements are adjusted by adding the $DCGL_W$ to each background reference area measurement, X_i; (i.e., $Z_i = X_i + DCGL$).
2. The number of adjusted background reference area measurements, m, and the number of survey unit measurements, n, are summed to obtain N, (N =m + n).
3. The measurements are pooled and ranked in order of increasing size from 1 to N. If several measurements have the same value, they are assigned the average rank of that group of measurements.
4. The ranks of the adjusted background reference area measurements are summed to obtain Wr.
5. The value of Wr, is compared with the critical value in Table I.4 of MARSSIM. If Wr, is greater than the critical

value, the survey unit meets the site release dose criterion. If Wr, is less than or equal to the critical value, the survey unit fails to meet the criterion.

5.6.1.3 *Sign Test*

The Sign test and Sign test Unity Rule are one-sample statistical tests used for situations in which the radionuclide of concern is not present in background, or is present at acceptable low fractions compared to the $DCGL_W$. If present in background, the gross measurement is assumed to be entirely from plant activities. This option is used when it can be reasonably expected that including the background concentration will not affect the outcome of the Sign test. The advantage of using the Sign test is that a background reference area is not necessary. The Sign test is conducted as follows:

1. The survey unit measurements, X_i, i = 1, 2, 3,...N; where N = the number of measurements, are listed.
2. X_i is subtracted from the $DCGL_W$ to obtain the difference $D_i = DCGL_W - X_i$, where i = 1, 2, 3,..., N.
3. Differences where the value is exactly zero are discarded and N is reduced by the number of such zero measurements.
4. The number of positive differences is counted. The result is the test statistic S+.

   Note: A positive difference corresponds to a measurement below the $DCGL_W$ and contributes evidence that the survey unit meets the site release criterion.

5. The value of S+ is compared to the critical value given in Table I.3 of MARSSIM. The table contains critical values for given values of N and α. The value of α is set at 0.05 during survey design. If S+ is greater than the critical value given in the table, the survey unit meets the site release criterion. If S+ is less than or equal to the critical value, the survey unit fails to meet the release criterion.

5.6.2 *Unity Rule*

5.6.2.1 *Multiple Radionuclide Evaluations*

The Cs-137 to Co-60 (or other gamma nuclide) ratio will vary in the final survey soil samples, and this will be accounted for using a "unity rule" approach as described in NUREG-1505 Chapter 11. Unity Rule Equivalents will be calculated for each measurement result using the surrogate adjusted Cs-137 DCGL and the Co-60 DCGL, as shown in the following Equation 5-13.

$$Unity\ Rule\ Equivalent\ \leq 1 = \frac{Cs-137}{DCGL_{Cs-137s}} + \frac{Co-60}{DCGL_{Co-60}} + \cdots + \frac{R_n}{DCGL_n}$$

Equation 5-13

where:

Cs-137 and Co-60 are the gamma results

$DCGL_{Cs\text{-}137\,s}$ = the surrogate Cs-137, DCGL, as applicable

$DCGL_{Co\text{-}60}$ = the Co-60 DCGL

R_n = any other identified gamma emitting radionuclide

$DCGL_n$ = the DCGL for radionuclide N

The unity rule equivalent results will be used to demonstrate compliance, assuming the DCGL is equal to 1.0 using the criteria listed in Tables 5-14 and 5-15. If the application of the WRS or Sign test is necessary, these tests will be applied using the unity rule equivalent results and assuming that the DCGL is equal to 1.0. An example of a WRS test using the unity rule is provided in NUREG-1505, Page 11-3; Section 11.4. (If the WRS Test was used, or background subtraction was used in conjunction with the Sign Test, background concentrations also would be converted to Unity Rule Equivalents prior to performing test). The Sign Test will be used without background subtraction if background Cs-137 is not considered a significant fraction of the DCGL.

Note that the surrogate Cs-137 DCGL will be used for both the statistical tests and comparisons with the criteria in Tables 5-13 and 5-14. The same general surrogate and unity rule methods described previously for soil would be applied to other materials, such as activated concrete, where sample gamma spectroscopy is used for final survey as opposed to gross beta measurements.

5.6.2.2 *Elevated Measurement Comparison Evaluations*

During final surveys, areas of elevated activity may be detected and they must be evaluated both individually and in total to ensure compliance with the release criteria. Each elevated area is compared to the specific $DCGL_{EMC}$ value calculated for the size of the specific elevated area. If the individual elevated area passes, then the elevated areas are combined and evaluated under the unity rule.

The average activity of each elevated area is determined as well as the average value for the survey unit. The survey unit average value is divided by the $DCGL_W$, the survey unit average value is subtracted from the elevated area average activity value, and the result is divided by the elevated area $DCGL_{EMC}$. Each elevated area net average activity is evaluated against its $DCGL_{EMC}$. The fractions are summed and the result must be less than unity for the survey unit to pass. This is summarized in Equation 5-14.

$$\frac{\delta}{DCGL} + \frac{\bar{C}_{elevated} - \delta}{(Area\ Factor) \times DCGL} < 1$$

Equation 5-14

Where:

δ = average concentration outside the elevated area

$\bar{C}_{elevated}$ = average concentration in the elevated area

A separate term will be used in the equation for each elevated area identified in a survey unit.

5.6.3 *Data Conclusions*

The results of the statistical tests, including application of the EMC, allow one of two conclusions to be made. The first conclusion is that the survey unit meets the site release dose criterion. The data provide statistically significant evidence that the level of residual radioactivity in the survey unit does not exceed the release criterion. The decision to release the survey unit is made with sufficient confidence and without further analysis. The second conclusion that can be made is that the survey unit fails to meet the release criterion. The data are not conclusive in showing that the residual radioactivity is less than the release criterion. The data are analyzed further to determine the reason for the failure.

Possible reasons include the following:

- The average residual radioactivity exceeds the $DCGL_W$.
- The average residual radioactivity is less than the $DCGL_W$; however the survey unit fails the elevated measurement comparison.
- The survey design or implementation was insufficient to demonstrate compliance for unrestricted release.
- The test did not have sufficient power to reject the null hypothesis (i.e., the result is due to random statistical fluctuation).

The power of the statistical test is a function of the number of measurements made and the standard deviation in measurement data. The power is determined from 1-β where β is the value for Type II errors. A retrospective power analysis may be performed using the methods described in Appendix I.9 of MARSSIM.

If the power of the test is insufficient due to the number of measurements, additional samples may be collected as directed by procedure. A greater number of measurements increase the probability of passing if the survey unit actually meets the release criterion. Retrospective power analyses will be developed for each HBPP survey unit, regardless if the unit passes FSS criteria or not.

If failure was due to the presence of residual radioactivity in excess of the release criterion, the survey unit shall be remediated and as necessary, reclassified. Survey unit failure due to inadequate design or implementation shall require investigation and re-initiation of the FSS process.

5.6.4 *Compliance*

The FSS is designed to demonstrate licensed radioactive materials have been removed from the HBPP site to the extent that remaining residual radioactivity is below the radiological criteria for unrestricted release. The site-specific radiological criteria presented in this plan demonstrate compliance with the criteria of 10 CFR 20.1402. If the measurement results pass the requirements of Section 5.6.1 and

5.6.1.2 and the elevated areas evaluated per Section 5.6.2.2 pass the elevated measurement comparison, the survey unit is suitable for unrestricted release. If survey measurements do not meet the criteria specified in Table 5-5, an investigation will be performed. Investigations will include an evaluation of survey design, instrumentation use, and calculations, as necessary. Investigations of this nature will be documented in accordance with the HBPP FSS QA Plan.

5.7 Final Status Survey Reporting Format

Survey results and a brief operating history are documented in the FSS Report. Other reports may be generated as requested by NRC.

5.7.1 *Operating History*

A brief operational history including relevant operational and decommissioning data is compiled. The purpose of the historical information is to provide additional, substantive data that form a portion of the basis for the survey unit classification, and hence, the level of intensity of the FSS. The historical information includes operating history that could affect radiological status, summarized scoping and site characterization data, and other relevant information, as deemed necessary.

5.7.2 *Final Status Survey Report*

Survey results will be described in a written report for each Survey Area and submitted to the NRC. Upon completion of each survey area the FSS report provides a summary of the survey results and the overall conclusions that demonstrate that the HBPP site meets the radiological criteria for unrestricted use. Information such as the number and type of measurements, basic statistical quantities, and statistical analysis results are included in the report. The level of detail is sufficient to describe clearly the FSS program and to certify the results. The format of the final report will contain, as a minimum, the following topics:

- Overview of the results
- Discussion of changes to FSS
- FSS Methodology
    - Survey unit sample size
    - Justification for sample size
- FSS Results

  - Number of measurements taken
  - Survey maps
  - Sample concentrations
  - Statistical evaluations
  - Judgmental and miscellaneous data sets
- Anomalous data
- Conclusion for each survey unit
- Any changes from initial assumptions on extent of residual activity

In accordance with plant procedure HBAP C-202, the final report will provide the unit specific or generic ALARA evaluation as well as any investigation performed, regardless of whether the survey unit failed or not. RAI 27

5.7.3 *Other Reports*

Other reports relating to FSS activities may be prepared and submitted as necessary.

5.8 Final Status Survey Program Quality (QAPP)

Quality is built in to each phase of the FSS Program and measures must be taken during the execution of the plan to determine whether the expected level of quality is being achieved. The FSS Program will ensure that the site will be surveyed, evaluated, and determined to be acceptable for unrestricted release if the residual activity results in an annual TEDE to the average member of the critical group of 25 mrem/year or less for all pathways and is ALARA. The following sections provide a description of applicable HBPP quality programs and specific quality elements of the FSS Program.

5.8.1 *FSS Quality Assurance Project Plan*

The objective of the FSS QAPP is to ensure the survey data collected are of the type and quality needed to demonstrate with sufficient confidence the site is suitable for unrestricted release. The objective is met through use of the DQO process for FSS design, analysis, and evaluation. The plan ensures the following items are accomplished:

- The elements of the FSS plan are implemented in accordance with approved procedures and survey instructions.
- Surveys are conducted by trained personnel using calibrated instrumentation.

- The quality of the data collected is adequate.
- All phases of package design and survey are properly reviewed, with management oversight provided.
- Corrective actions, when identified, are implemented in a timely manner and are determined to be effective.

The following sections describe the basic elements of the FSS QAPP.

5.8.1.1 *Project Management and Organization*

Compliance with the QAPP and the LTP shall be the responsibility of all personnel involved with FSS activities. The HBPP staff performs the following specific responsibilities. Outside vendors may be contracted to perform specific review activities such as the following:

- Perform surveillance of the implementation of the FSS
- Performing periodic audits of the FSS program
- Perform conformance reviews of selected FSS implementing procedures
- Perform conformance reviews of selected FSS reports

The HBPP FSS Organization is responsible for the quality of those activities necessary to achieve a final status of unrestricted use for the HBPP site.

The following are key FSS positions. The responsibilities for the key positions are described in HBAP C-225, "Final Status Survey Program" and responsibilities may be assigned to a designee as appropriate.

- HBPP Site Closure Manager
- FSS Supervisor
- FSS Engineers
- FSS Foreman

Figure 5-2 provides an organizational chart of the projected HBPP License Termination Organization.

5.8.1.2 *Program Controls*

Program Controls shall be established for performing specific FSS activities. Activities will be accomplished using suitable instructions, procedures, and drawings that incorporate appropriate regulatory and industry guidance.

Personnel conducting activities shall be appropriately trained and qualified. Training, qualification, and any appropriate maintenance of proficiency requirements shall be defined in administrative procedures or instructions. Professional resumes, other verifiable credentials, and/or discrete certification packages, as applicable, shall be used to document personnel qualifications.

5.8.1.3 *Design Controls*

Design control requirements are established to ensure that the applicable regulatory bases, codes, technical standards, and quality standards are identified in the FSS. Design controls also include independent verification and design interface control. These design controls will be implemented to determine the DCGLs, MDCs, area factors, and other DQO and FSS elements.

5.8.1.4 *Procurement Document Control*

Procurement documents related to the FSS shall be prepared in accordance with approved procedures and instructions. These procedures and instructions shall contain provisions to ensure that procurement documents include or reference applicable regulatory requirements and any other requirement necessary to guarantee adequate quality for the purchased service, equipment, or material.

5.8.1.5 *Instructions, Procedures, and Drawings*

The performance of the FSS will require procedures for personnel training, survey implementation, data collection, COC, instrument calibration and maintenance, verification, and record storage. These procedures will ensure compliance with the LTP and will meet applicable quality requirements. These quality requirements include the development and approval in accordance with the site controls.

5.8.1.6 *Document Control*

Instructions, Procedures, and Drawings shall be controlled as described in approved procedures or instructions. Controlled copies shall be available for use by personnel performing activities affecting the FSS Program. These controls shall ensure that only current information is issued and used. The results of the FSS will be retained at least for the duration of the 10 CFR 50 facility license.

5.8.1.7 *Control of Purchased Material, Items, and Services*

Vendors may be used for the performance of the FSS and laboratory activities. Quality related services, such as laboratory analysis, are procured from qualified vendors whose internal QA program is subject to approval in accordance with approved procedures. Additionally, audits and surveillances of these contractors should be performed to provide an adequate level of assurance that the quality activities are being effectively performed and conform to the requirements of the procurement document.

5.8.1.8 *Control of Special Processes*

Procedures will be developed to implement any special processes that may be used in support of FSS implementation. The special processes used will be validated and implemented by trained, qualified individuals using approved procedures.

5.8.1.9 *Inspections*

Inspections and verification activities will be delineated in implementing procedures. These programs and procedures will be used to verify that sampling and surveying protocols are appropriately performed. Appropriate members of the line organization that are qualified, or an independent organization as described in administrative procedures, may perform these inspections.

5.8.1.10 *Control of Measuring and Test Equipment*

Approved procedures will be developed for the control, use, calibration, and testing of the equipment used for the FSS, including both laboratory and field use equipment. These procedures will ensure confidence in the data obtained. Instrument calibrations will be performed periodically in accordance with appropriate industry standards.

5.8.1.11 *Handling, Storage and Shipping*

Some of the material samples will be transported to offsite laboratories for analysis. The process for controlling this material will be sufficient to ensure that a COC is maintained. Measures shall be established to ensure that samples are received, handled, stored, packaged, and shipped in accordance with approved procedures or instructions. These procedures or instructions shall be responsive to applicable industry or manufacturer's requirements and include controls for "shelf life" of sensitive products. Additionally, protocols must be established to ensure there is no cross-contamination between samples and sample packaging. Appropriate controls will be defined in administrative procedures to ensure that sample integrity is maintained.

5.8.1.12 *Control of Nonconformance*

During the performance of the FSS, non-conforming conditions may be identified with equipment or services. The data associated with the non-conforming condition will be controlled until such time that it is accepted, rejected, or reworked in accordance with an appropriate procedure. Nonconforming equipment will not be used until conformance with applicable requirements has been established.

5.8.1.13 *Corrective Action Program*

The existing HBPP Corrective Action Program will be used for the FSS Program to ensure conditions adverse to quality are promptly identified and corrected.

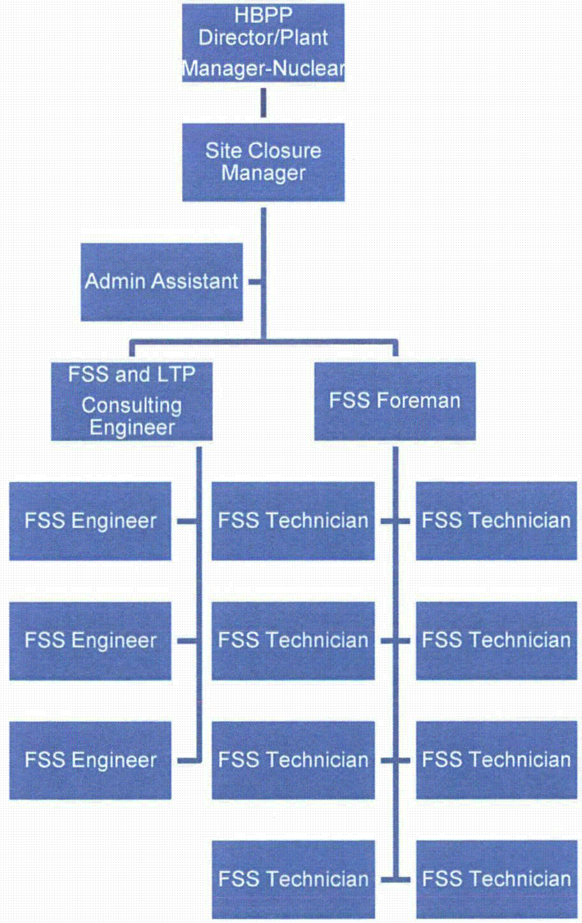
5.8.1.14 *Records*

Measures have been established that ensure that FSS records are maintained as quality records. These measures also include procedures by which the records are reviewed and approved, and procedures that ensure the records can be retrieved within a reasonable period. The controls shall also provide for the protection of the records to ensure they are not lost or subject to degradation over time.

5.8.1.15 *Audits*

Audits of FSS activities will be performed periodically, in accordance with approved procedures or instructions, to verify the implementation of quality activities.

Figure 5-2 Projected HBPP FSS Organizational Chart



5.9 References

5-1 U.S. Nuclear Regulatory Commission NUREG-1757, Vol. 2, "Consolidated NMSS Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria, Final Report," September 2003.

5-2 U.S. Nuclear Regulatory Commission NUREG-1575, Revision 1, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," August 2000.

5-3 U.S. Nuclear Regulatory Commission NUREG-1505, Revision 1, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys," June 1998 Draft.

5-4 U.S. Nuclear Regulatory Commission NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions," June 1998.

5-5 U.S. Nuclear Regulatory Commission NUREG-1700, Revision 1, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," April 2003.

5-6 U.S. Nuclear Regulatory Commission Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," January 1999.

5-7 U.S. Nuclear Regulatory Commission NUREG/CR-5512, Volume 1, Final Report, 1992.

5-8 International Organization for Standardization, ISO 7503-1, "Evaluation of Surface Contamination - Part 1: Beta Emitters and Alpha Emitters (first edition)," 1988.

5-9 Brodsky, A, 1992 “Exact Calculation of Probabilities of False Positives and False Negatives for Low Background Counting” Health Physics 63(2): 198-204.

5-10 HBPP TBD, *Instrument Efficiency Determination for use in Minimum Detectable Concentration Calculations in Support of the Final Status Survey at HBPP,* Revision 0.

5-11 HBPP TBD, *“Radiological Selection for DCGL Development,”* Revision 0.

5-12 Applicable Site Procedures for FSS:

- HBAP C-225 “Final Status Survey (FSS) Program,” Revision 0
- HBAP C-202 “FSS Quality Assurance Program Procedure (QAPP),” Revision 0
- RCP FSS-8 “Collection of Site Characterization and FSS Samples,” Revision 1
- RCP FSS-7 “Determination of the Number and Locations of FSS Samples,” Revision 0
- RCP FSS-4 “Isolation and Control of Areas for FSS,” Revision 0A
- RCP FSS-1 “Survey Unit Classification,” Revision 0
- RCP FSS-2 “Preparation of FSS Survey Plans,” Revision 0
- RCP-FSS-15 “Statistical Tests,” Revision 0
- RCP FSS-13 “Area Surveillance Following FSS,” Revision 0
- RCP FSS-14 “Data Quality Assessment,” Revision 0
- RCP FSS-17 “Preparation of FSS Reports,” Revision 0
- RCP FSS-11 “Split Sample Assessment for Final Status Survey,” Revision 0
- RCP FSS-18 “Computer Determination of the Number and Location of FSS Samples and Measurements,” Revision 0
- RCP FSS-3 FSS “Background Assessment,” Revision 0
- RCP FSS-16 “ALARA Evaluations for Final Status Survey Areas,” Revision 0

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ENG-HB-001, Rev. 0, RESRAD Input Parameter Sensitivity Analysis – Humboldt Bay
ENG-HB-002, Rev. 0, RESRAD-Build Input Parameter Sensitivity Analysis – Humboldt Bay
ENG-HB-003, Rev. 0, Humboldt Bay Soil Derived Concentration Guideline Levels
ENG-HB-004, Rev. 0, Humboldt Bay Building Surface Derived Concentration Guideline Levels
ENG-HB-005, Rev. 0, Area Factors for Use with Humboldt Bay Soil DCGLs
ENG-HB-006, Rev. 0, Area Factors for Use with Humboldt Bay Building Surface DCGLs

6. COMPLIANCE WITH THE RADIOLOGICAL CRITERIA FOR LICENSE TERMINATION

6.1. Site Release Criteria

6.1.1. Radiological Criteria for Unrestricted Use

The site release criteria for the Humboldt Bay Power Plant Unit 3 (HBPP) site are the NRC's radiological criteria for unrestricted use established in 10 CFR 20.1402 (Reference 6-1):

- **Dose Criterion**: The residual radioactivity that is distinguishable from background radiation results in a Total Effective Dose Equivalent (TEDE) to an average member of the critical group that does not exceed 25 mrem/yr, including that from groundwater sources; and
- **ALARA Criterion**: The residual radioactivity has been reduced to levels that are As Low As Reasonably Achievable (ALARA).

"Background radiation" in the previous criteria means radiation from cosmic sources, natural occurring radioactive material (including radon, except as a decay product of source or special nuclear material), and global fallout as it exists in the environment from the testing of nuclear explosive devices or from past nuclear accidents such as Chernobyl. All of these sources of radiation contribute to background radiation and are not under control of the licensee. Background radiation does not include radiation from source, byproduct, or special nuclear materials regulated by the commission.

6.1.2. Conditions Satisfying the Site Release Criteria

Derived concentration guideline levels (DCGLs) are radionuclide-specific activity concentrations that correspond to release criteria described in Section 6.1.1. DCGL values are derived from activity-dose relationships through the analysis of various exposure pathway scenarios. Section 6.2.3 discusses the potential radionuclides of concern for the HBPP site.

DCGL values for assessing residual radioactivity on building surfaces and in site soil have been calculated for each potential radionuclide of concern. The DCGLs form the basis for the following conditions which, when met, satisfy the site release criteria as prescribed in 10 CFR 20.1402:

- The average residual radioactivity above background is less than or equal to the DCGL. For mixtures of radionuclides, the sum of the fractions of the contaminant's concentration over the contaminant's DCGL must be less than or equal to one.
- Individual measurements representing small areas of residual radioactivity that exceed the DCGL, but do not exceed the elevated measurement comparison DCGL ($DCGL_{EMC}$).
- Where one or more individual measurements exceed the DCGL, but the average residual radioactivity passes the Wilcoxon Rank Sum or Sign statistical test.
- Remediation of contaminated areas is performed where ALARA considerations require that levels of residual radioactivity be below DCGLs.

The methods in Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (Reference 6-2) and the DCGLs may not be appropriate for nonstructural components. For those nonstructural components and systems to which MARSSIM does not apply, the current "no detectable" criteria, consistent with IE Circular 81-07, (Reference 6-17) and/or the Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual (MARSAME) (Reference 6-3) protocol will be applied to disposition these items. Similarly, the DCGLs are not appropriate for assessing embedded piping.

6.2. Dose Modeling Approach

6.2.1. Overview

Dose models allow the translation of residual radioactivity levels into potential radiation doses to the public. For the Humboldt Bay site, dose models have been developed based on the guidance found in NUREG/CR-5512 (Reference 6-4), Volumes 1, 2, and 3. The conceptual model reflects the anticipated site conditions at the time of license termination. The dose modeling approach for the Humboldt Bay site is consistent with the information for site-specific modeling provided in Appendix I of NUREG-1757 (Reference 6-6), including source term abstraction and scenarios, pathways, and critical groups.

There are three defining factors for a dose model: (1) the scenario, (2) the critical group, (3) and the exposure pathways. The scenarios described in NUREG/CR-5512, Volume 1, address the major exposure pathways of direct exposure to penetrating radiation and inhalation and ingestion of radioactive materials. The scenarios also identify the critical group, which is defined as the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity within the assumptions of a particular scenario. The design for scenarios and the site-specific modeling provide reasonable and conservative estimates of the potential doses associated with residual radioactivity.

The dose models supporting the building surface and soil DCGLs were developed using the approach outlined previously. The scenarios described in NUREG/CR-5512, Volume 1, were selected for the Humboldt Bay site to estimate potential radiation doses from radioactive material in buildings (building occupancy scenario) and soil (resident farmer scenario).

6.2.2. Site Conceptual Model

The site conceptual model (SCM) for the HBPP facility is the relationship between the sources of residual radioactivity, the areas where the sources are located, transport mechanisms, exposure routes, and the receptor (i.e., hypothetically exposed human). The SCM describes how residual radioactivity at the site might enter into the environment, how it moves around within the environment, and the routes used to expose humans. The SCM has three fundamental components that are needed to calculate (or model) the potential future dose to a receptor on or near the decommissioned facility. The first component is the physical characteristic(s) of the site. The second is the source term itself. The third is the range of realistic (plausible) human exposure scenarios that are described by factors that are associated with human behavior and metabolic processes. Each of these fundamental components is described in the following subsections.

6.2.2.1 Geology

Figures 6-1 and 6-2 provides the HBPP site location and the HBPP boundary respectively. The HBPP at Buhne Point is at the northern margin of the northeast-trending Eel River Geosyncline. Deposits in the geosynclines range in age from Cretaceous to Recent. Consolidated bedrock is overlain by approximately 3,000 to 4,000 ft of unconsolidated clay, silt, sand, and gravel in the Eel

River-Humboldt Bay area. The bedrock consists of the Franciscan assemblage; Yager Formation; and the Pullen, Eel River, Rio Dell, and Scotia Bluffs formations of the Wildcat Group. The unconsolidated sediments contain most of the groundwater in the region and are divided into dune sand, alluvium, terrace deposits, the Hookton Formation, and the Carlotta Formation of the upper Wildcat Group.

6.2.2.2 Site Stratigraphy

The HBPP site is underlain by sediments of the Hookton Formation. Borehole data indicate that to a depth of approximately 15-35 feet, the strata are compact clay, clayey sands, and clayey silt. Below this layer lies a sand body that becomes gravelly with depth, containing pea to cobble sized gravels in thin discontinuous lenses. The sand extends to a depth of approximately 110 feet; however, it is divided into two relatively clean sections by a clayey zone (2nd Bay Clay). The Lower Hookton Formation encounters the Wildcat Group at a depth of approximately 1,100 feet. The lower unit of the Hookton Formation consists of laterally persistent beds of alternating sand, silty sand, gravel, gravelly sand, silty clay, and clay. Figure 6-3 illustrates the various zones.

Figure 6-1 HBPP Site Location

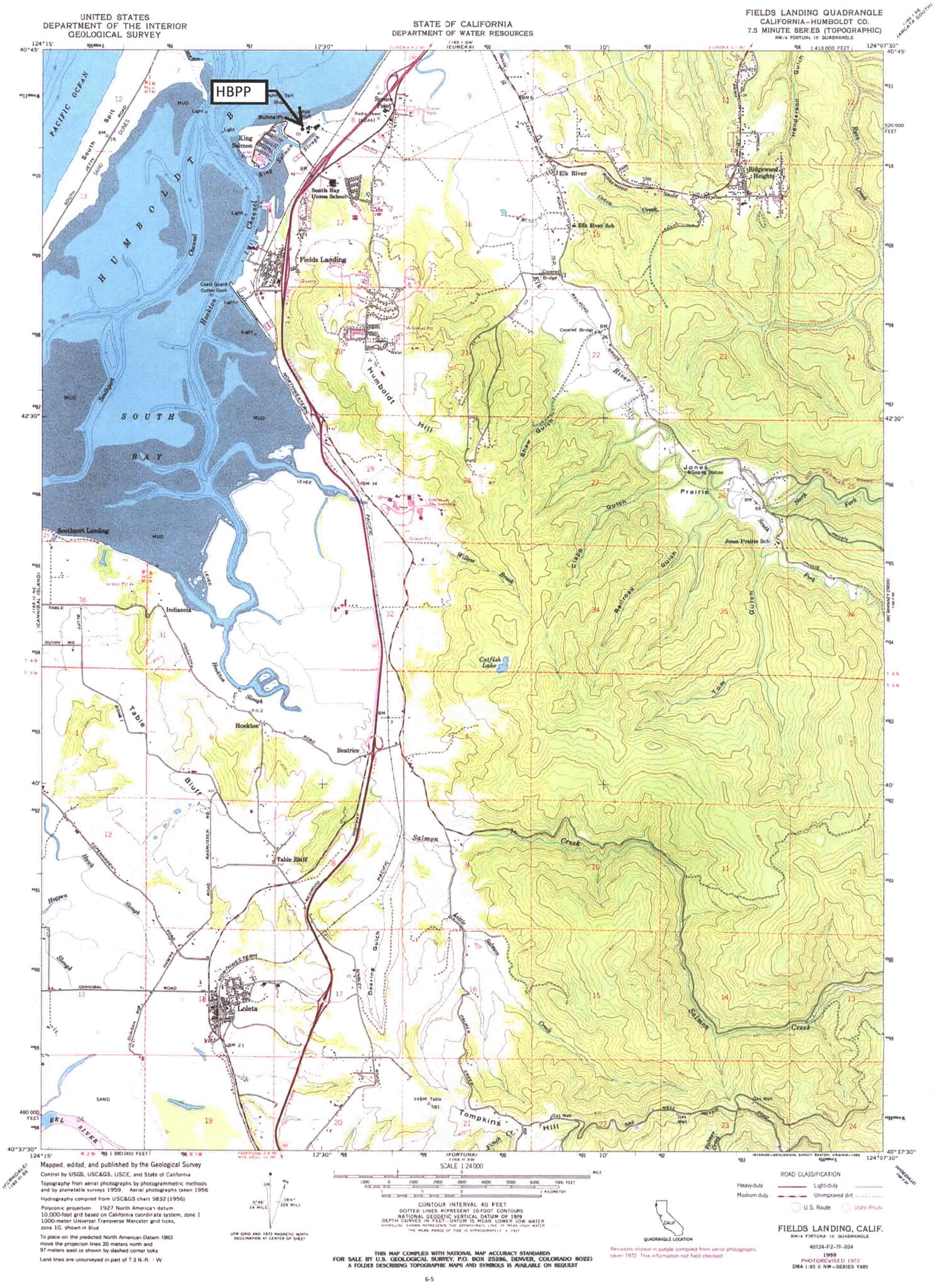


Figure 6-2 Aerial View of HBPP with Site Boundary

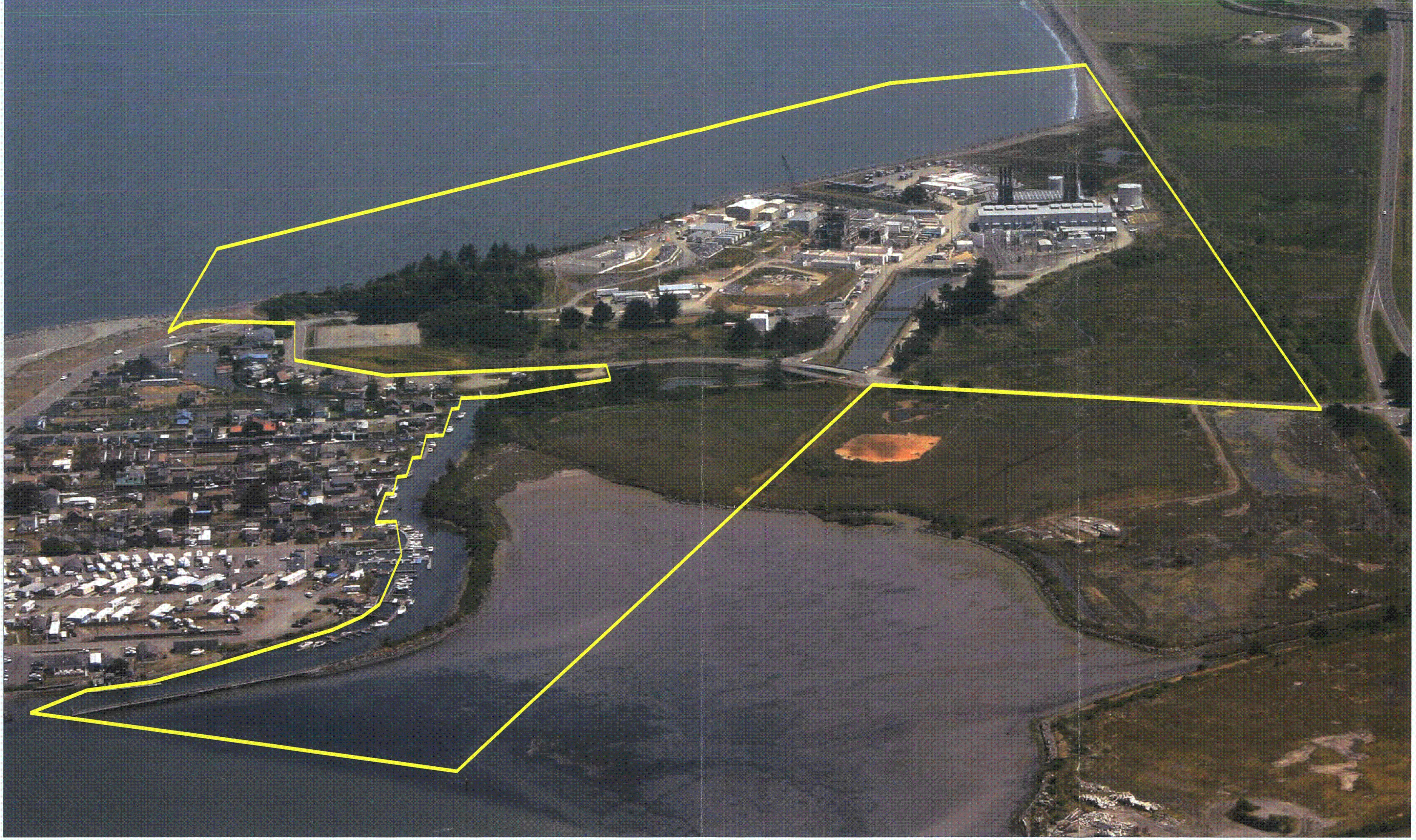
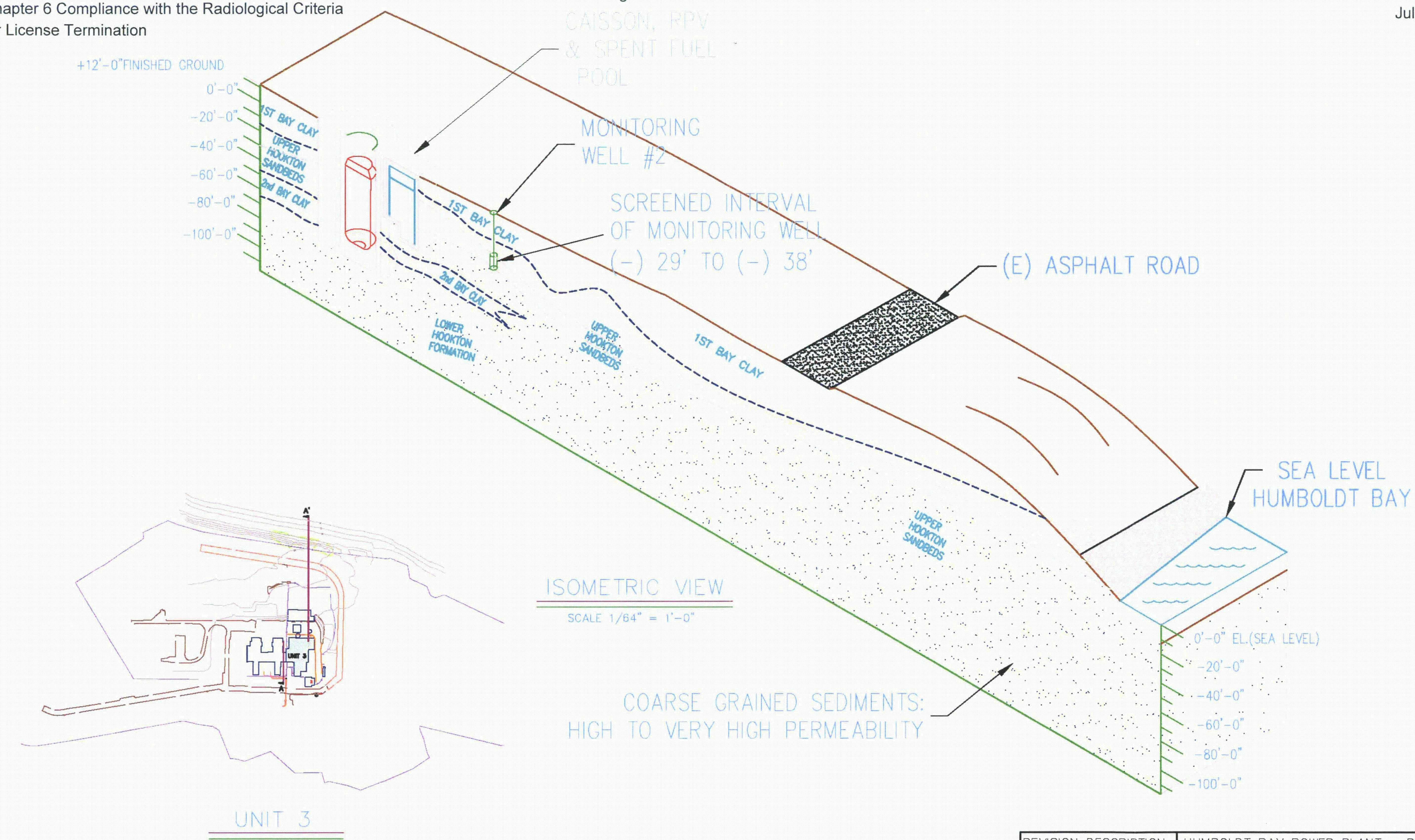


Figure 6-3 Cross-section of HBPP Subsurface



REVISION DESCRIPTION	HUMBOLDT BAY POWER PLANT – PG&E CO. UNIT 3 DECOMMISSIONING SUBSURFACE ISOMETRIC
DATE 11 APRIL 2013	DRAWING LTP-SUBSURFACE SITE ISOMETRIC / SHEET 1 of 1 / REV 2

6.2.2.3 *Groundwater Hydrology*

Groundwater in the Eel River-Humboldt Bay area is contained primarily in the loosely unconsolidated sediments of the dune sand, alluvium, and terrace deposits and the poorly consolidated sediments of the Hookton and Carlotta formations. HBPP is underlain by a portion of the Hookton aquifer, consisting of a predominately sand and gravel unit in the alluvium extending from about elevation Minus 8 feet Mean Sea Level (MSL) to about elevation Minus 100 feet (MSL).

Underlying the HBPP are three distinct water bearing zones:

1. Zone of Perched Groundwater in the Upper Hookton silt and clay beds; unconfined aquifer perched within the upper silt and clay beds of the First Bay Clay
2. Upper Hookton Aquifer; confined to semiconfined aquifer within the Upper Hookton Sand Beds between the First and Second Bay Clay
3. Aquifer between Unit F and Second Bay Clay; confined to semiconfined aquifer within the Lower Hookton, between the Second Bay Clay and Unit F Clay

Groundwater closest to the surface beneath HBPP is encountered in the interbedded fine grained deposits of the First Bay Clay, which extend to depths ranging from 16 to 28 feet below grade surface (BGS). Water in this unconfined aquifer is trapped within multiple intermixed sand and silt beds and is considered "perched."

The upper part of the Hookton aquifer zone is in relatively clean sand approximately 30 feet thick. The upper aquifer and lower aquifer are separated by a clayey zone. Wells in the aquifer have shown that the groundwater fluctuates with tidal cycles. Calculated tidal efficiencies have ranged from 46 percent for a well approximately 235 feet from the bay to 26 percent for a well approximately 605 feet from the bay. Permeability data for the shallow sand at the site present permeabilities of 3×10^{-3} to 1×10^{-2} cm/sec (3,100 ft/yr to

10,400 ft/yr) using data on tidal efficiency and tidal lag times from five monitoring wells.

6.2.2.4 *Groundwater Recharge, Flow and Discharge*

The Eel River Valley alluvial aquifer contours indicate that groundwater flows west, down the valley, toward the coast. Groundwater levels and flow direction at HBPP are influenced by topography, proximity to Humboldt Bay, tidal influence, seasonal variations, and heterogeneity in soil stratigraphy. While the flow of groundwater is predominately west to northwest, toward the bay, during rising tides the flow turns easterly. During wet winter months when the aquifer discharge is likely to be greater, the flow reversal is subdued with the predominant flow direction toward the bay. Additionally, during this time of the year, the alluvial aquifer is flushed by the high flows attributed to runoff. A downward vertical gradient exists within the first water bearing zone and between the first and second water bearing zones. Recharge to the alluvium deposits is by direct infiltration of precipitation, seepage from rivers and streams, and to some extent, by lateral seepage from the Hookton and Carlotta formations. Groundwater is discharged from the alluvium by seepage into tidal estuaries and Humboldt Bay, by evapotranspiration and by pumping. The maximum discharge by tidal seepage occurs at the low tidal cycle. Recharge to the Hookton and Carlotta aquifers is primarily through deep percolation of rainfall on the outcrop areas and subsequent lateral flow beneath the confining beds. Recharge potential at HBPP is low, due to the 15 to 35 feet of silt and clay at the surface. Discharge occurs primarily by seepage into the sea.

6.2.2.5 *Open Land Area Source Term*

The open land area source term for HBPP is the concentration of radioactivity that will be allowed to remain in the soils after remediation is complete. That concentration is bounded by an upper limit on radiation dose potential of 25 millirem TEDE. Chapter 2 describes the current characterization of the HBPP site by providing Survey Area by Survey Area concentrations of activity. The highest concentrations presently found on the HBPP site are found in Survey Areas NOL01 (the area within the Radiological Analysis (RA) boundary) and to a lesser degree in Survey Area OOL03 (the area north of Units 1 and 2). Contaminations in these areas are primarily due to unplanned events that have deposited contamination on the area surfaces. Contamination levels in these areas have shown that the contamination has migrated downward into the subsoil at depths up to 12 feet. The predominant radionuclides present are Cs-137 and Co-60.

Areas of significance at HBPP are as follows:

- Discharge Canal - Activity levels are greater in the first 90 percent lengthwise from the point where Unit 3 discharges into the canal. Activity levels in the headworks (southern end) portion average 8.7 and 1.0 pCi/g Cs-137 and Co-60, respectively, with the highest levels of 42.24 and 2.94 pCi/g, respectively. Levels taper off in the final 10 percent of the canal before entering the bay to an average of 1.2 and 0.2 pCi/g Cs-137 and Co-60, respectively. Activity levels appear to be confined to the top 2 feet of the sediment.
- Intake Canal - Activity levels are at their highest at the eastern end with the average concentration of 9.42 and 0.38 pCi/g Cs-137 and Co-60, respectively.
- RA Area - As seen in Table 2-4 of Chapter 2, activity in the soils within the RA vary considerably. Generally, levels of contamination in the first 0.5 foot from the surface average approximately 1 pCi/g Cs-137 and 0.12 pCi/g Co-60. However, areas where events have occurred demonstrate considerably greater contamination with levels as high as 30 pCi/g Cs-137, not only at the surface but at depths to 12 feet, or greater where levels of 3.5 pCi/g Cs-137 have been found.
- North Yard Drain Area - Activity levels where events have occurred range from 1 to 23.7 pCi/g Cs-137 and 0.06 to 0.48 pCi/g Co-60. The depth of activity is not as great as inside the RA; however, contamination is found at depths up to 4 feet.

6.2.2.6 Building Surface Area Source Term

The building surface area source term is composed of the contributions from activity present on the HBPP structure surfaces from primarily Cs-137. Few structures will remain at the time of license termination. The remaining structures were minimally impacted by Unit 3 operations. The primary impact was through the wet and dry deposition of activity due to stack releases. The residual activity, if present on the building surfaces, is fixed in nature and characterization data indicate levels of 213 to 1,126 dpm/100 cm^2. The average level of activity present on building surfaces is 411 dpm/100 cm^2 with a standard deviation of 182 dpm/100 cm^2. The activity is predominately located on the roofs of the structures

Areas where the potential exists for contaminants to migrate to subsurface locations (e.g., caisson) will undergo strict evaluations and, if a pathway is determined to exist from the inner surfaces to the soils beyond the structure, samples will be gathered to assess the extent of activity in these areas.

6.2.3. Potential Radionuclides of Concern

As part of the source-term extraction process, an analysis was performed in HBPP Technical Basis Document, "Radiological Selection for DCGL Development, Revision 0" (Reference 6-5) to identify a suite of radionuclides that could potentially be present on remaining site structural surfaces, in site soils, and in groundwater following completion of decommissioning activities. This document was developed using the HBPP Historical Site Assessment (HSA) (Reference 6-18) and the regulatory documents identified in the HSA. The HBPP HSA identified a suite of radionuclides that were the primary contaminants of concern for the HBPP site. The suite included Am-241, Cm-244, Co 60, Cs-137, Fe-55, Mn-54, Ni-63, Pu-238, Pu-239, Pu-240, and Sr-90. This suite was appropriate for the preliminary development of site-specific DCGLs.

Additional potential radionuclides were evaluated from NUREG/CR-3474 "Long-Lived Activation Products in Reactor Materials," (Reference 6-15) and NUREG/CR-4289 "Residual Radionuclide Contamination Within and Around Commercial Nuclear Power Plants" (Reference 6-16).

Radionuclides identified in NUREG/CR-3474 Table 5.14, Activity Inventory of Boiling Water Reactor (BWR) Internals at Shutdown, along with their half-lives in years and their decay modes, are provided in Table 6–1.

Table 6-1 NUREG/CR-3474 Identified Activation Product Radionuclides

Radionuclide	Half Life (Years)	Decay Mode
Ag-108m	4.18E+02	IT
Ag-110m	6.84E-01	β^-, γ
Ar-39	2.69E+02	β^-
Ba-133	1.05E+01	γ
C-14	5.73E+03	β^-
Ca-41	1.03E+05	β^+, γ
Ce-141	8.90E-02	β^-, γ
Cl-36	3.01E+05	β^-
Co-58	1.94E-01	β^+, γ
Co-60	5.27E+00	β^-, γ
Cr-51	7.58E-02	γ
Cs-134	2.06E+00	β^-, γ
Cs-135	2.30E+06	β^-
Cs-137	3.02E+01	β^-
Eu-152	1.36E+01	β^-, γ
Eu-154	8.80E+00	β^-, γ
Eu-155	4.96E+00	β^-, γ
Fe-55	2.70E+00	γ
Fe-59	1.22E-01	β^-
H-3	1.23E+01	β^-
Hf-178m	3.00E+01	IT
Ho-166m	1.20E+03	β^-, γ
I-129	1.57E+07	β^-, γ
Kr-81	2.10E+05	γ
Kr-85	1.07E+01	β^-, γ
Mn-53	3.70E+06	γ
Mn-54	8.56E-01	Γ, β^+
Mo-93	4.00E+03	γ
Nb-92m	2.78E-02	Γ, β^+
Nb-94	2.03E+04	β^-, γ
Ni-59	7.50E+04	Γ, β^+
Ni-63	1.00E+02	β^-
Pb-205	1.51E+07	γ
Pm-145	1.77E+01	γ
Pu-239	2.41E+04	α, γ
Sb-124	1.65E-01	β^-, γ
Sc-46	2.29E-01	β^-, γ
Se-79	1.13E+06	β^-
Sm-146	1.00E+08	α
Sm-151	9.30E+01	β^-, γ
Sn-121m	5.00E+00	β^-, γ
Sr-90	2.86E+01	β^-
Tb-158	1.50E+02	β^-, γ, β^+
Tc-99	2.13E+05	β^-, γ
U-233	1.59E+05	α, γ
Zn-65	6.69E-01	β^+, γ

Radionuclide	Half Life (Years)	Decay Mode
Zr-93	1.53E+06	β^-

α – Alpha decay
γ – Gamma decay
β- – Beta decay
IT – Isomeric transition
β+ – Positron decay

Radionuclides identified in NUREG/CR-4289 along with their half-lives in years and their decay modes, are provided in Table 6–2.

With the exception of Co-60, radionuclides with half-lives less than 5.4 years identified in NUREG/CR-4289 were discounted and not included in the list provided in Table 6–2. Based on the period from final shutdown of HBPP to the originally anticipated completion of license termination in 2016, it is highly unlikely that any activity from radionuclides with half-lives less than 5.4 years would remain significant. Although Co-60 has a half-life of 5.27 years, the HBPP HSA reported a September 1, 2006, inventory of 672.3 Ci of Co-60. Assuming a July 1, 2016, (estimated date at the TBD development) license termination, the Co-60 inventory at that time would still be approximately 172 Ci. Therefore, it is appropriate to retain Co-60 in the list of potential radionuclides.

Table 6-2 Radionuclides Identified in NUREG/CR-4289

Radionuclide	Half Life (Years)	Decay Mode
Am-241	4.32E+02	α, γ
C-14	5.73E+03	β-
Cm-244	1.81E+01	α, γ
Co-60	5.27E+00	β-, γ
Cs-137	3.02E+01	β-
Eu-152	1.36E+01	β-, γ
Eu-154	8.80E+00	β-, γ
H-3	1.23E+01	β-
I-129	1.57E+07	β-, γ
Nb-94	2.03E+04	β-, γ
Ni-59	7.50E+04	β^+,γ
Ni-63	1.00E+02	β-
Np-237	2.14E+6	α, γ
Pu-238	8.78E+01	α, γ
Pu-239	2.41E+04	α, γ
Pu-240	6.60E+03	α, γ
Sr-90	2. 86E+01	β-
Tc-99	2.13E+05	β-, γ

α – Alpha decay
β- – Beta decay

γ – Gamma decay

6.2.4. Discounting insignificant radionuclides

Since Tables 6–1 and 6–2 include trace elements that are not likely to be found in site area soils or on surfaces, due to their low abundance and/or short half-lives, an evaluation of radionuclides that may be discounted as being of potential importance was performed. The total inventory for each radionuclide was determined from generic activity inventories provided in Table 5.14 and Table 5.15 of NUREG/CR-3474. From this information, the percentage of total inventory for each radionuclide was calculated. The results of this evaluation are provided in Table 6–3.

6-3 Evaluation of NUREG/CR-3474 Total Activity Fractions (Reactor Vessel)

Radionuclide	Activity - Ci				Percent Total	Less than 0.1%?
	Shroud	Vessel Cladding	Vessel Walls	Total Activity		
Ag-108m	2.18E-01	1.79E-01	7.39E-06	6.41E-05	2.67E-04	Yes
Ar-39	2.68E-01	2.43E-01	2.73E-05	1.00E-03	3.64E-04	Yes
Ba-133	1.00E+01	9.24E-01	3.23E-05	2.03E-04	1.38E-03	Yes
C-14	1.03E+02	1.03E+02	2.79E-03	1.19E-02	1.53E-01	No
Ca-41	2.00E-02	2.00E-02	5.20E-07	2.00E-06	2.98E-05	Yes
Cl-36	2.24E+00	2.24E+00	5.70E-05	1.43E-04	3.34E-03	Yes
Co-60	4.50E+05	3.91E+03	1.20E-01	8.30E-01	5.83E+00	No
Cs-134	3.37E+01	1.80E-04	5.23E-09	1.87E-08	2.68E-07	Yes
Cs-135	3.80E-04	3.80E-04	3.67E-10	2.46E-09	5.67E-07	Yes
Cs-137	2.11E+00	9.22E-01	8.74E-06	6.03E-05	1.37E-03	Yes*
Eu-152	2.09E-02	4.91E-08	6.12E-04	2.70E-03	4.94E-06	Yes*
Eu-154	1.28E+01	7.46E-01	2.68E-05	2.62E-04	1.11E-03	Yes*
Eu-155	5.06E+00	3.27E-02	1.10E-07	1.21E-06	4.87E-05	Yes
Fe-55	9.29E+05	8.81E+01	2.24E-03	1.08E-02	1.31E-01	No
H-3	1.83E+02	2.40E+01	1.83E-03	7.98E-03	3.57E-02	Yes*
Hf-178m	5.21E-01	2.26E-01	1.87E-05	3.08E-04	3.37E-04	Yes
Ho-166m	3.93E-01	3.85E-01	1.08E-05	1.56E-04	5.74E-04	Yes
I-129	5.90E-07	5.90E-07	4.40E-12	1.88E-12	8.80E-10	Yes
Kr-81	2.24E-04	2.24E-04	5.40E-12	3.04E-11	3.34E-07	Yes
Kr-85	8.15E-01	7.87E-02	4.83E-07	2.12E-06	1.17E-04	Yes
Mn-53	6.51E-03	6.50E-03	8.00E-07	1.00E-05	9.71E-06	Yes
Mn-54	1.17E+04	2.39E-09	2.33E-13	2.60E-12	3.58E-12	Yes
Mo-93	1.08E+00	8.51E-04	3.47E-08	6.27E-07	1.27E-06	Yes
Nb-92m	6.36E-07	6.33E-07	2.20E-10	2.90E-09	9.49E-10	Yes
Nb-94	8.86E-01	8.85E-01	2.80E-05	7.19E-05	1.32E-03	Yes*

RAI 28
RAI 36

Radionuclide	Activity - Ci				Percent Total	Less than 0.1%?
	Shroud	Vessel Cladding	Vessel Walls	Total Activity		
Ni-59	6.04E+02	6.04E+02	1.80E-02	8.00E-02	9.01E-01	No
Ni-63	8.00E+04	6.23E+04	1.79E+00	7.44E+00	9.29E+01	No
Pb-205	4.00E-06	4.00E-06	2.58E-10	3.04E-09	5.97E-09	Yes
Pm-145	4.40E-03	1.07E-03	3.16E-08	2.29E-08	1.60E-06	Yes
Pu-239	3.81E-02	3.80E-02	3.00E-06	6.79E-05	5.67E-05	Yes*
Se-79	1.40E-03	1.40E-03	9.80E-08	1.00E-06	2.09E-06	Yes
Sm-146	4.08E-10	4.07E-10	4.50E-14	6.20E-13	6.08E-13	Yes
Sm-151	5.32E-02	4.05E-02	1.38E-05	1.11E-04	6.06E-05	Yes
Sn-121m	1.07E-02	7.19E-05	6.72E-09	9.41E-08	1.07E-07	Yes
Sr-90	2.11E+00	8.80E-01	5.84E-06	2.54E-05	1.31E-03	Yes*
Tb-158	5.31E-03	4.49E-03	5.34E-07	6.77E-06	6.70E-06	Yes
Tc-99	2.10E-01	2.10E-01	9.00E-06	1.59E-04	3.13E-04	Yes*
U-233	2.25E-03	2.25E-03	1.30E-07	2.00E-06	3.36E-06	Yes
Zn-65	1.55E+03	9.00E-14	2.38E-18	1.68E-18	1.34E-16	Yes
Zr-93	1.41E-04	1.41E-04	6.90E-09	8.10E-08	2.10E-07	Yes
Total	6.70E+04	1.94E+00	8.38E+00	6.70E+04	1.00E+02	
Total percent of activity discounted					6.57E-03	

* Radionuclides meet the criteria of contributing less than 0.1 percent of the total activity but cannot be discounted because they have other methods of production in addition to activation of reactor components and/or have been observed in 10 CFR Part 61 waste stream analyses or site characterization samples.

A comprehensive review of I-129 was performed to determine if it indeed should be included in the list of potential radionuclides. The following conclusions were reached:

1. I-129 contributed less than 0.1% of the total activity (i.e., 8.80E-10%) as shown in Table 6-3 of the LTP.

2. I-129 was screened using the DandD default parameters and input values were determined as outlined in HBPP TBD "Radionuclide Selection for DCGL Development." The dose attributed to I-129 were RAI 28
1.76E-07 mrem and 1.82E-07mrem for Residential and Occupancy RAI 36
respectively.

3. I-129 values are entered on certain HBPP radwaste shipment manifests. Certain waste burial sites require that values for all 10 CFR 61 radionuclides be entered on the manifest. Review of values entered determined that the MDC values were used for the I-129 concentrations. I-129 concentrations in 10 CFR 61 analyses have not been observed in the past at HBPP greater than their MDA values.

4. I-129 concentrations have not been observed above the MDA value in characterization sample analyses when analyzed at HBPP.

5. NUREG-4289 lists I-129 residual radionuclide concentrations in HBPP reactor component systems as insignificant (Table C.2.3)

Based upon the above review of I-129 at HBPP, it is appropriate to exclude I-129 from the list of site-specific radionuclides potentially present at the HBPP site.

RAI 28
RAI 36

Based on the previous evaluation, it was determined that individual radionuclides that contributed less than 0.1 percent of the total activity could potentially be discounted, providing that dose contributed by the sum of the those radionuclides does not exceed 1 percent of the total calculated dose. The total percentage of activity attributed to radionuclides that meet these criteria amounts to 0.007 percent

6.2.5. Site-Specific Suite of Radionuclides

Table 6–4 represents a list of radionuclides potentially present at HBPP, based on applying the described screening criteria to the combined list of potential radionuclides from regulatory guidance contained in NUREG/CR-3474 and NUREG/CR-4289 and historical 10 CFR 61 analyses.

Table 6-4 HBPP Site Specific Suite of Nuclides

Radionuclide	Half Life (Years)	Decay Mode
*Cm-243/244	1.81E+01	α, γ
*Cm-245/246	4.75E+03	α, γ
Am-241	4.32E+02	α, γ
C-14	5.73E+03	β-
Co-60	5.27E+00	β-, γ
Cs-137	3.02E+01	β-
Eu-152	1.36E+01	β-, γ
Eu-154	8.80E+00	β-, γ
H-3	1.23E+01	β-
Nb-94	2.03E+04	β-, γ
Ni-59	7.50E+04	γ
Ni-63	1.00E+02	β-
Np-237	2.14E+06	α, γ
Pu-238	8.78E+01	α, γ
Pu-239	2.41E+04	α, γ
Pu-240	6.60E+03	α, γ
Pu-241	1.44E+01	β-

RAI 28
RAI 36

Radionuclide	Half Life (Years)	Decay Mode
Sr-90	2. 86E+01	β-
Tc-99	2.13E+05	β-, γ

*Listed half-life is the shortest half-life for the radionuclides in the pair
α – Alpha Decay
β- – Beta Decay
γ – Gamma Decay

Samples will be taken of soils and building surfaces in areas deemed to have the highest activity present in those media. The samples will be analyzed for all the radionuclides in the site-specific suite. If any of the nuclides are not identified in the analyses then they may be deselected from the survey, however, the potential dose from the deselected nuclides will be determined using their MDC values decayed to a license termination date of September 5, 2019, as compared to their respective DCGLs. RAI 42 RAI 37

6.2.6. *Resident Farmer Scenario for Surface and Subsurface Soil Exposure*

6.2.6.1 *Resident Farmer Scenario Justification*

PG&E has no plans to release all or part of the facility for ownership by members of the public. Although the public does have access to portions of the site via the coastal walkway, there is no ready access to the majority of the site. The HBPP switchyard has been in continual use, and the site continues to be an important center of electrical supply from the Humboldt Bay Generating Station (HBGS).

It is unlikely that the HBPP site will be used for any purpose other than an industrial site; however, HBPP has chosen the conservative approach of remediating and surveying to the resident farmer scenario at license termination to allow for other uses following the expected 30-year life of the HBGS, which would be in 2040.

6.2.6.2 *Critical Group for Surface Exposure*

The average member of the critical group was determined to be the resident farmer who lives on the Humboldt Bay site following decommissioning, grows all or a portion of his/her diet onsite, and uses the water from a groundwater source on the site for drinking water and irrigation. The dose from residual radioactivity in soil is evaluated for the critical receptor as required by 10 CFR 20, Subpart E, and described in Appendix I to NUREG -1757.

6.2.6.3 *Conceptual Model and Site-Specific Exposure Pathways*

The conceptual model for this scenario is a residential farming family that lives onsite, raises crops and livestock for consumption, and drinks water from an onsite ground water source.

It is unlikely that any other set of plausible human activities that would result in a dose exceeding that calculated for the hypothetical resident farmer could occur on the Humboldt Bay site. It is more likely that the behavior of future occupants would result in a lower dose. For example, it is more likely that the Humboldt Bay site will be reused for industrial purposes rather than a site for a residential farmer. The hypothetical dose from residual contamination in the soil to an individual in these settings would be less than for a resident farmer because such an individual would not reside on the site and ingest food grown onsite. Therefore, the use of the resident farmer as the average member of the critical group is both conservative and bounding for the calculation of soil DCGLs. The following bullets list the potential exposure pathways that apply to the resident farmer, based upon those in NUREG/CR-5512, Volume 1:

- Direct exposure to external radiation from residual radioactivity
- Internal dose from inhalation of airborne radionuclides
- Internal dose from ingestion of the following items:
    - Plant foods grown in media containing residual radioactivity and irrigated with water containing residual radioactivity
    - Meat and milk from livestock fed with fodder grown in soil containing residual radioactivity and water containing residual radioactivity
    - Drinking water (containing residual radioactivity) from a well
    - Fish from a pond containing residual radioactivity

- Soil containing residual radioactivity

6.2.7. Building Occupancy Scenario for Building Occupancy Exposure

6.2.7.1 Building Occupancy Scenario Justification

The Building Occupancy scenario is described in NUREG/CR-5512, Volume 1. Modeling of this scenario provides an estimate of human radiation exposure to residual radioactivity on surfaces inside standing buildings and permits the determination of DCGLs for building surfaces. This scenario was selected as the modeling basis for building surface DCGLs.

The justification for the soils scenario (Section 6.2.6.1) also applies to the building surface scenario.

6.2.7.2 Critical Group for Structural Surface Exposure

The average member of the critical group is defined as an adult individual engaging in work within the buildings following decommissioning of the site. The person occupies and carries out light to moderate work activities inside the building for a full year of employment. The breathing rate applied in the sensitivity analysis was appropriate for light to moderate activity. For conservatism, a higher breathing rate (appropriate for moderate to heavy activity) was used in the development of the building surface DCGLs. The dose to the individual from residual radioactivity on building surfaces is evaluated as required by 10 CFR Part 20, Subpart E, and described in Appendix I to NUREG -1757.

6.2.7.3 Conceptual Model and Site-Specific Exposure Pathways

The conceptual model is a Humboldt Bay worker who occupies the building as a routine work area and performs light to moderate renovation activities for a full employment year, receiving radiation exposure via the following potential exposure pathways:

- Direct exposure to external radiation from the following sources:

- Material deposited on the room surfaces (i.e., walls, floor, and ceiling)
- Submersion in airborne dust

- Internal dose from inhalation of airborne radionuclides
- Internal dose from inadvertent ingestion of radionuclides

In the development of building surface DCGL values, the Building Occupancy scenario modeled for the Humboldt Bay site accounted for moderate to heavy renovation activities carried out inside Humboldt Bay site buildings through use of conservative input for breathing rate and inadvertent ingestion of surface contamination. This approach produced reasonably conservative estimates of annual doses associated with contaminated building surfaces.

6.3. Computational Model Used for Dose Calculations

6.3.1. Impacted Area Soils

The computer code RESidual RADioactive materials (RESRAD) v6.3, followed by v6.4 after its release during the winter of 2007, was selected to perform site-specific dose modeling of impacted area soils because of the ability to model subsurface soil contamination contained within the code. Argonne National Laboratory (ANL) developed the RESRAD computer code under the sponsorship of the U.S. Department of Energy (DOE). The code has been used widely by DOE and its contractors, the U.S. NRC, U.S. Environmental Protection Agency (EPA), U.S. Army Corps of Engineers, industrial firms, universities, and foreign government agencies and institutions. This code is a pathway analysis model designed to evaluate potential radiological doses to an average member of the specific critical group.

The NRC has adopted a risk-informed approach in assessing impacts on the health and safety of the public from radioactive contamination remaining at decommissioned sites. Therefore, the NRC tasked ANL to develop parameter distribution functions and parametric analysis for RESRAD for conducting probabilistic dose analysis. As part of this effort, external modules equipped with probabilistic sampling and analytical capabilities were developed for the RESRAD code. The modules also are equipped with user-friendly input/output interface features to accommodate numerous parameter distribution functions and to fulfill results display requirements.

The RESRAD database includes inhalation and ingestion dose conversion factors from the EPA's Federal Guidance Report (FGR) No. 11, direct external exposure dose conversion factors from FGR No. 12, and radionuclide half-lives from International Commission on Radiological Protection Publication 38 (References 6-7, 6-8 and 6-9, respectively).

6.3.2. *Impacted Structural Surfaces and Bulk Material*

RESRAD-BUILD v3.3 was selected to perform site-specific dose modeling of impacted structural surfaces and bulk material. RESRAD-BUILD is a computer code designed to evaluate the radiation doses from RESidual RADioactivity in BUILDings. The RESRAD-BUILD code was developed by ANL under sponsorship of the U.S. DOE and other federal agencies.

The RESRAD-BUILD computer code is a pathway analysis model designed to evaluate the potential radiological dose incurred by an individual who works or lives in a building contaminated with radioactive material. The transport of radioactive material within the building from one compartment to another is calculated with an indoor air quality model. The air quality model considers the transport of radioactive dust particulates and radon progeny due to air exchange, deposition and resuspension, and radioactive decay and ingrowth.

Seven exposure pathways are considered in the RESRAD-BUILD code:

(1) external exposure directly from the source

(2) external exposure from materials deposited on the floor

(3) external exposure due to air submersion

(4) inhalation of airborne radioactive particulates

(5) inhalation of aerosol indoor radon progeny (in the case of the presence of radon predecessors) and tritiated water vapor (the radon pathway was turned off because the NRC does not regulate dose received from radon and progeny)

(6) inadvertent ingestion of radioactive material directly from the source

(7) ingestion of materials deposited on the surfaces of the building compartments

Various exposure scenarios may be modeled with the RESRAD-BUILD code. These include, but are not limited to, office worker, renovation worker, building visitor, and residency scenarios. Both deterministic and probabilistic dose analyses can be performed with RESRAD-BUILD, and the results can be shown in both text and graphic reports.

6.4. Derived Concentration Guideline Levels

6.4.1. Computer Code Selection

The RESRAD Family of Codes has been selected for use in determining DCGL values at the Humboldt Bay site. The RESRAD computer codes are pathway-analysis models developed at ANL. This family of computer codes includes RESRAD-BUILD, used to analyze pathways associated with buildings, and RESRAD, used to analyze pathways associated with soil.

The RESRAD-BUILD computer code is a pathway analysis model designed to evaluate the potential radiological dose incurred by an individual who works in a building contaminated with radioactive material. Version 3.5 of the RESRAD-BUILD computer code was used in this analysis to consider four primary exposure pathways to occupants of a building:

- External exposure directly from the sources (walls, floors, and ceilings)
- External exposure due to air submersion
- Inhalation of airborne radioactive particulates
- Inadvertent ingestion of radioactive material directly from the sources

As with the RESRAD-BUILD code, the RESRAD computer code was developed by ANL as a multifunctional tool to assist in developing radiological criteria for unrestricted release and assessing the dose or risk associated with residual radioactive material. The RESRAD computer code is a pathway analysis model designed to evaluate the potential radiological dose associated with residual radioactive material in land areas. Version 6.5 of the RESRAD computer code was used in this analysis to consider three major exposure pathways to a resident farmer:

- Direct exposure to external radiation from soil containing residual radioactivity
- Internal exposure from inhalation of airborne radionuclides
- Internal exposure from ingestion of radionuclides

Both the RESRAD-BUILD and the RESRAD computer codes incorporate probabilistic modules that permit the user to perform a sensitivity analysis to identify those parameters that have the greatest impact on dose. In addition, the probabilistic modules allow the evaluation of dose as a function of parameter distributions. Information on the use of these codes and their applications are outlined in NUREG/CRs-6676, -6692, -6697, -6755 (References 6-20, 6-21, 6-10, and 6-12 respectively) and the Users Manual for RESRAD, Version 6.0 (Reference 6-22).

6.4.2. Sensitivity Analysis

6.4.2.1 Input Parameter Selection Process

The dose and conceptual models are quantified by a set of input parameters. Probabilistic modules that allow the evaluation of dose as a function of parameter distributions are incorporated within RESRAD-BUILD Version 3.5 and RESRAD Version 6.5. A schematic flow diagram of the parameter selection process is provided in Figure 6-4.

6.4.2.2 Classification (Type)

The input parameters were classified as behavioral, metabolic, or physical, consistent with NUREG/CR-6697. Behavioral parameters depend on the behavior of the receptor and the scenario definition. Metabolic parameters represent the metabolic characteristics of the receptor and are independent of the scenario definition. Physical parameters are those parameters that do not change with changes to the receptor.

6.4.2.3 Prioritization

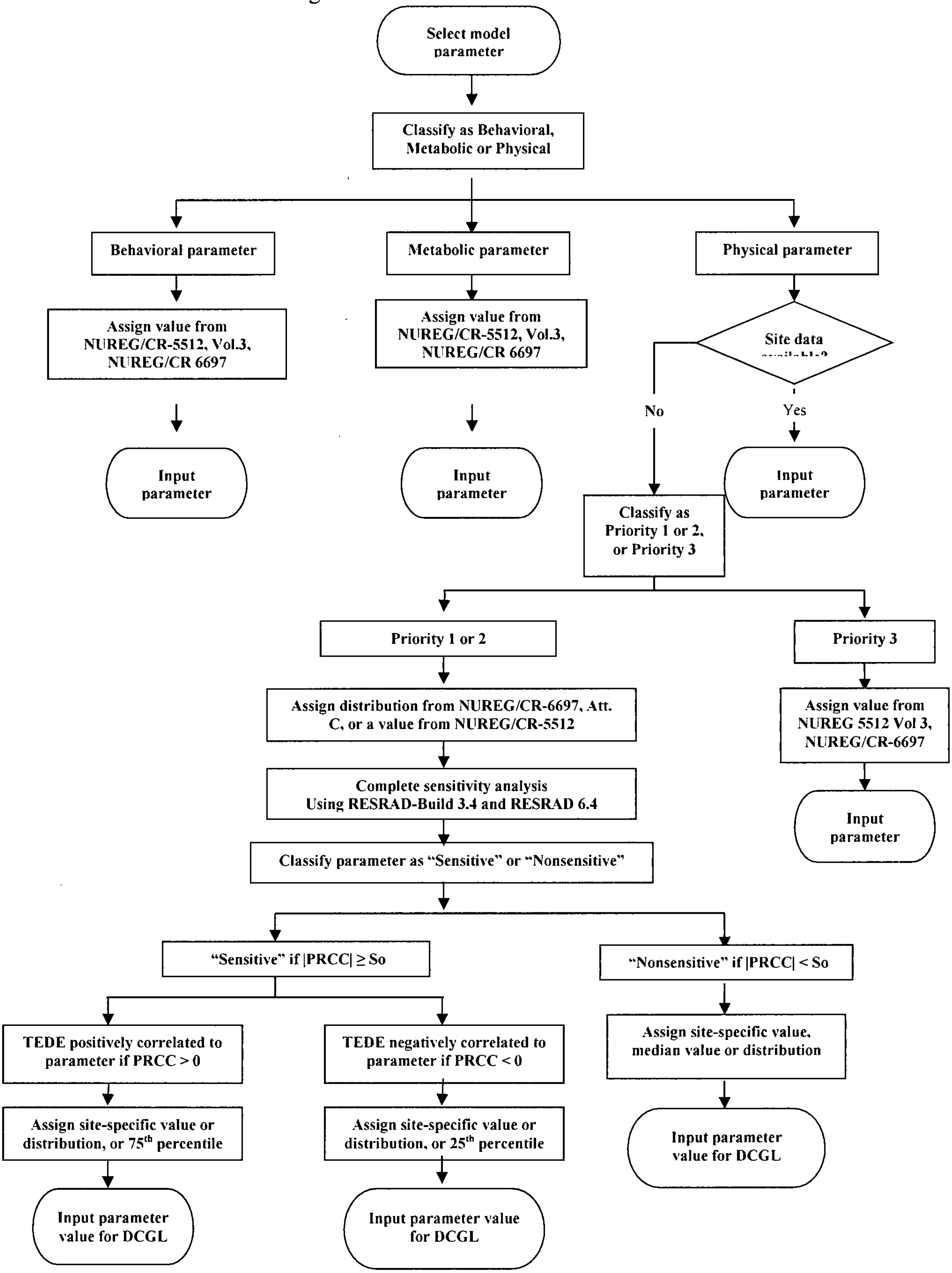
The parameters were prioritized in order of importance consistent with NUREG/CR-6697. Prioritization was based on the following items:

- The relevance of the parameter in dose calculations

- The variability of the dose as a result of changes in the parameter value
- The parameter type
- The availability of parameter-specific data

Priority 1 parameters are considered high priority; Priority 2 parameters are considered medium priority; and Priority 3 parameters are considered low priority.

Figure 6-4 Parameter Selection Process



6.4.2.4 *Treatment*

The parameters were treated as either deterministic or stochastic, depending on parameter type, priority, availability of site-specific data, and the relevance of the parameter in dose calculations. The deterministic modules of the code use a single value for input parameters and generate a single value for dose. The probabilistic modules of the code use probability distributions for stochastic input parameters and generate a range of doses.

The behavioral and metabolic parameters are treated as deterministic and were assigned values from NUREG/CR-5512, Volume 3, NUREG/CR-6697, or the applicable code's default library. Physical parameters for which site-specific data are available were also treated as deterministic.

The remaining physical parameters, for which no site-specific data are available to quantify, are classified as either Priority 1, 2, or 3. Priority 1 and 2 parameters are treated as stochastic and are assigned a probability distribution from NUREG/CR-6697 (Reference 6-10). The priority 3 physical parameters are treated as deterministic and are assigned values from NUREG/CR-5512, Volume 3, NUREG/CR-6697, or the applicable code's default library.

6.4.2.5 *Sensitivity Criteria*

In order to determine values for parameters not already assigned a value, a sensitivity analysis was performed to determine which of the stochastic parameters have an influence on the resulting dose and associated DCGLs. The analyses were performed using the probabilistic modules of RESRAD-Build, Version 3.5, and RESRAD, Version 6.5.

The stochastic parameters identified in the preceding paragraphs were generally assigned distribution types and corresponding distribution statistical parameters from NUREG/CR-6697, Attachment C. Sensitivity analyses were performed on the stochastic parameters using the assigned distributions. To perform the sensitivity analysis, the following information was required:

- **Sample Specifications:** The analyses were run using 300 observations for building surfaces, 2,000 observations for soils, and 1 repetition for both scenarios. The Latin Hypercube Sampling (LHS) technique was used to sample the probability distributions for each of the stochastic input parameters. The correlated or uncorrelated grouping option was used to preserve the prescribed correlation. Correlation coefficients were assigned to correlated parameters.
- **Sensitivity Indicator:** Sensitivity analyses were performed for each of the radionuclides. The Partial Rank Correlation Coefficient (PRCC) for the peak of the mean dose was used as a measure of the sensitivity of each parameter.
- **Sensitivity Thresholds:** For the building occupancy scenario, a parameter was identified as sensitive if the absolute value of its PRCC (|PRCC|) was greater than or equal to 0.10 and nonsensitive if the |PRCC| value was less than 0.10. For the resident farmer scenario, a parameter was identified as sensitive if the |PRCC| was greater than or equal to 0.25 and nonsensitive if the |PRCC| value was less than 0.25. These sensitivity thresholds (So) were selected based on the guidance included in NUREG/CR-6676 and NUREG/CR-6692.

6.4.2.6 Parameter Value Assignment for DCGL Determination

As discussed previously, behavioral and metabolic parameters were assigned values from NUREG/CR-5512 Volume 3, NUREG/CR-6697, or NUREG/CR-6755. When available, site data served as input for physical parameters. For Priority 3 physical parameters without site data, values from NUREG/CR-5512 Volume 3 or NUREG/CR-6697 were used.

Priorities 1 and 2 physical parameters were assigned values as follows:

- Priorities 1 and 2 physical parameters shown to be sensitive were assigned conservative values:
  - A site-specific value, or
  - The 25th or 75th percentile value of the distribution was used, respectively, depending on whether the parameter was positively or negatively

correlated with dose. Use of 25th and 75th percentiles values provides assurance that the DCGL calculations take into account the uncertainties associated with the sensitive input parameters.

- Priorities 1 and 2 physical parameters shown to be nonsensitive were assigned:
  - A distribution or site-specific value, or
  - The median value of the distribution

6.4.3. Code Output and Calculation of DCGL

RESRAD-BUILD code determines an average annual dose at the time of the peak dose in mrem/yr, whereas RESRAD code determines an annual peak of the mean dose in mrem/yr. By specifying a unit radionuclide concentration (i.e., 1 pCi/m^2 in RESRAD-BUILD or 1 pCi/g in RESRAD) to be used in conjunction with the parameters values determined by the process described previously, both codes produce a dose conversion factor (DCF). The DCF from RESRAD-BUILD is in units of mrem/yr per pCi/m^2 and the DCF from RESRAD is in units of mrem/yr per pCi/g. As suggested in NUREG-1757, DCFs based upon peak mean doses were used to calculate DCGLs with units of dpm/100 cm^2 for building surfaces and pCi/g for soil. The Humboldt Bay DCGLs correspond to the site release criterion, 25 mrem/y, and were calculated using the following equations:

For building surfaces,

$$DCGL(pCi/m^2) = \frac{25 mrem/y}{DCF\ mrem/yr/pCi/m^2}$$

Equation 6-1

$$DCGL(dpm/cm^2) = DCGL(pCi/m^2) \times 2.22 dpm/pCi \times (1m/100cm)^2$$

Equation 6-2

$$DCGL(dpm/100cm^2) = DCGL(pCi/m^2) \times 2.22 dpm/pCi \times (1m/100cm)^2 \times 100$$

Equation 6-3

For soil,

$$DCGL(pCi/g) = \frac{25mrem/y}{DCFmrem/y/pCi/g}$$

Equation 6-4

6.4.4. Calculation of Building Surface DCGL

6.4.4.1 Dose Model

The dose model used to calculate the building surface DCGLs is based upon the building occupancy scenario as defined in NUREG/CR-5512, Volumes 1, 2, 3, and NUREG-1757. The scenario assumes that the critical group consists of workers performing routine work activities in the building following license termination.

6.4.4.2 Conceptual Model

The conceptual model was based on site characteristics expected at the time of license termination. The model is composed of a room representative of rooms inside Humboldt Bay buildings expected to remain at the site. The model room was selected for the following reasons:

- Very little, if any, remediation will be required in this area and, therefore, will be most suited for occupancy
- The room is slated to be occupied by administrative personnel on the most continuous basis (i.e., will not leave to perform "rounds" or maintenance). This is the smallest room that will be continuously occupied. RAI 29

The four walls, floor, and ceiling of the room are assumed to be uniformly contaminated to equal levels. This is a conservative assumption as normally the amount of contamination on room walls and ceiling is less than that on the floor and would be expected to decrease as the distance from the floor increases.

6.4.4.3 Parameter Value Assignment

Appendix A provides the details for the determination of the room dimensions and the bases for other site-specific parameters affecting the modeling for building surfaces

DCGLs. The values and distributions assigned to all parameters for the sensitivity analyses and the bases for assigning such values and distributions are summarized in Appendix B. The time in which the maximum dose occurred was taken into account in the sensitivity analyses.

6.4.4.4 Sensitivity Analysis

The results of the sensitivity analysis performed for RESRAD-BUILD input parameters are provided in Appendix C.

6.4.4.5 DCGL Determination

The DCGL determination was performed using RESRAD-BUILD, Version 3.5. The input values, including the 25^{th} and 75^{th} percentile values for sensitive input parameters, are summarized in Appendix D. The resulting DCFs, based upon the average dose during the year that the maximum dose occurs, are provided in Appendix E. These DCGL values, which represent an annual dose of 25 mrem, were calculated using Equations 6-1 through 6-3. They are shown in Table 6-5 and provided in Appendix E.

6.4.5. Calculation of Soil DCGL

6.4.5.1 Dose Model

The DCGLs for soil were calculated using the resident farmer scenario. The residual radioactive materials were assumed to be contained in a soil layer on the property that can be used for residential and light farming activities. The average member of the critical group is the resident farmer that lives on the plant site, grows all of his/her diet onsite, and drinks water from a groundwater source onsite.

6.4.5.2 Conceptual Model

The conceptual model used in the code was based on the site characteristics expected at the time of release of the site. The model is composed of a contaminated zone underlain by an unsaturated zone underlain by a saturated zone. The contaminated zone is assumed to be at the ground surface with no cover material and the groundwater is initially uncontaminated. The model as described is consistent with that described in the RESRAD User's Manual.

6.4.5.3 *Parameter Value Assignment*

The evaluation of site/regional data and the justification of values assigned to the site-specific parameters are provided in Appendix F. The values/distributions assigned to all parameters for the sensitivity analyses and the basis for assigning such values/distributions are summarized in Appendix G.

6.4.5.4 *Sensitivity Analysis*

The results of the sensitivity analysis performed for RESRAD input parameters are provided in Appendix H.

6.4.5.5 *DCGL Determination*

The DCGL determination was performed using RESRAD Version 6.5. The input values, including the 25th and 75th percentile values for sensitive input parameters, are summarized in Appendix I. The resulting DCFs, based upon the peak of the mean doses, are provided in Appendix J. The DCGLs, which represent an annual dose equal to 25 mrem, were calculated using Equation 6-4. The DCGL values are shown in Table 6-5 and provided in Appendix J.

Table 6-5 DCGLs by Radionuclide and Medium Type

Nuclide	Building Surface DCGL (dpm/100 cm^2)	Soil DCGL (pCi/g)
Am-241	3.0E+03	2.5E+01
C-14	7.0E+06	6.3E+00
Cm-243	4.3E+03	2.9E+01
Cm-244	5.5E+03	4.8E+01
Cm-245	2.2E+03	1.7E+01
Cm-246	2.7E+03	2.5E+01
Co-60	1.3E+04	3.8E+00
Cs-137	4.6E+04	7.9E+00
Eu-152	2.7E+04	1.0E+01
Eu-154	2.5E+04	9.4E+00
H-3	1.8E+08	6.8E+02
I-129	4.9E+04	4.8E+00
Nb-94	1.9E+04	7.1E+00
Ni-59	6.3E+07	1.9E+03
Ni-63	2.4E+07	7.2E+02

Nuclide	Building Surface DCGL (dpm/100 cm^2)	Soil DCGL (pCi/g)
Np-237	2.4E+03	1.1E+00
Pu-238	3.4E+03	2.9E+01
Pu-239	3.1E+03	2.6E+01
Pu-240	3.1E+03	2.6E+01
Pu-241	1.4E+05	8.6E+02
Sr-90	9.7E+04	1.5E+00
Tc-99	9.6E+06	1.2E+01

6.5. Area Factors

6.5.1. Calculation of Area Factors

Area factors (AFs) for both building surface DCGLs and soil DCGLs may be required during final status survey activities. AF values are calculated in a step process. First, the total doses from all pathways are calculated for each radionuclide and for each area of contamination. Then, the AF values are determined from the ratio of the dose for the base case to the dose for each smaller area evaluated.

6.5.2. Calculation of Area Factors for the Building Surfaces

For the building occupancy scenario, an approach different from that used for the building surface DCGLs was applied in the computation of the area factors used to establish the $DCGL_{EMC}$. While the $DCGL_W$ is the average concentration over the entire surface area of the Humboldt Bay representative room, the $DCGL_{EMC}$ should reflect the exposure an occupant would receive from an area of elevated activity having dimensions that are much smaller than the total interior area of the room. The total surface area of contaminated sources for the Humboldt Bay representative room is 118 m^2, which includes the floor, four walls, and ceiling. The calculation of AFs assumed activity on a single surface that did not exceed 100 m^2. Elevated measurement comparisons (i.e., assessments of residual activity greater than the DCGL value) will occur only in Class 1 areas. Contamination levels exceeding DCGL values (or for a radionuclide mixture, a sum of the fraction exceeding one) are not expected in Class 2 or Class 3 survey units and, if found, would result in reclassification of the entire area (or a portion of the area) to Class 1. Accordingly, the recommended limit to the size of a Class 1 structure, 100 m^2, given in MARSSIM, was established as the upper bound (or base case) for sizes used to develop AFs for building surfaces.

The total doses for various areas of the contaminated source are calculated using RESRAD-BUILD Version 3.5. The model used in RESRAD-BUILD is similar to that used in the model for calculating building surface $DCGL_W$ values. However, only one source is modeled, instead of the five sources considered in calculating the building surface $DCGL_W$ values. The receptor is located at the source midpoint at a distance of 1 m away. All other input parameters and assumed active exposure pathways are the same as those used in the calculations for building surface DCGLs and are presented in Appendix K. Appendix L presents the radionuclide-specific area factors.

6.5.3. Calculation of Area Factors for the Soils

Area factors for the resident farmer are calculated using the RESRAD 6.5 computer code using the input parameters from the original soils analysis and a unit activity of 1 pCi/g. As the contaminated area decreases, some members of the set of ingestion pathway input parameters referred to as Contamination Fractions, also decrease, using the equations in the RESRAD Users Manual. A Contamination Fraction indicates the fraction of a person's total diet that is obtained from the contaminated area. As the contaminated area decreases below a certain size, it is reasonable to assume that the fraction of the person's total diet from the contaminated area will also decrease proportionately.

The contaminated fractions for drinking water, livestock water, irrigation water, and aquatic food are not allowed to decrease as the size of the contaminated zone decreases. Use of a value equal to 1.0 incorporates the assumption that all water used by the resident farmer comes from the site (i.e., residential well), regardless of the size of the contaminated area.

Adjustments to the contaminated fractions for plants, meat, and milk are made using equations from the RESRAD User's Manual. Values of the multiplier are listed in Appendix M as a function of the size of the contaminated zone. Appendix M provides contaminated fraction values as a function of the area of the contaminated zone.

The fraction of household water remains set at 1.0 for all sizes of contaminated zones, which is consistent with the RESRAD code input screen that does not allow deviation from the default value of 1.0.

As with buildings, elevated measurement comparisons (i.e., assessments of residual activity greater than the DCGL value) will occur only in Class 1 survey units. Contamination levels exceeding DCGL values (or for a radionuclide mixture, a sum of the fraction exceeding one) are not expected in Class 2 or Class 3 open land areas and, if found, would result in reclassification of the entire area (or a portion of the area) to a Class 1 open land survey unit. Accordingly, the recommended limit to the size of a Class 1 open land survey unit, 2,000 m^2 given in MARSSIM, was established as the upper bound (or base case) for sizes used to develop AFs for the soil DCGLs.

The total doses corresponding to the various areas of the contaminated zone are calculated using the input parameter values listed in Appendix M. Appendix N provides the soil AF values by radionuclide and area. Table 6-6 provides building surface area factors and Table 6-7 provides soil area factors.

Table 6-6 Building Surface Area Factors

	Area Factor Value:										
(m^2)	Am-241	C-14	Cm-243	Cm-244	Cm-245	Cm-246	Co-60	Cs-137	Eu-152	Eu-154	H-3
1	9.7E+01	9.7E+01	8.9E+01	1.0E+02	3.9E+01	6.4E+01	1.3E+01	1.5E+01	1.3E+01	1.3E+01	1.0E+02
2	4.9E+01	4.9E+01	4.5E+01	5.0E+01	2.1E+01	3.3E+01	7.2E+00	8.2E+00	7.2E+00	7.2E+00	5.0E+01
3	3.3E+01	3.3E+01	3.0E+01	3.3E+01	1.5E+01	2.3E+01	5.3E+00	6.0E+00	5.3E+00	5.3E+00	3.3E+01
4	2.5E+01	2.4E+01	2.3E+01	2.5E+01	1.2E+01	1.8E+01	4.3E+00	4.9E+00	4.3E+00	4.3E+00	2.5E+01
5	2.0E+01	2.0E+01	1.9E+01	2.0E+01	9.9E+00	1.5E+01	3.7E+00	4.2E+00	3.7E+00	3.7E+00	2.0E+01
6	1.6E+01	1.6E+01	1.6E+01	1.7E+01	8.6E+00	1.2E+01	3.3E+00	3.8E+00	3.3E+00	3.3E+00	1.7E+01
8	1.2E+01	1.2E+01	1.2E+01	1.2E+01	6.9E+00	9.7E+00	2.8E+00	3.2E+00	2.8E+00	2.8E+00	1.2E+01
10	9.9E+00	9.9E+00	9.5E+00	1.0E+01	5.9E+00	8.0E+00	2.5E+00	2.8E+00	2.5E+00	2.5E+00	1.0E+01
50	2.0E+00	2.0E+00	2.0E+00	2.0E+00	1.8E+00	1.9E+00	1.2E+00	1.3E+00	1.2E+00	1.2E+00	2.0E+00
100	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00
	Area Factor Value:										
(m^2)	I-129	Nb-94	Ni-59	Ni-63	Np-237	Pu-238	Pu-239	Pu-240	Pu-241	Sr-90	Tc-99
1	6.5E+01	1.3E+01	1.0E+02	1.0E+02	8.9E+01	1.0E+02	1.0E+02	1.0E+02	9.8E+01	9.0E+01	8.7E+01
2	3.4E+01	7.2E+00	5.0E+01	5.0E+01	4.5E+01	5.0E+01	5.0E+01	5.0E+01	4.9E+01	4.5E+01	4.4E+01
3	2.3E+01	5.3E+00	3.3E+01	3.3E+01	3.0E+01	3.3E+01	3.3E+01	3.3E+01	3.3E+01	3.0E+01	3.0E+01
4	1.8E+01	4.3E+00	2.5E+01	2.5E+01	2.3E+01	2.5E+01	2.5E+01	2.5E+01	2.5E+01	2.3E+01	2.3E+01
5	1.5E+01	3.7E+00	2.0E+01	2.0E+01	1.8E+01	2.0E+01	2.0E+01	2.0E+01	2.0E+01	1.9E+01	1.8E+01
6	1.3E+01	3.3E+00	1.7E+01	1.7E+01	1.6E+01	1.7E+01	1.7E+01	1.7E+01	1.7E+01	1.6E+01	1.5E+01
8	9.7E+00	2.8E+00	1.3E+01	1.2E+01	1.2E+01	1.3E+01	1.2E+01	1.2E+01	1.2E+01	1.2E+01	1.2E+01
10	8.0E+00	2.5E+00	1.0E+01	1.0E+01	9.4E+00	1.0E+01	1.0E+01	1.0E+01	9.9E+00	9.5E+00	9.4E+00
50	1.9E+00	1.2E+00	2.0E+00	2.0E+00	2.0E+00	2.0E+00	2.0E+00	2.0E+00	2.0E+00	2.0E+00	2.0E+00
100	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00

Table 6-7 Area Factors for Soils

ROC	Area Factor for Area Contaminated Zone (m^2):							
	2000	**1000**	**500**	**100**	**50**	**10**	**5**	**1**
Am-241	1.0E+00	1.0E+00	2.0E+00	8.7E+00	1.6E+01	4.9E+01	7.7E+01	1.9E+02
C-14	1.0E+00	1.5E+00	4.0E+00	4.2E+01	1.1E+02	1.0E+03	2.5E+03	1.8E+04
Cm-243	1.0E+00	1.0E+00	1.6E+00	3.4E+00	4.3E+00	7.3E+00	1.1E+01	3.2E+01
Cm-244	1.0E+00	1.0E+00	2.0E+00	9.7E+00	1.9E+01	7.6E+01	1.2E+02	2.8E+02
Cm-245	1.0E+00	1.0E+00	1.9E+00	6.2E+00	9.2E+00	1.9E+01	3.0E+01	8.1E+01
Cm-246	1.0E+00	1.0E+00	2.0E+00	9.7E+00	1.9E+01	7.6E+01	1.3E+02	2.8E+02
Co-60	1.0E+00	1.0E+00	1.1E+00	1.3E+00	1.4E+00	2.2E+00	3.3E+00	1.0E+01
Cs-137	1.0E+00	1.0E+00	1.3E+00	1.7E+00	1.9E+00	3.0E+00	4.5E+00	1.4E+01
Eu-152	1.0E+00	1.0E+00	1.0E+00	1.1E+00	1.3E+00	2.0E+00	3.0E+00	9.1E+00
Eu-154	1.0E+00	1.0E+00	1.0E+00	1.2E+00	1.3E+00	2.0E+00	3.0E+00	9.2E+00
H-3	1.0E+00	1.1E+00	2.1E+00	1.0E+01	2.1E+01	1.0E+02	2.0E+02	9.3E+02
I-129	1.0E+00	1.1E+00	2.2E+00	1.1E+01	2.2E+01	9.9E+01	1.9E+02	8.3E+02
Nb-94	1.0E+00	1.0E+00	1.0E+00	1.2E+00	1.3E+00	2.0E+00	3.0E+00	9.0E+00
Ni-59	1.0E+00	1.2E+00	2.3E+00	1.2E+01	2.3E+01	1.2E+02	2.3E+02	1.2E+03
Ni-63	1.0E+00	1.2E+00	2.3E+00	1.2E+01	2.3E+01	1.2E+02	2.3E+02	1.2E+03
Np-237	1.0E+00	1.0E+00	2.0E+00	9.0E+00	1.6E+01	5.6E+01	9.8E+01	3.5E+02
Pu-238	1.0E+00	1.0E+00	2.0E+00	9.7E+00	1.9E+01	7.6E+01	1.3E+02	2.8E+02
Pu-239	1.0E+00	1.0E+00	2.0E+00	9.7E+00	1.9E+01	7.6E+01	1.3E+02	2.9E+02
Pu-240	1.0E+00	1.0E+00	2.0E+00	9.7E+00	1.9E+01	7.6E+01	1.3E+02	2.9E+02
Pu-241	1.0E+00	1.0E+00	2.0E+00	8.8E+00	1.6E+01	4.9E+01	7.8E+01	1.9E+02
Sr-90	1.0E+00	1.0E+00	2.0E+00	1.0E+01	2.0E+01	9.9E+01	2.0E+02	9.6E+02
Tc-99	1.0E+00	1.0E+00	2.0E+00	1.0E+01	2.0E+01	1.0E+02	2.0E+02	1.0E+03

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